Fitzpatrick Station

2010 NRC Written Exam

RO Portion

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

100	90	80	70	60
50	40	30	20	10
9	8	7	6	5
4	3	2	1	0

FORM NO. 888-E

IMPORTANT

USE NO. 2 PENCIL ONLY

- MAKE DARK MARKS
- ERASE COMPLETELY TO CHANGE
- EXAMPLE: A B C D E

TO USE SUBJECTIVE SCORE FEATURE:

- Mark total possible subjective points
- Only one mark per line on key
- 1 to 3 points maximum

EXAMPLE OF STUDENT SCORE:

PART 1

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SUBJECT		TEST NO.	
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TEST RECORD	
PART 1	
PART 2	
TOTAL	

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	(T)	(F)			KEY
			3		
1	A		C	D	E
2		B	C	D	E
3	A	B		D	E
4		B	C	D	E
5	A	B		D	E
6	A	B	C		E
7	A		C	D	E
8	A	B		D	E
9		B	C	D	E
10	A	B		D	E
11		B	C	D	E
12	A	B		D	E
13		B	C	D	E
14	A		C	D	E
15	A	B	C		E
16		B	C	D	E
17	A	B	C		E
18	A	B		D	E
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43	A	B		D	E
44	A	B	C		E
45	A	B	C		E
46	A		C	D	E
47	A		C	D	E
48		B	C	D	E
49	A	B	C		E
50	A		C	D	E

FEED THIS DIRECTION

RO Exam Key

07 08 09 M12 3806 999 12 11 10 9 8 7 6 5 4 3 2 1

IMPORTANT

USE #2 PENCIL ONLY

- MAKE DARK MARKS
- ERASE COMPLETELY TO CHANGE
- EXAMPLE: A B D

TO USE SUBJECTIVE SCORE FEATURE:

- Mark total possible subjective points
- Only one mark per line on key
- 153 points maximum

EXAMPLE OF STUDENT SCORE:

NAME	RO Exam Key		
SUBJECT		TEST NO.	
DATE		PERIOD	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

PART 2

	(T)	(F)			KEY
	1	2	3	4	5
51	A	B	C	D	E
52	A	B	C	D	E
53	A	B	C	D	E
54	A	B	C	D	E
55	A	B	C	D	E
56	A	B	C	D	E
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93	A	B	C	D	E
94	A	B	C	D	E
95	A	B	C	D	E
96	A	B	C	D	E
97	A	B	C	D	E
98	A	B	C	D	E
99	A	B	C	D	E
100	A	B	C	D	E

FEED THIS DIRECTION

8 10054 00013 6

RO Exam Key

QUESTION 1.

The plant is operating at 90% power with both Recirc Pumps in operation, when a malfunction occurs in the recirculation controller system.

The plant stabilizes at the listed conditions:

- Reactor power is 60% and steady.
- Reactor Water Recirculation (RWR) flow is 47% of rated core flow and steady.

Which **ONE** of the following actions is required per AOP-8, "Loss or Reduction of Reactor Coolant Flow"?

A. Manually scram the reactor.

B. Raise recirculation pump speed or insert rods.

C. Shutdown per OP-65, "Startup and Shutdown".

D. Lower recirculation pump speed.

K&A # 295001 AK3.02
Importance Rating RO - 3.7; SRO – 3.8

QUESTION 1

K&A Statement: Knowledge of the operational implications of Power/Flow distribution as it applies to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: REACTOR POWER RESPONSE

Justification:

A. **Incorrect but plausible** if the operator uses the power flow map, then plant is in manual scram zone.

B. **Correct:** plant is in the buffer zone. Per AOP-8 exit restricted zone by raising recirc flow or lower power by inserting cram rods.

C. **Incorrect:** because AOP-8 requires the restricted zone to be exited and directs actions to either raise recirc flow or lower power by inserting the cram rods to 00. This answer is plausible because it would have the operator exit the restricted zone however it is not correct because it is not required to shutdown the reactor.

D. **Incorrect:** because AOP-8 requires the restricted zone to be exited and directs actions to either raise recirc flow or lower power by inserting the cram rods to 00.

References: AOP-8, "Loss or Reduction of Reactor Coolant Flow", Rev. 31

Student Ref: RAP – 7.3.16 (Rev. 44); Power/Flow Map with labeling removed

Learning Objective: SDLP-02HR14.pdf Reactor Recirc lesson plan 1.09
Given a set of plant conditions, describe the effect that a malfunction/loss of the Reactor Recirculation System may have on the following:
i. Reactor Power Versus Core Flow Operating Map Single Pump Trip

Question source: Modified NMP 1; 10/08 exam; question 53

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR 41.8

QUESTION 2.

The plant is operating at 100% power with 'D' EDG loaded for surveillance testing in a normal electrical lineup when the following annunciators alarm:

- 09-8-3-31 BUS 10400 RESERVE SUPP BKR 10412 TRIP
- 09-8-3-32 BUS 10400 NORM SUPP BKR 10402 TRIP
- 09-8-4-31 EDG D GEN LOCKOUT

Assume **NO** operator actions have been taken and all equipment functions as designed.

What will be the status of the 4 kV emergency bus breakers after this event?

A. ACB 10602 (EDG B Load Bkr) Closed,
ACB 10612 (EDG D Load Bkr) Open,
ACB 10404 (Bus 10400 – 10600 Tie Bkr) Open,
ACB 10614 (Bus 10400 – 10600 Tie Bkr) Open,
ACB 10604 (EDG B & D Tie Bkr) Open.

B. ACB 10602 (EDG B Load Bkr) Closed,
ACB 10612 (EDG D Load Bkr) Open,
ACB 10404 (Bus 10400 – 10600 Tie Bkr) Open,
ACB 10614 (Bus 10400 – 10600 Tie Bkr) Closed,
ACB 10604 (EDG B & D Tie Bkr) Closed.

C. ACB 10602 (EDG B Load Bkr) Closed,
ACB 10612 (EDG D Load Bkr) Open,
ACB 10404 (Bus 10400 – 10600 Tie Bkr) Closed,
ACB 10614 (Bus 10400 – 10600 Tie Bkr) Open,
ACB 10604 (EDG B & D Tie Bkr) Closed.

D. ACB 10602 (EDG B Load Bkr) Open,
ACB 10612 (EDG D Load Bkr) Closed,
ACB 10404 (Bus 10400 – 10600 Tie Bkr) Open,
ACB 10614 (Bus 10400 – 10600 Tie Bkr) Open,
ACB 10604 (EDG B & D Tie Bkr) Open.

K&A # 295003; AK3.01
Importance Rating RO – 3.3; SRO – 3.5

QUESTION 2.

K&A Statement: Knowledge of the reasons for the following as they apply to PARTIAL OR COMPLETE LOSS OF AC POWER: MANUAL AND AUTO BUS TRANSFER

Justification: Explanation: Stem results in a loss of power to the 10400 Bus, the EDGs B will auto start but will not force parallel because D EDG was already operating. (ACB 10604 closes initially then subsequently opens when a EDG > 90% and closes onto the 10600 bus) The resultant bkr alignment per OP-22 Step G.2.2 will be 10612 open, 10602 closed, 10404 open, 10604 open, 10614 open.

A. **Correct** - Breakers are in correct position per OP-22 Step G.2.2. except for 'D' EDG Load Breaker tripped open because of lockout.

B. **Incorrect** - ACB 10614 (Bus 10400 - 10600 Tie Bkr) should be Open vice Shut and 10604 should be open.

C. **Incorrect** - ACB 10604 (EDG B & D Tie Bkr) should be Open vice Shut and 10614 tripped open.

D. **Incorrect** - ACB 10602 should be closed vice open and 'D' EDG Load Breaker tripped open because of the lockout.

References: OP-22 Rev.53 Step G.2.2 Student Ref: None
AOP-17 Rev 14

Learning Objective: N/A

Question source: Modified to 'B' & 'D' EDGs

Question History: From JAF exam 3/08; question 2

Cognitive level: Comprehensive/Analysis:

10CFR 41.5/45.6

QUESTION 3.

The plant is in MODE 1 with reactor power at 100%.

All systems are in a normal lineup per procedures.

Subsequently, a loss of DC Power System 'A' occurs with a resulting reactor scram.

What is the reason for the reactor scram?

A. Main turbine trip from high RPV water level, due to loss of DC power to the selected Rx water level column.

B. Inboard MSIV closure due to loss of DC power to the solenoids.

C. Main turbine trip due to loss of DC power to EHC trip logic.

D. Low RPV water level due to loss of DC power to feedwater and steam flow instruments.

K&A # 295004; AK3.03
Importance Rating RO – 3.1; SRO – 3.5

QUESTION 3.

K&A Statement: Knowledge of the reasons for the following responses as they apply **to PARTIAL OR COMPLETE LOSS OF D.C. POWER: REACTOR SCRAM**

Justification: Per AOP-45 , Loss of DC Power System 'A- Section C- AUTOMATIC

A. **Incorrect but plausible:** Loss of DC power system1 A affects Rx Wtr Lvl 06LI-94A & C, not RX WTR LVL 06LI-94B, which is the normal selected column.

B. **Incorrect but plausible:** To lose power to the Inboard MSIV solenoids would require 29AOV-80A-D MSIV DC Inboard Solenoids powered from 71 DC-A2 & 29AOV-80A-D MSIV AC Inboard Solenoids powered from RPS "A" to be de-energized. Power has been lost to the DC solenoids however, RPS MG Set 1A is fed from MCC-251 fed via 10500 which would not lose power till the generator trips & the 10500 buss loses power as it will not transfer due to loss of DC control power to the 10500 breakers.

C. **Correct:** Main turbine trip from loss of DC power to EHC trip logic,
If Rx power >29 %, a Rx scram will occur from main turbine stop valve closure.

D. **Incorrect but plausible:** Loss of DC Power System 'A' results in a downscale failure of FDWTR flow 06FI-89A & Stm flow 06FI-88A & C. Control power is lost to RFP 'A' but RFP 'B' is not affected. The Rx would have scrammed due to response noted in the Correct answer prior to any scrams caused by water level from Steam /FW flow mismatch & loss of RFPT A speed control.

References: AOP-45 R 9 Student Ref: NONE
AOP-21 R 21, AOP-16 R 14

Learning Objective: SDLP-71BR8; DC 1.09 -8 & 9 — Given a set of plant conditions, describe the effect that a loss or malfunction of each of the following DC Electrical Systems may have on the listed systems: EHC and Main Turbine

Question source: Modified

Question History: From Fitz exam 3/08; question 3

Cognitive level: Comprehensive/Analysis:

10CFR 41.5 / 45.6

QUESTION 4.

Fill in the blanks.

Per OP-9, "Main Turbine", a reactor scram _____ required to be jumpered to conduct main turbine shell warming _____.

A. is;
because the turbine-initiated reactor scrams may be actuated by first stage pressure

B. is NOT;
because the main turbine trip signal is reset before admitting steam for shell warming

C. is;
to prevent a reactor scram when the turbine trip is tested

D. is NOT;
because the turbine-initiated reactor scrams are bypassed by first stage pressure

K&A # 295005; G2.1.32
Importance Rating RO - 3.8; SRO – 4.0

QUESTION 4.

K&A Statement: Ability to explain and apply system limitations and precautions as they apply to MAIN TURBINE GENERATOR TRIP: **MAIN GENERATOR TRIP**

Justification:

A. **Correct**: Shell warming could raise first stage pressure above the scram bypass setpoint. With first stage pressure greater than setpoint and turbine stop valves less than 90% open, a reactor scram will be initiated.

B. **Incorrect:** Jumpering of first stage pressure reactor scram is required by the shell warming procedure. **Plausible because main turbine trip signal is reset prior to admitting steam and the scram is based on automatically shutting down the reactor upon a main turbine trip in anticipation of potential rapid pressure spike causing power spike associated when voids collapse.**

C. **Incorrect**: The turbine trip is not tested during or following shell warming alignment. **Plausible because a scram is required to be jumpered and examinee may think the manual turbine trip capability is procedurally tested during or following shell warming alignment.**

D. **Incorrect:** Jumpering of first stage pressure reactor scram is required by the shell warming procedure. While this level transient will occur, it is not part of the basis for the scram. **Plausible because turbine-initiated reactor scrams are bypassed automatically by low first stage pressure. Examinee may not understand that shell warming has the potential to raise first stage pressure above the scram setpoint, initiating a scram because stop valves are not fully open.**

References: RPS; SDLP-05R16.pdf Student Ref: None
OP-9 R52

Learning Objective: RPS; SDLP-05R16 1.07.a.10 — State the expected plant response and appropriate operator actions for the following: Main Turbine Trip

Question source: Modified Question

Question History: From Limerick exam 10/06; question 1

Cognitive level: Memory or Fundamental Knowledge

10CFR (CFR:41.7 / 45.6)

QUESTION 5.

Given the following:

- The plant is operating at 15% power, with a plant startup in progress.
- A SINGLE control rod SCRAM just occurred as a result of a troubleshooting error while investigating a SCRAM test switch problem.

Which of the following groups of annunciators are consistent with this condition?

A. CRD ACCUM PRESS LO OR LVL HI (09-5-1-43)
ROD DRIFT (09-5-2-3)
SCRAM AIR HDR PRESS HI OR LO (09-5-1-54)

B. ROD DRIFT (09-5-2-3)
RWM ROD BLOCK RPIS INOP (09-5-2-01)
SCRAM AIR HDR PRESS HI OR LO (09-5-1-54)

C. CRD ACCUM PRESS LO OR LVL HI (09-5-1-43)
ROD DRIFT (09-5-2-3)
RWM ROD BLOCK RPIS INOP (09-5-2-01)

D. CRD ACCUM PRESS LO OR LVL HI (09-5-1-43)
ROD OVERTRAVEL (09-5-2-04)
RWM ROD BLOCK RPIS INOP (09-5-2-01)

K&A # 295006; AK2.03
Importance Rating RO-3.7; SRO- 3.8

QUESTION 5.

K&A Statement: Knowledge of the interrelations between SCRAM and the following: CRD HYDRAULIC

Justification:

A. **Incorrect but plausible:** SCRAM Air HEADER LOW PRESSURE will not come in on a single rod scram. 2 out of 3 are correct. SCRAM AIR HEADER LOW PRESSURE comes in on a full scram.

B. **Incorrect but plausible:** SCRAM Air HEADER LOW PRESSURE will not come in on a single rod scram. 2 out of 3 are correct. SCRAM AIR HEADER LOW PRESSURE comes in on a full scram.

C. **Correct:** CRD ACCUM TROUBLE will come in due to low accumulator pressure following the rod scram. ROD DRIFT will come in since the rod is moving without an RMCS command. ROD OUT MOTION BLOCK comes from the RWM and/or SDV not drained.

D. **Incorrect but plausible:** ROD OVERTRAVEL is indicative of an uncoupled control rod NOT a scrammed rod.

References: SDLP-03F R10; RMCS
Sdlp03C; CRD HYDRAULICS
OP-26, "RMCS" R23
Student Ref: None

Learning Objective: SDLP-03F R10; RMCS; 1.08
List the Reactor Manual Control System Administrative Limitation setpoints which effect the below listed components:
a. Rod Drift

Question source: From HC 9/07 exam question 75

Question History: None

Cognitive level: Comprehensive/Analysis

10CFR 41.8 to 41.10

QUESTION 6.

With the plant operating at 100% power a large smoky fire occurs in the Control Room and the CRS announces entry into AOP-43, "Plant Shutdown From Outside the Control Room", for a required control room evacuation.

Per AOP-43, which **ONE** of the following **subsequent** actions is required to be performed within 2 hours?

A. Verify closed 'A' Battery Charger DC Breaker.

B. Place the Reactor Mode Switch in Shutdown.

C. Transfer RPS MG Set to ALT per OP-18, "Reactor Protection System".

D. Align 'B' Battery/Charger Rooms emergency ventilation per OP-59A, "Battery Room Ventilation".

K&A # 295016; AA1.05
Importance Rating RO – 2.8 ; SRO – 2.9

QUESTION 6.

K&A Statement: Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT:
D.C. ELECTRICAL DISTRIBUTION

Justification:

A. **Incorrect but plausible**: guidance is for breaker 71BCB-2A-B03 is to be opened in AOP-43.

B. **Incorrect:** since the note in Attachment 1 of AOP-43(pg16) has the operator keep the mode switch in RUN **plausible**: if the candidate thinks that all normal AOP-1 actions are applicable.

C. **Incorrect but plausible:** guidance for RPS is to trip the running RPS MG sets.

D. **Correct** - guidance contained in AOP-43; step F.5

References: OP18 R28, OP-59A R8, AOP-43 R33 Student Ref: None

Learning Objective: N/A

Question source: MODIFIED

Question History: From last Fitz exam 3/08; question 6

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7/ 45.6

QUESTION 7.

The Plant is operating at 90% power with one Reactor Building Closed Loop Cooling (RBCLC) pump tagged out of service.

An electrical problem causes the two running RBCLC pumps to trip.

Operators have the ability to restore cooling via Emergency Service Water to **EACH** of the following **EXCEPT**:

A. Drywell Ventilation Coolers

B. Reactor Water Cleanup Pump Coolers

C. Control Rod Drive Hydraulic Pump Coolers

D. Drywell Equipment Drain Sump Cooler

K&A # 295018 AA1.01
Importance Rating RO -3.3; SRO 3.4

QUESTION 7.

K&A Statement: 295018 Partial or Complete Loss of Component Cooling Water Ability to operate and /or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: BACKUP SYSTEMS

Justification:

A. **Incorrect** – Can be manually aligned for cooling with ESW

B. **CORRECT** - RWCU Pump Coolers are **NOT** supplied by ESW

C. **Incorrect** - Can be manually aligned for cooling with ESW

D. **Incorrect** - Can be manually aligned for cooling with ESW

References: AOP-11 R15 Student Ref: None

Learning Objective: SDLP-12 R-16 RWCU 1.10 — Given a set of plant conditions, describe the effect that a loss of the following may have on Reactor Water Cleanup System:
a. RBCLC

Question source: From last Fitzpatrick exam 3/08;question 7

Question History: From last Fitzpatrick exam 3/08;question 7

Cognitive level: Memory or Fundamental knowledge:

10CFR 55.41 & 55.43

QUESTION 8.

The reactor is operating at 60% power during a rod sequence exchange when a loose fitting has resulted in a loss of instrument air to the in-service Control Rod Drive (CRD) Flow Control Valve.

Determine which of the following conditions could result from this instrument air loss.

A. Control Rod Drive accumulator alarms due to low pressure

B. High rod speeds during control rod withdraw

C. Control Rod Drive alarms due to high temperatures

D. Control Rods begin to drift due to excessive flow

K&A # 295019; AA2.02
Importance Rating RO – 3.6; SRO – 3.7

QUESTION 8.

K&A Statement: Ability to determine and/or interpret the following as thy apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: STATUS OF SAFETY RELATED INSTRUMENT AIR SYSTEM LOADS.

Justification:

A. **Incorrect but plausible:** The charging header maintains pressure on the accumulators, and the charging header taps off upstream of the FCV, so accumulator pressure will not go down.

B. **Incorrect but plausible:** The drive header is downstream of the FCV, so rod withdraw speeds will be slower rather than faster.

C. **Correct:** The FCV fails to the minimum position on a loss of air. The cooling water header is downstream of the FCV. So, the failure mode results in decreased flow to the cooling water header and temperatures will **RISE** resulting in CRD high temperature alarms.

D. **Incorrect but plausible:** The failure results in low flow. The failure mode does not cause an increased DP across the drive piston, so there is no motive force to cause the rods to drift.

References: SDLP-03C\SDLP-03C R15 page 20 and 21 Student Ref: None

Learning Objective: 1.05.a.3 Describe the purpose/function and operation of the CRD Hydraulic System components listed for the following conditions:
3. Flow Control Valves

Question source: INPO BANK

Question History: From HC exam 9/07 question 8

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.10/ 43.5/ 45.13

QUESTION 9.

The Unit is in MODE 5. The following conditions are present:

- A loss of shutdown cooling has occurred.
- The cavity is flooded.
- The cavity and spent fuel pool gates are installed.
- Radwaste is available.
- Condensate transfer keep-full is available.

For the conditions given, which of the following should be used as a method of alternate decay heat removal in accordance with AOP-30, "Loss of Shutdown Cooling"?

A. RWCU Blowdown Mode

B. Fuel Pool Cooling

C. Decay Heat Removal

D. Fuel Pool Cooling Assist

K&A # 295021; AA1.04
Importance Rating RO – 3.7 ; SRO – 3.7

QUESTION 9.

K&A Statement: Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: ALTERNATE HEAT REMOVAL METHODS

Justification:

A. **Correct:** Only available method of decay heat removal available. Gates installed, make up source available and radwaste available.

B. **Incorrect but plausible:** Fuel Pool Cooling gates are installed between cavity and spent fuel pool.

C. **Incorrect but plausible:** Decay heat removal needs gates removed between cavity and spent fuel pool.

D. **Incorrect but plausible:** Fuel Pool Cooling Assist mode, RHR and RHRSW must be available and gates are installed between cavity and spent fuel pool.

References: AOP-30; attachment 3 and 5, R19 Student Ref: None
OP-13D; RHR SHUTDOWN COOLING

Learning Objective: N/A

Question source: New

Question History: New

Cognitive level: Comprehensive/Analysis:

10CFR 41.7 / 45.6

QUESTION 10.

Plant conditions are as follows:

- MODE 5
- Fuel moves are in progress
- A fuel assembly immediately adjacent to SRM 'C' is being lowered into the core
- SRM 'C' count rate increases from 70 cps to 300 cps
- Remaining SRMs continue to indicate 70 to 80 cps
- Refueling Bridge SRO has just reported that the fuel bundle is half-way inserted to the seated position in the correct location

Which **ONE** of the following describes required action(s) of RAP 7.1.04C, "Neutron Instrumentation Monitoring During In-core Fuel Handling" and the basis for this action?

A. Monitor count rate while completing the bundle insertion to ensure rate remains below 560 cps. A count rate beyond 560 cps may reflect an inadvertent criticality.

B. Seat the fuel bundle and request Reactor Engineering to confirm that count rate is within expected range.

C. Notify Refuel Bridge SRO to stop lowering the bundle. This action addresses an unexpected count rate increase.

D. Evacuate personnel from line of sight of the reactor vessel. This action ensures personnel are safe from the effects of an inadvertent criticality.

K&A # 295023; AK1.02
Importance Rating RO – 3.2; SRO – 3.6

QUESTION 10.

K&A Statement: Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: SHUTDOWN MARGIN

Justification:

A. **Incorrect but plausible:** because the evolution should be stopped if count rate has tripled. Plausible if the candidate thinks that three doublings (560 cps) is acceptable.

B. **Incorrect but plausible:** because the evolution should be stopped if count rate has tripled. Plausible if the candidate does not know that this count rate is unexpected for inserting an assembly.

C. **Correct:** RAP-7.1.04C Step 8.5 is for loading fuel immediately adjacent to a SRM and states immediately stop the refueling evolution if the count rate triples.

D. **Incorrect but plausible:** since this would be the appropriate action if after stopping the fuel movement the count rate continued to rise. However, stable counts indicate a subcritical condition and evacuation would not be required.

References: RAP-7.1.04C, Rev 5
RAP-7.1.04B, Rev 24

Student Ref: None

Learning Objective: SDLP-8a R10; Refueling; L.O. 1.06.b and 1.13.b

Question source: Modified

Question History: Limerick exam 10/08; question 10

Cognitive level: Comprehensive/Analysis:

10CFR 41.7 / 45.6

QUESTION 11.

Which one of the following describes the consequences of spraying the drywell if conditions are in the unacceptable region of the Drywell Spray Initiation Limit Curve in EP-1, "EOP Entry and Use"?

A. Evaporative cooling results in an immediate, rapid, and large reduction in drywell pressure which could cause a loss of primary containment integrity due to drywell to atmosphere pressure being negative.

B. The cold spray water will put excessive thermal stress on the drywell, which may lead to structural failure of the primary containment.

C. Convective cooling results in an immediate, rapid, and large reduction in drywell pressure which could cause a loss of primary containment integrity due to drywell to atmosphere pressure being negative.

D. The steam produced by spraying cold water into a superheated atmosphere could over pressurize the primary containment.

K&A # 295024; EK3.01
Importance Rating RO – 3.6; SRO – 4.0

QUESTION 11.

K&A Statement: **EK3.01 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : Drywell spray operation**

Justification:

A. Correct – IAW with the reference, the initiation of drywell sprays will result in a large drop in primary containment pressure due to evaporative cooling. This drop in pressure can occur faster than can be compensated for by the vacuum relief system, and could result in challenging primary containment integrity.

B. Incorrect because the concern is not thermal stresses. Plausible if the candidate does not know the mechanism for drywell failure.

C. Incorrect because convective cooling does not cause a rapid lowering of drywell pressure, plausible if candidate does not realize the concern is if the drywell is superheated drywell spray droplets will evaporate instantaneously causing the large drop in drywell pressure.

D. Incorrect because drywell pressure will rapidly lower and not overpressurize the drywell. Plausible if the candidate does not know the failure mechanism.

References: MIT-301.11E Student Ref: None

Learning Objective: EA 4.05

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.5

QUESTION 12.

The plant was operating at 100% power when a transient was initiated by a turbine trip without bypass valves. Indicated reactor pressure peaked at 1350 psig.

The following plant conditions exist:

- The reactor failed to scram
- Reactor level is currently being maintained between +20 and +40 inches per EOP-3, "Failure To Scram"

Which **ONE** of the following is correct concerning Technical Specification safety limits?

A. No safety limits have been exceeded.

B. **ONLY** the safety limit for reactor water level has been exceeded.

C. **ONLY** the safety limit for reactor coolant system pressure has been exceeded.

D. The safety limits for reactor water level **AND** reactor coolant system pressure have **BOTH** been exceeded.

K&A # 295025; G.2.2.22
Importance Rating RO – 4.0; SRO – 4.7

QUESTION 12.

K&A Statement: G.2.2.22 Knowledge of limiting conditions for operations and safety limits: as it applies to High Reactor Pressure

Justification:

A. Incorrect because the safety limit for pressure has been exceeded. Plausible if the candidate does not know the safety limit for reactor pressure.

B. Incorrect because the safety limit for pressure has been exceeded. Plausible if the candidate does not know the safety limit for reactor reactor level is TAF and EOP-3 does not direct to lower below the safety limit.

C. Correct : The safety limit is 1325 psig per Technical Specifications and has been exceeded.

D. Incorrect because the safety limit for level has not been exceeded. Plausible if the candidate does not know the safety limit for reactor level.

References: TS 2.1.2, SDLP-02D Student Ref: None

Learning Objective: SDLP-02D,

Question source: Brunswick 2008 RO Exam

Question History: Q60 Brunswick NRC Exam

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.5

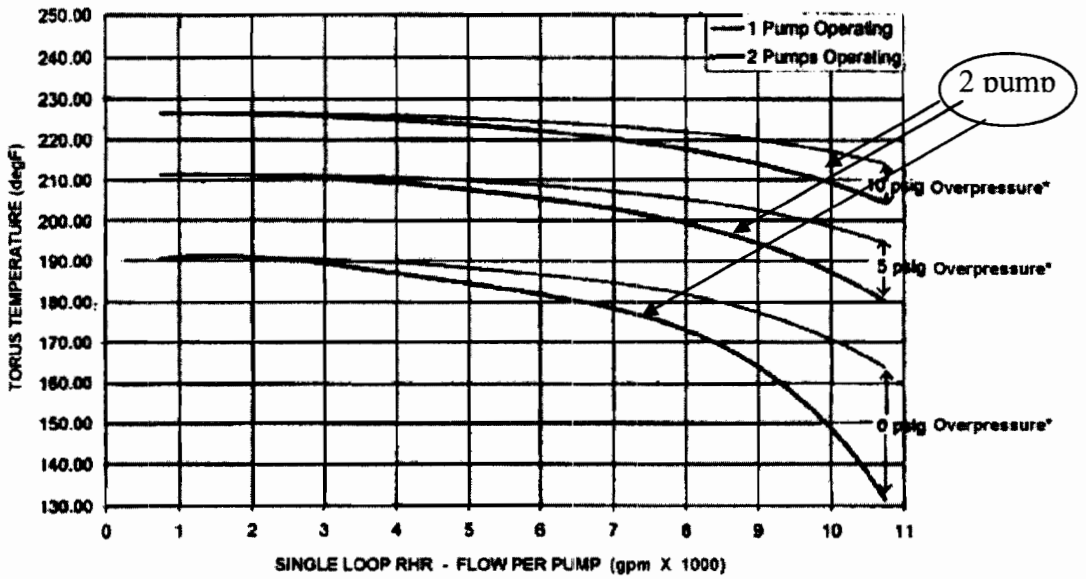
QUESTION 13.

The following plant conditions exist with RHR aligned for low pressure injection:

- Torus pressure 2.9 psig
- Torus level 10.72 feet
- RHR Pump 10P-3A flow 0 gpm
- RHR Pump 10P-3B flow 9000 gpm
- RHR Pump 10P-3C flow 0 gpm
- RHR Pump 10P-3D flow 9000 gpm

Given these conditions, what is the maximum Torus water temperature that will maintain net positive suction head within limits?

RHR PUMP NPSH LIMIT



***Torus Overpressure = Torus Pressure + 0.4 (Torus Water Level - 1.92)**

A. 194 °F

B. 202 °F

C. 213 °F

D. 220 °F

K&A # 295026; EK1.01
Importance Rating RO – 3.0; SRO – 3.4

QUESTION 13.

K&A Statement: Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE : Pump NPSH

Justification:

A. Correct – Overpressure is calculated from plant conditions and is 6.42 psig. Reading 2 pump running curve with 5 psig overpressure (can not interpolate) is 194 F.

B. Incorrect but plausible if the candidate uses the 1 pump running curve and 5 psig overpressure.

C. Incorrect but plausible if the candidate uses the 2 pump running curve and 10 psig overpressure.

D. Incorrect but plausible if the candidate uses the 1 pump running curve and 10 psig overpressure.

References: OP-13A Student Ref: None

Learning Objective: SDLP-10, EO-1.13.A

Question source: Fitz 03 modified

Question History: Fitz 03 NRC exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.8-41.10

QUESTION 14.

EPIC power is lost and **NO** Control Room computer screens are available.

The following indications exist:

- TORUS TEMP A, 16-1TR-131A is reading 93 °F.
- TORUS TEMP B, 16-1TR-131B is reading 95 °F.
- DW TEMP A, 16-1TR-108 is reading 130 °F.
- DW TEMP B, 16-1TR-107 is reading 138 °F.
- DW COOLER A TEMP INLET, 68TI-100 is reading 160 °F.
- DW COOLER A TEMP OUTLET, 68TI-100 is reading 120 °F.
- DW COOLER B TEMP INLET, 68TI-101 is reading 140 °F.
- DW COOLER B TEMP OUTLET, 68TI-101 is reading 120 °F.
- DW Cooling Fans 68FN-2A, B, C and 68FN-4A, B, D are running.

Which **ONE** of the following describes a criterion for entry into EOP-4, "Primary Containment Control" under these conditions in accordance with EP-1, "EOP Entry and Use"?

A. DW Cooler B inlet temperature

B. DW Cooler A average inlet and outlet temperature

C. DW Temp B temperature

D. Torus Temp A and Torus Temp B average temperature

K&A # 295028; EA2.01
Importance Rating RO – 4.0; SRO – 4.1

QUESTION 14.

K&A Statement: Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Drywell Temperature.

Justification: EP-1 lists parameters/instruments in their order of preference for use in determining EOP entry conditions. This question tests the applicant's ability to recall and apply EP-1 guidance for DW and Torus temperature evaluation against EOP entry criteria.

A. Incorrect because DW cooler inlet temperature is not a valid entry condition. Plausible if the candidate thinks that the DW cooler inlet being above 135 °F is an EOP-4 entry condition.

B. Correct – DW temperature entry condition is above 135 °F. Per EP-1 average inlet and outlet cooler temperature is a valid entry condition if at least one cooling fan is running. DW Cooler A average temperature is 140 °F and three cooling fans (normal configuration) are running on each DW cooler assembly.

C. Incorrect because a single drywell temperature is not a valid entry condition. Plausible if the candidate does not know that the average of the two DW temperatures is the correct indication per EP-1.

D. Incorrect because do not use average torus temperature for EOP entry. Also no reading is above 95 °F which is the entry condition. Plausible if the candidate does not know the EP-1 requirements for Torus temperature.

References: EP-1 5.2, Rev 10 Student Ref: EP-1; sections 5.1 and 5.2

Learning Objective: MIT-301.11E 4.02

Question source: Modified 03 Fitz

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

QUESTION 15.

A plant event has occurred with the following plant conditions:

- Reactor has been manually scrammed.
- Drywell pressure is 7.0 psig and rising.
- Torus level is steady at 5.5 feet.
- CST level is steady at 57 inches.
- Torus water temperature is steady at 150 °F.
- Emergency Depressurization is in progress using Group 2 pressure control systems.

In accordance with OP-19, "Reactor Core Isolation Cooling System", with these plant conditions, which **ONE** of the following is an RCIC operational concern?

A. At this Torus temperature, operating RCIC could cause equipment damage..

B. Operation of RCIC in pressure control mode will result in further level reduction in Torus level.

C. At this Torus level, operating RCIC will cause system exhaust to over-pressurize the Torus.

D. Operation of RCIC could result in vortexing at this Torus level.

K&A # 295030; EK2.02
Importance Rating RO – 3.7; SRO – 3.8

QUESTION 15.

K&A Statement: Knowledge of the interrelations between **LOW SUPPRESSION POOL WATER LEVEL** and the following: **RCIC**

Justification:

A. Incorrect because at 170 °F torus temperature, operating RCIC with the suction aligned to the Torus could cause equipment damage. Plausible if the candidate does not know the EOP limitations from section A.5.2 of OP-19.

B. Incorrect because RCIC operation in the pressure control mode pumps water from the CST to the CST. Plausible if the candidate does not realize that operating RCIC in pressure control mode will not lower torus level.

C. Incorrect because per EOP basis document, operation with the RCIC system exhaust discharge not submerged can not over-pressurize the Torus because the vent has the capacity to remove any steam discharged by RCIC. Plausible if the candidate does not realize that containment can not be overpressurized by RCIC turbine exhaust.

D. Correct – Per OP-19 Page 5, operating RCIC with suction form the Torus with torus level below 5.7 feet could cause vortexing. CST level below 59.5 inches RCIC suction will auto swap to the Torus.

References: OP-19 Student Ref: None

Learning Objective: N/A

Question source: Modified Fitz 02 Q16

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7 / 45.8

QUESTION 16.

Given the following:

- A failure to scram has occurred.
- Reactor power is 20%.
- Torus bulk water temperature is 115 °F.
- Drywell Pressure is 3.5 psig.
- **ALL** MSIVs are closed.
- Pressure is being controlled between 800 and 1000 psig with SRVs.
- Injection has been terminated and prevented from Condensate and Feedwater, HPCI, RHR, and Core Spray.
- RPV water level has been lowered to +100 inches in accordance with EOP-3, "Failure to Scram".

Which **ONE** of the conditions listed below would allow the operator to resume injection from these systems?

A. Reactor power drops below 2%.

B. RPV water level drops to +19 inches.

C. Drywell pressure is reduced below 2.7 psig.

D. RPV pressure is reduced such that one SRV can maintain pressure.

K&A # 295031; EA2.02
Importance Rating RO – 4.0, SRO-4.2

QUESTION 16.
K&A Statement: **Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : Reactor Power**

Justification:

A. Correct – Reactor Power < 2.5% is a condition per EOP-3 that will allow reinjection.

B. Incorrect because reactor level would need to reach 0 inches TAF to meet the conditions to re-inject. Plausible because level is a parameter that if the condition is met then reinjection is allowed.

C. Incorrect because the condition that would allow re-injection is all SRV's remain closed **AND** drywell pressure remains below 2.7 psig. Plausible because drywell pressure is a parameter in conjunction with SRV's would allow re-injecton.

D. Incorrect because all SRV's need to be closed to meet the condition to re-inject. Plausible because the amount of SRV's is a condition that will determine if reinjection can be met.

References: EOP-3 R9, EP-1 R10 Student Ref: None

Learning Objective: MIT-301 11D 1.07

Question source: Pilgrim Bank

Question History: Pilgrim 04 NRC

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

QUESTION 17.

The reactor failed to scram and the following conditions exist:

- RPV pressure is being controlled at 900-1000 psig with EHC.
- Standby Liquid Control (SLC) has been initiated and is injecting from an initial tank level of 84%.

What conditions must be met to lower RPV pressure per EOP-3, "Failure To Scram"?

A. The hot shutdown boron weight has been injected, indicated by the SLC tank level of <48%.

B. The hot shutdown boron weight has been injected, indicated by the SLC tank level of <38%.

C. The cold shutdown boron weight has been injected, indicated by the SLC tank level of <48%.

D. The cold shutdown boron weight has been injected, indicated by the SLC tank level of <38%.

K&A # 295037; EK1.05
Importance Rating RO – 3.4; SRO – 3.6

QUESTION 17.

K&A Statement: Knowledge of the operational implications of the following concepts as they apply to **SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:**
Cold Shutdown Boron Weight

Justification:

A. Incorrect but plausible because the Hot Shutdown Boron Weight does not ensure the reactor is shutdown under all conditions. Plausible because the HSBW is used in EOP-3 for controlling level. Level may be raised after HSBW is injected.

B. Incorrect but plausible because the Hot Shutdown Boron Weight does not ensure the reactor is shutdown under all conditions. Plausible because the HSBW is used in EOP-3 for controlling level. Level may be raised after HSBW is injected.

C. Incorrect because the cold weight boron is indicated by a 46% drop in SLC level, but plausible if the candidate does not know the specific number for CSBW.

D. Correct – Prior to lowering pressure during an ATWS with SLC injecting, the reactor must remain shutdown under all conditions. This is ensured by injecting the cold boron weight as indicated by a 46% drop in SLC tank level.

References: EOP-3 R9, MIT 301.11d, MIT 301.11b Student Ref: None

Learning Objective: MIT 301.11d, 1.07 and MIT-301.11b EO 1.02

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.8 to 41.1

QUESTION 18.

In preparation for the upcoming refueling outage, new fuel is being moved in the spent fuel pool per OP-66A, "Refueling Bridge Operation".

During the fuel moves, the grapple fails and the control room is notified that a new fuel bundle has fallen on irradiated fuel bundles and caused damage to the irradiated bundles.

The following alarms and indications are received in the Control Room:

- 9-3-1-20, REFUEL AREA ARM RAD HI is in alarm.
- 9-3-2-29, RX BLDG VENT RAD MON HI is in alarm.

The immediate actions of AOP-44, "Dropped Fuel Assembly" have been completed.

Which **ONE** of the following describes the Reactor Building Ventilation and SGTS alignment **AND** the reason for the system alignment?

	System Response	Reason
A.	Reactor Building Ventilation isolates **AND** SGTS starts.	RX BLDG VENT RAD MON HI is in alarm
B.	Reactor Building Ventilation isolates **AND** SGTS starts.	REFUEL AREA ARM RAD HI is in alarm.
C.	Reactor Building Ventilation is isolated **AND** SGTS is running.	Manually performed by the operator per AOP-44, "Dropped Fuel Assembly".
D.	Reactor Building Ventilation continues to run **AND** SGTS is in standby.	Automatic trip setpoint has **NOT** been exceeded and RB Ventilation is **NOT** required to be manually isolated.

K&A # 295038; EK3.02
Importance Rating RO – 3.9, SRO-4.2

QUESTION 18.

K&A Statement: **Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: System Isolations**

Justification:

A. Incorrect because the reactor building vent rad monitor at the Hi setpoint does not isolate the reactor building and start SGTS. Plausible if applicant does not know that Hi-Hi Rad starts system versus the Hi Rad alarm.

B. Incorrect because the refuel area ARM does not isolate the reactor building and start SGTS. Plausible if applicant does not know which ARMs start SGTS.

C. Correct - AOP immediate actions for a dropped fuel bundle are to isolate RB Ventilation per OP-51A. OP-51A has operator manually start SGTS prior to isolation of RB.

D. Incorrect because the AOP does direct realigning the system. Plausible if applicant does not know immediate actions of AOP.

References: AOP-44 R7, OP-51A R47, OP-20 R36 Student Ref: None

Learning Objective: LP-AOP 1.03

Question source: Modified NMP1 Q9

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.5, 41.6

QUESTION 19.
Given the following conditions:

- Fire pumps 76P-1 (West Diesel Fire Pump), 76P-2 (Electric Fire Pump), and 76P-4 (East Diesel Fire Pump) are all available for automatic start.
- A fire occurs in the plant and fire main header pressure lowers to 98 psig and is recovered due to the automatic start of the appropriate fire pump(s).
- Assume **NO** operator actions are taken and the system responds automatically per design.

Which of the following fire pump(s) are expected to be running based upon the above conditions?

A. 76P-1 only

B. 76P-2 only

C. 76P-1 and 76P-2 only

D. 76P-1, 76P-2, and 76P-4

K&A # 600000; AK2.01
Importance Rating RO – 2.6; SRO – 2.7

QUESTION 19.

K&A Statement: Knowledge of the interrelations between **PLANT FIRE ON SITE** and the following:
SENSORS/DETECTORS AND VALVES

Justification:

A. **Incorrect but plausible** - 76P-1 starts at 101 psig decreasing, however given the conditions in the stem, fire main header pressure lowers to 98 psig and is recovered, 76P-1 and 76P-2 will be running as 76P-2 starts at 109 psig decreasing and 76P-4 starts at 92 psig decreasing and will **NOT** be running.

B. **Incorrect but plausible** - 76P-2 starts at 109 psig decreasing, however given the conditions in the stem, fire main header pressure lowers to 98 psig and is recovered, 76P-1 and 76P-2 will be running as 76P-1 starts at 101 psig decreasing and 76P-4 starts at 92 psig decreasing and will **NOT** be running.

C. **Correct -** 76P-1 and 76P-2 will be running, 76P-1 starts at 101 psig decreasing, 76P-2 starts at 109 psig decreasing, however given the conditions in the stem, fire main header pressure lowers to 98 psig and is recovered, 76P-4 starts at 92 psig decreasing and will **NOT** be running.

D. **Incorrect but plausible** - 76P-1 and 76P-2 will be running, 76P-1 starts at 101 psig decreasing, 76P-2 starts at 109 psig decreasing, however given the conditions in the stem, fire main header pressure lowers to 98 psig and is recovered, 76P-4 starts at 92 psig decreasing and will **NOT** be running.

References: OP-33 Rev. 52 Student Ref: None

Learning Objective: SDLP-76 Objective 1.05c.1 and 2, 1.08.a and c

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 20.

The plant is operating at 87% power with the Main Generator supplying 820 MWe and 200 MVAR to the grid. It is 11:30 am on Wednesday, May 15th.

- The main generator voltage regulator is in AUTO.
- Generator hydrogen pressure is 57 psig.
- Fitz 345 kV line voltage is 360 kV.
- Main Gen terminal voltage is 24.4 kV.

Subsequently there is a grid disturbance resulting in the following plant conditions:

- Scriba R935 breaker is out of service.
- Fitz 345 kV Line voltage is 365 kV.

Which **ONE** of the following describes the required action per OP-11A, "Main Generator, Transformers and Isolated Bus Phase Cooling", considering the initial and final grid conditions?

A. Place the voltage regulator control switch to RAISE to increase MVAR to at least 200 MVAR (OUT).

B. Place the voltage regulator control switch to LOWER to decrease MVAR to less than 400 MVAR (OUT).

C. Place the voltage regulator control switch to RAISE to increase MVAR to at least 100 MVAR (OUT).

D. Place the voltage regulator control switch to LOWER to decrease MVAR to less than 200 MVAR (OUT).

K&A #	700000: AA1.03
Importance Rating	RO – 3.8; SRO – 3.7

QUESTION 20.

K&A Statement: Ability to operate and/or monitor the following as they apply to **GENERATOR VOLTAGE AND ELECTRICAL GRID DISTURBANCES: Voltage regulator controls**

Justification:

A. CORRECT. Per OP-11A Attachment 3, generator terminal voltage will be 24.4 kV under the given initial conditions of 360 kV on the Fitz 345 Line, with the main gen at 820 MWe and 200 MVAR out. Since the voltage regulator is in auto, it will maintain this generator terminal voltage thru the transient. Attachment 5, for on-peak hours with the R935 breaker out of service, should be used to determine the post-transient reactive load as 120 MVAR out when the Fitz 345 Line is at 365 kV and generator terminal voltage is at 24.4 kV. This same attachment specifies minimum generator terminal voltage and reactive load for real load less than 840 MWe at 24.7 kV and 200 MVAR. Therefore, the operator is required to raise voltage generator terminal voltage and MVAR out to > 24.7 kV and > 200 MVAR out, respectively.

B. Incorrect. If applicant believes MVAR would have increased in this situation it is plausible using attachment 2 that the MVARs need to be lowered below 400 for this power and generator pressure.

C. Incorrect. If applicant uses the 840-896 line in Attachment 5 that raising MVARs to 325 would be required. Plausible if candidate miss reads the graph and selects 175 MVAR.

D. Incorrect. If applicant believes MVAR would have increased in this situation it is plausible using attachment 2 that the MVARs need to be lowered below 200 for this power and generator pressure.

References:	OP-11A, "Main Generator, Transformers and Isolated Bus Phase Cooling", Rev 37 OP-11C, "Main Generator Hydrogen Cooling and Seal Oil System", Rev 41	Student Ref:	OP-11A, Section E.1 (pages 22 thru 25) and Attachments 2 through 12 (pages 55 thru 65)

Learning Objective:	SDLP-94D, Obj 1.13
Question source:	New
Question History:	None
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis: X
10CFR	41.5 and 41.10

QUESTION 21.

The plant is operating normally at 23 % power when the following occurs:

- Feedwater level control fails.
- Reactor level rose to 228" and is now lowering.

Assume **NO** operator actions have been performed.

Which **ONE** of the following identifies the plant response **AND** the correct operator actions to the conditions above?

A. Automatic scram due to turbine trip. Perform immediate actions of AOP-1, "Reactor Scram".

B. Reactor remains at power. Attempt to reset feed pumps and maintain RPV level in manual per AOP-42, "Feedwater Malfunction".

C. Automatic scram due to high Reactor pressure transient caused by turbine trip. Perform immediate actions of AOP-1, "Reactor Scram".

D. Reactor remains at power. Insert a manual scram per AOP-42, "Feedwater Malfunction".

K&A # 295008; G.2.1.7
Importance Rating RO – 3.7; SRO – 3.7

QUESTION 21.

K&A Statement: Ability to evaluate plant performance and make operational judgements as they apply to **HIGH REACTOR WATER LEVEL: REACTOR WATER LEVEL CONTROL**

Justification:

A. **INCORRECT but plausible** – With reactor power at 23%, at 222.5 inches the main turbine, RFPTs, HPCI and RCIC have tripped on high RPV water level. However reactor power is within the capacity of the Main turbine bypass valves. There is no automatic scram.

B. **INCORRECT but plausible:** At 222.5 inches the main turbine, RFPTs, HPCI and RCIC have tripped on high RPV water level. No guidance in AOP- 42 to reset the reactor feed pumps.

C. **INCORRECT but plausible** – With reactor power at 23%, at 222.5 inches the main turbine, RFPTs, HPCI and RCIC have tripped on high RPV water level. However reactor pressure is within the capacity of the Main turbine bypass valves. There is no automatic scram.

D. **CORRECT:** Once RFPTs have tripped RPV water level will begin to lower. Immediate actions of AOP-42, states with no RFPs in service which have all tripped due to level reaching 222.5 inches, then insert a manual scram.

References: AOP-41 R8
AOP-42 R12
AOP-1 R43

Student Ref: None

Learning Objective: SDLP-06 R12; FWLC; L.O. 1.08.a

Question source: New

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR 41.7 / 45.6

QUESTION 22.

The plant is initially operating at 80 percent power with both Recirc Pumps at 76% speed.

- I&C technicians are inspecting the "A" Recirc Pump speed controller control board connections when a fuse blows in the speed control circuit.
- The following annunciator is received: 09-4-3-31 RWR MG A SPEED CNTRL SIG FAILED.

SELECT the resulting Recirc Pump speeds if one Reactor Feed Pump trips, level decreases to 196.0 inches, and then is restored to +200 inches by feedwater.

	RWR "A" Speed	RWR "B" Speed
A.	44%	44%
B.	76%	30%
C.	30%	30%
D.	76%	44%

K&A # 295009; AK2.03
Importance Rating RO – 3.1; SRO – 3.2

QUESTION 22.

K&A Statement: Knowledge of the reasons for the following responses as they apply to **LOW REACTOR WATER LEVEL: RECIRCULATION SYSTEM**

Justification:

A. **INCORRECT:** RRP 'A' will remain at 76% speed. It's control circuit will receive a runback signal but the speed will not change because pump speed is locked by the speed control circuit fuse fault. RRP 'B' will runback to 44% since one RFPT has tripped. **Plausible since** the applicant may think that the conditions for 44% limiter runback exist for RRP 'A'.

B. **INCORRECT:** RRP 'A' will remain at 76% speed. It's control circuit will receive a runback signal but the speed will not change because pump speed is locked by the speed control circuit fuse fault. RRP 'B' will runback to 44% since one RFPT has tripped. **Plausible since** the applicant may think that the conditions for 30% limiter runback exist for RRP 'B'.

C. **INCORRECT:** RRP 'A' will remain at 76% speed. It's control circuit will receive a runback signal but the speed will not change because pump speed is locked by the speed control circuit fuse fault. RRP 'B' will runback to 44% since one RFPT has tripped. **Plausible since** the applicant may think that the conditions for 30% limiter runback exist for RRP 'A' and 'B'.

D. **CORRECT:** RRP 'A' will remain at 76% speed. It's control circuit will receive a runback signal but the speed will not change because pump speed is locked by the speed control circuit fuse fault. RRP 'B' will runback to 44% since one RFPT has tripped.

References: AOP-42 R12 Student Ref: None

Learning Objective: SDLP-2I R12; Recirc Flow Control; L.O. 1.10.b

Question source: Modified from SQ Exam 12/07

Question History: Modified

Cognitive level: Comprehensive/Analysis:

10CFR 41.8 / 45.8

QUESTION 23.

A reactor startup is in progress.

Reactor power is 2%, when a reactor scram occurs.

- Ten (10) control rods remain at position “48”

Which **ONE** of the following describes the required operator action?

A. Enter AOP-1, "Reactor Scram" and insert control rods with Reactor Manual Control System.

B. Enter EOP-2, "RPV Control" and insert control rods with Reactor Manual Control System.

C. Enter EOP-3, "Failure to Scram" and insert control rods per EP-3, "Backup Control Rod Insertion".

D. Enter AOP-1, "Reactor Scram" and insert control rods per EP-3, "Backup Control Rod Insertion".

K&A # 295015; AA1.03
Importance Rating RO – 3.6; SRO – 3.8

QUESTION 23.
K&A Statement: Ability to operate and/ or monitor the following as they apply to **INCOMPLETE SCRAM: RMCS**

Justification:

A. **CORRECT:** AOP-1 step F.1 directs if any control rod is not full in and EOP-3 entry is not present then insert control rods with RMCS.

B. **INCORRECT but plausible if candidates does not remember that** Stem conditions do not warrant entry into EOP-2.

C. **INCORRECT but plausible if candidates does not remember that** Stem conditions do not warrant entry into EOP-3.

D. **INCORRECT but plausible if candidates does not remember that** AOP-1 does not direct entry into EP-3.

References: EOP-2 R9, EOP-3 R9, EP-3 R7 Student Ref: None

Learning Objective: N/A

Question source: Modified from Limerick Exam 10/06 question 27

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR 41.7 / 45.6

QUESTION 24.

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

- 09-5-1-49, CRD PMP 3P-16A TRIP alarms.
- 09-5-1-09, CRD CHARGING WTR PRES LO alarms.
- 09-5-1-43, CRD ACCUM PRESS LO OR LVL HI alarms.
- Yellow ACCUM Light is **LIT** for Control Rod 02-31, which is at position 48.
- "**NO** other yellow ACCUM lights are lit at the 09-05 panel."

With these conditions, the procedurally required action is to:

A. Manually SCRAM the reactor IMMEDIATELY.

B. Manually SCRAM the reactor WITHIN 20 MINUTES.

C. Close 03CRD-56 CRD Charging Water Supply Header Isol Valve, and then start CRD Pump B.

D. Place the CRD Flow Controller in MANUAL, close the CRD Flow Control Valve and then start CRD Pump B.

K&A # 295022; AK2.03
Importance Rating RO – 3.4; SRO – 3.4

QUESTION 24.

K&A Statement: Knowledge of the interrelations between **LOSS OF CRD** and the following: **ACCUMULATOR PRESSURES**

Justification:

A. **Incorrect but plausible:** if charging water header pressure is not restored to greater than or equal to 940 psig within 20 minutes of a second accumulator light, then insert a manual scram. Only 1 accumulator alarm is in and 20 minutes have not passed.

B. **Incorrect but plausible:** if charging water header pressure is not restored to greater than or equal to 940 psig within 20 minutes of a second accumulator light, then insert a manual scram. Only 1 accumulator alarm is in.

C. **Incorrect but plausible:** preferred method for placing the CRD pump in service is to place the CRD Flow Controller in MANUAL, close the CRD Flow Control Valve, and then start CRD pump B.

D. **CORRECT:** IAW procedure guidance of AOP-69; step E-3.

References: AOP-69, "Control Rod Drive Pump Trouble" R7 Student Ref: None

Learning Objective:1.08a SDLP-03C-R15; CRD Hydraulics

Question source: From JAF exam bank

Question History: From JAF exam bank

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7/45.8

QUESTION 25.

The plant is operating at 100% power when a LOCA occurs, resulting in the following:

- The reactor has scrammed on High Drywell pressure of 7 psig, which is rising slowly.
- The reactor is shutdown.
- MSIV's have closed.
- HPCI and RCIC suction are aligned to the CST and both systems are injecting into the RPV.
- Reactor pressure is being controlled between 800 and 1000 psig with SRVs.
- Torus temperature is 120 °F and rising.
- Torus level is 17 feet and rising.
- Reactor level is being maintained 177 to 222.5 inches.

Which **ONE** of the following describes the operator response to the current conditions **AND** the reason for the response?

A. An Emergency Depressurization is required if Torus level **CANNOT** be restored and maintained below the SRV Tail Pipe Level Limit (STPLL), because operation of SRV's above the STPLL could lead to SRV discharge line damage and/or containment failure.

B. Depressurize the RPV to less than 200 psig, exceeding cooldown rates is authorized per EP-1, "EOP Entry and Use", because Torus temperature is at 120 °F.

C. Terminate injection into the RPV from sources external to Primary Containment, even if adequate core cooling is **NOT** assured, because when performing EOP-2 and EOP-4, "Primary Containment Control" concurrently, maintaining containment parameters takes precedent over adequate core cooling.

D. Initiate Drywell Sprays when Torus pressure exceeds 15 psig to ensure that the Primary Containment Pressure Limit is **NOT** exceeded.

K&A # 295029; EK3.01
Importance Rating RO – 3.5; SRO – 3.9

QUESTION 25.

K&A Statement: Knowledge of the reasons for the following responses as they apply to **HIGH SUPPRESSION POOL WATER LEVEL: EMERGENCY DEPRESSURIZATION**

Justification:

A. **CORRECT:** An emergency depressurization is required if Torus level can not be restored and maintained blow the SRV Tail Pipe Level Limit.

B. **Incorrect but plausible:** EP-1 does state to depressurize below 200 psig if Torus temperature is at 120 F, however normal cooldown rates are in effect.

C. **Incorrect but plausible:** Terminate injection into the RPV from sources external to Primary Containment is to be performed but only if adequate core cooling is assured. EP-1 states adequate core cooling takes precedent over containment parameters.

D. **Incorrect but plausible:** Drywell sprays are required when Torus pressure exceeds 15 psig, but this is to preclude chugging which could lead to fatigue failure of the junction at the junction of the downcomers and the vent header.

References: EOP-4 R8, EP-1 R10 Student Ref: None

Learning Objective:1.07.a SDLP-16A; R-12 and T/s Bases 3.6.2.2

Question source: New

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR 41.5 / 45.6

QUESTION 26.

With the unit at 100 percent power. The following conditions exist:

- Rx BLDG Sample Area Radiation Monitor is reading 900 mrem/hour.
- RWCU Pump Area Radiation Monitor is reading 900 mrem/hour.
- RWCU has failed to isolate on a **VALID** high temperature isolation signal.

Which **ONE** of the following describes what is occurring **AND** what action is required?

A. A Primary system is **NOT** discharging into the Reactor Building. Reset the area radiation monitor per ARP-09-3-1-40, RX BLDG ARM RAD HI.

B. A Primary system is **NOT** discharging into the Reactor Building. Continue power operations and attempt to isolate reactor water cleanup.

C. A Primary system is discharging into the Reactor Building. Shutdown the reactor per EOP-5, "Secondary Containment Control" **AND** EOP-2, "RPV Control".

D. A Primary system is discharging into the Reactor Building. Emergency depressurize the reactor per EOP-5, "Secondary Containment Control" **AND** EOP-2, "RPV Control".

REACTOR BUILDING AREA RADIATION LEVELS			
AREA	**INSTRUMENT**	**MAXIMUM NORMAL**	**MAXIMUM SAFE**
Spent Fuel Pool	18RIA-051-12	25 mr/hr	10^3 mr/hr
Reactor Building 344 ft elevation	18RIA-051-13	20 mr/hr	10^3 mr/hr
New Fuel Vault	18RIA-051-14	20 mr/hr	10^3 mr/hr
Cleanup Precoat Area	18RIA-051-15	80 mr/hr	10^3 mr/hr
RWCU Heat Exchanger Room	18RIA-051-16	50 mr/hr	10^3 mr/hr
Fuel Pool Pump Room	18RIA-051-17	300 mr/hr	10^3 mr/hr
Contaminated Equipment Storage	18RIA-051-18	50 mr/hr	10^3 mr/hr
RWCU Pump Area	18RIA-051-19	30 mr/hr	10^3 mr/hr
Rx BLDG Sample Area	18RIA-051-20	30 mr/hr	10^3 mr/hr
RBCLC Heat Exchanger Area	18RIA-051-21	5 mr/hr	10^3 mr/hr
Reactor Building Access 272 ft elev	18RIA-051-23	40 mr/hr	10^3 mr/hr
TIP Cubicle	18RIA-051-24	125 mr/hr	10^3 mr/hr

K&A # 295033; EA2.01
Importance Rating RO – 3.8; SRO – 3.9

QUESTION 26.
K&A Statement: Ability to determine and/or interpret the following as they apply to **HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: AREA RADIATION LEVELS**

Justification:

A. **Incorrect but plausible:** A primary system is discharging into the Reactor Building.

B. **Incorrect but plausible:** A primary system is discharging into the Reactor Building.

C. **CORRECT:** A primary system is discharging into the Reactor Building. Emergency depressurization must occur after 2 rooms exceed the maximum safer Area radiation Levels which is 1000 mrem/hr. Since radiation levels are only 900 mrem/hr, a unit Shutdown should be started. If radiation levels exceed 1000 mrem/hr an emergency depressurization is required.

D. **Incorrect but plausible:** A primary system is discharging into the Reactor Building. Emergency depressurization must occur after 2 rooms exceed the maximum safe area radiation levels, which is 1000 mrem/hr.

References: EOP-5 R7, EOP-2 R9 Student Ref: EOP-5 Table embedded into question

Learning Objective: 1.14.5, SDLP.17 R-10 Reactor Building Area radiation Monitors

Question source: NEW

Question History: NEW

Cognitive level: Comprehensive/Analysis:

10CFR 41.7/ 45.6

QUESTION 27.

A large break LOCA has occurred and the following conditions exist:

- Reactor Level is -60 inches and slowly rising.
- Drywell Pressure is 12 psig and slowly rising.

Which one of the following describes the availability of the Hydrogen and Oxygen Sampling Systems **AND** what actions are required?

The Hydrogen and Oxygen Sampling Systems are ...

A. Isolated. Direct Chemistry to sample with the Post Accident Monitoring System.

B. Available **AND** lined up. Verify proper operation at the H2/O2 Analyzer Cabinets.

C. Isolated. Direct operators to bypass the isolation interlocks to obtain a sample.

D. Available **AND NOT** lined up. Align H2/O2 Sampling to place it in service.

K&A # 500000; EA2.01
Importance Rating RO – 3.1; SRO – 3.5

QUESTION 27.

K&A Statement: Ability to determine and/or interpret the following as they apply to **HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: HYDROGEN MONITORING SYSTEM AVAILABILITY**

Justification:

A. **Incorrect but plausible:** if the candidate forgets the Post Accident Sampling System is used under severe LOCA conditions to sample the primary coolant.

B. **Incorrect but plausible:** if the candidate forgets that the RPV level and Drywell pressure signals have caused an isolation of the H2/O2 Analyzer Cabinet.

C. **CORRECT:** In accordance with EOP-4, defeat isolation with EP-2 place H2/O2 Analyzer Cabinets in service.

D. **Incorrect but plausible:** if the candidate forgets that this sampling method is only done when H2/O2 Analyzer Cabinet is not available. The cabinet can be restored with EP-2.

References: SDLP-16B R20, EP-2 R7, EOP-4 R8 Student Ref: None

Learning Objective: 1.05.3.a and b

Question source: Modified to JAF format

Question History: NMP 1; NRC 10/08; Q-65

Cognitive level: Memory or Fundamental Knowledge

10CFR 41.10/ 43.5 / 45.13

QUESTION 28.

The following describes the initial plant conditions:

- Reactor is at 100% power.
- RHR Loop 'A' is in Torus Cooling.

Subsequently, a reactor coolant leak occurs and the following conditions exist:

- Reactor pressure is 500 psig and dropping
- Reactor level is 60" and dropping
- Drywell pressure is 10.0 psig and rising

Which **ONE** of the following describes the **CURRENT** position of the following RHR Loop 'A' valves for LPCI injection:

- 10MOV-66A, RHR Heat Exch 'A' Bypass Valve
- 10MOV-39A, RHR 'A' Torus Cooling Isol Valve
- 10MOV-16A, RHR 'A' Min Flow Isol Valve
- 10MOV-25A, RHR 'A' LPCI Inbd Inj Valve

A.

10MOV-66A	Open
10MOV-39A	Closed
10MOV-16A	Open
10MOV-25A	Closed

B.

10MOV-66A	Closed
10MOV-39A	Closed
10MOV-16A	Open
10MOV-25A	Closed

C.

10MOV-66A	Open
10MOV-39A	Closed
10MOV-16A	Closed
10MOV-25A	Open

D.

10MOV-66A	Closed
10MOV-39A	Open
10MOV-16A	Closed
10MOV-25A	Closed

K&A # 203000; K4.01
Importance Rating RO – 4.2; SRO – 4.2

QUESTION 28.

K&A Statement: Knowledge of **RHR/LPCI: INJECTION MODE design feature(s) and/or interlocks which provide for the following: AUTOMATIC SYSTEM INITIATION/INJECTION**

Justification:

A. **CORRECT:** LOCA signal closes Torus return valve and Heat exchanger bypass valve remains open. RPV pressure is not below 450 psig. Min flow valve is open and LPCI injection is closed, but will open at 450 psig.

B. **Incorrect but plausible:** LOCA signal closes Torus return valve and Heat exchanger bypass valve remains open. RPV pressure is not below 450 psig. Min flow valve is open and LPCI injection is closed, but will open at 450 psig.

C. **Incorrect but plausible:** LOCA signal closes Torus return valve and Heat exchanger bypass valve remains open. RPV pressure is not below 450 psig. Min flow valve is open and LPCI injection is closed, but will open at 450 psig.

D. **Incorrect but plausible:** LOCA signal closes Torus return valve and Heat exchanger bypass valve remains open. RPV pressure is not below 450 psig. Min flow valve is open and LPCI injection is closed, but will open at 450 psig.

References: SDLP-10 R19; RHR OP-13 "Residual Heat removal System" R94 Student Ref: None

Learning Objective: SDLP-10R19;1.05.a.1.a

Question source: Modified to JAF format

Question History: Limerick 10/08 Exam; Q-28

Cognitive level: Comprehensive/Analysis:

10CFR 41.5 / 45.5

QUESTION 29.

The 'B' RHR loop is in Shutdown Cooling mode with 'A' RPS Bus on alternate power source and 'B' RPS Bus powered from 'B' RPS motor generator set.

Which **ONE** of the following conditions would result in a loss of shutdown cooling?

A. RPV water temperature rising to 300°F

B. Loss of 71MCC-251

C. RPV water level lowering to 187 inches

D. Loss of the 10600 bus

	K&A #	205000; A3.01
	Importance Rating	RO – 3.2; SRO – 3.1

QUESTION 29.

K&A Statement: Ability to monitor automatic operations of the **SHUTDOWN COOLING SYSTEM: VALVE OPERATION.**

Justification:

A. **Incorrect** because 300 F corresponds to a saturation pressure of approximately 55 psig which is below the setpoint of 75 psig which would close 10MOV-17 and 10MOV-18. Plausible because if the high temperature setpoint was reached shutdown cooling would isolate.

B. **Incorrect;** Since 'A' RPS is on it's alternate power supply 71MCC-252 and 'A" RPS bus will stay energized, **but plausible** if the candidate thinks loss of 71MCC-251 is going to cause a loss of 'A' RPS power supply which would cause 10MOV-17 and 18 Shutdown Cooling suction valves to close. See AOP-59 attachment 1.

C. **Incorrect but plausible:** Isolation occurs at an RPV level setpoint of 177".

D. **CORRECT:** Loss of 10600 will cause loss of B RPS bus, which will cause 10MOV-17 and 18 Shutdown Cooling suction valves to close. See AOP-60 attachment 1.

References:	AOP-30, "Loss of Shutdown Cooling" R19 AOP-59 "Loss of RPS Bus A Power" R7 SDLP-10, RHR System, Rev 19, Page 48 of 137	Student Ref: Steam tables
Learning Objective: SDLP-10 R19; RHR 1.05.4		N/A
Question source:		New
Question History:		None
Cognitive level:		Comprehensive/Analysis:
10CFR		41.7 / 45.7

QUESTION 30.

Given the following:

- The unit is operating at 90% power.
- Required quarterly HPCI surveillance test is in progress and taking longer than expected.
- 9-3-3-5, HPCI TORUS LVL HI alarm has just been received.

What is the expected response of HPCI to the Suppression Pool High Level alarm?

A. No effect since HPCI suction valves do **NOT** transfer on High Suppression Pool level.

B. HPCI Suppression Pool Suction Valves 23MOV-57 and 58 receive an open signal. When the valves are full open, the CST Suction Valve 23MOV-17 will go closed.

C. HPCI Suppression Pool Suction Valves 23MOV-57 and 58 receive an open signal **AND** simultaneously CST Suction Valve 23MOV-17 receives a closed signal.

D. HPCI Suppression Pool Suction Valves 23MOV-57 and 58 receive an open signal. CST suction valve 23MOV-17 does **NOT** receive any signal.

K&A # 206000; A4.09
Importance Rating RO – 3.8; SRO – 3.7

QUESTION 30.

K&A Statement: Ability to manually operate and /or monitor in the control room: **SUPPRESSION POOL LEVEL**

Justification: K/A match for suppression pool level based on during the performance of the HPCI ST, suppression pool water level will need to be monitored.

A. **Incorrect but plausible**: if the candidate forgets Torus water level ≤ 14 feet, the Torus suction valves 23MOV-57 and 58 will get an open signal. When the 23MOV-57 and 58 are open, CST suction 23MOV-17 will automatically close.

B. **CORRECT:** Torus water level ≤ 14 feet, the Torus suction valves 23MOV-57 and 58 will get an open signal. When the 23MOV-57 and 58 are open, CST suction 23MOV-17 will automatically close.

C. **Incorrect but plausible**: if the candidate forgets Torus water level ≤ 14 feet, the Torus suction valves 23MOV-57 and 58 will get an open signal. When the 23MOV-57 and 58 are open, CST suction 23MOV-17 will automatically close.

D. **Incorrect but plausible**: if the candidate forgets Torus water level ≤ 14 feet, the Torus suction valves 23MOV-57 and 58 will get an open signal. When the 23MOV-57 and 58 are open, CST suction 23MOV-17 will automatically close.

References: SDLP-23 R15; HPCI Student Ref: None

Learning Objective: 1.05.b.1

Question source: Modified to JAF

Question History: HC NRC Exam 11/05; question 39

Cognitive level: Memory

10CFR 41.7 / 45.5 to 45.8

QUESTION 31.

Given the following conditions at T= 0 seconds:

- An un-isolable small LOCA and a Loss of Offsite AC Power have occurred.
- Drywell pressure is 10 psig and rising.

At T=19 seconds, RPV pressure is 400 psig and slowly lowering.

Which **ONE** of the following identifies the Core Spray response at T=19 seconds to the conditions specified above?

A. Core Spray pumps are **NOT** running **AND** will **NOT** auto start later in the event.

B. Core Spray pumps are **NOT** running **AND** will auto start later in the event.

C. Core Spray pumps are running **AND** injecting.

D. Core Spray Pumps are running but **CANNOT** inject.

K&A # 209001; G.2.1.28
Importance Rating RO – 4.1; SRO – 4.1

QUESTION 31.

K&A Statement: Knowledge of the purpose and function of major system components and controls: **AUTOMATIC SYSTEM INITIATION**

Justification:

A. **INCORRECT but plausible if candidates do not remember that** Core Spray pumps receive an auto start signal. Injection valve gets power when the EDG output breaker closes and will be stroking open.

B. **CORRECT:** EDG output breakers close at T=10 seconds however the Core Spray pumps do not start until T=21 seconds.

C. **INCORRECT but plausible if candidates do not remember that** EDG output breakers close at T=10 seconds however the Core Spray pumps do not start until T=21 seconds. Injection valves are opening.

D. **INCORRECT.** Core spray pumps will not be running at 19 seconds. If CS pumps were running, they would be injecting since valves open at 450 psig and 400 is below pump shutoff head. **Plausible** because applicant may not remember the shutoff head.

References: SDLP-14; R12 Student Ref: None

Learning Objective: 1.05.c.1

Question source: Modified

Question History: NMP 1 10/08 exam; Q - 1

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7/ 43.5/ 45.4

QUESTION 32.

The plant was operating at 100% power when MSIV closure resulted in the following:

- Reactor Power is 15%.
- SRVs are cycling to control reactor pressure.
- 'A' SLC pump has been manually started from the Control Room.

Which one of the following describes indications that the squib valves have fired and are open?

A. White SQUIB VLVS READY light is ON.

B. Discharge pressure of the SLC pump is 1150 psig.

C. White SQUIB VLVS READY light is OFF.

D. Discharge pressure of the SLC pump is 1400 psig.

K&A # 211000; K5.04
Importance Rating RO – 3.1; SRO – 3.2

QUESTION 32.

K&A Statement: Knowledge of **STANDBY LIQUID CONTROL SYSTEM** design feature(s) and / or interlocks which provide for the following: **EXPLOSIVE VALVE OPERATION**

Justification:

A. **Incorrect but plausible:** The white squib vlvs ready light is on indicates there is continuity and the squib valves have not fired.

B. **CORRECT:** SLC is injecting because the discharge pressure of the pump is slightly above the SRV pressure setpoint.

C. **Incorrect but plausible:** 4.3 mA is normal indication of current, but after firing, the indication will be off-scale high (>5 mA) until the switch is taken to off. At this time the light goes off and indication of current ~0. Neither indicates that the squib valve actually opened.

D. **Incorrect but plausible:** squib valves are not open, pump is at dead head pressure of the SLC pump.

References: SDLP11 R12 Student Ref: None

Learning Objective: 1.05.9

Question source: New

Question History: Modified from NMP 1 exam 8/09; Q-34

Cognitive level: Comprehensive/Analysis:

10CFR 41.7

QUESTION 33.

The following describes the initial plant conditions:

- Reactor is at 100% power.
- A loss of 125 VDC power, 71DCB5 to 'B' ARI/RPT logic has occurred.

Subsequently, a Main Turbine trip occurs with a failure to scram.

Describe the capability of the ATWS RPT 'A' ARI/RPT logic?

A. Automatic initiation of ARI is available. Automatic initiation of RPT is **NOT** available.

B. Manual initiation of RPT is available by taking 1-CS-ARI to ACTUATE on the 09-5 panel.

C. Automatic initiation of ARI is available. Automatic initiation of RPT is available.

D. Manual initiation of ARI is available by taking 1-CS-ARI to ACTUATE on the 09-5 panel.

K&A # 212000; K6.04
Importance Rating RO – 2.8; SRO – 3.1

QUESTION 33.

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the **REACTOR PROTECTION SYSTEM** including: **DC ELECTRICAL DISTRIBUTION**

Justification: Manual RWR pump trip would be possible by tripping the 71-10210 breaker which is normally done in EOP-3; NOT by taking 1-CS-ARI to ACTUATE on the 09-5 panel. Therefore, as originally written, choice 'b' could be legitimately argued as correct; recommend making proposed changes to make choice 'd' the only correct response.

A. **Incorrect but plausible:** Both logic circuits must be energized in order for a complete ARI and RPT to occur. Loss of either 71DCA5 or 71DCB5 renders the automatic initiation of ARI and RPT completely inoperable. A loss of 'B' Logic power supply will still leave the manual ARI initiation operable because it is entirely powered by 'A' Logic.

B. **Incorrect but plausible:** Both logic circuits must be energized in order for a complete ARI and RPT to occur. Loss of either 71DCA5 or 71DCB5 renders the automatic initiation of ARI and RPT completely inoperable. A loss of 'B' Logic power supply will still leave the manual ARI initiation operable because it is entirely powered by 'A' Logic.

C. **Incorrect but plausible:** Both logic circuits must be energized in order for a complete ARI and RPT to occur. Loss of either 71DCA5 or 71DCB5 renders the automatic initiation of ARI and RPT completely inoperable. A loss of 'B' Logic power supply will still leave the manual ARI initiation operable because it is entirely powered by 'A' Logic.

D. **CORRECT:** A loss of 'B' Logic power supply will still leave the manual ARI initiation operable because it is entirely powered by 'A' Logic.

References: SDLP-05 R16, AOP-46 R13 Student Ref: None

Learning Objective: 1.05.c.3

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7 / 45.7

QUESTION 34.

A unit startup is in progress with Reactor power on IRM range 5, when the following occurs:

- The 'A' 24 VDC battery charger trips due to an electrical failure within the charger.

How long will the battery be able to supply normal loads **AND** what will be the response of the 'A' side IRMs as the battery is depleted?

A. 2 hours; failing upscale causing a rod block from 'A' side IRMs.

B. 4 hours; failing downscale causing a rod block **AND** a half scram of 'A' RPS.

C. 8 hours; failing downscale causing a rod block from 'A' side IRMs.

D. 12 hours; failing upscale causing a rod block **AND** a half scram of 'A' RPS.

K&A # 215003; K6.02
Importance Rating RO – 3.6; SRO – 3.8

QUESTION 34.

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the **INTERMEDIATE RANGE MONITORS (IRM) SYSTEM: 24/48 VOLT D.C. POWER**

Justification:

A. **Incorrect but plausible:** if the candidate forgets the 24 VDC battery is designed to provide power for 4 hours and IRM indications will fail downscale, causing a rod block and a half scram of'A' RPS side.

B. **Correct:** The 24 VDC battery is designed to provide power for 4 hours. IRM indications will fail downscale, causing a rod block and a half scram of 'A' RPS side.

C. **Incorrect but plausible:** if the candidate forgets the 24 VDC battery is designed to provide power for 4 hours and IRM indications will fail downscale, causing a rod block and a half scram of 'A' RPS side.

D. **Incorrect but plausible:** if the candidate forgets the 24 VDC battery is designed to provide power for 4 hours and IRM indications will fail downscale, causing a rod block and a half scram of 'A' RPS side.

References: SDLP-07B R11 Student Ref: None

Learning Objective: 1.10.c

Question source: Modified

Question History: HC exam 11/05: Q-36

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7 / 45.7

QUESTION 35.

The following plant conditions exist:

- Reactor Mode Switch is in START & HOT STBY.
- All IRMs are on Range 2.
- All SRMs are being withdrawn.
- SRM A is reading 60 cps.
- SRMs B, C and D are reading 8.3 x 10E4 cps.
- A rod block signal has been generated.

Considering the above conditions, which **ONE** of the following has caused the SRM rod block?

A. SRM Detector Not Full In

B. SRM Hi Flux

C. SRM Downscale

D. SRM Inop

K&A # 215004; A1.02
Importance Rating RO – 3.6; SRO – 3.7

QUESTION 35.

K&A Statement: Ability to predict and/or monitor changes in parameters associated with operating the **SOURCE RANGE MONITOR (SRM) SYSTEM** controls including: **REACTOR POWER INDICATION**

Justification:

A. **CORRECT:** 'A' SRM is not fully inserted and indicating < 100 cps which will result in a rod block if the associated IRM is on range 1 or 2.

B. **Incorrect but plausible if the:** candidate forgets that the SRM Hi Flux setpoint is 10E+5 cps and not 10E+4.

C. **Incorrect:** because the downscale setpoint is 5 cps. **Plausible if the:** candidate forgets the value of the SRM downscale setpoint.

D. **Incorrect because the conditions for an inop SRM are not given in the stem, but plausible if the:** candidate forgets what causes an inop rod block, Low voltage, Module unplugged, and switch out of operate.

References: SDLP-07B R11; Startup Range Neutron Monitoring Student Ref: None

Learning Objective: 1.14.a,b,c

Question source: Modified

Question History: HC exam 11/05 Q-37

Cognitive level: Analysis

10CFR 41.5/ 45.5

QUESTION 36.

The unit is on line at 100% power when "C' Flow Unit fails downscale.

What will be the response from RPS **AND** Control Rod Block?

A. **NO** response from RPS.
Rod block from flow unit comparator trip **ONLY**.

B. Half scram on 'A' RPS.
Rod block from 'C' APRM **ONLY**.

C. Half scram on 'A' RPS.
Rod blocks from 'A' & 'C' flow units, 'A' APRM, 'C' APRM, AND 'E' APRM.

D. **NO** response from RPS.
Rod blocks from 'A' & 'C' flow units, 'A' APRM, 'C' APRM, AND 'E' APRM.

K&A # 215005; A1.02
Importance Rating RO – 3.9; SRO – 4.0

QUESTION 36.

K&A Statement: Ability to predict and/or monitor changes in parameters associated with operating the APRM controls including: **RPS STATUS**

Justification:

A. **Incorrect but plausible:** if candidate forgets flow signal passes thru a low value gate. With 'A' Flow unit indicating normal, the flow signal from 'C' flow unit will be passed, causing a 10% comparator trip (rod block) from the flow unit and rod blocks and half scram trip from 'A', 'C' and 'E' APRMs.

B. **Incorrect but plausible:** if candidate forgets flow signal passes thru a low value gate. With 'A' Flow unit indicating normal, the flow signal from 'C' flow unit will be passed, causing a 10% comparator trip (rod block) from the flow unit and rod blocks and half scram trip from 'A', 'C' and 'E' APRMs.

C. **CORRECT:** flow signal passes thru a low value gate. With 'A' Flow unit indicating normal, the flow signal from 'C' flow unit will be passed, causing a 10% comparator trip (rod block) from the 'A' & 'C' flow units and rod blocks and half scram trip from 'A', 'C' and 'E' APRMs flow biased trips.

D. **Incorrect but plausible:** if candidate forgets flow signal passes thru a low value gate. With 'A' Flow unit indicating normal, the flow signal from 'C' flow unit will be passed, causing a 10% comparator trip (rod block) from the 'A' & 'C' flow units and rod blocks and half scram trip from 'A', 'C' and 'E' APRMs.

References: SDLP-07C R10 Student Ref: None
ARP-09-5-2-25 R4, OP-16 R27

Learning Objective: 1.05.a.2.a and 1.05.a.3.d

Question source: New

Question History: None

Cognitive level: Analysis

10CFR 41.7/ 45.5 to 45.8

QUESTION 37.

The unit is on line at 100% power with the following conditions:

- RX WTR LVL COLUMN SEL 06-S1 switch is in B-LEVEL.
- RCIC is in service for its quarterly surveillance test.
- A loss of DC Power System 'B' occurs.

Given these conditions, select the correct status for RCIC in accordance with AOP-46; LOSS OF DC POWER SYSTEM B.

A. RCIC is **NOT** available, a loss of power to all RCIC MOVs except 13MOV-15 (RCIC Steam Supply INBD ISOL Valve) has occurred.

B. RCIC is available, if RPV level approaches the high level trip, then manually close 13MOV-131 (RCIC Turbine Steam Inlet ISOL Valve).

C. RCIC is **NOT** available, a loss of power to RCIC logic 'B' has occurred.

D. RCIC is available, if RPV level approaches the high level trip, then RCIC will automatically trip.

K&A # 217000;K6.01
Importance Rating RO – 3.4; SRO – 3.5

QUESTION 37.

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the **REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): ELECTRICAL POWER**

Justification:

A. **Incorrect but plausible if the candidate** forgets that RCIC is available. This action is for a loss of DC Power System 'A'. AOP-45.

B. **CORRECT:** IAW AOP-46, this is an override for if RPV level approaches the high RPV water level trip, then close TURB STM SUPP VLV 13MOV-131.

C. **Incorrect but plausible if the candidate** forgets that RCIC is available. This action is for a loss of DC Power System 'A'. AOP-45.

D. **Incorrect but plausible.** RCIC is available for control of RPV water level and pressure; however, there is no high level trip for RCIC IAW AOP-46, a loss of DC Power System 'B'.

References: SDLP -13 R14; RCIC
AOP-46 R13, AOP-41 R8

Student Ref: None

Learning Objective: 1.10.b

Question source: New

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR 41.7 / 45.7

QUESTION 38.

The plant was operating at 100% power. Subsequently, a LOCA occurs at Time T=0 seconds.

At T=10 seconds:

- Reactor vessel water level is + 55 inches.
- HPCI and RCIC are injecting with discharge pressure at 1000 psig.
- ADS Normal/Override switches on Panel 09-4 are in the NORMAL position.
- Drywell pressure is 2.5 psig.
- A failure prevented the auto start of **ALL** ECCS pumps.
- 09-4-1-28, ADS TIMERS ACTUATED annunciator is lit.

At T=40 seconds:

- 'A' Core Spray Pump is manually started and is running on minimum flow at 130 psig.

At T=155 seconds, RPV water level is at + 70 inches.

Assume **NO** further operator action. What is the status of **ALL** ADS valves at T=190 seconds **AND** why?

A. Open. ADS valves opened at T=160 seconds.

B. Closed. Low pressure ECCS pump running logic is not satisfied.

C. Open. ADS valves opened at T=130 seconds.

D. Closed. ADS TIMER is reset.

K&A # 218000; K5.01
Importance Rating RO – 3.8; SRO – 3.8

QUESTION 38.

K&A Statement: Knowledge of the operational implications of the following concepts as they apply to **AUTOMATIC DEPRESSURIZATION SYSTEM: ADS LOGIC OPERATION**

Justification:

A. **Incorrect but plausible if the candidate** does not recognize that the ADS time delay relay has been reset since RPV water level is above 59.5 inches. The ADS valves will remain closed.

B. **Incorrect but plausible**. The low pressure ECCS pump running logic is satisfied with Core Spray discharge pressure at 120 psig. The candidate may not recognize that the ADS time delay relay has been reset since RPV water level is above 59.5 inches. The ADS permissives have not been satisfied. The ADS valves will remain closed.

C. **CORRECT:** The ADS time delay relay has not been reset since RPV water level did not go above 59.5 inches until T=155 seconds. The ADS valves will be open.

D. **Incorrect but plausible if the candidate** The ADS time delay relay has not been reset since RPV water level did not go above 59.5 inches until T=155 seconds. The ADS valves will be open.

References: SDLP-02 R12; ADS Student Ref: None

Learning Objective: 1.05.c.1 and 1.10.c

Question source: Modified

Question History: From JAF NRC 03/08: Q-38

Cognitive level: Comprehensive/Analysis:

10CFR 41.7/ 45.7

QUESTION 39.

During full power operation, I&C is conducting PCIS Group 1 surveillance testing on Transmitters 02-3LT-57A, 57B, 58A and 58B. The technician commences testing RPV Level 1 (LO-LO-LO) response by inserting a level signal of < 59.5" on Instrument 02-3LT-57B.

At the same time, Transmitter 02-3LT-58B fails downscale.

Which of the following describes the plant response **AND** the appropriate operator response?

	Plant Response	Operator Response
A.	**ONLY** outboard MSIV's close causing Reactor SCRAM.	Enter AOP-1, "Reactor Scram" **AND** AOP-15, "Isolation Verification and Recovery" **AND** EOPs.
B.	**ALL** MSIV's closed causing Reactor SCRAM.	Enter AOP-1, "Reactor Scram" **AND** AOP-15, "Isolation Verification and Recovery" **AND** EOPs.
C.	**NO** MSIVs close, half isolation signal occurs on PCIS 'B'.	Direct I&C to restore 02-3LT-57B per EN-AD-102, "Procedure Adherence and Level of Use" **AND** reset the isolation signal per AOP-15, "Isolation Verification and Recovery". Testing may continue after isolation is reset.
D.	**NO** MSIVs close, half isolation signal occurs on PCIS 'B'.	Direct I&C to restore 02-3LT-57B per EN-AD-102, "Procedure Adherence and Level of Use", **AND** stop further testing.

K&A # 223002;A2.05
Importance Rating RO – 3.3; SRO – 3.6

QUESTION 39.

K&A Statement: Ability to predict (a) predict the impacts of the following on the **PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF;** and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **NUCLEAR BOILER INSTRUMENTATION FAILURES**

Justification: Group 1 MSIV Logic includes RPV Low Level. To permit testing and guard against spurious instrument failure, the de-energized- to-function, fail-safe logic, initiates a I out of 2 taken twice arrangement, which is also divided between two divisions. The failure of any one transmitter (either LT-57A, 57B, 58A, or 58B) will not cause MSIV closure, just indication of partial logic actuation; the consequences will depend upon which ones are involved. Any combination of one on 'A' logic (LT-57A or 58A) and one on **'B'** logic (LT-57B or 58B) will fulfill complete logic and MSIV's (all) will automatically close. If both transmitters on the same logic train fail, it will result in no more than the half isolation logic actuation.

A. **Incorrect** because MSIVs do not close. P**lausible if the candidate** thinks a 'B' isolation signal will close half the valves.

B. **Incorrect** because MSIVs do not close. P**lausible if the candidate** forgets that only a partial isolation has occurred 'B' PCIS Logic.

C. **Incorrect** because the isolation signal cannot be reset. P**lausible** because applicant may think isolation can be reset after 57A is returned to service.

D. **CORRECT-** (1) Half isolation signal on PCIS 'B' (2) EN-AD-102, "Procedure Adherence and Level of Use" requires restoring conditions and terminating procedure performance.

References: SDLP-16c R08, AOP-1 R43, AOP-15 R26, EN-AD-102 R4, 09-5-1-7 R8 Student Ref: None

Learning Objective: 1.06.a

Question source: Bank

Question History: JAF NRC exam 7/03; Q-43

Cognitive level: Comprehensive/Analysis:

10CFR 41.5 / 45.5

QUESTION 40.

A manual scram has been initiated in response to a loss of vacuum while at 100% power. The reactor is presently shutdown with decay heat rejection to the torus. The reactor operator (RO) has direction to maintain RPV pressure between 800 and 1000 psig using SRVs.

Shortly after closing the only open SRV, the RO becomes distracted responding to an unrelated transient condition. Subsequently, the RO observes RPV pressure being maintained automatically at 1135 psig.

Assuming SRVs are operating as designed, RPV pressure response is indicative of which of the following system responses?

A. 'K' and 'L' SRVs are being operated by the SRV Electric Lift System.

B. 'K' and 'L' SRVs are being operated by their mechanical setpoint.

C. 'D' and 'E' SRVs are being operated by the SRV Electric Lift System.

D. 'D' and 'E' SRVs are being operated by their mechanical setpoint.

K&A # 239002; A3.02
Importance Rating RO – 4.3; SRO - 4.3

QUESTION 40.

K&A Statement: Ability to monitor automatic operations of the **RELIEF/SAEFTY VALVES** including:
SRV OPERATION ON HIGH REACTOR PRESSURE

Justification: All the safety/relief valves (SRVs) can also be operated by the SRV Electric Lift System (SRVELS). The SRVELS opens the SRV's electrically by energizing the solenoid valves on the pilot stage assembly located on each SRV. The electric lift initiation is designed to break any corrosion bonding on the valve seat to assist the existing mechanical relief in performing its intended function. The SRVELS functions only as an electrical back up to the mechanical setpoint and does not prevent the mechanical portion of the SRV from operating as designed.

A. **CORRECT -** 'K' and 'L' SRVs lift setpoint is 1135 psig which is below the 'D' and 'E' SRV lift setpoint of 1140 psig.

B. **Incorrect but plausible if the candidate forgets** 'K' and 'L' SRVs lift setpoint is 1135 psig which is below the 'D' and 'E' SRV lift setpoint of 1140 psig. Also the SRV Electric Lift System is a backup to the mechanical lift.

C. **Incorrect but plausible if the candidate forgets** 'D' and 'E' SRVs lift setpoint is 1140 psig which is above the 'K' and 'L' SRV lift setpoint of 1135 psig.

D. **Incorrect but plausible if the candidate forgets** 'D' and 'E' SRVs lift setpoint is 1140 psig which is above the 'K' and 'L' SRV lift setpoint of 1135 psig. Also the SRV Electric Lift System is a backup to the mechanical lift.

References: SDLP-02J R12; ADS Student Ref: None

Learning Objective: 1.05.a.1

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7 / 45.7

QUESTION 41.

The plant is operating at 100% power. The Feedwater Level Control System is maintaining RPV water level at the desired reactor water level in three-element control.

The "A" main feed line flow D/P instrument fails downscale to zero and remains at zero.

Which one of the following describes the direction **AND** magnitude of the RPV water level change?

A. Level lowers. The reactor scrams when level reaches the low level scram setpoint.

B. Level rises. The turbine trips when level reaches the high turbine trip setpoint.

C. Level lowers. Power operation continues with level controlling below the master feedwater controller level setpoint.

D. Level rises. Power operation continues with level controlling above the master feedwater controller level setpoint.

K&A # 259002; K3.01
Importance Rating RO - 3.8; SRO – 3.8

QUESTION 41.

K&A Statement: Knowledge of the effect that a loss or malfunction of the **REACTOR WATER LEVEL CONTROL SYSTEM** will have on the following: **REACTOR WATER LEVEL**

Justification:

A. **Incorrect but plausible:** if candidate thinks the mismatch will lower feed system flow which would cause a lowering level.

B. **Incorrect but plausible:** Total feed flow signal decreases, feed flow/steam flow mismatch will cause the feed system flow to rise, however the level error will balance the feed/steam mismatch error and level will rise about 11" to 212". The turbine trip is at 222.5" and therefore the turbine will not trip. Plausible if the candidate does not know the level error can compensate for the feed flow error signal.

C. **Incorrect but plausible:** if candidate thinks the mismatch will lower feed system flow, which would cause a lowering level.

D. **CORRECT:** Total feed flow signal decreases, feed flow/steam flow mismatch will cause the feed system flow to rise, however the level error will balance the feed/steam mismatch error and level will rise about 11", up to 212". The turbine trip is at 222.5" and therefore the turbine will not trip.

References: SDLP-06 R12; FWLC System Student Ref: None

Learning Objective: 1.08.b

Question source: Modified

Question History: From JAF exam bank; Q-33

Cognitive level: Comprehensive/Analysis:

10CFR 41.7/45.5; 41.7/45.5 to 45.8

QUESTION 42.

It is required to vent the Primary Containment following a LOCA.

Which **ONE** of the following paths is the preferred Primary Containment Vent and Purge strategy for these conditions?

A. Vent the Torus through **ONE** Standby Gas Treatment Train and purge the Drywell.

B. Vent the Drywell through **ONE** Standby Gas Treatment Train and purge the Torus.

C. Vent the Torus through **BOTH** Standby Gas Treatment Trains and purge the Drywell.

D. Vent the Drywell through **BOTH** Standby Gas Treatment Trains and purge the Torus.

K&A # 261000; K1.03
Importance Rating RO – 2.9; SRO – 3.1

QUESTION 42.
K&A Statement: Knowledge of physical connection and/or cause-effect relationship between **STANDBY GAS TREATMENT SYSTEM and the following**: **SUPPRESSION POOL**

Justification:

A. **CORRECT:** The preferred vent/purge flow path is to vent from the torus and purge through the drywell. This flow path has the following advantages:
- Maintains pressure suppression function
- Scrubs gases before release
- Ensures effective vent and purge of both drywell and torus

If Primary Containment will be vented to maintain torus pressure below PCPL or PSP, then perform Subsection 5.2 alone or concurrently with any one applicable gas control strategy. Ensure only one SGT train is in service.

B. **Incorrect but plausible:** if candidate forgets the Torus is the preferred path because by requiring the Drywell atmosphere to be pulled through the Torus to be vented, release rates can be minimized. Torus water level is low enough at 13.9 feet to allow this, but this is not the preferred path.

C. **Incorrect but plausible:** if candidate forgets that the venting is to be done with only one SGT Train. While venting the primary containment, the in service SGT train could rupture and cause a ground level release. Ensure only one SGT train is in service.

D. **Incorrect but plausible:** if candidate forgets the Torus is the preferred path because by requiring the Drywell atmosphere to be pulled through the Torus to be vented, release rates can be minimized. Torus water level is low enough at 13.9 feet to allow this; While venting the primary containment, radiation dose rates will rise in the following areas:
- Along primary containment vent piping in the Reactor Building
- Above underground vent piping between the Reactor Building and Stack
- In the vicinity of the Stack
- In the Standby Gas Treatment System Room

While venting the primary containment, the in service SGT train could rupture and cause a ground level release. Ensure only one SGT train is in service

References: SDLP-01B; Standby Gas Treatment Student Ref: None

Learning Objective: 1.06.b

Question source: NMP #1; NRC 10/08; question 5

Question History: NMP #1; NRC 10/08; question 5

Cognitive level: Memory/Fundamental knowledge:
10CFR 41.7/45.6

QUESTION 43.

Assuming 100% power and a normal electrical alignment, which **ONE** of the following conditions on Bus 10500 would cause a trip of the Bus 10300-10500 4kV Tie Breaker 10304?

A. Bus voltage at 3825 volts for 7 seconds with a LOCA signal present.

B. Bus voltage at 3950 volts for 10 seconds with a LOCA signal present.

C. Bus voltage at 3825 volts for 50 seconds with **NO** LOCA signal present.

D. Bus voltage at 3950 volts for 60 seconds with **NO** LOCA signal present.

K&A # 262001; K4.02
Importance Rating RO - 2.9; SRO – 3.3

QUESTION 43.

K&A Statement: Knowledge of **A.C. ELECTRICAL DISTRIBUTION** design feature (s) and /or interlocks which provide for the following: **CIRCUIT BREAKER AUTOMATIC TRIPS**

Justification:

A. **Incorrect but plausible:** if candidate believes required bus voltage 76% for 2.5 seconds will trip ACB 10304 and 'C' EDG voltage >75%.

B. **Incorrect but plausible:** if candidate believes < 92% for only 7 seconds.

C. **CORRECT:** Bus voltage < 93% (3870 volts) for 45 seconds with No LOCA signal present.

D. **Incorrect but plausible:** if candidate believes 'A' EDG is < 75% and bus voltage < 71.5%.

References: SDLP-71E 4160 AC Distribution
TECHNICAL SPECIFICATIONS
3.3.8.1 (Amendment 274) and Bases
OP-22, R56

Student Ref: None

Learning Objective: 1.12.a

Question source: New

Question History: None

Cognitive level: Comprehensive/analysis

10CFR 41.7

QUESTION 44.

The plant is at 100% power in a normal electrical lineup.

Given the following:

- MCC-262 is de-energized.
- **ALL** equipment operates as expected.

Which **ONE** of the following describes the operational implication regarding the Uninterruptible Power Supply (UPS) System loads?

A. Loads remain energized since the UPS bus remains on its normal power supply.

B. Loads are momentarily de-energized when the UPS bus transfers to its emergency 120 VAC power supply.

C. Loads are momentarily de-energized when the motor generator swaps to the DC motor supply.

D. Loads remain energized since the circuit automatically swaps to the DC motor supply.

K&A # 262002; K6.01
Importance Rating RO – 2.7; SRO – 2.9

QUESTION 44.

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the **UNINTERRUPTABLE POWER SUPPLY (A.C. D.C.) - A.C. ELECTRICAL POWER**

Justification:

A. **Incorrect but plausible:** if candidate does not remember that MCC-262 is the emergency power supply

B. **Incorrect but plausible:** if candidate does not remember that the loads ar NOT momentarily de-energized.

C. **Incorrect but plausible:** if candidate does not remember that the loads ar NOT momentarily de-energized.

D. **CORRECT:** MCC-262 is the normal power supply to the UPS AC motor upon a lost of the AC motor a flywheel will maintain generator speed and output voltage while the DC motor is automatically connected. The UPS System which supplies vital loads consists of a double motor-generator set as the power source and an associated distribution panel. The generator is driven by either an AC or DC motor, both of which are coupled to the generator shaft. The normal drive is from the AC motor supplied from the 600 VAC Emergency Power System. Upon loss of power to the AC motor, the DC motor is automatically connected to the 125 VDC System. A flywheel maintains generator speed and output voltage constant during the transfer. Transfer back to the AC motor is by manual transfer only. Upon loss of the MG set, AC power is automatically supplied to the distribution panel from the 600 VAC Emergency Power system through a 600-120/240 VAC, single-phase transformer. Transfer back to the double motor generator set is by manual transfer only.

References: SDLP-71F R3; 120 VAC Distribution Student Ref: None
AOP-21, R22

Learning Objective: 1.12.a

Question source: JAF Bank

Question History:

Cognitive level: Memory

10CFR 41.7/45.7

QUESTION 45.

The plant is operating at 25% power with the following equipment conditions:

- 'A' RFPT and the Main Turbine are in service.
- RX WTR LVL COLUMN SEL 06-S1 Switch is in A-Level.
- The electrical distribution system is in a normal alignment.

The following annunciators alarm:

- 09-8-1-19, 125VDC BATT CHGR A AC SUPP TROUBLE
- 09-8-1-20, 125VDC BATT A VOLT LO

Assuming all automatic actions function as designed, which **ONE** of the following actions will occur IAW AOP-45, "Loss of DC Power System A"?

A. 'A' RFPT will trip on high RPV water level.

B. UPS Bus transfers from AC to DC drive.

C. Outboard MSIVs close on loss of DC power.

D. ADS Logic 'A' transfers to DC Power System 'B'.

K&A # 263000; K1.02
Importance Rating RO – 3.2; SRO – 3.3

QUESTION 45.

K&A Statement: Knowledge of the physical connections and/or cause-effect relationships between **D.C. ELECTRICAL DISTRIBUTION** and the following: **BATTERY CHARGER AND BATTERY**

Justification:

A. **Incorrect but plausible:** if the candidate forgets there is a loss of power to 'A' RFPT trip and control power.

B. **Incorrect but plausible:** if the candidate forgets there is a loss of power to the DC drive motor.

C. **Incorrect but plausible:** Only the inboard MSIVs will close due to a loss of AC and DC solenoid power. The outboard MSIVs will remain open.

D. **CORRECT:** AOP-45, Attachment #6; ADS Logic 'A' transfers to DC Power system 'B'.

References: SDLP-71B R8; DC Distribution
AOP-45 R9
ARP 09-8-1-19 R5
ARP 09-8-1-20 R5

Student Ref: None

Learning Objective: 1.05.C.1

Question source: New

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR 41.2 to 41.9/45.7 to 45.8

QUESTION 46.

During the starting sequence of an Emergency Diesel Generator (EDG), what signal results in a "start" signal to the associated ESW pump?

A. The energization of the EDG DC fuel pump closes contacts in the starting circuits for the ESW pumps.

B. A speed switch on the EDG's tachometer starts the ESW pumps at 400 RPM.

C. A speed switch on the EDG's tachometer starts the ESW pumps at 200 RPM.

D. The energization of the air start solenoids closes contacts in the starting circuits for the ESW pumps.

K&A # 264000; K1.04
Importance Rating RO – 3.2; SRO- 3.3

QUESTION 46.

K&A Statement: **K1.04** Knowledge **of the physical connections and/or cause-effect relationship between EMERGENCY GENERATORS (DIESEL/JET)** and the following:
EDG COOLING WATER SYSTEM

Justification:

A. **Incorrect but plausible if the candidate:** does not remember the ESW pumps do not start when the EDG is given a start signal which energizes the DC fuel pump but are provided a ACB close signal at EDG speed of 400 rpm.

B. **CORRECT:** At 400 rpm, the field of the generator is flashed, the generator space heaters are de-energized and the ESW pump ACB is closed.

C. **Incorrect but plausible if the candidate:** does not remember that the ESW pump ACB is closed at 400 rpm, that at 200 rpm the air start motor stops and contacts in the force parallel circuit close the generator tie breaker.

D. **Incorrect but plausible if the candidate:** does not remember the ESW pumps do not start when the EDG is given a start signal which energizes the air start motors but are provided a ACB close signal at EDG speed of 400 rpm.

References: SDLP-93 R16, OP-22 R56, OP-21 R37 Student Ref: None

Learning Objective: 1.05.b.1

Question source: Modified (JAF Bank) Modified distractor A & B

Question History: None

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7/45.7

QUESTION 47.

The unit is at 100% power with Instrument Air Dryer 39AD-4A in service. Dryer 39AD-4B is out of service for maintenance.

Instrument air pressure begins to slowly degrade. Current instrument air pressure is 94 psig and instrument air pressure is degrading at 1 psig every 3 minutes.

The following indications are observed:

- 09-6-2-37 AIR COMPR DRYER TROUBLE is received.
- 09-6-1-31 SERV AIR HDR PRESS LO is received.
- Purge flow is 18 scfm.
- Prefilter 5A D/P is 10 psid.
- HIGH DEW POINT light is **NOT** illuminated.

Which **ONE** of the following conditions could have caused instrument air pressure to degrade **AND** which procedure must be used to correct that condition?

A. Dryer high moisture, the moisture trap is clogged. Perform actions of OP-39, "Breathing, Instrument and Service Air" to remove the affected dryer tower from service.

B. High filter D/P, filter is clogged and needs to be changed. Enter AOP-12, "Loss of Instrument Air".

C. A failure to switch dryers has occurred. Perform actions of OP-39, "Breathing, Instrument and Service Air" to return the affected dryer tower to service.

D. A loss of dryer purge has occurred. Enter AOP-12, "Loss of Instrument Air".

K&A # 300000; A2.01
Importance Rating RO – 2.9; SRO – 2.8

QUESTION 47.

K&A Statement: Ability to (a) predict the impacts of the following on the **INSTRUMENT AIR SYSTEM** and (B) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
AIR DRYER AND FILTER MALFUNCTIONS

Justification:

A. **Incorrect but plausible:** Moisture indicator light is not illuminated, the moisture trap is not clogged. If high moisture was present the red HIGH DEW POINT light would be on. The high moisture condition cannot be cleared then remove the affected dryer tower from service per section G of OP-39.

B. **Correct:** High d/p in the filters indicate that the filters are clogged and need to be isolated to change. Enter AOP-12.

C. **Incorrect but plausible:** The switching failure alarm is an alarm only, no operating impact on the dryer. Initiate a CCR for Maintenance to repair.

D. **Incorrect but plausible:** Purge flow is normal. If a loss of purge flow was to occur, the dryer tower heater should be immediately de-energized and Maintenance immediately notified.

References: SDLP-39 R14
AOP-12, 26
09-6-2-37 R4

Student Ref: None

Learning Objective: 1.15.a

Question source: New

Question History: None

Cognitive level: Analysis

10CFR 41.5/45.6

QUESTION 48.

An unidentified leak from the Reactor Building Closed Loop Cooling (RBCLC) system has resulted in a loss of level in the RBCLC Makeup Tank in excess of makeup capability. The following conditions exist:

- 46P-2A Emergency Service Water Pump is tagged for maintenance.
- RBCLC Makeup Tank level is two (2) feet and slowly lowering.
- ALL RBCLC Pumps are operating.
- RBCLC pressure is 38 psig and slowly lowering.
- RBCLC supply temperature is 93 °F and slowly rising.
- Operators have been dispatched to search for the location of the leak.

In accordance with AOP-11, "Loss of Reactor Building Closed Loop Cooling", which **ONE** of the following choices lists immediate actions required at this time?

A. Scram the Reactor **AND** trip the Reactor Recirc Pumps.

B. Trip all RBCLC Pumps **AND** attempt to isolate the leak.

C. Scram the Reactor **AND** trip Reactor Water Cleanup Pumps.

D. Cross-tie Emergency Service Water Loops to RBCLC.

K&A # 400000: A2.02
Importance Rating RO –2.8; SRO – 3.0

QUESTION 48.
K&A Statement:

Ability to (a) predict the impacts of the following on CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

HIGH/LOW SURGE TANK LEVEL

Justification:

A. **CORRECT:** IAW AOP-11 immediate actions are to insert a manual scram and trip the Recirculation pumps.

B. **Incorrect but plausible if candidate:** Placing all RBCLC pumps in Pull-to-lock is step 6 of the immediate actions however attempting leak isolation is a subsequent action of AOP-11.

C. **Incorrect but plausible if candidates:** Tripping RWCU pumps is a subsequent operator action. Scramming the reactor is IAW AOP-11.

D. **Incorrect but plausible if candidates:** Cross-tie ESW loops to RBCLC IAW AOP-11 is a subsequent action of AOP-11 and The 46P-2B ESW pump will be in service since RBCLC header pressure is < 45 psig is an auto start signal for the ESW pumps.

References: SDLP-15, R-13; RBCLC system
AOP-11, "Loss of RBCLC" R16
Student Ref: None

Learning Objective: 1.05.c.1

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.2 to 41.9/ 45.7 to 45.8

QUESTION 49.

The following describes the initial plant conditions:

- Reactor is at 100% power.
- LPCI independent power supply is in a normal lineup per OP-43C, "LPCI Independent Power Supply System".

Subsequently, the following occurs:

- Main battery breaker 1CB2 to LPCI Inverter 71INV-3B opens.
- A LOCA occurs. The reactor scrams and RPV pressure is rapidly lowering.

Which **ONE** of the choices below describes the correct operator response to address the opening of Breaker 1CB2 per OP-43C **AND** the reason for this response?

	Operator Response	Reason
A.	Place LPCI MOV B PWR SUPP switch in BYPASS	Prevent 1CB1 from opening on an Injection Signal. Power MOV bus from the inverter.
B.	Place LPCI MOV B PWR SUPP switch in ALT PULL TO LOCK	Prevent 1CB1 from opening on an Injection Signal. Power MOV bus from the inverter.
C.	Place LPCI MOV B PWR SUPP switch in BYPASS	Transfer MOV bus power to Alternate feed.
D.	Place LPCI MOV B PWR SUPP switch in ALT PULL TO LOCK	Transfer MOV bus power to Alternate feed.

K&A # 203000; K1.07
Importance Rating RO – 3.1; SRO – 3.3

QUESTION 49.

K&A Statement: Knowledge of the physical connections and/or cause-effect relationships between **RHR/LPCI: INJECTION MODE** and the following: **D.C. ELECTRICAL POWER**

Justification:

A. Incorrect because the LPCI MOV independent power supply inverter shall NOT be lined up to power the bus with the main battery breaker open per precaution c.2.5 of OP-43c but plausible if the candidate thinks the charger can directly supply the bus.

B. Incorrect because although the correct action is to put switch in ALT Pull to Lock, the reason is NOT to prevent 1CB1 from opening. Plausible if the candidate does not know that ALT Pull to Lock isolates the charger and lines up the alternate feed.

C. Incorrect because bypass does NOT transfer to the alternate feed. But plausible if the candidate thinks BYPASS, bypasses the inverter with the alternate feed.

D. CORRECT: per OP-43C The LPCI MOV Power Supply switch is placed in ALT Pull To Lock if there is a loss of the battery or inverter. This will line up the alternate feed to the MOV bus.

References: SDLP-71B
OP-43C R18
ARP 09-8-5-7 R6
AOP-45 R9

Student Ref: None

Learning Objective: 1.10.a.3

Question source: New

Question History: None

Cognitive level: Analysis

10CFR 41.2 to 41.9 / 45.7 to 45.8

QUESTION 50.

The HPCI turbine speed control system is designed to prevent the turbine overspeed during turbine starts. With the flow controller in AUTOMATIC, which choice below describes the design feature that controls turbine speed during startup?

HPCI Turbine Governor Valve 23HOV-2_____.

A. is controlled by the ramp generator upon turbine start, independent of HPCI Turbine Stop Valve 23HOV-1 position.

B. is controlled by the ramp generator when HPCI Turbine Stop Valve 23HOV-1 leaves its full closed position.

C. remains closed until the HPCI Turbine Stop Valve 23HOV-1 reaches full open position.

D. remains closed until the HPCI Turbine Stop Valve 23HOV-1 opens beyond its mid-position.

K&A # 206000; K5.05
Importance Rating RO – 3.3; SRO – 3.3

QUESTION 50.

K&A Statement: Knowledge of the operational implications of the following concepts as they apply to **HIGH PRESSURE COOLANT INJECTION SYSTEM**: **TURBINE SPEED CONTROL**

Justification:

A. **Incorrect** because ramp generator function is interlocked with HOV-1 position. **Plausible** because reasonable that ramp generator controlling HOV-2 would not be dependent on HOV-1 position.

B. **CORRECT.** To prevent turbine over speeding on startup, the electronics of the flow controller are such that, the hydraulic control valve ramp generator will limit valve position when it senses HOV-1 off its closed seat.

C. **Incorrect** because ramp generator function is interlocked with HOV-1 position. **Plausible** because the distractor describes a method that would work to limit overspeed.

D. **Incorrect** because ramp generator function is interlocked with HOV-1 position. **Plausible** because the distractor describes a method that would work to limit overspeed.

References: SDLP-23 R15; HPCI ESK-1.61-51 Student Ref: None

Learning Objective: 1.05.a.22

Question source: Modified

Question History: JAF LOI-08 exam bank Q-31

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.5 / 45.3

QUESTION 51.

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

- Electrical plant is in a normal lineup with **NO** equipment out of service.
- T=0 seconds. Bus 10500 voltage degrades to 3800 volts.
- T=13 seconds. Bus 10500 voltage is restored to 4160 volts.

At T=18 seconds, and with **NO** operator action, what are the status of Bus 10500 and RPS Bus A?

A. Bus 10500 is energized by the 10300 Bus. RPS Bus A is energized.

B. Bus 10500 is energized by the 10300 Bus. RPS Bus A is **NOT** energized.

C. Bus 10500 is energized by the EDGs. RPS Bus A is energized.

D. Bus 10500 is energized by the EDGs. RPS Bus A is **NOT** energized.

K&A # 212000; K2.01
Importance Rating RO – 3.2; SRO – 3.3

QUESTION 51.

K&A Statement: Knowledge of electrical power supplies to the following: RPS motor-generator sets

Justification:

A. CORRECT. Protective action is initiated on 10500 Bus degraded voltage after a 41 to 46 second time delay with **NO** LOCA signal present. Degraded voltage will increase the current drawn by the RPS MG Set drive motor. However, it will continue to rotate at rated speed, providing normal MG Set output voltage to the RPS Bus. Since the undervoltage clears prior to automatic protective action, 10500 and the RPS Bus **BOTH** remain energized from normal power sources.

B. Incorrect. RPS MG Set EPA monitors trip on undervoltage after 4 seconds. However, since they monitor output voltage, which is unchanged, they will **NOT** trip and the RPS Bus will remain energized from the MG Set.

C. Incorrect. The EDG time delay for degraded voltage without a LOCA is 41-46 seconds. Plausible if the candidate does think the EDG time delay has actuated.

D. Incorrect. The RPS MG Set EPA will **NOT** trip. Plausible if the candidate thinks degraded voltage on the 10500 will lower RPS Bus voltage.

References: SDLP 71-E, 71-F, Alarm 09-8-2-24 R6 Student Ref: None

Learning Objective: 71E-1.09.e

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 52.

The plant is operating normally at 100% power when the drywell instrument header is lost.

Which **ONE** of the following describes ADS valve operation capabilities in this condition?

A. ADS valves can self actuate on high reactor pressure **ONLY**.

B. ADS valves can self actuate on high reactor pressure **AND** auto initiate on ADS logic **ONLY**.

C. ADS valves can self actuate on high reactor pressure, auto initiate on ADS logic, **OR** can be manually cycled five times.

D. ADS valves can self actuate on high reactor pressure, auto initiate on ADS logic, **OR** can be manually cycled three times.

K&A # 218000; K4.04
Importance Rating RO – 3.5; RO -3.6

QUESTION 52.

K&A Statement: Knowledge of **AUTOMATIC DEPRESSURIZATION SYSTEM** design feature(s) and /or interlocks which provide for the following: Insures adequate air supply to ADS valves: Plant specific

Justification:

A. Incorrect because there is a nitrogen accumulator that will supply the pneumatic force for pilot actuation of the ADS valves, but plausible if the candidate thinks that only the self actuation on high pressure is available on a loss of nitrogen event.

B. Incorrect because there is a nitrogen accumulator that will supply the pneumatic force for pilot actuation of the ADS valves, but plausible if the candidate thinks that only the self actuation on high pressure or ADS logic is available on a loss of nitrogen event.

C. Correct – The ADS valve can actuate on reactor high pressure without nitrogen. Each ADS valve has an accumulator and double check valves that ensure that there is enough nitrogen to supply up to 5 cycles with 120 psig initial pressure. The valves can be both operated manually by operators or by the ADS system.

D. Incorrect because there is enough supply for 5 cycles. Plausible if the operator does not remember how many cycles can be accomplished with the accumulators.

References: SDLP02J, OP-68, OP-37 R77 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 53.

Assume the following conditions are observed at 10 seconds following a spurious LOCA initiation signal to an emergency diesel (EDG).

- Lube oil pressure has risen from 0 to 16 psig.
- Engine speed has risen from 0 to 190 rpm.

(1) Which one of the following describes the automatic response of the EDG **AND**
(2) Why?

	(1)	(2)
A.	EDG shuts down immediately.	EDG trips if RPM does not increase adequately following a start signal.
B.	EDG comes up to rated speed and **DOES NOT** connect to the bus.	Conditions will not trip EDG.
C.	EDG shuts down immediately.	EDG trips on inadequate lube oil pressure.
D.	EDG comes up to rated speed and connects to the bus.	Conditions will not trip EDG.

K&A # 264000; K6.01
Importance Rating RO – 3.8; SRO – 3.9

QUESTION 53.

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the **EMERGENCY DIESEL GENERATORS**: **STARTING AIR**

Justification:

A. **CORRECT:** EDGs trip on RPM < 200 RPM at > 10 seconds after a start signal.

B. **Incorrect.** The EDG will trip. **Plausible** because some trips defeated on LOCA signal and electrical conditions would not allow auto sync of the EDG with its bus.

C. **Incorrect.** The EDG will trip on low RPM. The low oil pressure trip is bypassed on a LOCA start. **Plausible** because lube oil pressure is below trip setpoint.

D. **Incorrect.** The EDG will trip. **Plausible** if applicant doesn't remember which trips are defeated on a LOCA start and thinks the EDG will load on a LOCA signal without a low voltage on the bus.

References: SDLP-93 R16; Emergency Power ARP- 93ECP-A-1 R2 Student Ref: None

Learning Objective: 1.10.e

Question source: Modified

Question History: NRC NMP 1 6/07 Q-34

Cognitive level: Comprehensive/Analysis:

10CFR 41.7 /45.7

QUESTION 54.

Which of the following describes the electrical power supply and function of the backup scram valve and ARI solenoid upon RPS and ARI initiation to actuate and depressurize the scram pilot air header?

	Backup Scram Valve	ARI Solenoid
A.	125 VDC energized	120 VAC de-energized
B.	120 VAC de-energized	125 VDC energized
C.	125 VDC energized	125 VDC energized
D.	120VAC de-energized	120 VAC de-energized

K&A # 201001; K2.02
Importance Rating RO – 3.6; SRO – 3.7

QUESTION 54.
K&A Statement: Knowledge of electrical power supplies to the following:
SCRAM VALVE SOLENOIDS

Justification:

A. **Incorrect but plausible if the** candidates forgets Backup Scram valves are powered from 125 VDC and need to be energized to open. ARI valves are also powered from 125 VDC and need to energized to open.

B. **Incorrect but plausible if the** candidates forgets Backup Scram valves are powered from 125 VDC and need to be energized to open. ARI valves are also powered from 125 VDC and need to energized to open.

C. **CORRECT:** Backup Scram valves are powered from 125 VDC and need to be energized to open. ARI valves are also powered from 125 VDC and need to energized to open.

D. **Incorrect but plausible if the** candidates forgets Backup Scram valves are powered from 125 VDC and need to be energized to open. ARI valves are also powered from 125 VDC and need to energized to open.

References: SDLP-05 R16; CRD Hydraulic Student Ref: None

Learning Objective: 1.04.c and d

Question source: Modified

Question History: JAF exam bank

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7

QUESTION 55.

When attempting to WITHDRAW control rod 22-07, the following indications are observed:

- Drive water flow initially indicates 4.0 gpm, and the control rod moves into the core as expected for a control rod withdrawal.
- THEN, drive water flow becomes 0.0 gpm, and the control rod settles at the initial position.

Which of the following solenoid operated directional control valve failures has caused the above indications?

Directional Control Valve:

A. 120, WITHDRAW EXHAUST AND SETTLE VALVE, is stuck open.

B. 121, INSERT EXHAUST VALVE, is stuck open.

C. 122, WITHDRAW SUPPLY VALVE, is stuck closed.

D. 123, INSERT SUPPLY VALVE, is stuck closed.

K&A # 201002; K3.01
Importance Rating RO – 3.4; SRO – 3.4

QUESTION 55.

K&A Statement: Knowledge of the effect that a loss or malfunction of the **REACTOR MANUAL CONTROL SYSTEM** will have on the following: **ABILITY TO MOVE CONTROL RODS**

Justification:

A. **Incorrect but plausible if** 120 was stuck open the control rods would not insert as drive water flow through the 123 valve would flow directly to the exhaust header through the open 120. Additionally withdraw flow would equal stall flow for the CRD.

B. **Incorrect but plausible if** the 121 valve was stuck open the control rod would insert and not withdraw but the withdraw flow would be high as drive water flow through the 122 valve would flow directly to the exhaust header through the open 121.

C. **CORRECT:** When the rod movement control switch is moved to the ROD OUT position, the RMCS timer opens the inlet drive water valve (123) and the exhaust valve (121) and the control rod moves into the core and off the collet fingers. RMCS then should open the withdraw valve (122) and exhaust valve (120). If the withdraw valve (122) does NOT open no pressure is applied to the collet fingers or the area above the drive piston the control rod will settle back onto the collet finger at it's original position. This is further indicated by the 0.0 gpm drive water flow.

D. **Incorrect but plausible if** 123 was failed closed the control rod would not insert and there would be no insert flow. Additionally withdraw flow would equal stall flow for the CRD.

References: SDLP-03F; RMCS Student Ref: None

Learning Objective: 1.10.e

Question source: NMP 1 NRC 10/08 Q-31

Question History: NMP 1 NRC 10/08 Q-31

Cognitive level: Comprehensive/Analysis:

10CFR 41.7/ 45.5 to 45.8

QUESTION 56.

Given the following conditions:

- The IRMs are on Range 8.
- Control Rod 22-51 was just moved from Position 04 to Position 08.
- Control Rod 22-51 now drifts in to Position 00.
- Alarm Typer message is received: "RWM Inoperable: Rod drift still indicated even though two full core rod scans have been performed in response to initial rod drift indication".

Which one of the following describes how the Reactor Manual Control System will respond?

A. Rods can be inserted **OR** withdrawn. RWM generates an insert error.

B. Rods **CANNOT** be inserted but can be withdrawn. RWM generates an insert block.

C. Rods can be inserted but **CANNOT** be withdrawn. RWM generates a withdraw block.

D. Rods **CANNOT** be inserted **OR** withdrawn. RWM generates insert **AND** withdraw blocks.

K&A # 201006;K4.01
Importance Rating RO – 3.4; SRO – 3.5

QUESTION 56.

K&A Statement: Knowledge of **ROD WORTH MINIMIZER SYSTEM** design features(s) and/or interlocks which provide for the following: **INSERT BLOCKS/ERRORS**

Justification:

A. **Incorrect but plausible:** if the candidate forgets that the RWM program aborts will generate withdraw and insert blocks. Takes 3 insert errors to generate an insert block.

B. **Incorrect but plausible:** if the candidate forgets that the RWM program aborts will generate withdraw and insert blocks. Takes 3 insert errors to generate an insert block.

C. **Incorrect but plausible:** if the candidate forgets that the RWM program aborts will generate withdraw and insert blocks. 1 withdraw error will generate a withdraw block.

D. **CORRECT:** The control rod drift will cause a program abort of the RWM. Withdraw and insert blocks will be applied.

References: SDLP-03D; RWM Student Ref: None

Learning Objective: 1.05.b.7

Question source: NMP #1; NRC 3/07; Q-37

Question History: NMP #1; NRC 3/07; Q-37

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7

QUESTION 57.

The plant is in MODE 4, preparing for a plant startup. RWR Pump 'A' is being started in accordance with Section D of OP-27, "Recirculation System".

All prerequisites have been completed and the procedure has been performed up to the step that directs the operator to start the pump.

The RO starts RWR Pump 'A'.

Select the choice that correctly completes the following description of expected MG set response **AND** required operator action for a normal start sequence.

The MG set speed demand signal ramps up to _____ % speed, the field breaker closes, the MG set speed runs back to 30%. The operator must open the 'A' RWR pump discharge valve within _____ seconds?

	Speed Demand	Seconds
A.	80	30
B.	80	55
C.	44	30
D.	44	55

K&A # 202002; A4.01
Importance Rating RO – 3.3; SRO – 3.1

QUESTION 57.

K&A Statement: **202002 Recirculation Flow Control System A4.01:**Ability to manually operate and/or monitor in the control room: MG sets

Justification:

A. Incorrect. The operator must open the discharge valve within 55 seconds to prevent pump trip on valve position interlock. Plausible if applicant thinks the interlock is 30 seconds, which is the magnitude of the #1 limiter speed signal.

B. Correct. When the pump is started, a signal generator generates a 44% speed signal which bypasses the flow controller. The pump will ramp to 80% until the field breaker closes and then the #1 limiter runs back speed to 30%. The discharge valve must be opened within 55 seconds or the MG set will trip.

C. Incorrect. When the pump is started, a signal generator generates a 44% speed signal which bypasses the flow controller. The MG set will ramp to 80%. Plausible because a 44% speed signal is initially applied and plausible if applicant thinks the interlock is 30 seconds, which is the magnitude of the #1 limiter speed signal.

D. Incorrect. Plausible if the candidate does **NOT** know when the pump is started, a signal generator generates a 44% speed signal which bypasses the flow controller. The MG set will ramp to 80%. The discharge valve must be opened within 55 seconds or the MG set will trip.

References: OP-27 R67 Student Ref: None

Learning Objective: SDLP-02H, 1.05a2

Question source: New

Question History: None

Cognitive level: Memory/Fundamental Knowledge X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 58.

The plant is currently operating at 100% power:

- L-16 is de-energized.

Subsequently, a full reactor scram occurs and RPV water level lowers to 160 inches before it is recovered by the Feedwater Level Control system.

Which of the following correctly describes the resultant status of the Reactor Water Cleanup (RWCU) System based on the conditions above?

	RWCU Pumps	RWCU System Valve Status
A.	Will trip	12MOV-15 RWCU Supply Inbd Isol Valve **CLOSED**. 12MOV-18 RWCU Supply Outbd Isol Valve **CLOSED**. 12MOV-69 RWCU Return Containment Isol Valve **OPEN**.
B.	Will **NOT** trip	12MOV-15 RWCU Supply Inbd Isol Valve **CLOSED**. 12MOV-18 RWCU Supply Outbd Isol Valve **CLOSED**. 12MOV-69 RWCU Return Containment Isol Valve **OPEN**.
C.	Will trip	12MOV-15 RWCU Supply Inbd Isol Valve **OPEN**. 12MOV-18 RWCU Supply Outbd Isol Valve **OPEN**. 12MOV-69 RWCU Return Containment Isol Valve **CLOSED**.
D.	Will **NOT** trip	12MOV-15 RWCU Supply Inbd Isol Valve **OPEN**. 12MOV-18 RWCU Supply Outbd Isol Valve **OPEN**. 12MOV-69 RWCU Return Containment Isol Valve **CLOSED**.

K&A # 204000;K6.05
Importance Rating RO -2.6; SRO – 2.6

QUESTION 58.

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the **REACTOR WATER CLEANUP SYSTEM: AC POWER**

Justification:

A. **CORRECT:** L-16 powers 12MOV-69. RWCU pumps will trip on closure of 12MOV-18 due to the Group 2 isolation signal from the scram. RWCU Supply isolation Valves,12MOV-15 and 12MOV-18 will close.

B. **Incorrect but plausible if the** candidates forget that L-16 powers 12MOV-69. RWCU pumps will trip on closure of 12MOV-18. RWCU Supply isolation Valves,12MOV-15 and 12MOV-18 will close.

C. **Incorrect but plausible if the** candidates forget that L-16 powers 12MOV-69. RWCU pumps will trip on closure of 12MOV-18. RWCU Supply isolation Valves,12MOV-15 and 12MOV-18 will close.

D. **Incorrect but plausible if the** candidates forget that L-16 powers 12MOV-69. RWCU pumps will trip on closure of 12MOV-18. RWCU Supply isolation Valves,12MOV-15 and 12MOV-18 will close.

References: SDLP-11; R12 - RWCU Student Ref: None

Learning Objective: 1.09.a

Question source: New

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR 41.5/ 45.5

Question #59

Which of the following APRM and RBM channel combinations will be affected by a downscale failure of FLOW UNIT 'D'?

	APRM Channel	RBM Channel
A.	A	A
B.	F	B
C.	D	A
D.	E	B

K&A # 215002: A3.04
Importance Rating RO – 3.6; SRO – 3.5

QUESTION 59.

K&A Statement: Ability to monitor automatic operations of the **ROD BLOCK MONITOR SYSTEM** including: **VERIFICATION OR PROPER FUNCTIONING/OPERABILITY OF ROD BLOCK MONITOR**

Justification:

A. **Incorrect but plausible** - Flow channels A and C provide inputs to APRM A and RBM A

B. **Correct** - Flow channels B and D provide inputs to APRM B and RBM B

C. **Incorrect but plausible** Flow channels A and C provide inputs to APRM C and RBM A

D. **Incorrect but plausible** Flow channels A and C provide inputs to APRM E and RBM A

Reference SDLP-07C Student Ref: None

Learning Objective: 1.05.a.3.d

Question source: Modified Question from JAF Bank

Question History: Modified Question from JAF Bank

Cognitive level: Memory/Fundamental knowledge:

10CFR 41.7 / 45.7

QUESTION 60.

The plant is operating at 80% power when a fault on the grid causes a 50% mismatch between the main generator electrical output and turbine power to occur.

Assume **NO** operator actions.

What are the turbine and reactor responses to this condition?

A. Main turbine control valves close at normal rate. Reactor trips on high RPV pressure.

B. Main turbine control valves close at rapid rate. Reactor trips on high RPV pressure.

C. Main turbine control valves close at normal rate. Reactor remains on line.

D. Main turbine control valves close at rapid rate. Reactor trips on turbine control valve fast closure.

K&A # 245000; G2.1.28
Importance Rating RO – 4.1; SRO 4.1

QUESTION 60.

K&A Statement: **G2.1.28 Knowledge of the purpose and function of major system components and controls as it applies to Main Turbine Gen/Aux**

Justification:

A. Incorrect. Plausible if the applicant does not know turbine fast closure load reject circuitry. Reactor would trip on high pressure if control valves closed to reduce load by 50%.

B. Incorrect. Plausible because control valves do close rapidly from load reject circuitry. However, reactor will trip on the fast closure interlock before actuation of the high pressure trip.

C. Incorrect. Plausible if applicant does not know turbine fast closure load reject circuitry and final load is within turbine bypass capability.

D. Correct – Turbine control valves trip closed by fast acting solenoid due to load imbalance > 40%, reactor will trip on the fast closure interlock .

References: SDLP-94A, SDLP-94C, TS 3.3.1 Bases

Student Ref: None

Learning Objective: SDLP-94C 1.07.c

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 61.

A main steam line break has occurred.

Secondary containment has isolated on high radiation in the reactor building vent exhaust.

What operator actions are required per AOP-40, "Main Steam Line Break"?

Ensure Control Room and Relay Room ventilation are_____.

A. purged within 30 minutes

B. isolated within 30 minutes

C. purged within 60 minutes

D. isolated within 60 minutes

K&A # 290003; A4.04
Importance Rating RO – 2.8; SRO – 3.0

QUESTION 61.

K&A Statement: **Control Room HVAC A4.04 Ability to manually operate and/or monitor in the control room: Environmental conditions**

Justification:

A. Incorrect- When there is contamination the correct mode for CR HVAC is isolate. Purge is used for smoky environments. Plausible because the purge is a mode of CR HVAC.

B. Correct-CR and relay room must be isolated within 30 minutes per AOP-40.

C. Incorrect- When there is contamination the correct mode for CR HVAC is isolate. Purge is used for smoky environments. Plausible because the purge is a mode of CR HVAC.

D. Incorrect- the immediate actions require isolation within 30 minutes, plausible because AOP-40 requires TSC to be isolated within 60 minutes.

References: AOP-40 R10 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: N/A

Cognitive level: Memory: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 62.

Given the following conditions:

- The plant is in MODE 1, two weeks after a refueling outage.
- The Fuel Pool Cooling and Cleanup (FPCC) System is operating with one pump and heat exchanger in service.
- **NO** makeup water sources are available.
- Assume **NO** evaporative losses.

Which of the following is the effect on Spent Fuel Pool water temperature **AND** level if a leak develops on the common FPCC pump suction?

	Spent Fuel Pool Temperature	Spent Fuel Pool Water Level
A.	Remain stable	Water level will lower slightly then stabilize
B.	Will rise	Water level will continuously lower
C.	Remain stable	Water level will continuously lower
D.	Will rise	Water level will lower slightly then stabilize

K&A # 233000; K3.01
Importance Rating RO – 3.2; SRO – 3.4

QUESTION 62.

K&A Statement: Knowledge of the effect that a loss or malfunction of the **FUEL POOL COOLING AND CLEANUP** will have on the following: **FUEL POOL TEMPERATURE**

Justification:

A. **Incorrect but plausible** The skimmer surge tank will drain and the FPCC pumps will trip low-low skimmer surge tank level. Fuel pool level will drain to the bottom of the weir overflow pipe then stop. Water temperature will rise because the FPCC pumps are tripped. Temperature rise causes fuel pool water to expand, level is maintained at the weir.

B. **Incorrect but plausible** The skimmer surge tank will drain and the FPCC pumps will trip low-low skimmer surge tank level. Fuel pool level will drain to the bottom of the weir overflow pipe then stop. Water temperature will rise because the FPCC pumps are tripped. Temperature rise causes fuel pool water to expand, level is maintained at the weir.

C. **Incorrect but plausible** temperature will rise since there is no cooling and water level will not continuously lower. Fuel pool is designed for zero leakage. Temperature will rise level is maintained at the weir.

D. **CORRECT:** The skimmer surge tank will drain and the FPCC pumps will trip low-low skimmer surge tank level. Fuel pool level will drain to the bottom of the weir overflow pipe then stop. Water temperature will rise because the FPCC pumps are tripped. Temperature rise causes fuel pool water to expand, level is maintained at the weir.

References: SDLP-19 R12; Fuel Pool Cooling Student Ref: None

Learning Objective: 1.09.a

Question source: HC NRC exam 11/05; Q-65

Question History: HC NRC exam 11/05; Q-65

Cognitive level: Comprehensive/Analysis:

10CFR 41.7 / 45.6

Question 63

The plant is conducting a refueling outage with the following conditions:

- Fuel Handling evolutions are in progress
- All IRM Range Switches are on Range 1
- The Reactor Mode Switch is in the **REFUEL** position

Given the above conditions, which of the following conditions will cause a **ROD BLOCK**?

A. Refuel Bridge over core **with** a Hoist Unloaded Signal

B. One control rod at position 02 **and** a second control rod selected

C. Refuel Bridge over core **with** grapple FULL UP

D. One control rod at position 48 **and** SRMs are all indicating 8 cps

K&A # 234000; K5.02
Importance Rating RO – 3.1; SRO – 3.7

QUESTION 63.

K&A Statement: Knowledge of the operational implications of the following concepts as they apply to **FUEL HANDLING EQUIPMENT: FUEL HANDLING EQUIPMENT INTERLOCKS**

Justification:

A. **Incorrect but plausible** if the candidate does not recall that the hoist requires a load to cause the rod block.

B. **CORRECT:** Rod blocks are enforced through the Rod Movement Control System. This circuitry will prevent selection of control rods or inhibit their motion. These blocks are Mode Switch dependent.
With the Mode Switch in Refuel:
- Refueling bridge near or over the core with a sensed load on any refueling platform hoist or fuel grapple not fully up.
- Selection of a second control rod for movement with any other control rod withdrawn from the fully inserted position.

C. **Incorrect but plausible** if the candidate does not recall that the hoist not being full up will cause the rod block

D. **Incorrect but plausible** if the candidate does not recall that the SRM downscale rod block occurs with SRM counts $\leq$ 5 cps

References: SDLP-08B; R07, Student Ref: None

Learning Objective: 1.05.b.2.a

Question source: Modified JAF Bank Question

Question History: Modified JAF Bank Question

Cognitive level: Comprehensive/Analysis:

10CFR 41.5 /45.3

QUESTION 64.

During a plant startup, with reactor power at 15%, Control Rod 06-23 was selected and the following indications occurred:

- 09-5-2-2, ROD WITHDRAWAL BLOCK is in alarm.
- 09-5-2-1, RWM ROD BLOCK RPIS INOP is in alarm.
- **ALL** rod position lights on the Four Rod Display are lost.
- **ALL** red Full-Out **AND** green Full-In lights on Full Core Display are lost.

Which **ONE** of the following malfunctions would explain **ALL** of these indications?

A. Loss of Panel 71AC10 Relay Room Distribution Panel

B. Loss of Panel 71RBACB5 Reactor Building Distribution Panel

C. Loss of Panel 71ACUPS Relay Room Uninterruptible Bus Distribution Panel

D. Loss of Panel 71ESSA1 Relay Room Safeguard Power Distribution Panel

K&A # 214000: A3.02
Importance Rating RO – 3.2; SRO – 3.1

QUESTION 64.

K&A Statement: **A3.02** Ability to monitor automatic operations of the **ROD POSITION INFORMATION SYSTEM** including: Alarm and indicating lights

Justification:

A. **Incorrect but plausible because** power supply 71AC10 powers Reactor Manual Control Rod Block Relays which if lost could give a rod block.

B. **Incorrect but plausible because** power supply 71RBACB5 powers CRD accumulator panel which if lost could give a rod block.

C. **CORRECT** – The loss of the UPS panel 71ACUPS will cause a loss of rod position indication which inputs into the RWM, which will give the rod block.

D. **Incorrect but plausible because power supply** 71ESSA1 powers various safety related logic circuits which are 120 VAC.

References: AOP-21 R22, OP-46B R27 Student Ref: None

Learning Objective: SDLP-03F R10; RMCS 1.04a

Question source: Fitzpatrick NRC Exam

Question History: 2003 Q58

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 65.

Given the following:

- The plant is critical in Mode 2
- RPV Pressure is 550 psig
- 'A' CRD pump is out of service for maintenance.
- 'B' CRD pump tripped at 10:00 due to a motor fault.
- At 10:05 5 different CRD trouble alarms were received due to low accumulator pressure. The associated control rods are fully withdrawn.

Which **ONE** of the following describes the current control rod insertion capability?

A. Manual insertion capability is available using Reactor Manual Control System (RMCS). Scram function is fully assured.

B. Manual insertion capability is **NOT** available using RMCS. Scram function is fully assured.

C. Manual insertion capability is available using RMCS. Scram function **MAY** be degraded.

D. Manual insertion capability is **NOT** available using reactor RMCS. Scram function **MAY** be degraded.

K&A # 201003; K1.01
Importance Rating RO – 3.2; SRO – 3.3

QUESTION 65.

K&A Statement: **Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROD AND DRIVE MECHANISM and the following: Control drive hydraulic system**

Justification:

A. **Incorrect** but plausible if the applicant thinks that manual insertion is available without drive pressure.

B. **Incorrect** but plausible if the candidate thinks that the rods with the low accumulator pressure will still scram at normal speeds with only 550 psig reactor pressure.

C. **Incorrect** but plausible if the applicant thinks that manual insertion is available without drive pressure.

D. **CORRECT.** With both CRD pumps off, there is no drive pressure, so manual insertion using RMCS is not possible. Also CRD charging water pressure is lost. The 5 accumulators that have low pressure are inoperable and with reactor pressure less than 800 psig, reactor pressure alone may not insert the rods within the tech spec limits, so the scram function is degraded.

References: SDLP-03C R15 TS Bases 3.1.4 Student Ref: None

Learning Objective: NA

Question source: Bank Hope Creek

Question History:

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41

Question #66

The following plant conditions exist:

- Reactor Mode Selector Switch is in "SHUTDOWN"
- **ALL** reactor vessel head closure bolts are fully tensioned
- Average Reactor Coolant Temperature is 215°F

Which of the following correctly describes the **MINIMUM** shift manning requirements for the above conditions IAW EN-OP-115?

a.	1 Shift Manager (SRO Licensed) 1 Licensed Reactor Operator 1 Non-Licensed Operator
b.	1 Shift Manager (SRO Licensed) 1 Control Room Supervisor (SRO Licensed) 1 Field Support Supervisor / Shift Technical Advisor 3 Licensed Reactor Operators 5 Non-Licensed Operators
c.	1 Shift Manager (SRO Licensed) 1 Control Room Supervisor (SRO Licensed) 1 Field Support Supervisor / Shift Technical Advisor 2 Licensed Reactor Operators 2 Non-Licensed Operators
d.	1 Shift Manager (SRO Licensed) 1 Licensed Reactor Operator 4 Non-Licensed Operator

K&A # G.2.1.5
Importance Rating 2.9

QUESTION 66.
K&A Statement: **Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.**

Justification:

A. Incorrect – This is the minimum staff, taking into consideration tech specs and 10CFR50.54 for mode 4 and 5.

B. Correct – This is the minimum staff, taking into consideration tech spec and safe shutdown for mode 3 , IAW EN-OP-115.

C. Incorrect – This is the minimum staff, taking into consideration tech specs and 10CFR50.54 for mode 3.

D. Incorrect – This is the minimum staff, taking into consideration tech specs and fire brigade mode 4 and 5.

References: EN-OP-115 Student Ref: EN-OP-115; addendum 10.5 shift manning

Learning Objective: LP-AP 46.01f

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10

Question # 67

- The Reactor is critical and a plant heat up is underway with RPV pressure at 500 psig.
- AOP-24, Stuck Control Rod, is being performed due to a stuck control rod.
- The Reactor Operator is cautioned by AOP-24 that drive water differential pressure is limited to less than 600 psid with reactor pressure below 650 psig.

Adhering to this caution is intended to **specifically** prevent which one of the following from occurring?

A. control rod double notching

B. drive mechanism seal damage

C. violating RWM sequence

D. CRD pump damage

K&A # G.2.1.32
Importance Rating 3.8

QUESTION 67.
K&A Statement: Ability to explain and apply system limits and precautions.
Justification:

A. **Incorrect but plausible** if the candidate believes control rod double notching is the concern instead of damage to the drive mechanism seals which would result in increased scram times. Elevated drive water D/P could result in a mis-positioned control rod due to increased control rod speed, this could occur at any elevated drive water D/P when it is raised in 50 psi increments up to 400 psid but per the stem, the concern is when the drive water D/P is greater than 600 psid and RPV pressure is less than 650 psig.

B. **Correct** - damage to the drive mechanism seals can occur which would result in increased scram times.

C. **Incorrect but plausible** if the candidate believes violating the RWM sequence (due to the potential for a rod to move greater than one notch as a result of the elevated drive water D/P) if <10% power is the concern instead of damage to the drive mechanism seals which would result in increased scram times. Elevated drive water D/P could result in a mis-positioned control rod which could result in violating the RWM sequence due to increased control rod speed, this could occur at any elevated drive water D/P but per the stem, the concern is when the drive water D/P is greater than 600 psid and RPV pressure is less than 650 psig.

D. **Incorrect but plausible** if the candidate believes raising the drive water D/P will cause CRD pump damage (raising the drive water D/P does NOT place a significantly greater load on the CRD Pump) vice the concern instead of damage to the drive mechanism seals which would result in increased scram times.

References: AOP-24 R9 Student Ref: None

Learning Objective: LP-AOP, OBJECTIVE: 1.04, 1.10

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

QUESTION 68.

Plant conditions are as follows:

- Reactor power is 80%.
- Control Rod 22-47 is at Position 08.

Control Rod 22-47 is required to be withdrawn to Position 24 for a rod pattern adjustment.

The Reactor Engineer recommends continuous control rod withdrawal. Control Rod 22-47 is **NOT** identified as **FAST** on the list of CRD deficiencies.

Given the recommendation by the Reactor Engineer, WHICH **ONE** of the following choices completes the statement below to describe the procedural restrictions for continuous control rod withdrawal of Control Rod 22-47 in accordance with OP-26, "Control Rod Drive Manual Control System"?

Continuously withdraw the rod and release the rod control switches at least _____.

A. 2 notches before the intended position, then single notch withdraw to position 24

B. 2 notches before the intended position, then continuously withdraw to position 24

C. 3 notches before the intended position, then single notch withdraw to position 24

D. 3 notches before the intended position, then continuously withdraw to position 24

K&A # G.2.1.20
Importance Rating RO- 4.6, SRO-4.6

QUESTION 68.
K&A Statement: Ability to interpret and execute procedure steps.
Justification:

A. In-Correct but plausible if applicant believes that continuous withdraw does not need to be released until 2 notches prior to the intended target.

B. In-Correct but plausible if applicant believes that continuous withdraw does not need to be released until 2 notches prior to the intended target.

C. Correct – In accordance with OP-26, Section C.2.6, for continuous withdraws, the withdraw signal is removed at least three notches before its target position. Single notch withdraws are then performed to the target position.

D. In-Correct but plausible if applicant knows that continuous must be stopped three notches before target but believes that continuous can be used again to achieve target position.

References: OP-26 R23 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10

QUESTION 69.

Plant conditions 24 hours ago were as follows:

- Reactor power stable at 80%
- Total RCS LEAKAGE at 6.7 gpm
- Identified RCS LEAKAGE at 3.5 gpm
- Unidentified RCS LEAKAGE at 3.2 gpm

Current conditions are as follows:

- Reactor power stable at 75%
- Total RCS LEAKAGE at 12.1 gpm
- Identified RCS LEAKAGE at 5.7 gpm
- Unidentified RCS LEAKAGE at 6.4 gpm

Which **ONE** of the following describes the impact of these changed conditions on limits of Tech Spec 3.4.4, "RCS Operational Leakage?"

A. Unidentified RCS LEAKAGE is within limits; however, the increase in unidentified RCS LEAKAGE exceeds Tech Spec limits.

B. Total RCS LEAKAGE exceeds Tech Spec limits **AND** the unidentified RCS LEAKAGE exceeds Tech Spec limits.

C. Identified RCS LEAKAGE is within limits; however, the increase in the identified RCS LEAKAGE exceeds Tech Spec limits.

D. Unidentified RCS LEAKAGE **AND** the increase in unidentified RCS LEAKAGE **BOTH** exceed Tech Spec limits.

K&A # G.2.2.38
Importance Rating 3.6

QUESTION 69.
K&A Statement: Knowledge of conditions and limitations in the facility license
Justification:

A. Incorrect-because the unidentified leakage is not within the Tech Spec limit. Plausible if the candidate does not know the Tech Spec limit for UILR.

B. Incorrect because the Total leakage is within the Tech Spec limit. Plausible if the candidate does not know the Tech Spec limit for Total Leakage.

C. Incorrect– There is no Tech Spec limit on increase in identified leakage. Plausible if the candidate does not know the difference between identified and unidentified limits.

D. Correct- because the UILR is >5 gpm and the increase in UILR is >2 gpm.

References: TS 3.4.4 (Amendment 274) Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 70.

A plant startup is in progress with the Mode Selector Switch in STARTUP. Control rods are being withdrawn.

- The Rod Worth Minimizer (RWM) has just failed with 11 control rods withdrawn.

Per OP-64, "Rod Worth Minimizer", what actions are required to continue the startup with the 11 control rods withdrawn?

A. Verify >10 control rods have been withdrawn, bypass the RWM, verify all further control rod movements are in compliance using a second individual, licensed operator **ONLY**.

B. Verify by administrative methods that startup with RWM inoperable has not been performed in the past 365 days, bypass the RWM, verify all further control rod movements are in compliance using a second individual, licensed operator **ONLY**.

C. Verify >10 control rods have been withdrawn, bypass the RWM, verify all further control rod movements are in compliance using a second individual, licensed operator **OR** Shift Technical Advisor.

D. Verify by administrative methods that startup with RWM inoperable has not been performed in the current calendar year, bypass the RWM, verify all further control rod movements are in compliance using a second individual, licensed operator **OR** Shift Technical Advisor.

K&A # G.2.2.1
Importance Rating RO- 4.5, SRO-4.4

QUESTION 70.

K&A Statement: Ability to perform pre-startup procedures for the facility/including operating those controls associated with plant equipment that could affect reactivity.

Justification:

A. Incorrect because >10 rods is incorrect. If 12 rods had been withdrawn then this action would be correct. Plausible if candidate does not know that need >12 rods to continue. Also a STA can be the second individual.

B. Incorrect because only one startup with RWM inoperable in a calendar year is allowed by TS. Plausible if the candidate believes that the startup can continue if a startup with the RWM has not been performed in the past 365 days. Also a STA can be the second individual.

C. Incorrect because >10 rods is incorrect. If 12 rods had been withdrawn then this action would be correct. Plausible if candidate does not know that need >12 rods to continue.

D. Correct – OP-64 and TS 3.3.2.1 allow continuing startup if administratively verify that startup with inoperable RWM has not been performed this calendar year and station second qualified individual, licensed operator or STA.

References: OP-64 R18 Section E
TS 3.3.2.1 (Amendment 274)

Student Ref: None

Learning Objective: SDLP-3D, EO-1.15.A

Question source: Modified Fitzpatrick 03 exam

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.5
&10

QUESTION 71.

Personnel are preparing to enter the Drywell at power to investigate a problem.

Which **ONE** of the following correctly completes the statement regarding the requirements of AP-12.02, "Drywell Entries During Primary Containment"?

Personnel may enter the Drywell when reactor power is less than or equal to __(1)__ and oxygen concentration is greater than __(2)__.

A. (1) 15%
(2) 19.5%

B. (1) 20%
(2) 19.5%

C. (1) 15%
(2) 17.5%

D. (1) 20%
(2) 17.5%

K&A # G.2.3.12
Importance Rating 3.2

QUESTION 71.

K&A Statement: Knowledge of radiological safety principles pertaining to licensed operator duties such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Justification:

A. Correct – IAW AP-12.02 max power may be 15%. The drywell must be de-inerted which requires oxygen to be 19.5% or above.

B. Incorrect because power must be at or below 15%. Plausible if the candidate does not know the limit is 15%.

C. Incorrect because minimum oxygen concentrations are 19.5%. Plausible if the candidate does not know what constitutes a safe minimum oxygen level.

D. Incorrect because power must be at or below 15%. Plausible if the candidate does not know the limit is 15%.

References: AP-12.02 R17 Student Ref: None

Learning Objective: LP-AP EO 45.04, and 45.05

Question source: VY 09 Q70

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.12

QUESTION 72.

The plant is operating at 100% power when the following events occur:

- 09-3-2-38, OFF GAS RAD MON HI-HI alarm is received.
- The Reactor Operator reports **BOTH** Offgas Radiation Monitors are at the Hi-Hi alarm setpoint.

Which **ONE** of the following states the immediate actions per AOP-3, "High Activity in Reactor Coolant or Off-Gas"?

A. Ensure 01-107AOV-100, Off Gas Discharge to Stack is closed.

B. Insert a manual scram per AOP-1, "Reactor Scram".

C. Secure hydrogen addition flow to reduce radiation levels.

D. Lower reactor power as necessary to control radiation levels.

K&A # G.2.3.11
Importance Rating RO 3.8, SRO

QUESTION 72.
K&A Statement: Ability to Control Radiation Releases
Justification:

A. Incorrect because the isolation has **NOT** occurred when the high offgas alarm comes in, (timer needs to time out) but plausible if the candidate thinks the system has isolated. This is an override if the timer has timed out.

B. Incorrect because AOP-3 does **NOT** direct a scram **UNTIL** the timer has timed out. However, plausible if the candidate thinks this action is required as an immediate action. This is an override if the timer has timed out.

C. Incorrect because this is a subsequent action of AOP-3 (step F.2.3) and is **NOT** an immediate action. Plausible because a reduction in hydrogen injection will reduce the radiation levels.

D. Correct – With an off gas hi-hi rad alarm in, AOP-3 directs to reduce power to control radiation levels below the alarm setpoint.

References: AOP-3 R16 Student Ref: None

Learning Objective: N/A

Question source: NMP1 09 q71

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.11

QUESTION 73.

The plant is operating at 100% power when conditions develop which requires entry into EOP-5, "Secondary Containment Control".

IF	THEN
A primary system is discharging into the reactor building **AND** Any Reactor Building Area Temperature is above its Max Normal value	**BEFORE** any Reactor Building Area Temperature reaches its Maximum Safe value, enter EOP-2, "RPV Control," and execute it concurrently with this procedure.

Per BWROG Emergency Procedure Guidelines/Severe Accident Guidelines, what is the basis for entering EOP-2, "RPV Control," and initiating a reactor scram before any Reactor Building Area Temperature reaches its Maximum Safe value?

A. To terminate the fissioning process since Emergency Depressurization is required.

B. To lower reactor power to decay heat to reduce the energy that may be discharged to the Secondary Containment.

C. To ensure the reactor is shutdown prior to the leak getting larger.

D. To allow the steam leak to depressurize the reactor thereby reducing the release rate.

K&A # G.2.4.6
Importance Rating 3.7

QUESTION 73.
K&A Statement: Knowledge of EOP mitigation strategies
Justification:

A. **Incorrect but plausible if candidate forgets** that only a single area is approaching it's maximum safe area temperature. Emergency de-pressurization requires 2 or more Reactor Building (RB) area temperatures above their maximum safe operating (MSO) values. Should RB temps exceed their MSO values in more than one area and a primary system is discharging outside primary containment (PC), the RPV must be depressurized to preclude further temp increases. RPV depressurization places the primary system in its lowest possible energy state, rejects heat to the torus in preference to outside the containment, and reduces the driving head and flow of primary systems that are unisolated and discharging into the RB and preserving Secondary Containment (SC).

B. **CORRECT:** If any RB area temps listed in EOP-5 approach their maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EOP actions can no longer be assured. EOP-2 RPV Control must be entered to make certain the reactor is scrammed. This reduces the energy that the RPV may be discharging to the SC to decay heat levels. An explicit direction to scram the reactor is not provided in this step. The MSO temp for RCIC is 137 °F.

C. **Incorrect Plausible if candidate forgets** that only a single area is approaching it's MSO temp. Shutting down the reactor prior to the leak getting larger is not a concern per the bases. The reactor shutdown is based on room temperatures approaching their MSO temp.

D. **Incorrect Plausible if candidate forgets** that the RCIC room is not an enclosed by watertight doors. If a steam leak were to occur it would spread to other rooms (e.g. the west crescent area, RCIC drywell entrance, and RHR Heat exchanger room).
Only a single area is approaching its MSO temp. Should RB temps exceed their MSO values in more than one area and a primary system is discharging outside PC, the RPV must be depressurized to preclude further temp increases. RPV depressurization places the primary system in its lowest possible energy state, rejects heat to the torus in preference to outside the containment, and reduces the driving head and flow of primary systems that are unisolated and discharging into the RB and preserving SC.

References: MIT-301.11F Student Ref: None
EOP-5 R7

Learning Objective: 1.07

Question source: HC 2/09; Q-61

Question History: HC 2/09; Q-61

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10

QUESTION 74.

A plant transient has resulted in the following plant conditions:

- Reactor is shutdown.
- Emergency Depressurization has been performed.
- Reactor Pressure is 50 psig and steady.
- RPV water level is -25 inches and steady.
- A Core Spray subsystem is injecting at 4825 gpm.

Is core cooling adequate and why?

A. Yes. Steam cooling with injection **AND** reactor level above -31.5 inches ensures adequate core cooling.

B. No. Steam cooling with injection requires reactor level above -19 inches. Since level is at -25 inches adequate core cooling **CANNOT** be ensured.

C. Yes. Core Spray injecting at or above 4725 gpm **AND** reactor level above -44.5 inches ensures adequate core cooling.

D. No. Spray cooling requires Core Spray injecting at or above 4925 gpm. Since Core Spray is injecting at 4825 gpm adequate core cooling **CANNOT** be ensured.

K&A # G.2.4.17
Importance Rating 3.9

QUESTION 74.
K&A Statement: Knowledge of EOP terms and definitions
Justification:

A. Incorrect because steam cooling with injection requires RPV level to be above -19 inches. Plausible if the candidate does not know the minimum level for steam cooling with injection.

B. Incorrect because adequate core cooling is ensured with spray cooling. Plausible if the candidate does not know the spray cooling requirements.

C. Correct - Spray cooling is defined to exist when a Core Spray loop is injecting at or above 4725 gpm. With level above -44.5 inches and core spray injecting above the minimum flow adequate core cooling is ensured.

D. Incorrect because the correct flow is 4725 gpm. This answer is a plausible distractor if the candidate does not know the minimum core spray flow for spray cooling.

References: EOP-2 R9
MIT 301.11C
Student Ref: None

Learning Objective: MIT 301.11C LO 3.0

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

QUESTION 75.

A plant startup is in progress after a refueling outage, with the following conditions:

- RPV pressure is 970 psig.
- Main Turbine Bypass Valve BPV-1 is 50% open.
- Reactor power is being raised per OP-65, Reactor Power Ascension, to achieve BPV-1 near full open.
- Preparations are being made to calibrate APRM's per ST-5D.
- All plant parameters are within normal range.
- All equipment is operable.

Then the following conditions occur:

- PREFERRED POWER (red) light on Panel 09-43 is off.
- EPIC-D-912 INTERPOSING RLY SYS LOSS OF AC is in alarm.
- 09-5-1-58, INTERPOSING RELAY SYS ON DC PWR is **NOT** in alarm.

Based on these indications, what is the annunciator status, and what is required in accordance with AOP-65, Loss of Control Room Annunciators?

A. The Control Room annunciators are available on backup power. The power ascension **AND** APRM calibration may continue as directed by the Shift Manager.

B. The Control Room annunciators are available on backup power. The power ascension **AND** APRM calibration may **NOT** continue.

C. The Control Room annunciators are **NOT** available. The power ascension **AND** APRM calibration must be suspended until control room annunciators are returned to service.

D. The Control Room annunciators are **NOT** available. The power ascension must be suspended until control room annunciators are returned to service, however the APRM calibration may continue.

K&A # G.2.4.32
Importance Rating 3.6

QUESTION 75.
K&A Statement: Knowledge of operator response to loss of all annunciators
Justification:

A. Incorrect because a loss of AC power only would cause Annunciator 09-5-1-58 INTERPOSING RELAY SYS ON DC PWR to alarm, but plausible because the candidate may conclude that power is still available.

B. Incorrect a loss of AC power only would cause Annunciator 09-5-1-58 INTERPOSING RELAY SYS ON DC PWR to alarm, but plausible because the candidate may conclude that power is still available

C. Correct – AOP-65 immediate actions require to avoid any transients and secure all surveillance/testing activities

D. Incorrect but plausible if the candidate thinks that the APRM calibration must be done in this situation because of the power level.

References: AOP-65 R3
OP-65 R109
Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

Fitzpatrick Station

2010 NRC Written Exam

SRO Portion

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EXAMPLE OF STUDENT SCORE:

PART 2

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52	A	B		D	E
53		B	C	D	E
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96	A	B	C		E
97		B	C	D	E
98	A	B		D	E
99		B	C	D	E
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- Mark total possible subjective points
- Only one mark per line on key
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39	A	B	C		E
40		B	C	D	E
41	A	B	C		E
42		B	C	D	E
43	A	B		D	E
44	A	B	C		E
45	A	B	C		E
46	A		C	D	E
47	A		C	D	E
48		B	C	D	E
49	A	B	C		E
50	A		C	D	E

FEED THIS DIRECTION

07 08 09 M12 3806 999 12 11 10 9 8 7 6 5 4 3 2 1

SRO Exam Key

QUESTION 76.

Plant conditions are as follows:

- Unit startup is in progress per OP-65, "Startup and Shutdown".
- Reactor Power is 24%.
- Main Generator is on-line.

The Main Turbine bearing header pressure drops to 16 psig.

Which **ONE** of the following choices describes the expected plant response to the above **AND** the procedure to respond to the event?

A. (1) Main Turbine Trips.
(2) Reactor Power rises.
(3) Enter AOP-2, "Main Turbine Trip Without a Scram"

B. (1) Main Turbine remains on line.
(2) Shutdown the Main Turbine per OP-9, "Main Turbine".

C. (1) Main Turbine Trips.
(2) Reactor Scrams.
(3) Enter AOP-1, "Reactor Scram".

D. (1) Main Turbine Trips and Reactor Recirculation Pumps Trip.
(2) Reactor Scrams.
(3) Enter EOP-2, "RPV Control".

K&A # 295005 A2.05
Importance Rating 3.9

QUESTION 76.

K&A Statement: **AA2.05** Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP **: REACTOR POWER**

Justification:

A. **CORRECT:** Under the current conditions with reactor power less than 24% the turbine willl trip. But instead of a Reactor Scram, reactor power will rise due to increased feedwater subcooling since feedwater heating is lost when extraction steam is isolated by the turbine trip.

B. **Incorrect but plausible if the candidate forgets** The Main Turbine bearing header pressure drops to 17 psig will cause a turbine trip.

C. **Incorrect but plausible if the candidate forgets** The Main Turbine bearing header pressure drops to 17 psig will cause a turbine trip.

D. **Incorrect but plausible if the candidate forgets** a Reactor Scram and Recirculation pump trip are not expected since Reactor Power is below 29%.

References: OP-9 R52, AOP-32 R10
ARP 09-5-2-7 R6, ARP 09-5-2-28 R4
SDLP-35R12, Feedwater Heating

Student Ref: None

Learning Objective: 1.09.a

Question source: Modified Limerick 2008

Question History: NRC Exam Q 77

Cognitive level: Memory/Fundamental knowledge:

10CFR55.43(b)(6)

QUESTION 77.

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

1300 on 5/03/10: 'A' Emergency Diesel Generator is declared INOPERABLE
1600 on 5/03/10: Line 4 offsite AC circuit is declared INOPERABLE
1700 on 5/03/10: 'B' Emergency Diesel Generator is declared INOPERABLE

When will the plant be required to enter MODE 3, in accordance with Technical Specifications?

A. 0500 on 5/04/10

B. 0600 on 5/04/10

C. 0700 on 5/04/10

D. 1600 on 5/04/10

K&A # 295003; G.2.2.22
Importance Rating SRO – 4.7

QUESTION 77.

K&A Statement: Knowledge of limiting conditions for operations and safety limits in regards to: **PARTIAL OR COMPLETE LOSS OF AC**

Justification:

A. **INCORRECT:** because this time represents only 12 hours for mode 3 entry. With 3 AC sources inoperable, required per TS 3.8.1 Action G to enter T/S 3.0.3 immediately. Action shall be initiated to place the plant in Mode 3 within 13 hours **Plausible if the candidate bases** shutdown only on T/S 3.8.1.

B. **CORRECT:** With 3 AC sources inoperable, required per TS 3.8.1 Action G to enter T/S 3.0.3 immediately. Action shall be initiated within 1 hour to place the plant in Mode 3 within 13 hours.

C. **Incorrect:** because this time represents the action time to mode 3 based on 2 EDG being inoperable. **Plausible if the candidate bases** the shutdown actions on 2 EDGs being inoperable in TS3.8.1 Action E and not a loss of 3 AC sources. A total time of 14 hours for mode 3 entry.

D. **Incorrect:** because this time represents 1 AC offsite and 1 EDG being inoperable. **Plausible if the candidate bases** the shutdown on TS 3.8.1 Action D. A total time of 24 hours for mode 3 entry.

References: SDLP-93 R16 "Emergency AC power" Student Ref: T/S 3.0.3 only

TS 3.8.1 (Amendment 284)

Learning Objective: 1.16 and 1.18

Question source: New

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR55.43(b) (2)

QUESTION 78.

The reactor is operating normally at 100% power when the following sequence of events occurs:

0323 - EPIC-D-124 SEISMIC EVENT ALARM actuates
- NMP-2 reports 0.10g horizontal ground acceleration
- Operators notice ground motion and smoke in JAF control room
- Offsite power is lost
- EDGs A and C automatically connect to Bus 10500
- EDGs B and D do **NOT** connect to Bus 10600

0324 Smoke and fire observed behind the panels. Shift Manager orders control room evacuation

0325 Operators take required **INSIDE** control room actions and then exit the control room

0342 Fire brigade extinguishes the fire in the control room

0345 Bus 10600 energized from EDGs

0359 LPCI injection to RPV commenced, vessel level rising

0405 Plant walkdown complete, only minor equipment damage

Which of the following (1) describes field action(s) required per AOP-43, "Plant Shutdown From Outside the Control Room", and (2) identifies the highest EAL classification?

A. (1) Trip RWR MG Set A Motor Breaker 71-10110
(2) ALERT

B. (1) Open ESW Sys B Injection Valve 46MOV-101B
(2) ALERT

C. (1) Start RHRSW Pump A and RHR Pump C
(2) SITE AREA

D. (1) Align LPCI MOV Bus B to Alternate Feed
(2) SITE AREA

K&A # 295016; G.2.4.12
Importance Rating SRO – 4.3

QUESTION 78.

K&A Statement: Knowledge of general operating crew responsibilities during emergency operations in regards to:
CONTROL ROOM ABANDONMENT

Justification:

A. **Incorrect.** Site Area is the correct classification **but plausible** because the ATC trips RWR MG Set A Motor Feed Breaker 71-10110 per AOP-43 and AOP-43 directs minimum EAL classification of Alert.

B. **Incorrect.** Site Area is the correct classification **but plausible** because the ESW Sys B Injection Valve is opened by the SNO or CRS at Panel 25ASP-3 and AOP-43 directs minimum EAL classification of Alert.

C. **Incorrect.** RHR Pumps A and C motor breakers are tripped by the SNO or CRS at the breakers **but plausible** because RHRSW Pp B and RHR Pump D breakers are closed by the SM at Panel 25RSP. Site Area is the proper classification when vessel injection is not restored within 30 minutes after evacuating the control room.

D. **Correct:** LPCI MOV Bus B is placed on its alternate feed per subsequent steps. The fire safe shutdown strategy uses Bus 10600 equipment to establish vessel level control. Site Area is the proper classification when vessel injection is not restored within 30 minutes after evacuating the control room.

References: AOP-43 R33 Student Ref: EAL Matrix

Learning Objective: LP-AOP; 1.03.a & b

Question source: New

Question History: None

Cognitive level: Analysis

10CFR55.43(b) (5)

QUESTION 79.

The plant is shutting down for a refueling outage. Shutdown Cooling has been in service for 1 hour.

T=1200 RPV temperature is 162 °F.

Then, a complete loss of Shutdown Cooling occurs. After 20 minutes, the operators determine that RPV temperature is rising at 18 degrees every 10 minutes.

T=1220 RPV temperature is 196 °F.

Which **ONE** of the following describes how the heatup, if it continues at the rate stated above, will affect the plant Operational Condition and the Technical Specification (TS) heatup limits?

A. After T=1230, a MODE change will occur. At T=1300, the TS heatup rate limit will be exceeded.

B. After T=1230, a MODE change will occur. At T=1300, the TS heatup rate limit will **NOT** be exceeded.

C. Before T=1230, a MODE change will occur. At T=1300, the TS heatup rate limit will be exceeded.

D. Before T=1230, a MODE change will occur. At T=1300, the TS heatup rate limit will **NOT** be exceeded.

K&A # 295021; AA2.01
Importance Rating SRO – 3.6

QUESTION 79.

K&A Statement: Ability to determine and/or interpret the following as they apply to **LOSS OF SHUTDOWN COOLING: REACTOR WATER HEATUP/COOLDOWN RATE**

Justification: MODE change occurs at>212 degrees F per TS definitions. The TS heatup limit is 100 degrees in a one hour period. Although the rate is > 100 degrees per hour the limit is not exceeded until the one hour time period has been met (1200-1300)

A. **Incorrect but plausible** MODE change occurs at>212 degrees F per TS definitions. Based on the heatup rate the MODE change occurred before 1230 and not after 1230. The TS heatup rate will be exceeded at T=1300.

B. **Incorrect but plausible** MODE change occurs at>212 degrees F per TS definitions. Based on the heatup rate the MODE change occurred before 1230 and not after 1230. The TS heatup rate will be exceeded at T=1300.

C. **CORRECT:** MODE change occurs at>212 degrees F per TS definitions. Based on the heatup rate the MODE change occurred before 1230 and not after 1230. The TS heatup rate will be exceeded at T=1300.

D. **Incorrect but plausible** MODE change occurs at>212 degrees F per TS definitions. Based on the heatup rate the MODE change occurred before 1230 and not after 1230. The TS heatup rate will be exceeded at T=1300.

References: TS 3.4.9, TS 1.1 definitions Student Ref: None

Learning Objective: SDLP-02AR11; 1.07.a

Question source: Modified TS for JAF

Question History: HC NRC 02/09; SRO-76

Cognitive level: Memory or Fundamental Knowledge

10CFR55.43(b) (1)
10CFR 41.10 / 43.5 / 45.13

QUESTION 80.

The unit was shutdown 144 hours ago to begin a refueling outage. The refueling cavity is flooded and irradiated fuel moves are in progress. Both doors in the drywell personnel air lock have been open to support drywell inspections. Both Standby Gas Treatment (SGT) trains have just been declared INOPERABLE because of a safety qualification issue related to control components in both SGT trains.

Per Technical Specifications what operator actions must be taken in reference to the irradiated fuel moves?

A. **NO** actions are required, irradiated fuel movements may continue.

B. Suspend movement of irradiated fuel assemblies in secondary containment until **ONE** train of SGT is OPERABLE.

C. Suspend movement of irradiated fuel assemblies in secondary containment until **BOTH** trains of SGT are OPERABLE.

D. Manually initiate and run **ONE** Standby Gas Treatment subsystem until the SGT automatic initiation logic is returned to an OPERABLE status.

K&A #	295003 G.2.2.22
Importance Rating	SRO 4.7

QUESTION 80.

K&A Statement: **295003 Refuel ACC G.2.2.22 Knowledge of limiting conditions for operations and safety limits**

Justification: Knowledge of TS bases definition of "**recently** irradiated" is required to implement numerous TS requirements, such as TS 3.3.6.2, 3.6.4, 3.7.3, 3.7.4, 3.8.2, 3.8.5 and 3.8.8. Tech Specs whose applicability is based on "recently irradiated" are not applicable at the 6 day point.

A. Correct –per TS bases B3.6.4.1 recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 96 hours. Since this is day 6 after shutdown, TS 3.6.4.3 is not applicable.

B. Incorrect but plausible because this would be the correct action statement if the TS was applicable.

C. Incorrect but plausible if the candidate thinks both systems need to be OPERABLE to exit 3.6.4.3.D.

D. Incorrect but plausible if the candidate thinks that by manually starting a system, this would make it OPERABLE, and fuel movements could continue.

References: TS 3.6.4.3 (Amendment 276) Student Ref: TS 3.6.4.3

Learning Objective: SDLP-01B R13 LO 1.17.b

Question source: Modified Fitz 02 NRC Exam

Question History: SRO Q24

Cognitive level: Memory/Fundamental knowledge: x
Comprehensive/Analysis:

10CFR 43.2

QUESTION 81.

After a long period of full power operation, an instantaneous loss of **ALL** AC power occurs and is **NOT** corrected. The HPCI System failed to start.

- Assume that decay heat over the next hour is 6.2×10^8 Btu/hr

With **NO** initial operator action, over the next hour you would expect ___(1)___ and, per AOP-49, "Station Blackout", operators should attempt to maintain RPV cooldown rate less than _(2)_ ?

A. (1) SRV's to open and close periodically on overpressure
(2) 80 °F/ hr

B. (1) SRV's to open and close periodically on overpressure
(2) 20 °F/ hr

C. (1) SRV initial operation with RCIC operation precluding the need for further SRV operation
(2) 80 °F/ hr

D. (1) SRV initial operation with RCIC operation precluding the need for further SRV operation
(2) 20 °F/ hr

K&A # 295025 EA2.05
Importance Rating SRO 3.6

QUESTION 81.

K&A Statement: **EA2.05 Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Decay heat generation**

Justification:

A. Incorrect because although station normal cooldown is 80 °F/hr per OP-13D, AOP-49 is the correct procedure and it has a more limiting cooldown rate. Plausible because the limit cooldown rate is 80 °F/hr per OP-13D.

B. Correct – Candidate must recognize that RCIC steam demand is far less than decay heat generation. Per AOP-49 cooldown is to be <20 °F per hour.

C. Incorrect because candidate must recognize that RCIC steam demand is far less than decay heat generation which will cause the SRV's to continue to cycle without operator action. Plausible if the candidate does not know that the decay heat is beyond the capacity of the RCIC system.

D. Incorrect because candidate must recognize that RCIC steam demand is far less than decay heat generation which will cause the SRV's to continue to cycle without operator action. Plausible if the candidate does not know that the decay heat is beyond the capacity of the RCIC system.

References: AOP-49 R14
OP-13D R21
EOP-2 R9

Student Ref: None

Learning Objective: LP-AOP, EO-1.10

Question source: Fitz 03 NRC

Question History: Q S30 Fitz NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 43.5

QUESTION 82.

The plant was operating normally at power. A loss of coolant accident occurs with complications, resulting in an automatic reactor scram. Conditions shortly after the scram are as follows:

- Reactor is shutdown.
- Drywell pressure is 9.5 psig and rising slowly.
- Drywell temperature is 190 °F and rising slowly.
- RPV water level is 45 inches and slowly lowering.
- Torus water level is 9.6 feet and dropping.
- RPV pressure is being controlled between 700 to 800 psig.
- Torus water temperature is 110 °F and rising slowly.

Per the EOPs, WHICH of the following is required **AND** why?

A. Place RHR in Drywell Spray to mitigate the effects of a deflagration.

B. Cycle SRV's to maintain RPV pressure below the heat capacity temperature limit.

C. Secure Core Spray pumps to prevent cavitation damage of the pumps.

D. Emergency depressurize due to compromise of pressure suppression capability.

K&A # 295030; G.2.1.27
Importance Rating 4.0

QUESTION 82.

K&A Statement: Knowledge of system purpose and/or function in regards to: **LOW SUPPRESSION POOL WATER LEVEL**

Justification:

A. **Incorrect but plausible if the candidate forgets** that Drywell conditions do not support placing RHR in the Drywell Spray mode. Drywell sprays are used to effect a reduction in drywell temperature and pressure, to control hydrogen and oxygen concentrations in the drywell, and to mitigate the effects of a deflagration.

B. **Incorrect but plausible if the candidate forgets** HCTL is not about to be exceeded. With RPV pressure between 700 and 800 psig, HCTL limit is 120°F.

C. **Incorrect but plausible if the candidate forgets** the Vortex Limit is defined to be the lowest torus water level above which air entrainment is not expected to occur in RCIC or an ECCS pump taking suction on the torus. ECCS pumps could handle 20% entrainment of air, but > 4% air by volume can noticeably reduce pump capacity. Also should a pump trip under these conditions air would collect at system high points. Subsequent restarts could damage system components due to water hammer. This limit is a function of pump flow, and is imposed to preclude system damage due to air entrainment. 8.92 ft for flows between 0 to 20,800 gpm.

D. **CORRECT:** Per EOP bases the downcomer vents can begin to uncover. This can lead to pressurization of the Torus air space.

References: MIT-301.11E
EOP-4 R8

Student Ref: EOP-11

Learning Objective: EO-4.05

Question source: Modified

Question History: NMP #1; NRC3/07; Q-63

Cognitive level: Comprehensive/Analysis:

10CFR55.43(b) (1)

QUESTION 83.

The plant is refueling.

- Refueling is 50% complete and fuel moves are in progress.
- 'A' SRM is INOPERABLE.
- The intended core location for the next bundle is in the 'B' SRM quadrant.

Prior to inserting the next bundle in the 'B' SRM quadrant, the Reactor Operator reports the following SRM readings:

- 'B' SRM reading 102 cps
- 'C' SRM reading 2 cps
- 'D' SRM reading 110 cps

In accordance with RAP-7.1.04C, "Neutron Instrumentation Monitoring During In-Core Fuel Handling", which of the following is correct (**see Core Map on next page**)?

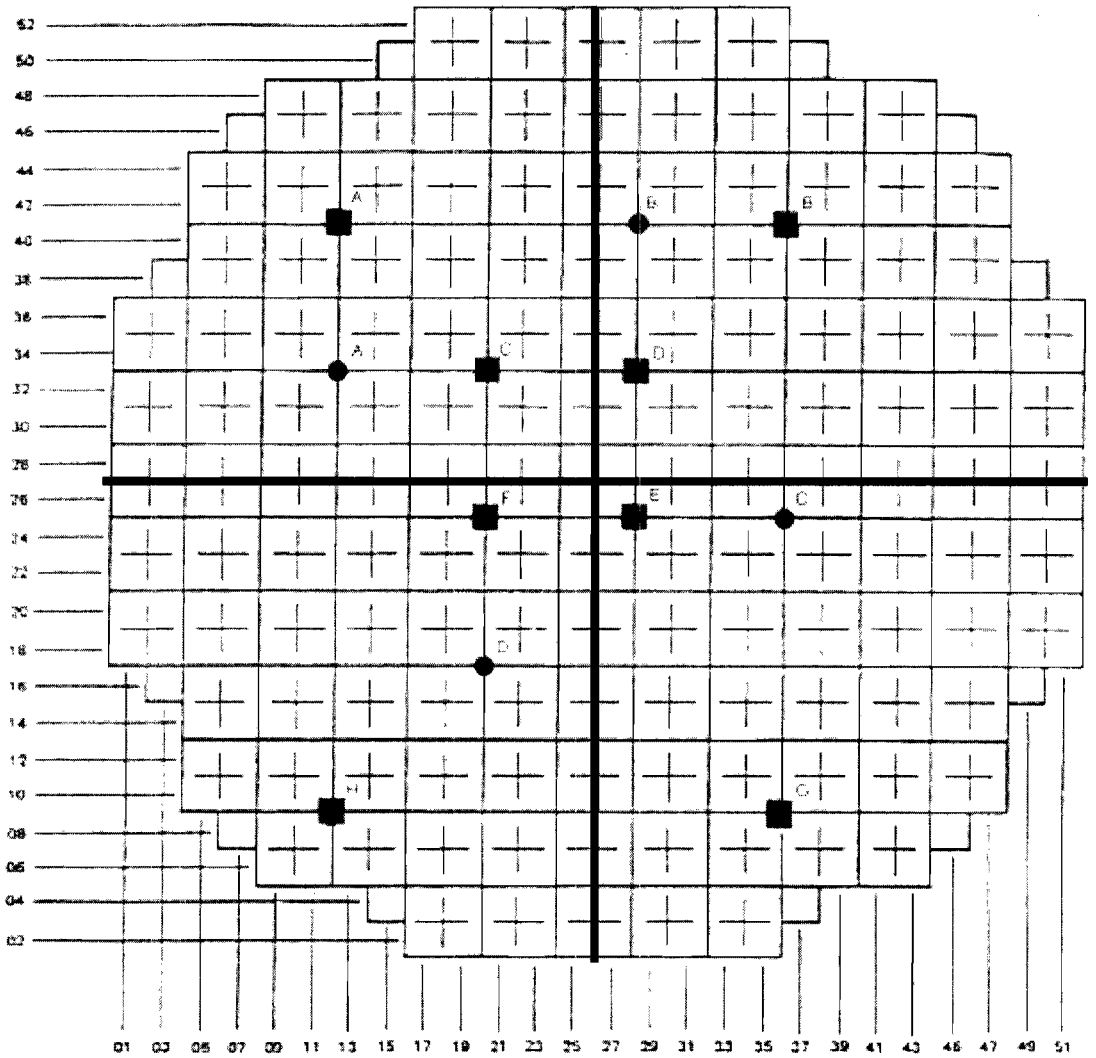
A. Core alterations may continue because 'B' and 'D' SRMs are still OPERABLE.

B. Core alterations shall be halted in the 'C' SRM quadrant **ONLY**.

C. Core alterations may continue because 'C' SRM indicates $\geq$ 1 cps.

D. **ALL** core alterations shall be halted.

CORE MAP



● SRM LOCATIONS

■ IRM LOCATIONS

K&A # 215004; G.2.1.40
Importance Rating SRO – 4.2

QUESTION 83.

K&A Statement: Knowledge of refueling administrative requirements as they apply to **SRM**

Justification:

A. **Incorrect:** because this does not meet RAP-7.1.04C guidance of 3 required operable adjacent SRMs **but plausible if the candidate forgets** 3 SRMs required operable.

B. **Incorrect:** because this does not meet RAP-7.1.04C guidance by have the adjacent "A' and 'C' SRM operable and performing core alterations in the 'B' SRM quadrant **but plausible if the candidate forgets** 3 SRMs operability is based on indications > 3 cps and 2 SRMs have to be adjacent to each other.

C. **Incorrect:** because 'C' SRM has to indicate > than 3 cps **but plausible if the candidate forgets** 3 SRMs operability is based on indications > 3 cps and 2 SRMs have to be adjacent to each other.

D. **CORRECT:** IAW RAP-7.1.04C the 3 operable SRMs have to indicate > 3 cps and 2 SRMs have to be adjacent to each other. With 'A' and 'C' SRMs inoperable core alterations must be stopped. RAP-7.1.04C is more conservative than T/S 3.3.1.2.4 RAP-7.1.04C step 6.2 states **IF** Neutron Instrumentation reads **LOWER THAN** 3 **CPS, THEN** core alterations shall immediately stop until it can be determined if the following is met: Technical Specifications SR 3.3.1.2.4.

References: RAP-7.1.04B R24, RAP-7.1.04C R5 Student Ref: None
SRM/IRM Core Map
T/S 3.3.1.2.E
SDLP-08A R10; Refueling Equipment
HOSDLP-007B Fig 2 (Core Map)

Learning Objective: 1.10.a

Question source: OC 08/01 Q-14

Question History: OC 08/01 Q-14

Cognitive level: Memory/Fundamental knowledge:

10CFR 55.43(b) (2)

QUESTION 84.

A LOCA has occurred and the following conditions exist:

- Reactor is shutdown.
- Reactor pressure is 900 psig and lowering slowly.
- Torus temperature is 135 °F and rising slowly.
- Torus water level is 10 feet and steady.
- Torus pressure is 6 psig and lowering slowly.
- Drywell pressure is 7 psig and lowering slowly.
- Drywell temperature is 310 °F and rising slowly.
- Drywell hydrogen concentration is 0.5% and steady.
- Drywell oxygen concentration is 4% and steady.
- RPV level is 141 inches and lowering at 2 inches/min.

Based on the above conditions, which **ONE** of the following actions is required?

A. Vent the drywell.

B. Emergency depressurize the RPV.

C. Spray the drywell.

D. Lower RPV pressure maintaining <100 °F/hour cooldown rate.

K&A # 295010; A2.06
Importance Rating SRO – 3.6

QUESTION 84.

K&A Statement: Ability to determine and /or interpret the following as they apply to **HIGH DRYWELL PRESSURE : DRYWELL TEMPERATURE**

Justification:

A. **Incorrect but plausible if the candidate** does not remember venting the drywell occurs when drywell hydrogen concentration is above .6%. Also drywell pressure will not exceed the Primary Containment Pressure Limit.

B. **CORRECT:** IAW drywell temperature leg of EOP-4 before drywell temperature reaches 309 degs, spray the Drywell. However, parameters are not within the drywell spray initiation limit and therefore the next step if you cannot spray is to ED.

C. **Incorrect but plausible if the candidate** does not recognize that they are on the bad portion of the drywell spray initiation limit.

D. **Incorrect but plausible** if the candidate does not recognize not on bad portion of HCTL and pressure is lowering, Lowering pressure to maintain parameters within the HCTL is not required at this time.

References: SDLP-10 R19; RHR Student Ref: EOP-4, EOP-11

Learning Objective: 1.05.a.3.b & d

Question source: New

Question History: New

Cognitive level: Comprehensive/Analysis:

10CFR55.43(b) (5)

QUESTION 85.

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

- Offgas flow is 150 SCFM and lowering.
- Vacuum is 26.50 inches Hg and lowering.
- 09-6-1-23, OFF GAS LINE FILTER DIFF PRESS HI is in alarm.

Reactor power is being lowered per RAP-7.3.16, "Plant Power Changes" to maintain condenser vacuum within the Normal Operating Range.

Which **ONE** of the following describes the correct CRS directions to the Reactor Operators?

A. Insert a manual scram and trip the recombiner.

B. Place the RECOMBINER SYS Switch in OFF and ensure Hydrogen Injection trips.

C. Ensure main turbine seal steam pressure is between 1 and 4 psig.

D. Place steam jet air ejectors in a 4 first stage – 3 second stage lineup per OP-24C, "Condenser Air Removal".

K&A # 295002; AA2.04
Importance Rating SRO – 2.9

QUESTION 85.

K&A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM**: OFFGAS SYSTEM FLOW**

Justification:

A. **INCORRECT because** inserting a scram is the immediate action for AOP-4, Explosion in air ejector discharge piping. Plausible if the candidate thinks the high d/p alarm is an indication of explosion in air ejector discharge piping.

B. **CORRECT**: AOP-5 Combustion in SJAE After Condenser symptoms are a lowering offgas flow with offgas line filter DP Hi alarm. Immediate actions are to place recombiner to off and ensure Hydrogen Injection trips.

C. **INCORRECT because** AOP-31 Loss of condenser vacuum would have these actions performed if the cause of the loss of vacuum was due to air inleakage. Plausible if the candidate does not realize air inleakage would make offgas flow go up, not down. Lower offgas flow is an indication of blockage in offgas or the gasses are recombining by combustion.

D. **INCORRECT because** AOP-31 Loss of condenser vacuum would have these actions performed if the cause of the loss of vacuum was due to air inleakage. The 4 first stage – 3 second stage lineup is an incorrect lineup per OP-24C. Plausible if the candidate does not realize air inleakage would make offgas flow go up, not down. Lower offgas flow is an indication of blockage in offgas or the gasses are recombining by combustion.

References: AOP-31 R17
AOP-4 R5
AOP-5 R11
OP-24C R29

Student Ref: None

Learning Objective:

Question source: New

Question History:

Cognitive level: Analysis

10CFR55.43(b) (5)

QUESTION 86.

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

- 09-3-1-1, CORE SPRAY HDR A PIPE BREAK DETECTOR ALARM is received.
- Report received that the D/P indicating switch reading is +4.0 psid.

Which **ONE** of the following describes the LOCATION of this piping break **AND** the Technical Specification implication of this failure?

A. Break is in CS piping BETWEEN the Reactor Pressure Vessel wall and the Core Shroud. Enter a 7 day LCO.

B. Break is in CS piping INSIDE the Core Shroud. Enter a 14 day LCO.

C. Break is in CS piping INSIDE the Core Shroud. Enter a 7 day LCO.

D. Break is in CS piping BETWEEN the Reactor Pressure Vessel wall and the Core Shroud. Enter a 14 day LCO.

K&A # 209001; G.2.1.45
Importance Rating SRO – 4.3

QUESTION 86.

K&A Statement: Ability to identify and interpret diverse indications to validate the response of another indication for **LOW PRESSURE CORE SPRAY SYSTEM LINE BREAK PROTECTION**

Justification:

A. **Correct:** A Line Break annunciator detects a line break BETWEEN the Reactor Pressure Vessel and the Core Shroud. With spray function not assured, a 7 day LCO is required by Technical Specification 3.5.1.

B. **Incorrect but plausible:** Break INSIDE Core Shroud will NOT be annunciated.

C. **Incorrect but plausible:** Break INSIDE Core Shroud will NOT be annunciated.

D. **Incorrect but plausible if** does not recognize 7 day LCO is applicable for the Core Spray function.

References: SDLP-14 R12; Core Spray
TS 3.5.1 (Amendment 284)
Student Ref: None

Learning Objective: 1.05.a.13 and 1.18.a

Question source: Modified to JAF

Question History: NMP #2 NRC Exam 3/8; Q-87

Cognitive level: Comprehensive/Analysis

10CFR55.43(b) (5)

QUESTION 87.

Plant conditions are as follows:

- Reactor Power is 100%.
- I&C reports that Narrow Range Level Transmitters 02-3LT-101A and 02-3LT-101D (Reactor Water Vessel Water Level-Low (Level 3)) have failed their Surveillance Test.
- **ALL** other Narrow Range Level Transmitters are OPERABLE.

WHICH **ONE** of the following identifies the required action with the **SHORTEST** completion time per Technical Specifications that is applicable for the above conditions?

A. Restore isolation capability within 1 hour.

B. Restore RPS trip capability within 1 hour.

C. Place **ONE** Channel in the tripped condition within 6 hours.

D. Place **ONE** Channel in the tripped condition within 12 hours.

K&A # 212000; G.2.2.40
Importance Rating 4.7

QUESTION 87.

K&A Statement: Ability to apply Technical Specifications for a system: **RPS**

Justification: LT-101A inputs to RPS A1 Logic & A1 Group II Isolation Logic. LT-102D inputs to RPS B2 Logic & B2 Group II Isolation Logic. The RPS function is 1 out of 2 taken twice. Therefore, with one channel inop on each subsystem, the trip capability is still maintained. The PCIS Group II logic is 2 out of 2 once. Therefore, with one channel inop on each subsystem, isolation capability is lost.

A. **Correct** because 3.3.6.1.B is applicable in this condition because the isolation capability is not functional with one transmitter in each trip system inoperable. Placing one channel in trip will restore isolation functionality.

B. **Incorrect** because 3.3.1.1.C is not applicable in this condition because the RPS trip system can still function. Plausible if the candidate thinks that RPS trip capability is not maintained with 2 channels inoperable.

C. **Incorrect** because 02-3LT-101A and D inoperable means that TS 3.3.1.1.B is applicable. There is one function with one or more channels inoperable in both trips systems. One channel must be tripped within 6 hours, however 3.3.6.1 is more limiting.

D. **Incorrect** because although 3.3.1.1.A would be an entry condition, the question asked which condition has the shortest completion time. This is plausible if the candidate does not realize that 3.3.1.1.B is applicable because A and D are in separate trip systems.

References:	SDLP-02B R16; Rx Vessel Level instrumentation OP-27A R10 T/S 3.3.1.1 and 3.3.6.1	Student Ref:	TS 3.3.6.1 TS 3.3.1.1 Bases Pg B 3.3.1.1-24 Bases Pg B 3.3.6.1-26 Bases Pg B 3.3.6.1-27 OP-27A Pgs 79 and 80

Learning Objective: 1.16 and 1.17

Question source: Modified

Question History: Limerick 10/06; Q-86

Cognitive level: Comprehensive/Analysis: X

10CFR55.43(b) (5)

QUESTION 88.

While operating at 60% Reactor Power, a Reactor Scram on low reactor water level occurs but **ALL** control rods remain at their pre-trip conditions.

Plant conditions 30 minutes after the transient are:

- Power level is 5% and lowering.
- RPV pressure is controlled between 600 to 800 psig.
- RPV level is 110 inches.
- Torus level is 14.00 feet.
- Torus temperature is 175 °F and rising.
- Drywell pressure is 15 psig and steady.
- Torus pressure is 13.5 psig and steady.
- Main steam tunnel temp is 200 °F and rising.

Which one of the following is required for the conditions above?

A. Maintain RPV pressure at 800 psig.

B. Lower pressure using bypass valves, maintain < 100 °F/hr cooldown rate.

C. Reduce RPV pressure using SRVs, can exceed 100 °F/hr cooldown rate.

D. Emergency depressurize the reactor vessel.

K&A # 295015; G.2.1.7
Importance Rating 4.7

QUESTION 88.

K&A Statement: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior and instrument interpretations as they apply to: **INCOMPLETE SCRAM**

Justification:

A. **Incorrect** because with Torus temperature rising, the HCTL has been exceeded. RPV pressure needs to be reduced **but plausible if the candidate forgets** RPV pressure needs to be reduced.

B. **Incorrect** because there is indication of a steam line break in the steam tunnel **but plausible if the candidate forgets** the isolation setpoint of 193 °F.

C. **CORRECT:** With Torus temperature rising. The HCTL has been exceeded. RPV pressure needs to be reduced.

D. **Incorrect** because Torus pressure does not meet the criteria for Emergency Depressurization **but plausible if the candidate forgets** the required torus pressure for Emergency depressurization.

References: EOP-3 R9
MIT-301.11.D;

Student Ref: EOP-11 Graphs

Learning Objective: 1.07

Question source: HC 2/09: Q-80

Question History: HC 2/09: Q-80

Cognitive level: Comprehensive/Analysis:

10CFR55.43(b) (5)

QUESTION 89.

The plant is shutdown for a Refueling Outage and the following conditions exist:

- The 115 kV System is providing site electrical power.
- The 115 kV System is in a normal alignment, **EXCEPT** for North-South Bus Disconnect 10017, which is OPEN.
- A **AND** C EDGs are tagged out for maintenance.

From these conditions, Circuit Breaker 10012 (NMP-Fitz 115 kV Line 4 Breaker) trips.

Which **ONE** of the following identifies the correct **INITIAL** procedural response?

A. AOP-16, "Loss of 10300 Bus" **AND** AOP-18, "Loss of 10500 Bus"

B. AOP-17, "Loss of 10400 Bus" **AND** AOP-19, "Loss of 10600 Bus"

C. AOP-57, "Recovery from Residual Bus Transfer"

D. AOP-49A, "Station Blackout in Cold Condition"

K&A # 262001; A2.06
Importance Rating SRO – 2.9

QUESTION 89.
K&A Statement:

Ability to(a) predict the impacts of the following on the **A.C. ELECTRICAL DISTRIBUTION;** and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
DE-ENERGIZING A PLANT BUS

Justification:

A. **CORRECT.** With the 10017 open, a trip of the 10012 breaker results in a loss of the T-3 transformer. With T-3 de-energized busses 10100, 10300 & 10500 will be lost. AOP-18 is entered for the loss of the 10500 Bus.

B. **Incorrect because** With the 10017 open, a trip of the 10012 breaker results in a loss of the T-3 transformer. With T-3 de-energized busses 10100, 10300 & 10500 will be lost. AOP-18 is entered for the loss of the 10500 Bus. Bus 10600 will not be de-energized and AOP-19 is not required to be entered.

C. **Incorrect because** With the 10017 open, a trip of the 10012 breaker results in a loss of the T-3 transformer. With T-3 de-energized busses 10100, 10300 & 10500 will be lost. AOP-18 is entered for the loss of the 10500 Bus. AOP-57 is not required to be entered as a loss of Buss 10600 did NOT occur.

D. **Incorrect because** With the 10017 open, a trip of the 10012 breaker results in a loss of the T-3 transformer. With T-3 de-energized busses 10100, 10300 & 10500 will be lost. AOP-18 is entered for the loss of the 10500 Bus. Station Blackout conditions do not exist. AOP-49A is inappropriate because Bus 10600 is still energized from Line 3 via Transformer T-4.

References: SDLP-71D
AOP-17 R17
OP-44 R17

Student Ref: None

Learning Objective: 1.05.a, 1.06, 1.09

Question source: Bank (JAF)

Q was an SRO from LOI-03-01 NRC Exam. Modified 5/3/05.

Question History: JAF Exam May 2005

Cognitive level: Comprehensive/Analysis:
10CFR55.43(b) (5)

Question 90

The plant is operating at 100% power with all LCO's met.

A feed and bleed of the drywell is in progress per OP-37 with SBGT Train "A".

The following alarms are received:

- 09-4-0-5 CONT HI RANGE RAD MON B ALERT
- 09-4-0-6 CONT HI RANGE RAD MON B HI-HI
- All other Plant conditions are **NORMAL** with **NO** other alarms present

Which of the following is the required procedures **and** directions that the CRS should provide to respond to the above conditions?

A. AOP-9, Loss of Primary Containment, secure SBGT Train "A" per OP-20, Standby Gas Treatment System

B. AOP-15, Isolation Verification and Recovery, secure SBGT Train "A" per OP-20, Standby Gas Treatment System

C. AOP-9, Loss of Primary Containment, secure SBGT Train "A" **and** "B" per OP-20, Standby Gas Treatment System

D. AOP-15, Isolation Verification and Recovery, secure SBGT Train "A" **and** "B" per OP-20, Standby Gas Treatment System

K&A # 223002; A2.04
Importance Rating SRO – 3.2

QUESTION 90.

K&A Statement: Ability to (a) predict the impacts of the following on the **PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF** (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **PROCESS RADIATION MONITORING SYSTEM FAILURES**

Justification:

A. **Incorrect but plausible** if candidate does not recall that the first indications for a loss of coolant are high drywell temperatures or pressure and that "B" SBGT does NOT receive a start signal from 27RM-104A or B. The symptoms presented in the stem are consistent with a failure of the 27RM-104B monitor only which will result in a PCIS Partial Group Isolation that will isolate the OP-37 feed and bleed lineup.

B. **CORRECT:** ARP 09-4-0-6 CONT HI RANGE RAD MON B HI-HI directs the operator to AOP-15. The symptoms presented in the stem are consistent with a failure of the 27RM-104B monitor only which will result in a PCIS Partial Group Isolation that will isolate the OP-37 feed and bleed lineup. SBGT Train "A" should be secured per OP-20.

C. **Incorrect but plausible** if candidate does not recall that the first indications for a loss of coolant are high drywell temperatures or pressure and that "B" SBGT does NOT receive a start signal from 27RM-104A or B. The symptoms presented in the stem are consistent with a failure of the 27RM-104B monitor only which will result in a PCIS Partial Group Isolation that will isolate the OP-37 feed and bleed lineup.

D. **Incorrect but plausible** ARP 09-4-0-6 CONT HI RANGE RAD MON B HI-HI directs the operator to AOP-15. The symptoms presented in the stem are consistent with a failure of the 27RM-104B monitor only which will result in a PCIS Partial Group Isolation that will isolate the OP-37 feed and bleed lineup. SBGT Train "A" should be secured per OP-20. SBGT does NOT receive a start signal from 27RM-104A or B and the "B" SBGT Train should not have received a auto start signal based upon the conditions presented in the stem.

References: AOP-3, 15, 39, 60, ARP-09-4-04, ARP-09-4-05, ARP-09-4-06,
ISP-95B R1
SDLP-17

Student Ref: None

Learning Objective: 1.14.D.15 &16 SDLP-17, "Area Process Rad Monitors"

Question source: NEW

Question History: NONE

Cognitive level: Comprehensive/Analysis:

10CFR 55.43 (b) (2) & (5)

QUESTION 91.

Plant conditions are as follows:

- Reactor Power = 37%.
- Reactor Pressure = 997 psig.
- Control rod scram time testing is in progress IAW RAP-7.4.01, "Control Rod Scram Time Evaluation (IST)".
- **ALL** control rods are OPERABLE.
- Rods 30-35 and 30-39 are the only rods that are declared SLOW.
- Rod 30-31 has been withdrawn from position "12" to position "48" to support scram time testing.

The following sequence of events occur (time in seconds):

0.0	Rod 30-31 toggle switch at Panel 9-16 in DOWN SCRAM
0.3	Rod 30-31 at notch position 46
1.2	Rod 30-31 at notch position 36
5.4	Rod 30-31 at notch position 26
8.2	Rod 30-31 at notch position 06
8.4	Rod 30-31 FULL IN indication received on full core display
15.1	Rod 30-31 toggle switch at Panel 9-16 in UP SCRAM

For the conditions described above, which **ONE** of the following meets Tech Spec required actions?

A. Fully insert **either** Rod 30-31 **or** Rod 30-39 **or** Rod 30-35 within 3 hours, **AND** isolate the associated inserted rod's HCU within 4 hours.

B. Declare Rod 30-31 as INOPERABLE. Ensure the rod remains fully inserted, **AND** disarm Rod 30-31 HCU within 6 hours.

C. Fully insert Rod 30-31 within 3 hours, **AND** remove power from the rod's HCU directional control valves within 4 hours.

D. Declare Rod 30-31 as SLOW. Ensure < 11 rods are declared SLOW **AND** restore Rod 30-31 position to notch position 12.

K&A # 201003; A2.10
Importance Rating SRO – 3.4

QUESTION 91.

K&A Statement: Ability to (a) predict the impacts of the following on the **CONTROL ROD AND DRIVE MECHANISM;** and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
EXCESSIVE SCRAM TIME FOR A GIVEN DRIVE MECHANISM

Justification:

A. **Incorrect.** Rods 30-35 and 30-39 are adjacent rods. TS 3.1.4 requires no more than 2 adjacent rods declared as SLOW. Rod 30-31 is also an adjacent rod but its scram time is greater than 7 seconds to notch position 04, which requires declaration as INOPERABLE. Must therefore take action to maintain the rod fully inserted per TS 3.1.3. Plausible to take actions for > 2 adjacent slow rods since rod times are slower than required by TS Table 3.1.4-1.

B. **Incorrect.** Control Rod 30-31 scram time is greater than 7 seconds requires actions for INOPERABLE rod. **Plausible if applicant** thinks disarm required within 6 hours rather than 4 hours.

C. **CORRECT.** Control Rod 30-31 is inoperable IAW T/S 3.1.3 and 3.1.4 since the scram time is greater than 7 seconds to position 04. Position 04 time is not provided in stem but can be inferred scram time exceeds 7 seconds since time to notch position 06 is greater than 8 seconds. T/S actions required are to fully insert the control rod within 3 hours and disarm the HCU directional control valves (or isolate the HCU) within 4 hours. References are not provided with this question. However, it is reasonable to expect an SRO applicant to know from memory 1) INOPERABLE if > 7 seconds to position 04, and 2) withdrawal capability defeated within 4 hours.

D. **Incorrect.** Control Rod 30-31 scram time exceeds SLOW criteria. Therefore, **plausible if the candidate doesn't realize** INOPERABLE and doesn't apply limit of no more than 2 adjacent SLOW rods.

References: TS 3.1.3 (Amendment 291)
TS 3.1.4 (Amendment 291)
RAP-7.4.01; Control Rod Scram Time Evaluation, Rev. 23

Student Ref: TS 3.1.3 and TS 3.1.4

Learning Objective: SDLP03A R8; 1.17 and 1.18

Question source: Limerick NRC 10/06; Q-94

Question History: Limerick NRC 10/06; Q-94

Cognitive level: Comprehensive/Analysis:

10CFR55.43(b) (6)

QUESTION 92.

The plant is in a refueling outage in MODE 5. There is **NO** fuel movement in progress. The "A" CREVAS System is OPERABLE.

The 24 month Surveillance for "B" Control Room Emergency Ventilation Air Supply (CREVAS) System has just been completed at 0400 with the following test results:

- HEPA filter inplace test penetration and system bypass flow is 0.9% at a flow rate of 990 scfm.
- Charcoal adsorber inplace test penetration and system bypass flow is 0.8% at a flow rate of 980 scfm.

The refuel schedule has an activity to perform a recirculation pump suction valve replacement using a freeze seal beginning at 0800.

You are the CRS, tasked with evaluating Tech Specs to determine if this maintenance activity can be performed as scheduled.

Which **ONE** of the following describes:
(1) whether this work can be authorized to commence **WITH** the existing plant conditions **AND WITHOUT** any additional evaluations, and
(2) the basis for that decision?

	(1)	(2)
A.	Maintenance activity **CANNOT** be authorized.	LCO 3.7.3 is currently **NOT** met and entry into OPDRV specified condition can **ONLY** be done when two CREVAS subsystems are OPERABLE.
B.	Maintenance activity can be authorized.	LCO 3.0.4.a allows entry into OPDRV specified condition because TS 3.7.3.D.1 allows continued operation for an unlimited period with an OPERABLE CREVAS subsystem in isolate mode.
C.	Maintenance activity **CANNOT** be authorized.	LCO 3.7.3 is currently **NOT** met and entry into OPDRV specified condition would require a LCO 3.0.4.b risk assessment be performed prior to performing the maintenance.
D.	Maintenance activity can be authorized.	Maintenance activity is **NOT** an OPDRV and Technical Specification 3.7.3 is not applicable in MODE 5.

K&A # 290003; G2.2.38
Importance Rating SRO – 4.5

QUESTION 92.
K&A Statement: **CONTROL ROOM HVAC G2.2.38 Knowledge of conditions and limitations in the license.**

Justification:

A. Incorrect because with one subsystem operable the work can commence without any additional evaluations. LCO 3.0.4.a allows entry into a specified condition if the actions have an unlimited completion time. With one subsystem inoperable 3.7.3.A would be the required action and if the 7 days ran out action 3.7.3.D would allow placing the operable subsystem for an unlimited time, therefore the maintenance can be performed and the specified condition can be entered. Plausible if the candidate does not consider LCO 3.0.4. and thinks the only way to meet LCO 3.7.3 is to restore the B subsystem to operability.

B. CORRECT: because the "B" CREVAS is inoperable per TS 5.5.8.b. and LCO 3.0.4.a is applicable. LCO 3.7.3 requires 2 subsystems to be operable. LCO 3.7.3 is not met and entry into a specified condition (during OPDRVS) can only be made IAW LCO 3.0.4. LCO 3.0.4.a allows entry into a specified condition if the actions taken have an unlimited time which per 3.7.3.D with the operable system in isolate OPDRV's can continue for an unlimited time. Therefore the OPDRV specified condition can be entered without restoring the inoperable system or by performing a risk assessment.

C. Incorrect because with one subsystem operable the work can commence without any additional evaluations. Although the "B" CREVAS is inoperable per TS 5.5.8.b. and therefore 3.7.3 is not met, the risk assessment is not required because LCO 3.0.4.a allows entry into the specified condition. Plausible if the candidate does not consider LCO 3.0.4.a to allow entry into the specified condition of an OPDRV.

D. Incorrect because the maintenance activity is an OPDRV. Maintenance activities that open a reactor coolant pressure boundary below the RPV normal water level and actions are required to maintain the isolation is an OPDRV. The suction valve is below the normal RPV level and the freeze seal requires actions to maintain the isolation. Plausible if the candidate does not know the definition of an OPDRV per AP-10.09.

References: TS 3.7.3 and 5.5.8 Student Ref: TS 3.0.4(only),3.7.3 and 5.5.8

AP-10.09, Rev 26

Learning Objective:

Question source: NEW

Question History: None

Cognitive level: Comprehensive/Analysis:

10CFR55.43 (1)

QUESTION 93.

The plant is operating at steady-state conditions at 85% power when #2 Turbine Control Valve drifts closed slowly.

Which **ONE** of the following identifies an appropriate procedural response to mitigate the transient?

A. Reduce reactor power to approximately 75%, then evaluate thermal limits using 3D Monicore report per AOP-6, "Malfunction of EHC Pressure Regulator".

B. Reduce reactor power as necessary to maintain power at pre-transient level per RAP-7.3.16, "Plant Power Changes".

C. Reduce reactor power to $\leq$ 65%, then insert rods until power below 70% rod line per AOP-62, "Loss of Feedwater Heating".

D. Reduce reactor power as necessary to ensure the bypass valves close per AOP-32, "Unexplained/Unanticipated Reactivity Change".

K&A # 241000; A2.04
Importance Rating SRO – 3.8

QUESTION 93.
K&A Statement:

Ability to(a) predict the impacts of the following on the **REACTOR/TURBINE PRESSURE REGULATOR** and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **FAILED OPEN/CLOSED CONTROL/GOVERNOR VALVE**

Plant response: Closure of #2 TCV was demonstrated in the Fitzpatrick simulator at 85% power, the power level at which the station performs normal TCV closure surveillance testing. The simulator showed TCVs 1 and 3 go full open, TCV 4 goes to 60% open and one BPV remains open to control pressure at new, higher equilibrium value. RPV pressure peaked at 1030 psig during the transient. Feedwater temperature lowered from 402°F to 391°F.

A. **Incorrect.** AOP-6 directs a 5% power reduction, not a 10% power reduction. Also, AOP-6 is written to specifically address a pressure regulator malfunction. This type of malfunction would affect all TCVs, not just one. **Plausible** because AOP-6 entry condition is met (change in reactor pressure with no operator action), because AOP-6 does direct a power reduction and because AOP-6 directs thermal limit evaluation. Applicants may not remember that the AOP directs a specific percentage power reduction but may not remember it is 5% not 10%.

B. **Incorrect.** The transient causes a BPV to open and remain open, resulting in a reduction in normal feedwater heating. Power level must be lowered to close the BPV to restore normal feedwater temperature. **Plausible** because it is generally desirable to stabilize power at pre-transient level pending further assessment of conditions and procedural requirements. RAP 7.3.16 does provide guidance for power changes.

C. **Incorrect.** The transient causes a BPV to open and remain open, resulting in a reduction in normal feedwater heating. Power level must be lowered. However, the feedwater heating problem is resolved when BPVs are closed. Guidance in AOP-62 addresses a sustained loss of heating due to extraction steam isolation or heater level control problems. Also, action to reduce to below the 70% rod line is only required for loss of feedwater heating with initial core flow less than 55%. **Plausible** because there is a feedwater temperature issue when BPVs are open. Also, AOP-32 refers the operator to AOP-62 for a loss of feedwater heating.

D. **CORRECT.** Entry conditions and symptoms are met for AOP-32, Unexplained / Unanticipated Reactivity Change. The procedure provides guidance for lowering reactor power to the desired level. In the given situation, the desired level would be that power level which would result in closure of the BPV(s).

References: SDLP-94C, AOP-6 R7, AOP-32 R10, AOP-62 R8
RAP 7.3.16 R44

Student Ref:
None

Learning Objective: 1.05.a.4 & 5

Question source: Modified to JAF
Question History: Limerick NRC 10//06; Q-90
Cognitive level: Comprehensive/Analysis:
10CFR55.43(b) (5)

QUESTION 94.

The plant is in MODE 5 with core alterations in progress.

- You are the oncoming refuel bridge SRO, preparing to continue core alterations.
- The time is 1600.
- The offgoing refuel bridge SRO briefs you on the status of refueling activities.
- RPV water temperature is 67 °F.
- All control rods are inserted.
- Refueling Checklist Attachment 9 was last performed at 0300.
- Current on-shift ops staffing consists of Refuel Bridge SRO, Bridge Operator, Control Room Refuel Operator, Fuel Move Spotter, Reactor Engineer, Shift Manager, CRS, another RO and 3 NPOs.
- The on-duty CRS has just left the site due to illness. The Shift Manager has called another CRS to come to the site. The CRS will arrive in 2 hours.

Based on these conditions, which describes your expected actions as the oncoming Refuel Bridge SRO IAW RAP-7.1.04B, "Refueling Procedure"?

A. Continue with planned moves, all refueling administrative requirements are met.

B. Perform Refueling Checklist Attachment 9, checklist has not been performed within required periodicity.

C. Perform work stand down, refueling team is not properly staffed.

D. Halt fuel moves, water temperature limit for refueling is not met.

K&A # G.2.1.35
Importance Rating SRO – 3.9

QUESTION 94.
K&A Statement: Knowledge of fuel-handling responsibilities of SROs
Justification:

A. **Incorrect because** the minimum temperature of 68 degrees is not met. **Plausible if the candidate** does not remember this requirement.

B. **Incorrect because** the refuel checklist is required per step 8.1 of RAP 7.1.04B to be completed for the current day. The turnover checklist (attachment 20 requires within the past 24 hours.) **Plausible if the candidate** does not remember this requirement.

C. **Incorrect because** this requirement is listed in RAP-7.1.04B step 7.1. The refueling team is properly staffed. **Plausible if the candidate** does not remember this requirement.

D. **CORRECT:** Temperature band for core alterations is 68°F to 135°F IAW RAP-7.1.04B Refueling checklist to ensure proper plant conditions exist and Tech Specs are satisfied during refuel activities. With RPV water temperature at 67°F this temperature is outside on the established temperature band for refueling.

References: RAP-7.1.04B R24
TS 4.3.1.1.a
SDM TS Definition
SDM TS 3.1.1.e

Student Ref: None

Learning Objective: LP AP R9 73.02 SRO Refuel Responsibilities

Question source: New

Question History: New

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55.43(b)(6)

QUESTION 95.

The plant is in MODE 1. It is January 25th 2010.

Prior to assuming the shift as the Control Room Supervisor (CRS), you are informed that you are **NOT** qualified.

Which **ONE** of the following conditions would have caused you to be disqualified as a CRS?

A. SRO License was issued in December of 2005.

B. Last quarter you stood three 12 hour watches as CRS and five 12 hour watches as Shift Manager.

C. NRC Form 396 Certification of Medical Examination by Facility Licensee completed in December of 2007.

D. Successfully completed a re-qualification operating exam in October 2008 and then a re-qualification written and operating exam in December 2009.

K&A # G.2.1.4
Importance Rating SRO – 3.8

QUESTION 95.

K&A Statement: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Justification:

A. Incorrect because license expires in 6 years, December of 2011but plausible if candidate does not know how long licenses are issued.

B. Incorrect because either position CRS or shift manager performs SRO duties and is therefore proficient. But plausible if the candidate thinks that standing only 3 watches previous quarter does not satisfy proficiency watches for quarter.

C. Correct – Require a medical exam every two years. Expired December 2009

D. Incorrect because exceeded annual operating test but completed within training cycle.

References: 10CFR55, ODSO-30 Student Ref: None

Learning Objective: N/A

Question source: Perry 2009 NRC Exam

Question History: Q SRO 19

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55.43(b) (2)

QUESTION 96.

Given the following:

- I&C is performing APRM SYSTEM 'A' CHANNEL FUNCTIONAL TEST.
- The next section of the test procedure contains several discrepancies.

Which **ONE** of the following changes is **PROHIBITED** as an **EDITORIAL** change to the procedure in accordance with AP-02.04, "Control of Procedures"?

A. Correct a title of a specific, referenced component.

B. Delete a redundant caution from a section of the functional test.

C. Fix a setpoint range, where the correction is justified by other steps in the same procedure.

D. Add a commitment and associated step annotations that alter existing text in a procedure step.

K&A # G.2.2.6
Importance Rating SRO – 3.6

QUESTION 96.
K&A Statement: Knowledge of the process for making changes to procedures.
Justification:

A. Incorrect because per AP-.04 attachment 4 correcting errors such as equipment numbers or titles is editorial. Plausible if applicant does not know the definition of editorial.

B. Incorrect because per AP-.04 attachment 4 deleting cautions is editorial. Plausible if applicant does not know the definition of editorial.

C. Incorrect because per AP-.04 attachment 4 correcting errors such as typographical errors in setpoints is editorial. Plausible if applicant does not know the definition of editorial.

D. Correct-per AP-02.04 adding commitments that do NOT alter existing text would be an editorial change. Therefore altering text would be prohibited per the procedure.

References: AP-02.04 R44 Student Ref: None

Learning Objective: AP-02.04 LO4.02.a

Question source: HC NRC Exam 2007

Question History: Q97 HC 2007

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55.43(b) (3)

Question 97

The plant is operating at 100% power.

- At time 0930 Annunciator 09-5-2-56 "MAIN TURBINE HI VIB ALERT" alarms
- EPIC indicates that vibration levels for Main Turbine bearings 2 and 3 are stable at 9.5 mils
- All other Main Turbine bearings indicate vibration levels that are < 7mils.

The time is now 0937 and Main Turbine Bearing vibration levels are the same as indicated above.

Which of the following statements correctly describes the AOP-66 "Main Turbine High Vibration" response to this condition?

A. Reduce load per RAP-7.3.16 to establish vibration < 9 mils.

B. Manually Scram the Reactor and enter AOP-1.

C. If alarms continue for > 15 minutes, refer to section B of OP-9 for operation with elevated vibration.

D. Trip the Main Turbine per section G of OP-9.

K&A # G.2.2.44
Importance Rating SRO – 4.4

QUESTION 97.
K&A Statement: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Justification:

A. Correct – AOP-66 Step E.3 Table regarding the response to high vibration states: Any bearing >9 mils for < 15 minutes, reduce load per RAP-7.3.16 to establish vibration < 9 mils.

B. Incorrect but plausible is not applicable per AOP-66; At >29 % power, if the bearing vibration is ≥ 10 mils for bearings 1, 2, 9 or 10 **OR** ≥ 12 mils for bearings 3, 4, 5, 6, 7 or 8 for greater than 15 minutes, then Scram the Rx and execute AOP-1.

C. Incorrect but plausible per AOP-66 Step F.2, a Reactor Scram would be required if the alarms continued for greater than 15 minutes as power is currently > 29%

D. Incorrect but plausible is not applicable per AOP-66, At <29% power, if the bearing vibration is >9 mils >15 minutes, then trip the turbine and execute AOP-2.

References: AOP-66, Rev. 6; ARP 09-5-2-56 (Rev. 0) Student Ref: None

Learning Objective: LP-AOP; 1.01 and 1.03

Question source: New

Question History: New

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR55.43(b) (5)

QUESTION 98.

A canal discharge is required to reduce level in Waste Sample Tank 'A'. The radwaste effluent radiation monitor is out of service for maintenance.

Select the choice which correctly fills in the blanks in the statement below to identify
1) who approves the permit **AND**
2) what additional actions must be documented to initiate the discharge with the radiation monitor out of service per OP-49, Liquid Radioactive Waste System

The ____(1)_____ approves the permit, and the permit must document that ___(2)____ shall verify discharge line valving, and that two independent representative samples are obtained and analyzed.

	(1)	(2)
A.	Shift Manager	one technically qualified facility staff member
B.	General Manager – Operations	one technically qualified facility staff member
C.	Shift Manager	two technically qualified facility staff members
D.	General Manager – Operations	two technically qualified facility staff members

K&A # G.2.3.6
Importance Rating SRO – 3.8

QUESTION 98.
K&A Statement: Ability to approve release permits.

Justification:

A. Incorrect because OP-49 requires two individuals perform the discharge line valving, but plausible if the candidate does not know the special requirements for an unmonitored release.

B. Incorrect because OP-49 requires two individuals perform the discharge line valving and the the shift manager must approve the permit. Plausible if the candidate does not know the special requirements for an unmonitored release and the Radwaste Supervisor obtaining General Manager – Operations concurrence to perform a canal discharge.

C. Correct – OP-49 requires that the permit document that two qualified members of the staff shall verify the discharge valving. The Shift Manager approves the permit.

D. Incorrect but plausible because although the GMO must concur for a discharge the shift manager approves the permit. Plausible if the candidate does not know the special requirements for an unmonitored release and the Radwaste Supervisor obtaining General Manager – Operations concurrence to perform a canal discharge.

References: OP-49 R58, ODCM Table 2.1-1 Student Ref: None

Learning Objective: 2.3.6 Knowledge of the requirements for reviewing and approving release permits.

Question source: Modified Fitzpatrick NRC Exam

Question History: Q93 2002 NRC Exam

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55.43(b)(4)

QUESTION 99.

A plant event has led to the following conditions:

Time	Condition
0800	Plant conditions support declaration of an **ALERT**.
0805	The Shift Manager has assessed plant conditions and informs the Control Room staff that he has declared an **ALERT**.
0808	Plant conditions have improved. Plant conditions **NO LONGER** support declaration of an **ALERT**. Plant conditions support declaration of a **NOTIFICATION OF UNUSUAL EVENT (NUE)**. The initial notification for the event has **NOT** yet been signed or transmitted.

Which **ONE** of the following describes the IAP-2, "Classification of Emergency Conditions", notification to be made in response to these conditions at time 0808?

A. Declare and report an ALERT. Terminate the event when terminate criteria are met.

B. Declare and report a NUE. Terminate the event when terminate criteria are met.

C. Declare and report an ALERT. Submit a separate notification to indicate the change in classification to a NUE.

D. Declare and report a NUE. No mention of the momentary ALERT condition is required.

K&A # G.2.4.29
Importance Rating SRO - 4.4

QUESTION 99.
K&A Statement: Knowledge of the emergency plan.

Justification:

A. **CORRECT:** IAW IAP-2 step 5.1.4. Classifying transient events. For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses (e.g. coolant radiochemistry sampling) may be necessary. Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

B. **Incorrect but plausible if the candidates forgets that** IAW IAP-2 step 5.1.4. Classifying transient events. For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses (e.g. coolant radiochemistry sampling) may be necessary. Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

C. **Incorrect but plausible if the candidates forgets that** IAW IAP-2 step 5.1.4. Classifying transient events. For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses (e.g. coolant radiochemistry sampling) may be necessary. Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

D. **Incorrect but plausible if the candidates forgets that** IAW IAP-2 step 5.1.4. Classifying transient events. For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses (e.g. coolant radiochemistry sampling) may be necessary. Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

References: IAP-2, Rev 28; step 5.1.4 Student Ref: None

Learning Objective: EP-12.6 LO 1.10
EP-12.5.4.2 LO 2.05

Question source: Modified

Question History: NMP #1 10/08; Q-98

Cognitive level: Memory/Fundamental knowledge:

10CFR55.43(b) (5)

QUESTION 100.

Initial plant conditions are the plant is operating at 100% power. A LOCA occurs. Drywell pressure increases to 3.0 psig with **NO** rod motion.

Subsequently, plant conditions are as follows:

- A manual scram is inserted resulting in **NO** rod movement.
- Reactor Pressure is being controlled 800-1000 psig with SRV's.
- Drywell Pressure is 5 psig and rising.
- RCS leakage is 15 GPM.

15 minutes later, plant conditions are as follows:

- Reactor Pressure is being controlled 800-1000 psig with SRV's.
- Reactor water level has been lowered to +100 inches.
- Reactor Power is 2.0%.
- Torus temperature is 135 °F and rising slowly.
- RCS leakage is 25 GPM.
- Drywell Pressure is 8 psig and rising.
- Torus level is 10 feet and rising slowly.

Which **ONE** of the following describes the **HIGHEST** EAL classification required for the conditions described above?

A. Notification of Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

K&A # G.2.4.41
Importance Rating SRO – 4.6

QUESTION 100.
K&A Statement: Knowledge of the emergency action level thresholds and classifications.

Justification:

A. Incorrect because the ATWS is classified above a NUE. Plausible if the candidate thinks that the RCS leakage above 10 GPM is the highest classification due to the wording in the alert for the ATWS saying automatic scram, however the reactor should have scrammed on the 5 psig drywell pressure.

B. Incorrect because the ATWS combined with the initial power being greater than 2.5% means that the correct classification is SAE. Plausible if the candidate thinks that power being reduced to 2.0% means that the alert is the correct classification. The highest EAL should be declared.

C. Correct: Because the manual and automatic scrams did not lower power to below 2.5%. Lowering level did reduce power to <2.0% however the procedure states to declare the highest EAL classification.

D. Incorrect because the ATWS plus exceeding the HCTL would be the GE classification. However RPV pressure can be maintained <HCTL. The pressure band can be lowered to meet the HCTL curve. Plausible if the candidate does not realize pressure can be reduced.

References: EAL Matrix Student Ref: EAL Matrix + HCTL and BIIT curves

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Analysis

10CFR55.43(b) (5)

2010 JAF NRC Written Exam Answer Key

1.	B	26.	C	51.	A	76.	A
2.	A	27.	C	52.	C	77.	B
3.	C	28.	A	53.	A	78.	D
4.	A	29.	D	54.	C	79.	C
5.	C	30.	B	55.	C	80.	A
6.	D	31.	B	56.	D	81.	B
7.	B	32.	B	57.	B	82.	D
8.	C	33.	D	58.	A	83.	D
9.	A	34.	B	59.	B	84.	B
10.	C	35.	A	60.	D	85.	B
11.	A	36.	C	61.	B	86.	A
12.	C	37.	B	62.	D	87.	A
13.	A	38.	C	63.	B	88.	C
14.	B	39.	D	64.	C	89.	A
15.	D	40.	A	65.	D	90.	B
16.	A	41.	D	66.	B	91.	C
17.	D	42	A	67.	B	92.	B
18.	C	43.	C	68.	C	93.	D
19.	C	44.	D	69.	D	94.	D
20.	A	45.	D	70.	D	95.	C
21.	D	46	B	71.	A	96.	D
22.	D	47.	B	72.	D	97.	A
23.	A	48.	A	73.	B	98.	C
24.	D	49.	D	74.	C	99.	A
25.	A	50.	B	75.	C	100.	C