1ST 50 Q'S
"DRAFT"

ES-401 **Written Examination Quality Checklist** Form ES-401-6

Facility: **Limerick Units 1 & 2** Date of Exam: **10/31/2008** Exam Level: RO **X** SRO **X**

	***Note: This checklist covers the first 50 questions developed, as listed below.				Initial		
	Item Description				a	b*	c#
1.	Questions and answers are technically accurate and applicable to the facility.					N/A	
2.	a. NRC K/As are referenced for all questions. b. Facility learning objectives are referenced as available.					N/A	
3.	SRO questions are appropriate in accordance with Section D.2.d of ES-401					N/A	
4.	The sampling process was random and systematic (If more than 4 RO or 2 SRO questions were repeated from the last 2 NRC licensing exams, consult the NRR OL program office).						
5.	Question duplication from the license screening/audit exam was controlled as indicated below (check the item that applies) and appears appropriate: __ the audit exam was systematically and randomly developed; or __ the audit exam was completed before the license exam was started; or X the examinations were developed independently; or __ the licensee certifies that there is no duplication; or __ other (explain)					N/A	
6.	Bank use meets limits (no more than 75 percent from the bank, at least 10 percent new, and the rest new or modified); enter the actual RO / SRO-only question distribution(s) at right.	Bank: 16 / 2	Modified: 10 / 1	New: 21 / 0		N/A	RO only
7.	Between 50 and 60 percent of the questions on the RO exam are written at the comprehension/ analysis level; the SRO exam may exceed 60 percent if the randomly selected K/As support the higher cognitive levels; enter the actual RO / SRO question distribution(s) at right.	Memory: 19 / 0	C/A: 28 / 3			N/A	
8.	References/handouts provided do not give away answers or aid in the elimination of distractors.					N/A	
9.	Question content conforms with specific K/A statements in the previously approved examination outline and is appropriate for the tier to which they are assigned; deviations are justified.					N/A	
10.	Question psychometric quality and format meet the guidelines in ES Appendix B.					N/A	
11.	The exam contains the required number of one-point, multiple choice items; the total is correct and agrees with the value on the cover sheet.				N/A	N/A	N/A

	Printed Name / Signature	Date
a. Author	J. Tomlinson /	8/8/08
b. Facility Reviewer (*)	Not Applicable – NRC Developed Exam	
c. NRC Chief Examiner (#)	P. Presby (CE – UI) / Peter Presby	8/8/08
d. NRC Regional Supervisor	J. Caruso (Acting BC) /	8/8/08

Note: * The facility reviewer's initials/signature are not applicable for NRC-developed examinations.
\# Independent NRC reviewer initial items in Column "c"; chief examiner concurrence required.

*** This 401-6 covers 50 Q #s: 1,4,5,7,8,9,10,12,13,14,15,16,17,21,22,23,24,25,26,28,29,30,31,33,34,35,37,38,39,40,42,43,44,45,46,48,50,51,52,54,56,57,58,59,64,66,71,76,85,92

Questions 1-75 are RO
Questions 76-100 are SRO ONLY

NRC-developed Limerick 2008 Initial NRC Written Exam

Partial Submittal (first 50 questions) to Facility for Review on 8/8/08

Questions in Package (1-75 are RO, 76-100 are SRO):

1	15	29	42	56
4	16	30	43	57
5	17	31	44	58
7	21	33	45	59
8	22	34	46	64
9	23	35	48	66
10	24	37	50	71
12	25	38	51	76
13	26	39	52	85
14	28	40	54	92

QUESTION 1

Unit 1 plant conditions are as follows:

- 50% power
- 65% total core flow on recorder XR-042-1R613
- A recirc pump speed at 57%
- B recirc pump speed at 60%
- Core dp 4.7 psid on recorder XR-042-1R613

"B" recirc pump trips.

New plant conditions are:

- 39% total core flow on recorder XR-042-1R613
- A recirc pump speed at 57%
- Core dp 1.5 psid on recorder XR-042-1R613

Assume no operator actions are taken.

WHICH ONE of the following identifies the approximate core power level and recirculation flow configuration one minute after the recirc pump trip based on automatic actions and the above conditions?

A. 40% power; idle loop flow is reverse

B. 40% power; idle loop flow is forward

C. 37% power; idle loop flow is reverse

D. 37% power; idle loop flow is forward

K&A #	295001 Partial or Complete Loss of Core flow
Importance Rating	3.3

QUESTION 1

K&A Statement: K1.02 – Knowledge of the operational implications of power to flow distribution as it applies to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION

Justification:

A. In-Correct but plausible. Idle loop flow is always subtracted, but for recirc pump speed less than 60%, actual flow is forward through the jet pumps. Since actual flow is forward, to determine actual total core flow, the flow through the idle loop should not be subtracted. The correct reactor power is therefore determined based on core dp measurement. This choice provides the correct power level; however, the basis for the answer is incorrect.

B. Correct - Total core power is determined based on core dp and rodline. Actual idle loop flow is forward through the jet pumps when the operating recirc pump is less than 60% speed.

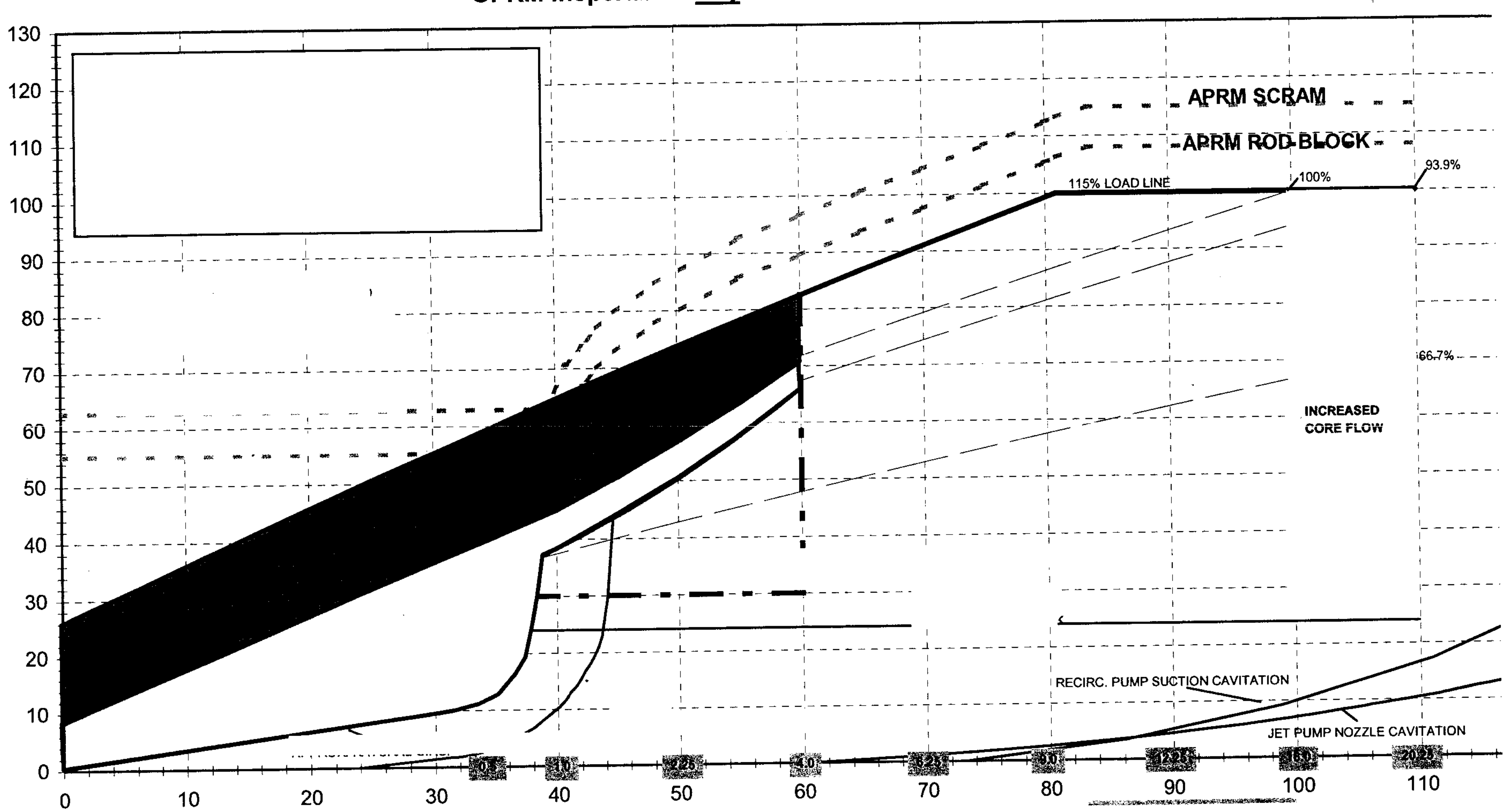
C. In-Correct but plausible if the applicant does not understand that indicated flow is less than actual flow when in single loop operation with the recirc pump speed less than 60%. Using indicated flow instead of core dp results in underpredicting actual core power.

D. In-Correct but plausible if applicant does not understand indicated flow is less than actual flow, but understands flow is forward in the idle loop.

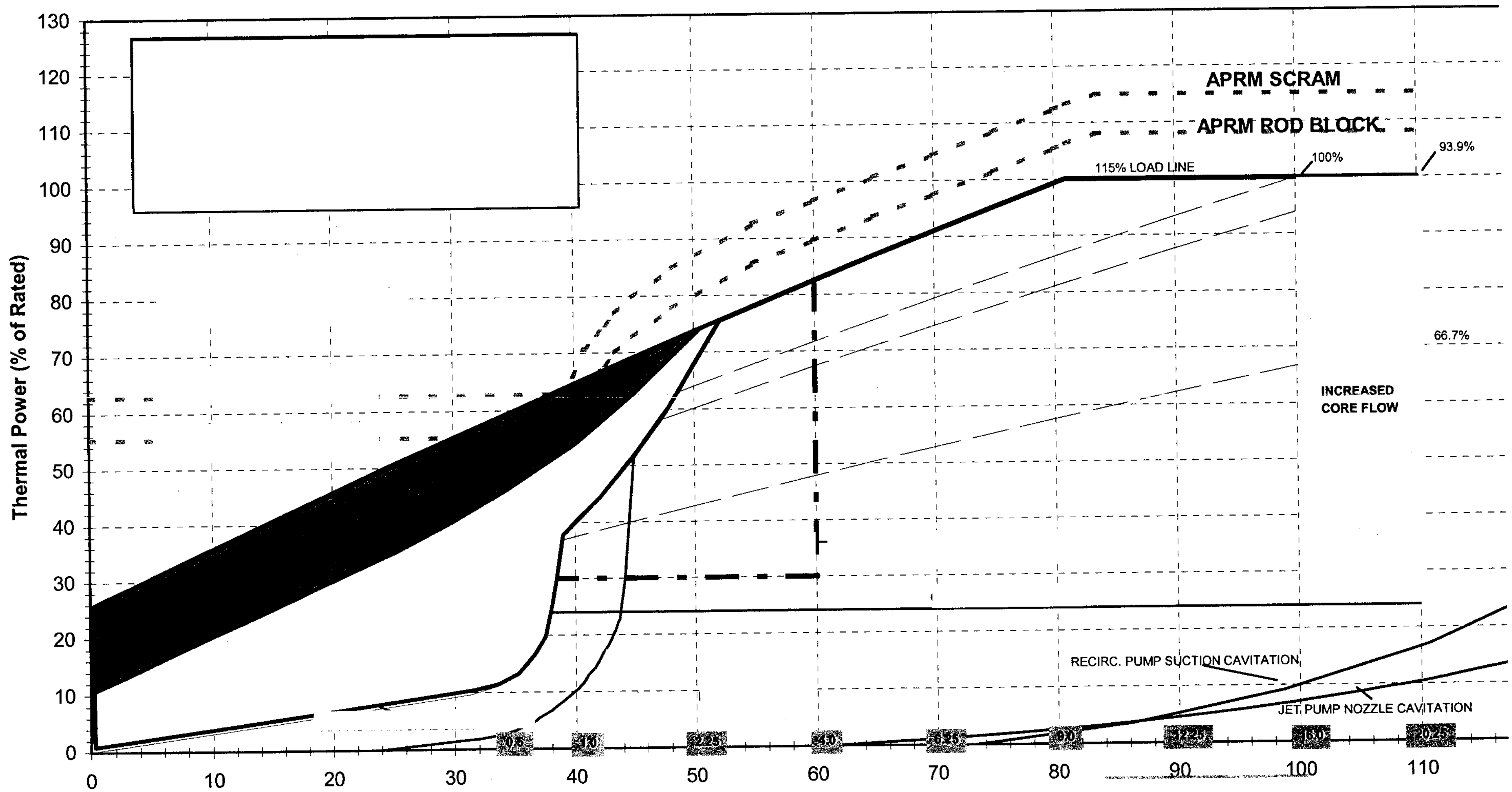
References:	OT-112 pg.4, Attachment 1, LLOT275	Student Ref: required	Yes, power to flow map w/o labeling

Learning Objective:	N/A	
Question source:	new	
Question History:	None	
Cognitive level:	Memory/Fundamental knowledge:	
	Comprehensive/Analysis:	X
10CFR	55.41(5) X	
Comments:	Created/Modified by: Tomlinson Reviewed by:	

LGS Power Flow Operation Map
OPRM Inoperable - Any Feedwater Heater Out Of Service



LGS Power Flow Operation Map
OPRM Inoperable - ALL Feedwater Heaters In Service



QUESTION 4

Unit 2 plant is at 18% power and shutting down in accordance with GP-3, "Normal Plant Shutdown" for a refueling outage following a 2 year operating cycle. Reactor Power was reduced from 48% to 18% in the previous 30 minutes. The turbine has just been tripped per step 3.1.33.1 of GP-3.

- There is a simultaneous trip of one of the two running feedwater pumps

Assume no additional operator action.

(1) WHICH ONE of the following describes the expected plant response over the next 15 minutes and
(2) WHICH ONE also provides a correct basis?

	(1)	(2)
A.	Reactor power lowers to a new stable value	Change in xenon concentration
B.	Reactor power lowers to a new stable value	Change in recirc pump speed
C.	Reactor power rises to a new stable value	Change in feedwater temperature
D.	Reactor power rises to a new stable value	Change in RPV pressure

K&A #	295005 Main Turbine Generator Trip
Importance Rating	3.8

QUESTION 4

K&A Statement: K2.01 – Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and Feedwater temperature

Justification:

A. In-Correct but plausible. Xenon would initially build in due to the power decrease resulting in further reducing reactor power. Incorrect because xenon is not decaying at this point. Plausible distractor because xenon is a poison and will cause power to change depending on if it is building in or burning out. The applicant needs to understand the characteristics of xenon during power changes.

B. In-Correct but plausible since recirc runbacks do occur on feed pump trips. However at 18% power the recirc pumps would already be at min speed.

C. Correct – Following a turbine trip, a loss of extraction steam to the feedwater heaters would lower feedwater temperature. FW temperature lowering would add positive reactivity to the core.

D. In-Correct but plausible. Expect RPV pressure to change slightly when removing the turbine from service. Several minutes after turbine is tripped pressure should return to normal based on BPVs taking steam loads.

References:	GP-3 "Normal Plant Shutdown", LLOT0540 pg. 26; LGSOPS0001A pg. 51	Student Ref: required	N

Learning Objective: LGSOPS0001A IL8f

Question source: SSSE bank

Question History: None

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

10CFR 55.41(5) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 5

Unit 1 conditions are as follows:

- Reactor Power is 100%
- Reactor level is 35"
- "FWLCS TROUBLE" alarm is received
- Alarm "1XX-FW301.1SFE", "Steam flow SMS Err" is noted on the FWLC Alarm List

A Main Turbine trip results in the following:

- RPS actuates and scram valves open
- 150 control rods fail to insert due to hydraulic lock
- Reactor power remains at 50%

WHICH ONE of the following describes the FWLC response, and the impact on Reactor water level after 30 seconds?

A. SCRAM Profile will activate, reactor water level will go up

B. SCRAM Profile will not activate on incomplete SCRAM, reactor water level will remain constant

C. SCRAM Profile will activate, reactor water level will go down

D. SCRAM Profile will not activate on incomplete SCRAM, reactor water level will go down

K&A # 295006 SCRAM
Importance Rating 3.9

QUESTION 5

K&A Statement: A1.02– Ability to operate and/or monitor Reactor water level control system as it applies to SCRAM

Justification:

A. In-Correct but plausible if applicant does not understand the system response based on the SCRAM profile.

B. In-Correct but plausible if applicant does not understand the conditions under which the SCRAM profile activates.

C. Correct – SCRAM profile activates by C71-K14 RPS relays located in panels 609 and 611. SCRAM profile is not dependent on rod position. SCRAM profile will reduce feedwater flow 6%/sec after the initial 10 seconds. Feedwater flow reduction will result in RPV level reduction.

D. In-Correct but plausible if applicant does not understand the conditions under which the SCRAM profile activates.

References: LGSOPS0550 pg. 31, 32, 33 Student Ref: required N

Learning Objective: N/A

Question source: Limerick Bank

Question History: None

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

10CFR 55.41(3) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 7

Unit 1 plant conditions are as follows:

85% power
"1A" Stator Water Cooling pump is tripped & in PULL-TO-LOCK
"1B" Stator Water Cooling pump is running
Stator Cooling outlet temperature is 76 C
Stator Cooling water inlet pressure is 23 psig
Stator Cooling Storage Tank level is 4" below normal level

WHICH ONE of the following describes the status of the Main Turbine due to the conditions above?

A. Will trip if generator stator current is 8,000 amps after 3.5 minutes

B. Will trip if generator stator current is 22,000 amps after 2 minutes

C. Will trip immediately if total feedwater flow is >50%

D. Will remain on-line

K&A # 295018 CCW
Importance Rating 3.4

QUESTION 7

K&A Statement: K2.02 - Knowledge of the interrelations between PARTIAL OR TOTAL LOSS OF CCW and **Plant operations**

Justification:

A. Correct – Plant conditions indicate a loss of cooling to the generator based on low stator coolant flow supply pressure, 43 psig. If the current carried by the main generator is not reduced to 7468 amps in 3 1/2 minutes a turbine trip will be generated.

B. In-Correct but plausible if the applicant does NOT remember the settings for the first checkpoint. A trip signal would be generated at two minutes if load was not reduced to 26,173 amps.

C. In-Correct but plausible if the applicant relates the turbine trip to a feedwater setpoint.

D. In-Correct but plausible if the applicant believes that stator cooling is adequate based on the "B" pump running.

References: LLOT0630 pages 14, 15, 16 Student Ref: required No

Learning Objective: N/A

Question source: Limerick Bank

Question History: None

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

10CFR 55.41(4) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Johnson

QUESTION 8

Plant conditions are as follows:

Instrument Air and Service Air are in a normal configuration

Instrument Air header pressure has dropped to below 65 psig

WHICH ONE of the following describes the status of Instrument Air headers and Service Air Compressor output due to the above conditions?

A. One Instrument Air header is pressurized from Service Air and Service Air Compressor output is directed to Instrument Air loads only.

B. One Instrument Air header is pressurized from Service Air and Service Air Compressor output is directed to both Service Air and Instrument Air loads.

C. Both Instrument Air headers are pressurized from Service Air, and Service Air Compressor output is directed to Instrument Air loads only.

D. Both Instrument Air headers are pressurized from Service Air, and Service Air Compressor output is directed to both Service Air and Instrument air loads.

K&A # 295019 Partial or Total Loss of Inst. Air

Importance Rating 3.6

QUESTION 8

K&A Statement: A2.02 – Ability to determine and/or interpret Status of safety-related instrument air system loads as it applies to PARTIAL OR TOTAL LOSS OF INSTRUMENT AIR

Justification:

A. Correct – Service Air will automatically backup Instrument Air when Instrument Air header pressure drops below the Service Air header pressure. When both Instrument Air headers drop to 70 psig, PV-015-*67 closes to isolate service air header from the service air compressor. Service Air feeds the selected Instrument Air header through a manually positioned valve.

B. In-Correct but plausible if the applicant does NOT remember that when both Instrument Air headers are less than 70 psig, Service Air header isolates from Service Air compressor.

C. In-Correct but plausible if the applicant does NOT remember that Service Air compressor feeds only the selected Instrument Air header.

D. In-Correct but plausible if the applicant does NOT remember that when both Instrument Air headers are less than 70 psig, Service Air header isolates from Service Air compressor.

References: LLOT0730 pages 4, ON-119 Student Ref: required No

Learning Objective: LLOT0730 #2

Question source: Limerick Bank

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Johnson

QUESTION 9

Unit 1 Plant conditions are as follows:

> OPCON 4
> "1A" RHR is in Shutdown Cooling
> No Reactor Recirc pumps are running

The "1A" RHR Pump trips

WHICH ONE of the following RPV level indication provides assurance that proper natural circulation will exist per S51.8.B, SHUTDOWN COOLING/REACTOR COOLANT CIRCULATION OPERATION START-UP AND SHUTDOWN based on the above conditions?

A. Upset level indication at 66 inches

B. Shutdown level indication at 64 inches

C. Wide Range level indication at 58 inches

D. Narrow Range level indication at 56 inches

K&A # 295021 Loss of Shutdown Cooling
Importance Rating 3.9

QUESTION 9

K&A Statement: K1.03 – Knowledge of the operational implications of Adequate core cooling as it applies to LOSS OF SHUTDOWN COOLING

Justification:

A. In-Correct but plausible. Vessel level must be maintained above 78 inches on the Upset level indication.

B. Correct – Vessel level must be maintained above 60 inches on the Shutdown level indication

C. In-Correct but plausible if the applicant does NOT remember wide range indication could be off scale. Must use either Upset level or Shutdown level indication.

D. In-Correct but plausible if the applicant does NOT remember narrow range indication would be off scale.

References: S51.8.B, Rev.66 pg.2; Drawing 042-01 Student Ref: required No

Learning Objective: LLOT

Question source: Limerick Bank

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Johnson

QUESTION 10

Unit 1 plant conditions are as follows:

- OPCON 5
- Control rod 50-27 is removed from the core but operable.
- Fuel bundle 43-20 is being lowered into the core
- SRM 'C' count rate increases from 70 cps to 300 cps and has stabilized.
- Remaining SRMs continue to indicate 70 to 80 cps
- Fuel Floor has just reported that fuel bundle 43-20 is approaching its seated position in the correct location.

Based on the above conditions, Which one of the following is a required action in accordance with ON-120, "Fuel Handling Problems" and provides an accurate basis for the action?

A. Insert all insertable control rods. This action ensures that an adequate shutdown margin exists under all conditions.

B. Direct raising of fuel bundle 43-20 from its current position. This action will reduce the sub-critical multiplication in the core.

C. Request Reactor Engineering to determine if count rate is within expected range. This action verifies higher count rate is consistent with a fuel bundle positioned adjacent to an SRM.

D. Evacuate the Fuel Floor. This action ensures personnel are safe from the effects of a potential criticality.

K&A # 295023 Refueling Accident
Importance Rating 3.2

QUESTION 10

K&A Statement: K1.02 – Knowledge of the operation implications of **shutdown margin** as it applies to Refueling Accident.

Justification:

A. In-Correct but plausible since this would be the appropriate action if an increasing trend were observed.

B. Correct – ON-120 directs the raising of the fuel assembly if still grappled, even before determining count rate trend. Count rate doubling is the basis for the actions in ON-120. However, stable counts indicate reactor is still subcritical. Withdrawal of the fuel assembly will reduce subcritical multiplication and increase the shutdown margin of the core.

C. In-Correct but plausible since ON-120 states that a doubling of count rate should be expected for loading of fuel assemblies adjacent to SRM detectors.

D. In-Correct but plausible since this would be the appropriate action if an increasing trend were observed. However, stable counts indicate a subcritical condition and evacuation would not be required.

References: LLOT0760, Rev 14
LLOT1550, Rev.13
ON-120, Rev.017

Student Ref. required No

Learning Objective: Obj 2, (LLOT1550)

Question source: Modified (Limerick) Changed conditions and distractors. Changes resulted in new answer.

Question History: NRC-05 (LGS), OYS CERT-04

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 12

Unit 1 plant conditions are as follows:

- Unit 1 scrammed from rated power.
- Turbine has tripped.
- SRVs are cycling to control pressure.
- Turbine bypass valves are approximately 3% open.

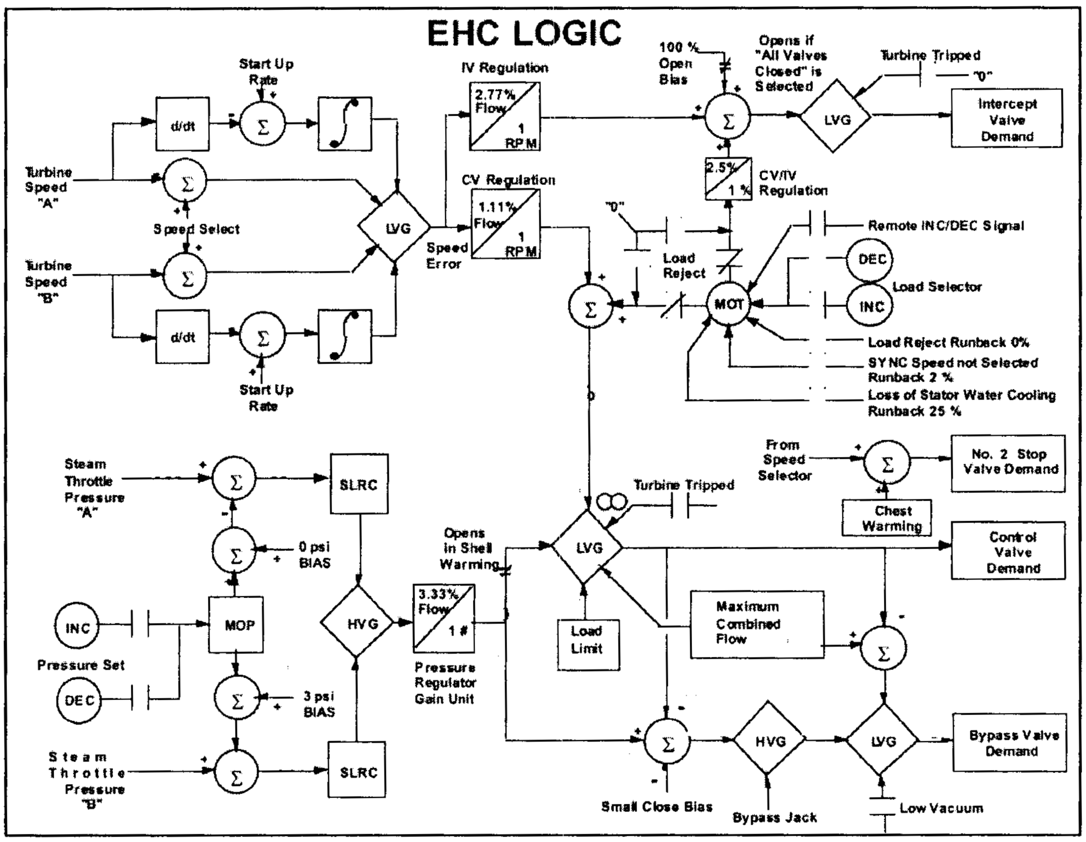
Given the above conditions and using the logic diagram below, which one of the following is consistent with the above conditions?

A. Bypass Jack adjusted to 3%.

B. Maximum Combined Flow Limiter adjusted to 103%.

C. Controlling pressure regulator output has failed low.

D. Pressure Set has failed high to 1099 psi.



K&A #	295025 High Reactor Pressure
Importance Rating	3.5

QUESTION 12

K&A Statement: K3.08 – Knowledge of the reasons for Reactor/Turbine pressure regulating system operation as it applies to High Reactor Pressure.

Justification:

A. In-Correct but plausible if the applicant doesn't understand that the high value gate would select the much higher output (>100%) coming from the Pressure Regulator Gain Unit over the input from the Bypass Jack.

B. In-Correct but plausible if applicant believed that the lower than normal setting (103% -vs- 115%) on the Maximum Combined Flow Limiter will limit bypass valve demand signal to 3%.

C. In-Correct but plausible if the applicant doesn't understand that the backup regulator output would ramp and become the output of the high value gate as steam throttle pressure increases.

D. Correct – With a Steam Throttle Pressure Transmitter range of 0 to 1100 psi and RPV pressure at SRV lift setpoint (~1170 psi), a Pressure Set setpoint of 1099 would generate a 1 psi mismatch. The range of the Pressure Set unit is 0 to 1100 psi. Once converted by the Pressure Regulator Gain Unit, an equivalent percent demand signal of 3.33% (~ 3%) would be sent to the Turbine Bypass Valves.

References: LGSOPS0031B Rev000 — Student Ref. required: No

Learning Objective: IL 2, IL 5

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 13

Unit 2 plant conditions are as follows:

- Unit 2 is at rated power.
- Quarterly HPCI flow testing is in progress IAW ST-6-055-230-2 "HPCI Pump, Valve and Flow Test".
- Plant is configured to support testing.
- Suppression pool level is 22' 10".
- Div 1 SPOTMOS indicates 92°F and slowly trending higher.
- Div 2 SPOTMOS indicates 90°F and slowly trending higher.
- Highest individual suppression pool temperature sensor indicates 97°F.

Based on these current conditions, which one of the following courses of action is required for these conditions?

A. Enter T-102 "Primary Containment Control" AND immediately suspend testing.

B. Suspend testing when average Suppression Pool temperature reaches 95°F.

C. Enter T-102 "Primary Containment Control" AND suspend testing before highest individual suppression pool temperature reaches 105°F.

D. Suspend testing before average Suppression Pool temperature reaches 105°F.

K&A # 295026 Suppression Pool High Water Temperature

Importance Rating 3.9

QUESTION 13

K&A Statement: A1.03 – Ability to operate and/or monitor **temperature monitoring** as it applies to Suppression Pool High Water Temperature.

Justification:

A. In-Correct but plausible if applicant doesn't recognize that entry into T-102 is based on the "average" suppression pool temperature reaching 95°F versus the "highest individual" temperature indication AND is unaware of exception during testing which allows heat to be added to the Suppression Pool until the maximum "average" Suppression Pool temperature approaches 105°F.

B. In-Correct but plausible if the applicant is unaware of exception during testing which allows heat to be added to the Suppression Pool until the maximum average Suppression Pool temperature approaches 105°F.

C. In-Correct but plausible if applicant doesn't recognize that entry into T-102 is based on the "average" suppression pool temperature reaching 95°F versus the "highest individual" temperature indication AND is unaware of the exception during testing which allows heat to be added to the Suppression Pool until the maximum "average" Suppression Pool temperature approaches 105°F.

D. Correct – Average Suppression Pool Temperature is below 95°F, therefore entry into T-102 is not required yet. An exception allows maximum Suppression Pool Temperature to be increased from 95°F to 105°F during testing which adds heat to the Suppression Chamber.

References: LLOT0130 Rev016 Student Ref. required No

Learning Objective: Obj. 4

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 14

Given the following initial conditions:

- The plant is stable at 85% power
- RPV dome pressure is 1032 psig
- Drywell temperature is 124°F
- Drywell pressure is 0.22 psig

During the subsequent power increase to 100%, a loss of drywell cooling causes elevated drywell temperature and pressure.

Current plant conditions are as follows:

- The plant is at 100% power. Power, pressure and indicated level are stable.
- RPV dome pressure is 1045 psig
- Drywell temperature is 135°F
- Drywell pressure is 0.28 psig

WHICH ONE of the following correctly completes the statement to describe Narrow Range Level instrumentation indication relative to actual RPV level? Do not assume operator action or any subsequent RPS actuations or plant transients.

"Indicated level is __________ actual level and indicated level will trend ________ actual level if drywell temperature increases to 139°F."

A. less than, toward

B. less than, away from

C. equal to, away from

D. greater than, toward

K&A # 295028 High Drywell Temperature

Importance Rating 3.6

QUESTION 14

K&A Statement: K2.03 – Knowledge of interrelations between HIGH DRYWELL TEMPERATURE and Reactor water level indication

Justification:

A. In-Correct but plausible. NR level is non-density compensated and is calibrated for 1045 psig PRV pressure and 135 F drywell temperature. Plausible that indicated is less than actual if applicant assumes the increased variable leg density at 1045 psig RPV pressure has de-calibrated the instrument, since lower density water column on the variable leg will lower the indicated level.

B. In-Correct but plausible. NR level is non-density compensated and is calibrated for 1045 psig PRV pressure and 135 F drywell temperature. Plausible, and correct, that indicated level will trend away from actual level if drywell temperature continues to rise.

C. Correct – NR level is non-density compensated and is calibrated for 1045 psig PRV pressure and 135 F drywell temperature. NT indicated level will trend away from actual level as drywell temperature continues to increase beyond the instrument calibration value of 135 F because density of the reference leg will decrease as it heats up, reducing the reference leg pressure on the d/p sensor.

D. In-Correct but plausible. NR level is no-density compensated and is calibrated for 1045 psig RPV pressure and 135 F drywell temperature. Plausible that indicated level is greater than actual level is applicant assumes instrument was at calibrated conditions prior to the drywell temperature increase to 135 F.

References: LGSOPS0042 pg. 37, 39 IL 7G, Dwg. 042-01, 006-04 Student Ref: required N

Learning Objective: LGSOPS0042, #7

Question source: New

Question History:

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

10CFR 55.41(5) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 15

Unit 2 plant conditions are as follows:

- An Emergency Blowdown in progress.
- Suppression Pool level is 14' and lowering.
- "2A" and "2B" RHR are in Suppression Pool Cooling with suction temperature at 115°F.
- "2A" and "2B" Core Spray loops are injecting.
- Division 1 SPOTMOS indicates 131°F
- Division 2 SPOTMOS is de-energized.

Which one of the following is the actual value of Suppression Pool temperature and the status of ECCS NPSH limits?

	Suppression Pool <u>Temperature</u>	ECCS NPSH <u>Limits</u>
A.	131°F	Met
B.	131°F	Not Met
C.	115°F	Met
D.	115°F	Not Met

K&A #	295030 Low Suppression Pool Water Level
Importance Rating	3.6

QUESTION 15

K&A Statement: A1.01 – Ability to operate and/or monitor ECCS systems (NPSH considerations) as it applies to Low Suppression Pool Water Level.

Justification:

A. In-Correct but plausible if the applicant doesn't recognize that the Suppression Pool temperature thermocouples are uncovered below 17.8 ft.

B. In-Correct but plausible if the applicant doesn't recognize that the Suppression Pool temperature thermocouples are uncovered below 17.8 ft and doesn't recognize current Suppression Pool level is above the NPSH limit of 13.5 ft.

C. Correct – Note 2 associated with the step SP/L-4 of T-102 "Primary Containment Control" informs the procedure user that below 17.8 ft the RHR suction temperature should be used as a valid indicator of Suppression Pool temperature. Note 3 associated with step SP/L-5 informs the procedure user that 13.5 ft is the minimum level for NPSH and Vortex limits.

D. In-Correct but plausible if the applicant doesn't recognize that current Suppression Pool level is above the NPSH limit of 13.5 ft.

References:	LLOT1560 T-102 Bases "Primary Containment Control – Bases", Rev. 022, pp.79 & 80	Student Ref: required	Yes T-102

Learning Objective: IL3

Question source: Limerick Bank

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 16

Unit 1 plant conditions are as follows:

- A reactor startup is in progress.
- The unit is at 18% power.
- FWLC is in semi-automatic
- 'A' RFP is in service with a feed flow of 1.7 Mlb/hr
- 'B' RFP is in service with a feed flow of 0.6 Mlb/hr
- Subsequently, a reactor scram occurs on low reactor water level.

Which one of the following failures would be consistent with the above conditions?

A. Total steam flow signal fails low.

B. RFP 'A' suction pressure transmitter fails low.

C. Total feedwater flow demand signal fails low.

D. 6B Feedwater Heater feed flow transmitter fails low.

K&A # 295031 Reactor Low Water Level.

Importance Rating 4.3

QUESTION 16

K&A Statement: A1.13– Ability to operate and/or monitor the **Reactor Water Level Control** as it applies to Reactor Low Water Level.

Justification:

A. In-Correct but plausible since total steam flow is not an input when FWLC is in single-element control (i.e., below 21% power).

B. In-Correct but plausible since the suction pressure is one of the inputs to the "Calculated" feedwater flow signal. However, this signal is not used as a control input to the FWLC system

C. Correct –Total feedwater flow demand signal is compared to actual feedwater flow. If the signal fails low, feedwater flow will be reduced resulting in a low RPV level.

D. In-Correct but plausible since 6B feed heater flow transmitter failing low will generate an open signal for the 'B' RFP minimum flow valve. However, the 'A' RFP will increase speed automatically to control level at setpoint. Feedwater flow is not an input when FWLC is in single-element control.

References: LLOT0550, Rev. 018 Student Ref. required No

Learning Objective: Obj 10

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 17

Unit 1 is at 100% power with all systems normal

- I&C are calibrating the "B" RPS HI Drywell Pressure Switches
- Four RPS Solenoid white status lights are lit at PNL C603

Subsequently, the following alarm is received at 120 D11: 1A RPS & UPS DISTR PNL TROUBLE

- The [BOP] reports RECW Isolation
- The [CRO] reports he has completed applicable portions of OT-117, RPS FAILURES
- You observe various process Rad Monitor Trips and Alarms

Current plant conditions are as follows:

- Unit 1 is 100% power
- RPV pressure is 1038 psig
- RPV level is 32"

Considering the above conditions:

(1) From the following, select the appropriate procedure to implement next, and
(2) Select the appropriate basis.

	(1)	(2)
A.	ON-117 Loss of TECW	Loss of 1AY160 causes loss of TECW
B.	ON-104 Control Rod Problems	CRD Flow Controller has lost power
C.	OT-100 Reactor Low Level	Failure of Feed Pump Speed Control
D.	OT-214 Manual Initiation of ARI	Unit 1 has experienced an ATWS

K&A #	295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
Importance Rating	4.1

QUESTION 17

K&A Statement: K2.03-Knowledge of the interrelations between SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown and ARI/RPT/ATWS

Justification:

A. In-Correct The I&C surveillance has caused a half scram on Channel B (only 4 white lights lit). Per E-1AY160 there has been a loss of 1A RPS UPS power. There should have been a FULL SCRAM. Plausible if applicant does not associate response to loss of 1AY160 with SCRAM on "A" side and/or does not recognize that ATWS takes precedence over TECW.

B. In-Correct – Loss of 1AY160 will not affect CRD Flow Controllers (E-1AY160 Section 1.0). Plausible if some other power supply is assumed lost or if associates with RECW/RWCU isolations.

C. In-Correct but plausible if the applicant does not know that Feedwater has it's own, separate UPS.

D. Correct The I&C surveillance has caused a half scram on Channel B (only 4 white lights lit). Per E-1AY160 there has been a loss of 1A RPS UPS power. There should have been a FULL SCRAM. With the reactor at 100% power and with the operator completing "applicable portions of OT-117", the Mode Switch should be in "SHUTDOWN" and actions of T-101 in progress. With NO scram functions working, the next appropriate action (directed in T-101) is to manually initiate ARI.

References: E-1AY160 page 1
OT-117 page 3
LGSOPS0071 pages 40 and 41

Student Ref: required No

Learning Objective: N/A

Question source: Limerick Bank

Question History: None

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

10CFR 55.41(5) X

Comments: Created/Modified by: Johnson
Reviewed by: Presby

QUESTION 21

Plant conditions are as follows:

- 85% power
- "C' Inboard MSIV has inadvertently closed and RPV pressure peaked at 1060 psig.
- OT-102, High Reactor Pressure has been entered.

Which one of the following is the basis for reducing reactor power to less than 75% for the failed closed MSIV?

A. Ensures reactor power does not increase to greater than 100%.

B. Restore the margin between actual steam flow and the Group 1 isolation trip point.

C. Ensures reactor pressure transient will not result in SRV actuation.

·D. Restore the balanced steam admission characteristics on the HP turbine.

K&A #	295015 High Reactor Pressure
Importance Rating	3.8

QUESTION 21

K&A Statement: K2.02 – Knowledge of the interrelations between High Reactor Pressure and reactor power.

Justification:

A. In-Correct but plausible since the void collapse would result in a power increase.

B. Correct – The basis discussion for step 3.4 of OT-102 "High Reactor Pressure" states this concern as the reason for reducing power to 75%.

C. In-Correct but plausible if candidate is unsure of pressure setpoint where SRVs actuate.

D. In-Correct but plausible since this is the basis for power reduction in the event of a TCV or Stop Valve closure.

References: LLOT1540, Rev. 09
OT-102 Bases, "Reactor High Pressure – Bases", pg. 4, Rev. 17

Student Ref. required No

Learning Objective: IL5

Question source: Bank (Limerick) Modified distractor "c"

Question History: NRC-05, Oyster Crk. Cert - 04

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 22

Unit 2 is at 50% power with 'A' and 'C' RFPs in service when the following indications are received in the MCR:

- 207 D-4, FWLCS FAILURE is lit
- 202 C-2, 2C RFPT TRIP is lit

Investigation reveals that power has been lost to both AF100 buses. Currently, Reactor water level is +24 inches and decreasing.

Which one of the following choices correctly identifies the mode for Reactor Feedwater Control and the response of the Reactor Recirculation system?

A. Feedwater Control via 'A' RFP Manual Speed Controller AND Reactor Recirculation Pump speed runs back to 42%.

B. Feedwater Control via 'A' RFP M/A Station in Manual mode AND Reactor Recirculation Pump speed runs back to 42%.

C. Feedwater Control via 'A' RFP Manual Speed Controller AND Reactor Recirculation Pump speed runs back to 28%

D. Feedwater Control via 'A' RFP M/A Station in Manual mode AND Reactor Recirculation Pump speed runs back to 28%

K&A # 295009
Importance Rating 3.9

QUESTION 22

K&A Statement: A1.01 – Ability to operate and/or monitor Reactor Feedwater as it applies to LOW REACTOR WATER LEVEL.

Justification:

A. In-Correct but plausible since on a loss of the AF100 buses the FWLCS will transfer to the Manual Speed Controller. However, rather than RRP runback to 42% speed which would be expected for any RFP flow less than 1.8 Mlb/hr with RPV level less than +27.5 inches, a RRP runback to 28% speed should have occurred due to the loss of the AF100 buses.

B. In-Correct but plausible if the applicant doesn't know that control has transferred to the MSC on failure of the AF100 buses. Also the applicant may believe that only the conditions for a runback to 42% RRP speed is required as a result of the one feed pump tripping (i.e., flow less than 1.88 Mlb/hr).

C. Correct – As the result of loss of both AF100 buses RFP control transfers to the Manual Speed Controller (MSC) because the analog output from the turbine governor would be 0 amps which would indicate a loss of control signal and result in a transfer to MSC. Also, as a result of a loss of both AF100 buses, the RRP should have runback to 28% speed.

D. In-Correct but plausible if the applicant doesn't know that control is transferred to the MSC on failure of the AF100 buses.

References: OT-100, "Low Reactor Level"
LLOT0550, Rev. 017
LLOT0540, Rev.024
S06.1.HU/2, "Responding to Alarms and Selected Events at the Feedwater Level Control System Operator Station", Rev. 4

Student Ref. required No

Learning Objective: Obj. 10 (LLOT0550)

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 23

Unit 1 plant conditions are as follows:

- A suction line break has occurred on RRP 'A'.
- Core Spray is unavailable.
- Reactor vessel level is being maintained using both loops of RHR in LPCI mode.

The current conditions are as follows:

- RPV level is +22 inches and slowly rising.
- RPV pressure is 230 psig and stable.
- Suppression Pool temperature is 103°F and rising.

Step SP/T-5 states,

"**IF** NOT required for core cooling,

THEN operate 2 loops of Supp Pool Cooling."

For the above conditions, which one of the actions below meets the intent of Step SP/T-5?

A. Align 'B' Loop to Suppression Pool Cooling. If RPV level approaches +12.5", then realign 'B' Loop to LPCI mode and inject to the vessel.

B. Align Loops 'A' and 'B' to Suppression Pool Cooling. If RPV level approaches +12.5", then realign Loops 'A' and 'B' Loop to LPCI mode and inject to the vessel.

C. Align 'B' Loop to Suppression Pool Cooling. If RPV approaches TAF, then realign 'B' Loop to LPCI mode and inject to the vessel.

D. Align Loops 'A' and 'B' to Suppression Pool Cooling. If RPV level approaches TAF, then realign 'B' Loop to LPCI mode and inject to the vessel.

K&A # 295013 High Suppression Pool Temperature

Importance Rating 3.6

QUESTION 23

K&A Statement: K2.01 – Knowledge of the interrelations between High Suppression Pool Temperature and Suppression Pool Cooling.

Justification:

A. In-Correct but plausible since the basis for step RC/L-4 of T-101 states that maintaining reactor level between +12.5" and +54" provides **assurance** of adequate core cooling. However, the opening discussion of the basis document for T-101 states, *"RPV level control actions establish adequate core cooling by maintaining the core submerged".* Therefore maintaining the level greater than +12.5 inches would be overly conservative on level control and would severely limit the use of RHR for Suppression Pool Cooling.

B. In-Correct but plausible since the basis for step RC/L-4 of T-101 states that maintaining reactor level between +12.5" and +54" provides assurance of adequate core cooling. However, the opening discussion of the basis document for T-101 states, *"RPV level control actions establish adequate core cooling by maintaining the core submerged".* In addition, swapping both loops from LPCI mode would mean periods of no injection to the reactor vessel with a known leak. A failure during realignment back to LPCI could result in the inability to inject to the core when needed.

C. Correct – The opening discussion of the basis document for T-101 states, *"RPV level control actions establish adequate core cooling by maintaining the core submerged".* In addition, the basis for step SP/T-5 to T-102 states, "*Maintaining adequate core cooling takes precedence over maintaining suppression pool temperature below 95°F since catastrophic failure of the primary containment is not expected to occur at this temperature…Therefore, only if continuous operation of* ***an RHR pump in an RPV injection mode is not required*** *to assure adequate core cooling is it permissible to use that pump for suppression pool cooling. Step SP/T-5 does, however, permit* ***alternating*** *the use of RHR pumps between the RPV injection mode and the suppression pool cooling mode* ***as the need for each occurs*** *and so long as adequate core cooling is maintained.*" As long as the core is maintained covered, the intent of Step SP/T-5 is meet by having one loop in Suppression Pool Cooling and the other injecting to the vessel.

D. In-Correct but plausible since maintaining the core covered provides adequate core cooling. However, per the discussion associated with step SP/T-5 of T-102 (See discussion in Choice C.) the bases allows alternating of individual pumps, but not realignment of both pumps simultaneously. Swapping both loops from LPCI mode would mean periods of no injection to the reactor vessel with a known leak. A failure during realignment back to LPCI could result in the inability to inject to the core when needed. A failure during realignment back to LPCI could result in the inability to inject to the core when needed.

References: LLOT1560, Rev. 12
T-102 Bases, Rev. 022

Student Ref. required Yes
T-101
T-102

Learning Objective:	IL5 (LLOT1560)
Question source:	New
Question History:	
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis: X
10CFR	55.41 X
Comments:	Created/Modified by: M. Riches Reviewed by: P. Presby

QUESTION 24

Unit 1 plant conditions are as follows:

- SCRAM signal due to steam leak in drywell.
- Drywell pressure is 3.0 psig
- Reactor power is 10%
- RPV water level is 35 inches
- SRVs are all closed
- Main Turbine is on line
- Suppression Pool temperature is 97°F

Which one of the following RPV level bands should be issued based on the above conditions?

A. +12.5 to +54 inches

B. -186 to +54 inches

C. -100 to -60 inches

D. -186 to +35 inches

K&A # 295015 Incomplete SCRAM
Importance Rating 3.8

QUESTION 24

K&A Statement: G2.4.9 – Knowledge of low power/shutdown implications in accident mitigation strategies, as it relates to Incomplete SCRAM.

Justification:

A. In-Correct but plausible if the applicant doesn't recognize that they must exit RC/L leg of RPV Control.

B. In-Correct but plausible since a subsequent step (LQ-16) directs lowering level between this level band.

C. Correct – Step LQ-5 of T-117 "Level/Power Control" directs lowering RPV level below -50".

D. In-Correct but plausible since a subsequent step (LQ-17) directs lowering level between -186" and level to which it was previously lowered.

References:	LLOT1560, Rev. 12 T-117 Bases, "Level/Power Control – Bases", pg. 7, Rev. 12	Student Ref. required	Yes T-101 T-102 T-117

Learning Objective: IL6

Question source: Limerick Bank

Question History: NRC-05, Oyster Crk. Cert - 04

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 25

Unit 2 plant conditions are as follows:

- 'B' Containment H_2 Recombiner is in operation for post-maintenance testing.
- Wide range level transmitter LT-042-2N081D has failed such that the associated Main Control board indicator shows -42".
- Subsequently, LT-042-2N081C fails offscale low.

Which one of the following identifies the current status of these valves associated with the Drywell Unit Coolers and Hydrogen Recombiners?

A. 'A' Drywell Chilled Water Return – OPEN
'B' Drywell Chilled Water Supply – OPEN
Drywell Supply to 'B' Recombiner – OPEN

B. 'A' Drywell Chilled Water Return – OPEN
'B' Drywell Chilled Water Supply – OPEN
Drywell Supply to 'B' Recombiner – CLOSED

C. 'A' Drywell Chilled Water Return – CLOSED
'B' Drywell Chilled Water Return – CLOSED
Drywell Supply to 'B' Recombiner – OPEN

D. 'A' Drywell Chilled Water Supply – CLOSED
'B' Drywell Chilled Water Supply – CLOSED
Drywell Supply to 'B' Recombiner – CLOSED

K&A #	295020 Inadvertent Containment Isolation
Importance Rating	3.1

QUESTION 25

K&A Statement: K2.03 – Knowledge of the interrelations between Inadvertent Containment Isolation and **Drywell/containment ventilation/cooling**.

Justification:

A. In-Correct but plausible if the applicant believes that both the Hydrogen Recombiners and cooling water to the Drywell Cooler Units isolate on a Level 1 (-129").

B. Correct – Cooling to the Drywell Cooler Units isolates on a Level 1 (-129") signal on actuation of either trip channels A & B or C & D. Level 1 has only been reached on trip channel C (LT-042-2N081C). Drywell Supply to Hydrogen Recombiner 'B' isolates on Level 2 (-38") signal on trip channel 'D'.

C. In-Correct but plausible if the applicant believes cooling to the Drywell Cooler Units isolate on Level 2 and that Drywell Supply to Hydrogen Recombiner 'B' receives a close signal either when Level 1 is reached or a Level 2 is reached on trip channel 'C'.

D. In-Correct but plausible if the applicant believes that both systems isolate on Level 2 signal.

References: LLOT0180, Rev. 015
SSN Drawing # 058-01, Rev. 2
LGSOPS0042, Rev. 000
GP-8.1, Automatic Actuations by Isolation Signals, Rev.14
S58.1A, Placing Containment Hydrogen Recombiners in Ready Mode" Rev,. 008

Student Ref. required: No

Learning Objective: Obj 2, (LLOT0180)
IL-9 (LGSOPS0042)

Question source: Modified Bank (Limerick) — (Modified) Changed trip signal and systems affected

Question History: NRC-05, Oyster Crk. Cert - 04

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 26

Limerick Generating Station experienced a seismic event. Unit 2 was shutdown in OPCON 4 and did not experience any noticeable transients. Unit 1 was at 100% when the seismic event occurred. Current plant conditions on Unit 1 are as follows:

- Reactor scrammed on low level in the reactor vessel.
- Reactor level dropped to -195" before a cooling source could be aligned to the core.
- Reactor level is currently at -175" and stable.
- Reactor pressure is 600 psig and stable.
- Div 1 and Div 2 125 VDC have been lost.
- RCIC is inoperable.
- The HPCI equipment room is filled with steam.
- HV-55-1F002 "HPCI Steam Line Inboard Isolation Valve" failed to isolate.

Conditions in the HPCI Equipment room are as follows:

- Room temperature is 230°F and rising.
- Radiation levels are 0.8 mr/hr and rising.

Given the above conditions and assuming no operator actions, which one of the following is correct concerning the radioactive release from the Unit 1 HPCI Equipment Room?

A. Steam and radiation levels will quickly stabilize in the HPCI Equipment Room. The release was terminated by the automatic isolation of the HPCI Steam Supply Line on high room temperature.

B. Steam and radiation levels will continue to rise in HPCI Equipment Room, but are confined to the room. The RE HVAC Steam Flooding Isolation Dampers associated with HPCI Equipment Room closed on high d/P in the HPCI Equipment Room exhaust duct.

C. Steam and radiation levels will continue to rise in HPCI Equipment Room, but are confined to the room. The RE HVAC Steam Flooding Isolation Dampers associated with HPCI Equipment Room closed on loss of power.

D. Steam and radiation levels will continue to rise and be transported to the RE HVAC system, but will be confined within the RE HVAC system and routed through a filtered release to the South Stack. The RE HVAC Supply and Exhaust dampers closed on a loss of power.

K&A # 295032 High Secondary Containment Temperature

Importance Rating 3.6

QUESTION 26

K&A Statement: K1.02 – Knowledge of the operational implications of **Radiation releases** as it applies to HIGH SECONDARY CONTAINMENT TEMPERATURE.

Justification:

A. In-Correct but plausible since normally HV-55-F003 "HPCI Steam Line Outboard Isolation Valve" would close on high room temperature. However, the signal for closing HV-55-F003 is powered from Div 2.

B. In-Correct but plausible since the Steam Flooding Isolation Dampers normally close on a high d/P signal. However, with the loss of Div 1 and Div 2 the solenoids to the actuating arm will be de-energized. These solenoids need to energize to release the actuating arm for the dampers.

C. In-Correct but plausible since most all of the dampers associated with the RE HVAC system close on a loss of DC power. However, the solenoids associated with the Steam Flooding Isolation Dampers need to energize to release the actuating arm for the dampers.

D. Correct – Conditions within the HPCI equipment room will continue to worsen. HV-155-F003 receives its isolation signal from Div 2, which means with HV-55-1F002 failed open, an unisolable leak path exists from the ruptured HPCI steam supply line into the HPCI Equipment Room. The RE HVAC system dampers are supplied by 125V DC from Div 1 and Div 2 and on a loss of power the exhaust and supply dampers would close. This would confine the steam and radiation within the RE HVAC and eventually would reach the South Stack through the filtered flowpath provided by the SGTS.

References: LLOT0200, Rev. 018
LLOT0340, Rev. 25

Student Ref. required No

Learning Objective: Obj 7, (LLOT0200)
Obj. 14 (LLOT0340)

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 28

The following describes the initial Unit 2 plant conditions:

- Reactor is at 100% power.
- SRV testing is in progress.
- RHR Loop 'A' is running and aligned in Suppression Pool Cooling mode with HV-C-51-2F048A, "2A RHR Htx Shell Side Bypass Valve" closed.
- RHR 'A' discharge pressure is 215 psig.
- RHR Loop 'B' is aligned for LPCI injection.

Subsequently, a LOCA occurs on Unit 2 and the following conditions exist:

- Reactor pressure is 500 psig and dropping
- Reactor level is minus (-) 100"
- Drywell pressure is 15.3 psig

Given these current conditions and NO operator actions taken to align RHR, which one of the following describes the current position of the following RHR Loop 'A' valves:

- HV-C-51-2F048A, 2A RHR Htx Shell Side Bypass Valve
- HV-51-2F024A, 2A RHR Pp Full Flow Test Return Valve
- HV-51-2F007A, 2A RHR Pp Min Flow Valve
- HV-51-2F017A, 2A RHR LPCI Injection PCIV

A. F048A - **Open**
F024A - **Closed**
F007A - **Open**
F017A - **Closed**

B. F048A - **Closed**
F024A - **Closed**
F007A - **Open**
F017A - **Closed**

C. F048A - **Open**
F024A - **Closed**
F007A - **Closed**
F017A - **Open**

D. F048A - **Closed**
F024A - **Open**
F007A - **Closed**
F017A - **Closed**

K&A # 203000 RHR/LPCI: Injection Mode

Importance Rating 4.2

QUESTION 28

K&A Statement: K4.01 – Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for **Automatic system initiation/ injection**.

Justification:

A. In-Correct but plausible since this would be the status of these valves if just high drywell pressure was an initiation signal.

B. In-Correct but plausible if the applicant believes high drywell pressure was an initiation signal and that the signal only repositioned valves that rob flow from LPCI flowpath. Following this premise, the heat exchanger bypass and LPCI injection valve would not reposition until conditions were met for LPCI injection valve opening (i.e., 74 psid across injection valve).

C. In-Correct but plausible if the applicant believes that high drywell pressure was an initiation signal and that the LPCI Injection valve opens immediately on an initiation signal.

D. Correct – LPCI initiation needs either reactor level less than -129" **OR** Drywell pressure greater than 1.68 psig along with reactor pressure less than 455 psig. At the current conditions, none of the valves would have repositioned yet.

References: LLOT0370, Rev. 017 Student Ref. required No

Learning Objective: Obj. 6, Obj. 8

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 29

Given the following Unit 2 plant conditions:

- RHR loop "B" is operating in the Shutdown Cooling (SDC) Mode
- Reactor coolant temperature is at 290° F and slowly increasing
- RPV pressure is 45 psig and slowly increasing

The following alarms are received:

- 222 D22 A2 201 D22 Bus breaker trip
- 222 D22 A1 D22 bus diff/overcurrent lockout
- 222 D22 B1 D22 safeguard bus undervoltage
- 222 D22 B3 D224 load center xfmr breaker trip
- 222 D22 C4 D22 Diesel Running

SELECT the statement that describes the automatic response of the Shutdown Cooling Suction Isolation Inboard and Outboard Valves (HV51-2F009 and HV51-2F008), if/when reactor pressure exceeds 75 psig.

Assume no operator action is taken.

A. HV-51-2F009 will remain in the current position. HV-51-2F008 will close.

B. HV-51-2F009 and HV-51-2F008 will remain in their current positions.

C. HV-51-2F009 and HV-51-2F008 will both close.

D. HV-51-2F008 will remain in its current position. HV-51-2F009 will close.

K&A #	205000 Shutdown Cooling
Importance Rating	3.3

QUESTION 29

K&A Statement: K6.05 - Knowledge of the effect that a loss or malfunction of AC electrical power will have on the SHUTDOWN COOLING system

Justification:

A. In-Correct but plausible if the applicant does not associate the loss of Bus 22 to a failure of F008; rather than F009.

B. In-Correct but plausible if applicant thinks both valves are powered from D22.

C. In-Correct but plausible if applicant thinks neither valve is powered from D22 or if applicant thinks the EDG has powered the bus.

D. Correct – Alarms indicate lockout conditions on Bus D22 with the bus de-energized. 2F008 is powered from D22 bus and will not operate. RPV pressure will increase from decay heat input with loss of Loop B RHR cooling. 2F009 will isolate on pressure interlock at 75 psig RPV pressure.

References: LLOT0370 pages 25 Student Ref: required No

Learning Objective:

- LLOT0370 #14
- S55.1.7.B, Defeating The Rhr Shutdown Cooling Auto Isolation, Rev 007, Steps 4.2.3, 4.2.4 (power supplies)

Question source: Bank (Perry) Chgd pwr supply to fit Limerick

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 30

Unit 2 is operating normally at 100% power. The following sequence of events occurs.

- Rupture in the feedwater pump suction line results in loss of all reactor feedwater pumps on low suction pressure.
- Shortly thereafter, the reactor scrams on low level.
- One minute after the scram, a DC distribution panel low voltage alarm is received (2PPB1/2PPB3 125 VDC DIST PANELS UNDERVOLTAGE – Window G-4 on AR-222).
- An operator in the vicinity of Distribution Panel 2BD102 reports an acrid odor and that Panel 2BD102 de-energized.
- The current time is 4 minutes after the scram.

Given the situation and assuming no operator action, which of the following correctly identifies expected plant conditions?

A. HPCI turbine is rotating, RPV level is minus (-) 52 and slowly lowering

B. HPCI turbine is rotating, RPV level is plus (+) 25 and slowly lowering

C. HPCI turbine is tripped, RPV level is minus (-) 52 and slowly lowering

D. HPCI turbine is tripped, RPV level is plus (+) 25 and slowly lowering

K&A # 262000 HPCI
Importance Rating 2.8

QUESTION 30

K&A Statement: K2.03 - Knowledge of the electrical power supplies to Initiation logic

Justification:

A. Correct – Div II provides HPCI initiation logic power. HPCI will have started post trip as expected on level decrease below Level 2 (-38.5 inches) based on a low level scram from full power at Level 3 (+12.5 inches) with loss of both reactor feedwater pumps. The subsequent loss of DC power to 2BD102, Division II distribution Panel will disable automatic HPCI initiation on low level, render HPCI turbine trips inoperable and remove control power from the HPCI flow controller. The loss of control power will result in a zero percent speed demand, throttling down on the HPCI turbine steam control valve. Turbine speed will decrease to the low speed limit (750 rpm). Level would be expected low and decreasing slowly. RCIC would be supplying flow to the vessel, but not at a sufficient rate to overcome steam loss for decay heat removal.

B. In-Correct but plausible. Level would not have been restored to +25 inches with RCIC, given the initiating event because RCIC flow cannot leep up with the immediate post trip decay heat steaming rate.

C. In- Correct but plausible. HPCI would have auto-started immediately after the scram at Level 2 (-38 inches). The subsequent loss of Div II power will disable auto and manual turbine trips.

D. In-Correct but plausible. Level would not have restored to +25 inches with RCIC, given the initiating event. HPCI would have auto started immediately after the scram at Level 2 (-38.5 inches). The subsequent loss of Div II DC power will disable auto and manual turbine trips.

References: ARC-MCR-222 G4; LLOT0340 rev. 25; LLOT0690, rev.12 Student Ref: required No

Learning Objective: LLOT0690 Obj. 6c; LLOT0340 obj. 14a

Question source: Modified Bank (Limerick)

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 31

Unit 1 is at 100% power when the reactor scrams on low reactor level. A HPCI auto initiation signal is received, however no flow is observed on the system. The following indications are observed in the MCR:

- HPCI flow controller in AUTO with a setpoint of 5600 gpm.
- All auto initiation valves have repositioned to their expected positions.
- HV-56-1F012, Turbine Stop Valve is Open.
- FV-56-111, HPCI Turbine Control Valve is Closed.

Which of the following explains the HPCI system response?

A. Speed Feedback to Turbine Speed Governor Controller fails low.

B. Ramp Generator output signal fails low.

C. Auxiliary Oil Pump failed to start.

D. Trip drain valve is aligned to the oil drain.

K&A # 206000 High Pressure Coolant Injection

Importance Rating 3.4

QUESTION 31

K&A Statement: K4.11 – Knowledge of HIGH PRESSURE COOLANT INJECTION design feature(s) and/or interlocks which provide for **Turbine Speed Control**.

Justification:

A. In-Correct but plausible since if the applicant did not understand how a failure of this would effect the Turbine Speed Governor Controller. However, a low failure of the speed feedback signal would result in generating in an open signal to the HPCI Turbine Control Valve (TCV) and speed and flow to increase.

B. Correct – The ramp generator overrides the flow controller during turbine startup to allow a controlled rate of acceleration. Once the ramp generator output exceeds the signal from the flow controller, the controller takes over and maintains control until the ramp generator is reset. The ramp generator is reset whenever the turbine stop valve is fully closed. With the ramp generator signal failed low, its output will never exceed the flow controller and will not reset.

C. In-Correct but plausible since failure of the Auxiliary Oil pump would lead to no control oil getting to the TCV and the Turbine Stop Valve (TSV). However, since the initial conditions listed in the question state that the Turbine Stop Valve is open then this couldn't be the cause of the problem.

D. In-Correct but plausible since the trip drain valve aligned to the oil drain would prevent control oil pressure from increasing. It would prevent the TCV and TSV from opening. However, since the initial conditions listed in the question state that the Turbine Stop Valve is open then this couldn't be the cause of the problem.

References: LLOT0340, Rev. 025 Student Ref. required No

Learning Objective: Obj. 6b, Obj 10

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 33

Unit 1 plant conditions are as follows:

- D13 bus is locked out.
- Reactor level is -140".
- Reactor pressure is 370 psig

Which one of the following describes the status of "1B" Loop Core Spray?

A. Injecting approximately 3,000 gpm.

B. Injecting approximately 6,000 gpm.

C. Not injecting, only one Core Spray pump is running on min flow.

D. Not injecting, both Core Spray pumps are running on min flow.

K&A # 209001 Low Pressure Core Spray

Importance Rating 3.7

QUESTION 33

K&A Statement: A3.04 – Ability to monitor automatic operations of the Low Pressure Core Spray controls including **system flow**.

Justification:

A. In-Correct but plausible since this would be the approximate flow if either of the "1B" Loop pumps received power from the D13 bus and if applicant confuses the setpoint for the injection valves opening (455 psig) with the pump shutoff head (~330 psig), then design flow could be reached at reactor pressure equal to 370 psig.

B. In-Correct but plausible since this would be the approximate flow if both the "1B" Loop pumps were injecting into the vessel and if applicant confuses the setpoint for the injection valves opening (455 psig) with the shutoff head (~330 psig) then design flow could be reached at reactor pressure equal to 370 psig.

C. In-Correct but plausible since this would be the status of the pumps if either of the "1B" Loop pumps received power from the D13 bus.

D. Correct-Neither the '1B' or '1D' Core Spray pumps would be effected by the loss of D13 bus. However, the current reactor pressure exceeds the shutoff head (~330 psig) of the Core Spray pumps.

References: LLOT0350, Rev. 015 Student Ref. required No

Learning Objective: Obj 5, Obj 9

Question source: Modified Bank (Limerick) (Modified) Changed affected Core Spray loop ("1B"), changed reactor pressure from 230 psig to 370 psig, and removed any reference to a specific pumps within distractors.

Question History: NRC-05, Oyster Crk. Cert - 04

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 34

Unit 2 has scrammed but only 50% of the control rods are at position 00. Post-scram plant conditions are as follows:

- Reactor power is 15%.
- MSIVs are closed with SRVs cycling to control reactor pressure.

All three SLC pumps have been manually started from the Control Room with the following system indications:

- SLC Tank level is 3,800 gallons
- SLC Pump discharge pressure are:
  'A' = 1100 psig 'B' = 1195 psig 'C' = 1405 psig
- SQUIB READY status lights indicate as follows:
  'A' = OFF 'B' = OFF 'C' = OFF

Five minutes following manual initiation of SLC, annunciator MCR 108 I-2, STANDBY LIQUID TANK HI/LO LEVEL, is received. SLC Tank level is 3,585 gallons.

Given these conditions, which one of the following describes the SLC pumps that are injecting boron solution into the RPV?

	Pump A	Pump B	Pump C
A.	Injecting	Injecting	Not Injecting
B.	Injecting	Injecting	Injecting
C.	Not injecting	Not injecting	Injecting
D.	Not injecting	Injecting	Not injecting

K&A # 211000 Standby Liquid Control

Importance Rating 3.8

QUESTION 34

K&A Statement: K4.04 – Knowledge of STANDBY LIQUID CONTROL system design features and/or interlocks which for **indication of fault in explosive valve firing circuits.**

Justification:

A. *NOTE: For this event, 'A' SLC pump's squib valve opened but its discharge pressure is too low to inject into the reactor vessel and 'C' squib valve has lost continuity but the squib valve did not fire.*

In-Correct but plausible since the SQUIB READY status lights for squib valves 'A' and 'B' are out which is one possible indication that boron is injecting. However, 'A' SLC pump discharge pressure is below reactor pressure, so it is not injecting. For 'C' SLC pump although its continuity light is out, the high discharge pressure of 1405 psi would indicate that flow is through the discharge relief valve which lifts at 1400 psi. This would indicate that flow is not getting through 'C' SLC injection line.

B. In-Correct but plausible if the applicant used only the status of the SQUIB READY status lights being off as confirmation that all SLC pumps are injecting into the core. Also if the applicant did not recognize that with reactor pressure being controlled by SRVs that at a minimum SLC pressure must exceed the SRV lift setpoints (1170 to 1190 psig).

C. In-Correct but plausible since it is accurate for 'A' SLC pump given that reactor pressure is greater than 'A' SLC pump discharge pressure. Even though its continuity light is out, 'A' SLC pump would not be injecting. Also, if the applicant were confused about where reactor pressure would actually be with reactor pressure being controlled by SRVs than he could believe that 'B' SLC pump would not be injecting into the core as well. With only one pump injecting to the RPV, this would be consistent with the level change observed on the SLC tank (See discussion below.).

D. Correct - Only 'B' SLC pump is injecting into the core. In addition to the abnormal pressures with 'A' and 'C' discussed previously, confirmation that only one pump is injecting into the vessel is volume change of the SLC tank. The volume change over the five minute injection period is 215 gallons (3800 – 3585) and each pump's design flow rate is 43 gpm. This would then indicate that only one SLC pump can be injecting into the vessel (i.e. 43 gpm x 5 min = 215 gallons.)

References: LGSOPS0048, Rev. 000 Student Ref. required No

Learning Objective: EO4, IL2, IL6

Question source: New

Question History:

Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	 X
10CFR	55.41 X	
Comments:	Created/Modified by: M. Riches Reviewed by: P. Presby	

QUESTION 35

Unit 1 plant conditions are as follows:

- Reactor power is at 97% power coasting down for a refueling outage
- RPS SCRAM Functional testing is in progress
- Group “A1” and “A4” white lights are extinguished on panel 10C603
- Group “A2” and “A3” white lights are lit on panel 10C603

Group “B1” RPS solenoid power fuse blows.

WHICH ONE of the following identifies the status of Control Rods ten (10) seconds after the fuse blows, assuming no operator action?

A. No rods inserted

B. 45 rods inserted

C. 48 rods inserted

D. 93 rods inserted

K&A # 212000 RPS
Importance Rating 3.2

QUESTION 35

K&A Statement: A4.04 – Knowledge of the effect that a loss or malfunction of REACTOR PROTECTION SYSTEM will have on RPS logic channels.

Justification:

A. In-Correct RPS K14 relay is de-energized for Group "A1" and "A4" control rods based on white light indication. Blown fuse on Group "B1" circuit results in venting of air from scram pilot valves for Group 1 control rods (45 rods SCRAM). Plausible if applicant does not recall the RPS logic or impact of blown fuse.

B. Correct – In a coast-down all rods would be expected to be full out (position 48). RPS K14 relay is de-energized for Group "A1" and "A4" control rods based on white lights being extinguished. "A" side scram pilot valves have repositioned for control rod Groups 1 and 4. Blown fuse on "B1" results in de-energizing "B1"relay. This repositions the second set of scram pilot valves on Group 1 control rods and vents air from Group 1 scram valves. Venting air from the scram valves results in scramming of Group 1 control rods. There are 45 rods in Group 1.

C. In-Correct – RPS K14 relay is de-energized for Group "A1" and "A4" and has repositioned "A" scram pilot valves for Group 1 control rods. Blow fuse on "B1" de-energizes "B1" relay only and results in repositioning of "B1" scram pilot valves. Plausible if applicant does not understand that the blown fuse does not affect the "B4" control rod circuitry (which would result in scram of 48 rods).

D. In-Correct – RPS K14 relay is de-energized for Group "A1" and "A4" control rods and has repositioned "A" side scram pilot valves Blow fuse on "B1" de-energizes "B1" relay only (45 rods insert). Plausible if applicant thinks both Group 1 and 4 SCRAM.

References: LGSOPS0071, Dwg. 071-01a, 071-03a Student Ref: required N

Learning Objective: LGSOP0071 EO4

Question source: Limerick bank modified

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 37

IRM Channel 'D' failed upscale earlier in the shift during reactor startup on Unit 2. The channel was declared inoperable and all appropriate actions were taken. I&C has been informed and is developing a work package to troubleshoot and repair.

Currently, IRM 'B' is reading 30 on Range 4 when the operator selects Range 3 on the IRM 'B' Range Switch.

Which of the following describes the expected response and the system(s) affected by the above conditions?

A. "Retract Permit" is generated by bypassing of IRM Channel 'D' as the only response (i.e., no rod blocks or scram signals generated).

B. Control Rod block is generated by Reactor Manual Control System AND Half-scram generated by Reactor Protection System.

C. Control Rod block is generated by Reactor Manual Control System AND Reactor scram is generated by the Reactor Protection System.

D. Control rod block is generated by the Reactor Manual Control System as the only response (i.e., no half-scram or scram signals generated).

K&A # 215003 IRM
Importance Rating 3.6

QUESTION 37

K&A Statement: K3.02 – Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR system will have on **Reactor Manual Control**.

Justification:

A. In-Correct but plausible since this is true. However, it is not the only response to the event. Placing IRM 'B' to Range 3 would raise the value by a factor of 3.16 (3.16 x 30 = 94). The new value of 94 would exceed the upscale setpoint of 85/125 for IRM 'B' and a control rod block would be generated by RMCS.

B. In-Correct but plausible since a rod block would occur and if the applicant believes with one IRM inop (IRM 'D') and that IRM 'B' exceeded the scram setpoint (120/125) then a half-scram would be generated on RPS trip system 'B' . However, with IRM 'D' bypassed it is removed from the trip circuit and the new value of 94 would not reach the scram setpoint (120/125) for IRM 'B'. In addition, the two IRMs are on the same trip system.

C. In-Correct but plausible since a rod block would occur and if the applicant believes with one IRM inop (IRM 'D') and that IRM 'B' exceeded the scram setpoint (120/125) then a full -scram would be generated with a tripped detector one each RPS trip system the same . However, with IRM 'D' bypassed it is removed from the trip circuit and the new value of 94 would not reach the scram setpoint (120/125) for IRM 'B'. In addition, the two IRMs are on the same trip system.

D. Correct – IRM'D' is bypassed. Therefore no trips or blocks will be generated from IRM 'D". IRM 'B' will indicate 94 of 125 scale, which is greater than the 85/125 scale for a rod block signal generated by RMCS but less than the 120/125 scram signal generated by RPS.

References: LLOT0250, Rev. 012 Student Ref. required No

Learning Objective: Obj 10

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 38

Unit 2 is in a refueling outage with the following conditions established:

- Mode switch is in REFUEL
- Control rod 30-31 is at position 24; all others are at 00
- Fuel is being moved in the spent fuel pool
- Shorting links are installed

An I&C Technician, troubleshooting a problem on the "A" Source Range Monitor (SRM), moves the drawer Mode switch out of OPERATE

(1) Which of the following is the effect of the above conditions on SRMs?
(2) Which of the following is the effect of the above conditions on Control Rod 30-31?

	(1)	(2)
A.	SRM Downscale Alarm	No effect on Control Rod 30-31
B.	SRM Downscale Alarm	Control Rod 30-31 cannot be withdrawn
C.	SRM Upscale/Inop Alarm	Control Rod 30-31 cannot be withdrawn
D.	SRM Upscale/Inop Alarm	Control Rod 30-31 rapidly inserts on SCRAM signal

K&A # 215004 SRM
Importance Rating 3.3

QUESTION 38

K&A Statement: K3.02- Ability to monitor automatic operations of the Source Range Monitor system including annunciator and alarm conditions.

Justification:

A. In-Correct but plausible if the applicant does not understand the SRM out of OPERATE will result in an Upscale alarm.

B. In-Correct but plausible if the applicant does not understand the SRM out of OPRATE will result in an Upscale alarm. Downscale alarm does not cause a Rod Block.

C. Correct – With the shorting links installed, one SRM upscale will cause a Rod Block.

D. In-Correct but plausible if the applicant does not understand that the control logic requires 2 out of 4 SRM channels upscale to initiate a SCRAM

References: LLOT0240 pages 16 Student Ref: required No

Learning Objective: N/A

Question source: Bank (Limerick)

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 39

Unit 1 plant conditions are as follows:

- Reactor power is 100%
- APRM "1" is bypassed to support I/V LPRM data plotting.

Which one of the following describes the effect if the High Voltage Power Supply (HVPS) associated with APRM "2" chassis was inadvertently adjusted to 150 VDC?

A. Reactor scram.

B. Half-scram.

C. Upscale alarm on APRM "2".

D. Auto transfer APRM "2" to the alternate HVPS.

K&A # 215005 APRM/LPRM
Importance Rating 3.6

QUESTION 39
K&A Statement: A4.06 – Ability to manually operate and/or monitor in the control room: **Verification of proper functioning/operability**.

Justification:

A. In-Correct but plausible if the applicant believes the bypassed channel inserts an "inop" signal to the RPS logic (i.e. any two upscale or inop APRMs will generate a reactor scram.)

B. In-Correct but plausible if the applicant believes the "1" and "2" APRM channels feed the same RPS channel. "1" APRM feeds RPS channel "A"; "2" APRM feeds RPS channel "B".

C. Correct – Will result in an upscale alarm on "2" APRM (Ann. 108, C-3). LLOT0275 states that HVPS normally provides100 VDC to the LPRM detectors. The bypassed APRM changes the RPS "2 out of 4"logic to "2 out of 3" logic.

D. In-Correct but plausible since each APRM chassis has two HVPSs but there is not an automatic selection logic.

References: LLOT0275, Rev. 004 Student Ref. required No

Learning Objective: Obj 10, Obj 17

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 40

Unit 1 plant conditions are as follows:

- Reactor power is 85%
- Recirc Loop "A" flow, as indicated on FR-043-1R614 on Panel 10C602 is 39,200 gpm.
- Recirc Loop "B" flow, as indicated on FR-043-1R614 on Panel 10C602 is 38,600 gpm.

Subsequently, Reactor Recirc pump 'A' trips. The following Recirc Loop flows are observed after conditions stabilize:

- Recirc Loop "A" flow, as indicated on FR-043-1R614 on Panel 10C602 is 0 gpm.
- Recirc Loop "B" flow, as indicated on FR-043-1R614 on Panel 10C602 is 39,100 gpm.

Assuming 100%Total Recirc Drive Flow equals 88,000 gpm, which one of the following identifies the APRM flow-biased scram setpoints **before** AND **after** the Recirc pump trip?

(Assume setpoint values have been rounded to nearest whole number).

	BEFORE	AFTER
A.	117%	87%
B.	121%	87%
C.	117%	92%
D.	121%	92%

K&A # 215005 APRM/LPRM
Importance Rating 3.7

QUESTION 40

K&A Statement: K4.07 – Knowledge of Average Power Range Monitor/Local Power Range Monitor system design feature(s) and/or interlocks which provide for: **Flow-biased trip setpoints**.

Justification:

A. Correct – Before: APRM Flow-biased setpoint is clamped at 116.6% (~117%). After: For single-loop operation, the setpoint uses the following algorithm
$$0.66\,[(39{,}100/88{,}000)100 - 7.6] + 62.8 = \sim 87\%$$

B. In-Correct but plausible since using the correct algorithms for two-loop and single-loop operation will yield these results.

C. In-Correct but plausible if the applicant uses the incorrect ΔW (0.0% –vs- 7.6%) to calculate the setpoint for single loop operation.

D. In-Correct but plausible if the applicant uses the incorrect ΔW (0.0% –vs- 7.6%) to calculate the setpoint for single loop operation.

References: LLOT0275, Rev. 004 Student Ref. required No

Learning Objective: Obj 14

Question source: Limerick Bank Modified lesson plan question

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 42

Unit 2 plant conditions are as follows:

- Reactor is at 100% power
- RCIC monthly pump surveillance is in progress
- Then a loss of 125 VDC Bus C power occurs

Subsequently, you receive the following alarms on Panel 116:

- RCIC OUT OF SERVICE
- RCIC TURB EXH DIAPHRAGM RUPTURED

You observe RCIC room fire detectors in alarm.

WHICH ONE of the following is correct operator response? (assume all systems operate as designed)

A. Enter Fire Safe Shutdown Guide for the RCIC room.

B. Reset Turbine Trip valve when turbine speed decreases to 0.

C. Enter T-103 since there is a primary system discharging into Secondary Containment.

D. Close the steam line inboard (F007) valve; there is an incomplete isolation of RCIC with a valid isolation signal present.

K&A # 217000 RCIC
Importance Rating 3.3

QUESTION 42

K&A Statement: A2.14 – Ability to (a) predict the impacts of Rupture disc failure: Exhaust Diaphram on the REACTOR CORE ISOLATION COOLING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations.

Justification:

A. In-Correct Entry into the Fire Safe Shutdown Guides is only after a fire is confirmed by the Fire Brigade. Plausible if applicant does NOT associate the rupture disk with the fire alarm.

B. In- Correct but plausible since a turbine trip will occur from Div 1 However there is no direction to reset the trip at this point.

C. In-Correct but plausible. Entry into T-103 is required due to initiation of steam leak detection (Div 1), however there is NO primary system discharging into secondary containment since the steam leak has been isolated by the closure of F008 and F076 (as well as Turbine Trip valve).

D. Correct – On high equipment room temperature, Div 1 and Div 3 isolation signals are generated. The RCIC OUT OF SERVICE alarm should indicate an isolation has initiated. Div 1 closes F008 and F076; Div 3 closes F007. Since Div 3 power is lost, F007 failed to close automatically. Appropriate operator action, per "Roles and Responsibilities" is to complete the RCIC system isolation by manually closing F007 from the control board. F007 is an AC powered motor operated valve.

References: LLOT0380 pg. 11, 12, 13, 15, 16, 17
OP-AA-101-111, Roles And Responsibilities Of On-Shift Personnel, Section 4.6.2.5

Student Ref: required N

Learning Objective: N/A

Question source: Bank (Limerick)

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 43

Unit 2 was operating at 100% power when Div 3 125 VDC power was lost. Subsequently, a small-break LOCA (SBLOCA) occurred with reactor level dropping below -130".

Two minutes have elapsed since the above events occurred. Current plant conditions are as follows:

- Reactor vessel level is -148 inches.
- HPCI is injecting with discharge pressure at 1090 psig.
- Drywell pressure is 2.5 psig.
- "A" Core Spray pump discharge pressure is 140 psig
- "B" Core Spray pump discharge pressure is 285 psig.
- "D" Core Spray pump discharge pressure is 300 psig.
- "A" Loop RHR/LPCI discharge pressure is 115 psig.
- "B", "C", and "D" Loop RHR/LPCI discharge pressures are 0 psig.
- ADS "NORM-INHIBIT" switches are in the "NORM" position.

Which one of the following describes the response of the ADS SRVs to the above conditions AND the effect on the initiation logic?

A. OPEN – Initiation logic initiated on input from Drywell pressure.

B. OPEN – Initiation logic only needs input from either Div 1 or Div 3.

C. CLOSED – Initiation logic needs input on the availability of low pressure ECCS pumps.

D. CLOSED – Initiation logic needs input on the status of the High Drywell Pressure Bypass Timer.

K&A #	218000 Automatic Depressurization System
Importance Rating	3.9

QUESTION 43

K&A Statement: K6.01 – Knowledge of the effect that a loss or malfunction of **RHR/LPCI system pressure** will have on the Automatic Depressurization System.

Justification:

A. In-Correct but plausible if the applicant does not know that the initiation logic hasn't been satisfied (i.e., low head pumps available) to open the ADS valves.

B. In-Correct but plausible since the system does have independent and redundant logic trains that can initiate ADS automatically. However, the ADS valves would be closed given the lack of input to the initiation logic concerning availability of low pressure ECCS.

C. Correct – Div 3 initiation logic is inoperable with the loss of Div 3 125 VDC. Therefore, only the inputs to the Div 1 initiation logic will result in automatic initiation of ADS. While the initiation logic inputs for High Drywell Pressure (+1.68 psig) Low Reactor Water Level (-129") and 105 second timer (i.e., two minutes have elapsed since going below Level 1) are met; the inputs for low pressure pump availability are not. Div 1 initiation logic looks at the following combination of low pressure ECCS pumps:

Channel A	Channel E	Setting
RHR Pumps "A" or "C' running	RHR Pumps "A" or "C" running	125 psig
Core Spray Pump "A" running	Core Spray Pump "C" running	145 psig

"A" Loop Core Spray, which consists of "A" and "C" Core Spray pumps, only indicates 140 psig. "A" Loop RHR/LPCI, which consists of "A" and "C" RHR pumps only indicates 115 psig. With these low pressures on Div 1 logic and the loss of Div 3 logic, the ADS valves will not receive an open signal.

D. In-Correct but plausible since the High Drywell Pressure Bypass Timer initiates when RPV level of -129" is reached. Whether this timer times out or not, conditions still have not been met due to the lack of a valid signal concerning low pressure ECCS availability.

References: LLOT0330, Rev. 009 Student Ref. required No

Learning Objective: Obj 8

Question source: New

Question History:

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

10CFR 55.41 X

Comments:	Created/Modified by: M. Riches Reviewed by: Johnson

QUESTION 44

Unit 1 was in OPCON 4 with '1B' Loop RHR in Shutdown Cooling when a valving error resulted in a RPV level transient. Conditions have stabilized and the following conditions are observed:

- Reactor vessel level is +15 inches and slowly increasing.
- MCR Annunciator REACTOR 107 H-1, "REACTOR WATER LEVEL BELOW LEVEL 3 TRIP" is lit.
- MCR Annunciator REACTOR 107 H-2, "REACTOR HI/LO LEVEL" is lit.
- HV-51-1F009, RHR Shutdown Cooling Suction Inboard OPEN
- HV-51-1F008, RHR Shutdown Cooling Suction Outboard CLOSED
- HV-51-1F015B, RHR Shutdown Cooling Return Outboard CLOSED
- HV-51-151B, RHR Shutdown Cooling Test Check Equalizing Line Inboard OPEN
- HV-51-1F050B, RHR Shutdown Cooling Testable Check CLOSED

Based on the above conditions, which one of the following describes the status of the associated NSSSS logic relays?

	Channel A	Channel B	Channel C	Channel D
A.	Energized	Energized	Deenergized	Deenergized
B.	Energized	Deenergized	Energized	Deenergized
C.	Deenergized	Deenergized	Energized	Energized
D.	Deenergized	Energized	Deenergized	Energized

K&A # 223002 PCIS / Nuclear Steam Supply Shutoff System

Importance Rating 2.6

QUESTION 44

K&A Statement: A1.04 – Ability to predict and/or monitor changes in parameters associated with operating the PCIS/ Nuclear Steam Supply Shutoff System including **Individual system relay status**.

Justification:

A. Correct – A Group IIA isolation has occurred that only affected the outboard isolation valves. This means that the 'B' trip relay channel tripped which was due to trip conditions on Channels 'C' and 'D'. The NSSSS logic uses de-energized state to achieve a trip condition.

B. In-Correct but plausible if the applicant believes the 'B' and 'D' channels are required to trip to affect the outboard isolation valves.

C. In-Correct but plausible if the applicant believes the 'A' and 'B' channels are required to trip to affect the outboard isolation valves.

D. In-Correct but plausible if the applicant believes the 'A' and 'C' channels are required to trip to affect the outboard isolation valves.

References: LLOT0180, Rev. 015 Student Ref. required No

Learning Objective: Obj 2

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 45

Div 1 125 VDC has been lost. Which one of the following describes the impact on manual operation of the non-ADS SRVS from the Main Control Room (MCR) and the Remote Shutdown Panel (RSP)?

Manual operation is....

	MCR	RSP
A.	Available	Available
B.	Unavailable	Available
C.	Unavailable	Unavailable
D.	Available	Unavailable

K&A # 239002 Safety Relief Valves

Importance Rating 2.8

QUESTION 45

K&A Statement: K2.01 – Knowledge of electrical power supplies to **SRV solenoids**.

Justification:

A. In-Correct but plausible if the applicant believed that the controls switches at both locations were powered from Div 3 125 VDC.

B. In-Correct but plausible if the applicant believed Div 1 125 VDC powered SRV control switches in the MCR and Div 3 125 VDC powered SRV control switches in the RSP.

C. Correct – The manual switches for the non-ADS SRVs in the MCR and RSP are both supplied from Div 1 125 DC.

D. In-Correct but plausible if the applicant believed Div 3 125 VDC powered SRV control switches in the MCR and Div 1 125 VDC powered SRV control switches in the RSP.

References: LLOT0120, Rev. 014 Student Ref. required No

Learning Objective: Obj 3

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 46

Unit 1 plant conditions are as follows:

100% power
"1D" Narrow Range Level Transmitter has failed upscale

Subsequently, "1C" Feedwater Narrow Range Level Transmitter fails upscale

WHICH ONE of the following identifies the status of the RFP Turbines and the Main Turbine?

	RFP Turbines	Main Turbine
A.	operating	operating
B.	tripped	tripped
C.	tripped	operating
D.	operating	tripped

K&A # 295002 Rx Water Level Control

Importance Rating 3.1

QUESTION 46

K&A Statement: K5.03 - Knowledge of operational implications of Water level measurement as it applies to REACTOR WATER LEVEL CONTROL SYSTEM

Justification:

A. Correct – There are four narrow range level instruments (A, B, C &D). All four signals are used to determine reactor level for control and for +54" trips. The system takes the four reactor water level inputs and checks for signal validity and then averages the valid ones. (I.E. The output is the average of the valid signals.) The system only provides a good output when there are 2 or more valid input signals. If there is only one valid input signal, a LEVEL SIGNAL FAILURE condition exists. For each level transmitter, a failure is defined as: a deviation of greater than 4 inches from the SMS output for 3 second OR a hardware failure, eg. Outside 4-20mA, on any sensing element used in the calculation. A failure results in an automatic bumpless disconnection of the signal from the soft majority selector. The detected error is annunciated as a trouble alarm in the MCR. More detailed alarm information is available at the operator work station. Bumpless reconnection of the repaired signal is performed automatically. If 3 out of 4 reactor level signals are in error, or if two errors occur simultaneously, a FWLC failure will occur.

B. In-Correct but plausible if the applicant does NOT remember the required level instruments to initiate a high level trip.

C. In-Correct but plausible if the applicant does NOT remember the required instruments to initiate a high level trip.

D. In-Correct but plausible if the applicant does NOT remember the required instruments to initiate a high level trip.e

References: LLOT0550 pages 14, 15 Student Ref: required No

Learning Objective: N/A

Question source: Limerick Bank

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Johnson

QUESTION 48

Unit 1 plant conditions are as follows:

100% power
Normal electrical lineup

The 101 Safeguard Transformer Feeder Breaker trips.

WHICH ONE of the following identifies the response of the D114 Load Center feeder breaker to the above condition?

A. Trips and recloses 3 seconds after 201-D11 breaker closes

B. Trips and recloses 3 seconds after D11 EDG output breaker closes

C. Remains closed and supplies power to D114 Load Center when 201-D11 breaker closes

D. Remains closed and supplies power to D114 Load Center when D11 EDG output breaker closes

K&A # 262001 AC Electrical Distribution

Importance Rating 3.8

QUESTION 48

K&A Statement: K3.01 - Knowledge of the effect that a loss or malfunction of the AC ELECTRICAL DISTRIBUTION system will have on Major System loads

Justification:

A. In-Correct but plausible since on a LOCA condition Load Center Transformer Breakers trip and then re-close after a three second time delay

B. In-Correct but plausible since the on a LOCA condition Load Center Transformer Breakers trip and then re-close after a three second time delay

C. Correct – D114 Load Center Transformer Breaker (4.16kV) is closed. Power is supplied by D*1 bus via a 4.16 KV/480 V transformer. Load Center Transformer Breakers remain closed on undervoltage. All Load Center MCC Feeder Breakers (480 V) remain closed except D*14-G-D and D*24-G-D NON-Safeguard MCC feeder breakers.

D. In-Correct but plausible EDG does not supply D114 Load Center.

References: LLOT0650 pages 13, LGSOPS 092A, Student Ref: required No

Learning Objective: N/A

Question source: Limerick Bank

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Johnson

QUESTION 50

A piece of scaffolding inadvertently falls across the positive and negative output terminals of Battery Bank 1B1 and short circuits the bank.

Which one of the following describes the immediate impact this event will have on the Div 2 DC Distribution system?

The result will be...

A. a complete loss of 125 VDC loads and undervoltage on 250 VDC loads.

B. a complete loss of 125 VDC loads and normal voltage on 250 VDC loads.

C. a partial loss of 125 VDC loads and undervoltage on 250 VDC loads.

D. a partial loss of 125 VDC loads and normal voltage on 250 VDC loads.

K&A # 263000 DC Electrical Distribution

Importance Rating 3.2

QUESTION 50

K&A Statement: K1.02 - Knowledge of the physical connections and/or cause-effect relationships between DC ELECTRICAL DISTRIBUTION SYSTEM and **Battery charger and battery**.

Justification:

A. In-Correct but plausible if the applicant did not understand how 125 VDC is generated on Div 2 using two battery banks. The second half of the answer is correct for the effect on 250 VDC.

B. In-Correct but plausible if the applicant did not understand how the loss of a battery bank would effect the generation of 125 and 250 VDC on Div 2.

C. Correct – The Div 2 DC system uses two battery banks / battery chargers to generate the 125 VDC and 250 VDC power supply. Output from one battery charger/battery bank feeds one terminal within the Div 2 DC Fuse Box. Output from the other battery charger/ battery bank feeds the other terminal of the fuse box. Each battery bank/ charger generates 125 VDC. 250 VDC power is generated by tapping off each of these positive and negative 125 VDC strips, while 125 VDC power is generated by tapping off one or the other of the strips within the fuse box to a neutral leg. Loss of a battery bank and its associated charger will result in a loss of power to one of the strips. Those 125 volt DC loads tapping off the terminal fed by the faulted battery will lose power and all the 250 VDC loads would be supplied at only 125 volts.

D. In-Correct but plausible since the first half of the answer is correct concerning partial loss of 125 VDC loads and if the applicant did not understand how the 250 VDC signal is generated.

References: LLOT0690, Rev. 012 Student Ref. required No

Learning Objective: Obj. 2

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 51

Unit 2 is at 100% power when the following concerns developed:

- While performing ST-6-092-364-0, "D24 Diesel Generator Operability Verification", an internal Jacket Water Cooling leak is observed on EDG D24 and the diesel is declared inoperable.
- A team has been sent out to perform ST-6-092-366-0, "Inoperable Unit 2 Safeguard Power Supply Actions for Both Units" to verify operability of remaining Unit 2 diesels.
- Visual inspection of remaining EDGs reveals leaks at the same locations on EDG D21 and D22.
- Results of inspection of Jacket Water Cooling on EDG D11, D12, D13, D14 and D23 were satisfactory.
- Based on the results of the inspection, operations management has declared EDG D21 and D22 inoperable.

Which one of the following choices contains the correct action(s) and completion times per Technical Specifications that must be taken based on the above events?

A. Only performance of Section 4.3, "One Hour Actions for Loss of Offsite Feed" of ST-6-092-366-0 within one hour is required.

B. Only performance of Section 4.5, "Plant Systems with Four Subsystems" of ST-6-092-366-0 within one hour is required.

C. Performance of Section 4.3, "One Hour Actions for Loss of Offsite Feed" of ST-6-092-366-0 within one hour AND performance of ST-6-092-363-0, D23 Diesel Generator Operability Verification" within one hour is required.

D. Performance of Section 4.5, "Plant Systems with Four Subsystems" of ST-6-092-366-0 within one hour AND performance of ST-6-092-363-0, D23 Diesel Generator Operability Verification" within one hour is required.

K&A #	264000 Emergency Diesel Generators
Importance Rating	3.9

QUESTION 51

K&A Statement: G.2.2.39 – Knowledge of less than or equal to one hour Technical Specification action statements for systems as they relate to EDGs.

Justification:

A. In-Correct but plausible since this surveillance (SR 4.8.1.1.1.a.) would be required anytime a diesel is lost.

B. In-Correct but plausible since the operability of LPCI is required anytime two or more diesels are declared inoperable (action e), but it has a two hour completion time rather than a one hour completion time.

C. Correct – Condition 'C' of LCO 3/4.8.1 "AC Sources Operating" states that SR 4.8.1.1.1a should be performed for the two required offsite circuits (ST-6-092-366-0, Section 4.3) within 1 hour and that SR 4.8.1.1.2.a (ST-6-092-363-0) should be performed for the last diesel within 1 hour.

D. In-Correct but plausible since the operability of LPCI is required anytime two or more diesels are declared inoperable(action e), but it has a two hour completion time rather than a one hour completion time. Performance of ST-6-092-363-0 within one hour is correct.

References: LLOT0670, Rev. 012 — Student Ref. required: No

Learning Objective: Obj. 13

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 52

The following sequence of events occurs on Unit 2:

- Bus D21 de-energizes due to spurious trip of 201-D21
- Bus D21 Feeder Breaker 101-D21 remains open.
- MCR alarm 218 G-3, TURB BLDG COOLING WATER HTX OUTLET PRESS LO, is received.
- The D21 EDG starts and provides power to Bus D21.
- Subsequently, 'A' Instrument Air compressor automatically starts due to a system demand signal.
- A short time later, MCR alarm 218 B-1, 2A INST AIR COMPRESSOR TROUBLE, is received.
- An EO has been dispatched and reports observing the following indications on the operating 'A' Instrument Air compressor:
    - Lube oil temperature is 187°F
    - Intercooler outlet air temperature is 303°F
    - Compressor cooling water outlet temperature is 123°F
    - Air temperature downstream of the aftercooler is 108°F.

Which one of the following would explain the conditions observed on the 'A' Instrument Air compressor?

A. 'B' TECW pump failed to start on its auto start signal.

B. Compressor Motor Start interlock failed on 'A' Instrument Air compressor.

C. The actuator on the TECW Temperature Control Valve to 'A' Instrument Air compressor developed an air leak.

D. 'A' TECW pump failed to restart once the undervoltage condition cleared on Bus D21.

K&A # 300000 Instrument Air
Importance Rating 2.8

QUESTION 52

K&A Statement: K1.04 – Knowledge of the physical connections and/or cause-effect relationships between Instrument Air System and **Cooling water to the compressors**.

Justification:

A. In-Correct but plausible since 'B' TECW pump should have started on low pressure in the TECW supply. However, based on the temperature indications on the 'A' Instrument Air compressor, the aftercooler temperatures are within a normal range. Since the compressor and aftercooler have separate TECW supplies. A loss of TECW pump would result in loss of cooling to both the aftercooler and the compressor.

B. Correct – A solenoid operated valve is located on the TECW supply line to each Instrument Air compressor. Each Instrument Air compressor starting circuit is designed with a "motor start interlock" that opens this valve when the compressor starts. Failure of this valve to open would prevent cooling from reaching the compressor, however cooling flow to the aftercooler would not be affected since it has its own flowpath.

C. In-Correct but plausible since the TECW TCV is located on the TECW supply lines to each Instrument Air compressor. However, an air leak on the TCV would cause it open increasing cooling flow to the compressor. In addition, a bypass line around the TCV is normally throttled open to ensure adequate cooling gets to the compressor.

D. In-Correct but plausible since the 'A' TECW pump would receive an auto-start signal once power was returned to its affected load center. However, based on the temperature indications on the 'A' Instrument Air compressor, the aftercooler temperatures are within a normal range. Since the compressor and aftercooler have separate TECW supplies. A loss of TECW pump would result in loss of cooling to both the aftercooler and the compressor, which from the indications didn't occur.

References: LLOT0730, Rev. 014
LGSOPS0014, Rev. 000

Student Ref. required No

Learning Objective: Obj. 9, 10 (LLOT0730)
EO 6, IL 4, (LGSOPS0014)

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 54

Unit 1 is operating normally at rated power when the instrument air line breaks at the actuator of the in-service CRD Flow Control Valve, FVC4-1F002A.

Which one of the following describes the impact of the air line break on the CRD system?

A. The flow control valve opens.
CRD pump will trip on high flow.

B. The flow control valve closes.
CRD mechanism temperatures will increase.

C. The flow control valve opens.
CRD mechanism will move more than one notch in response to a RMCS withdrawal command.

D. The flow control valve closes.
CRD mechanism will not move in response to a RMCS withdrawal command.

K&A # 201003 Control Rod Drive and Mechanism

Importance Rating 3.2

QUESTION 54

K&A Statement: K1.01 – Knowledge of the physical connections and/or cause-effect relationships between Control Rod Drive and Mechanism System and Control Rod Drive Hydraulics System.

Justification:

A. In-Correct but plausible if the applicant doesn't know which direction the flow control valve fails. Also, failure of the valve will result in a low flow / high pressure condition on the discharge of the CRD pumps.

B. Correct – Flow control valve will close (~20% open with mechanical gag) on loss of air. With the Drive Water Pressure Control Valve, F003 throttled to maintain Drive Water pressure higher than reactor pressure, the already limited flow (0.2 – 0.3 gpm) providing cooling flow to each CRD mechanism will become more restricted and CRD mechanism temperatures will increase.

C. In-Correct but plausible if the applicant doesn't know which direction the flow control valve fails.

D. In-Correct but plausible since drive header pressure is maintained because the flow control valve cannot fully close (~20% open).

References: LGSOPS0046 Rev001 Student Ref. required No

Learning Objective: IL 9

Question source: Modified (Susquehanna) Changed stem from loss of power to controller to loss of air to flow control valve.

Question History: NRC-07 (SSES)

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 56

The “A” Recirc Motor Generator (MG) Set is being started IAW S43.1.A, “Startup of Recirculation System”, Step 4.3.6.

For the listed parameters, WHICH ONE of the following sets of approximate values is correct 30 seconds after the “A” MG Set Drive Motor Breaker is closed:

Assume NO operator action after placing the “A” MG Set Drive Motor Control (MOTOR) to “START” at *OC602.

	Speed Demand	MG Set Generator Voltage (nominal)
A.	40%	1610 volts
B.	40%	805 volts
C.	20%	1610 volts
D.	20%	805 volts

K&A # 202002 Recirculation Flow Control

Importance Rating 3.3

QUESTION 56

K&A Statement: A4.01 – Ability to manually operate and/or monitor the control room: MG sets

Justification:

A. In-Correct. After 30 seconds the speed controller has reduced speed demand to approximately 20% and voltage would be approximately 805. Voltage increase does occur so answer is plausible if the applicant does not recall the "settle speed" of the MG set start sequence.

B. In-Correct. After 30 seconds the speed controller has reduced speed demand to approximately 20%. Plausible since voltage is correct for the 20% "settle speed".

C. In-Correct. After 30 seconds the speed controller has reduced speed demand to approximately 20% and voltage would be approximately 805. Voltage increase does occur so answer is plausible if the applicant does not recall the voltage (805) associated with the "settle speed".

D. Correct - The speed demand is initially from the startup signal generator which positions the scoop tube between 35%-45% speed to provide sufficient torque for breakaway MG set startup. After the Generator Field breaker closes, speed control transfers to the speed controller, which reduces speed demand to approximately 20%. Generator voltage increases when the field breaker closes and is controlled by the MG Set voltage regulator to maintain a 70V/Hz relationship. Six pole sync generator speed varies from 230 rpm to 1150 rpm depending on scoop tube controlled drive coupling. Voltage = (70V/Hz)*(N*P)/120, where N=gen spd in rpm, P=number of poles. Voltage approx equal to 805V at 20% min speed of 230 rpm.

References: LGSOPS0043B pg. 6, 7, 10-12; S43.1.A, rev.59 pg.13, 14, S43.9A LGSOPS0043A pg. 13, 14

Student Ref: required No

Learning Objective: N/A

Question source: Limerick bank

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 57

Unit 2 plant conditions are as follows:

- Reactor Power is at 75%
- Traversing In-core Probe (TIP) scans are in progress
- The "B" TIP is stuck at the indexer

A main turbine trip occurred, resulting in the following:

- RCIC and HPCI automatically start on a valid signal

WHICH ONE of the following describes the expected position of the TIP Shear and Ball Valves for the "B" TIP two (2) minutes later? No operator actions are taken.

	Shear Valve	Ball Valve
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

K&A # 215001 Traversing In-Core Probe

Importance Rating 3.1

QUESTION 57

K&A Statement: K6.04 - Knowledge of the effect that a loss or malfunction of Primary Containment Isolation System will have on the TRAVERSING IN-CORE PROBE SYSTEM

Justification:

A. Correct - A containment isolation signal is present, so the TIP system should automatically shift into reverse, withdraw detectors and close associated ball valves. Since the "B" detector is stuck at the indexer, the "B" TIP is not withdrawn into its shield and the ball valve will not automatically close. The shear valve requires operator action to close. Therefore for the given conditions, both valves remain open.

Note: The K/A is matched since the TIP malfunction causes a PCIS function to NOT be satisfied. That is, the ball valve cannot close with the TIP partially inserted. The result is that the shear valve is the only action left. The author chose to NOT include manual action (to fire the shear valve) in the stem to avoid potentially conflicting answers.

B. In-Correct – The ball valve will not automatically close on the isolation signal if the TIP is not withdrawn into its shield during a containment isolation. Plausible if the applicant does not recall the interlock between the automatic closure of the ball valve and the TIP detector location or it the applicant thinks that the TIP detector being at the indexer will allow the ball valve to close.

C. In-Correct – The shear valve requires operator action to close. Plausible if the applicant thinks that under an isolation condition, the shear valve would receive an automatic signal.

D. In-Correct – The ball valve will not automatically close on an isolation signal if the TIP detector is not withdrawn into its shield. The shear valve requires operator action. Plausible if the applicant thinks that the shear valve would receive an automatic signal to allow the ball valve to close on a containment isolation signal.

References: LLOT0290 pg. 8, 9, 12, 17, 18 Student Ref: required No

Learning Objective: N/A

Question source: Limerick bank- modified

Question History:

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

10CFR 55.41(9) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 58

Plant conditions are as follows:

- Reactor power is at 31%.
- Control Rod 28-29 is selected.

Subsequently, the 'C' level detector on the LPRM string located at 24-33 fails downscale. Which one of the following describes the effect this will have on the Rod Block Monitoring (RBM) system?

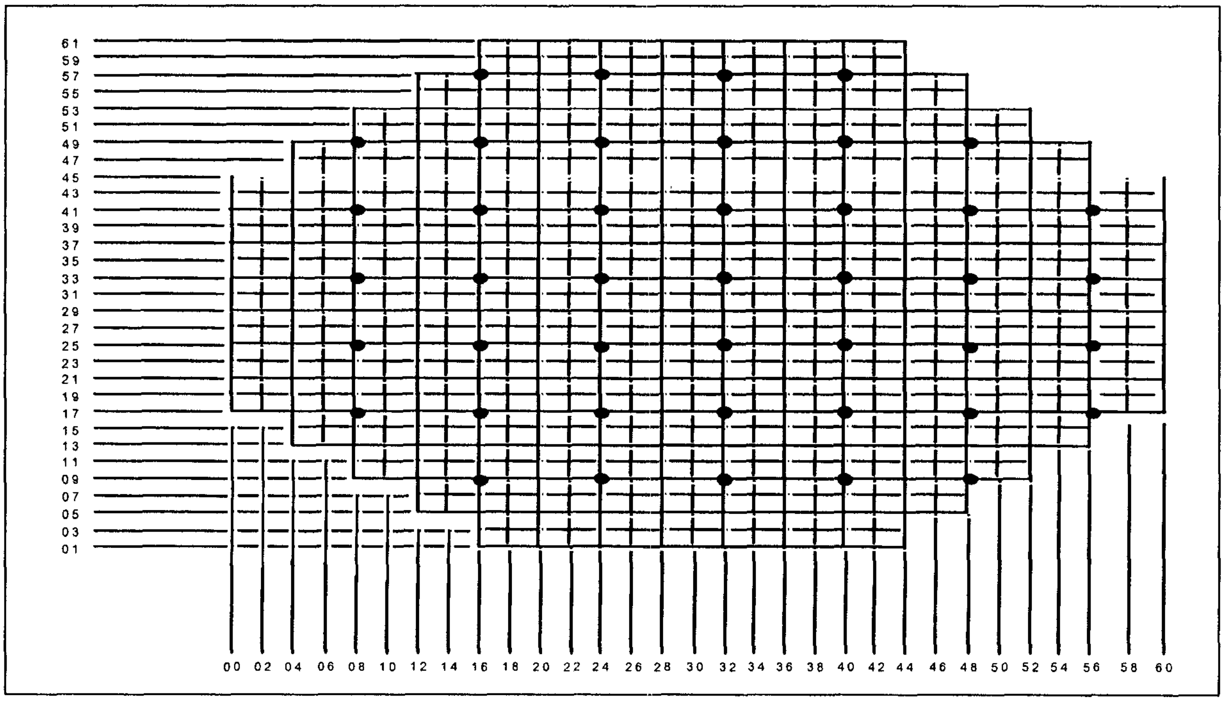
Core Map provided to locate positions of control rod and LPRM string.

A. RBM system will be unaffected since level 'C' LPRM detectors are not part of the RBM circuit.

B. RBM 'A' will register a change in the "Average Flux" value since the control rod has been selected.
RBM 'B' will be unaffected since the level 'C' LPRM detectors are not part of the RBM 'B' circuit

C. RBM 'B' will register a change in the "Average Flux" value since the control rod has been selected.
RBM 'A' will be unaffected since level 'C' LPRM detectors are not part of the RBM 'A' circuit

D. Both RBM channels will register a change in the "Average Flux" value since level 'C' LPRM detectors are used by both channels to generate an "Average Flux" value.



K&A # 215002 Rod Block Monitor
Importance Rating 2.8

QUESTION 58

K&A Statement: K6.05 – Knowledge of the effect the a loss or malfunction LPRM detectors will have on the Rod Block Monitoring System.

Justification:

A. In-Correct but plausible since 'A' level detectors are excluded from the averaging circuit.

B. In-Correct but plausible since RBM 'A' would be effected and RBM 'B' would be unaffected if the failure was on a 'D' level detector.

C. In-Correct but plausible since RBM 'B' would be effected and RBM 'A' would be unaffected if the failure was on a 'B' level detector.

D. Correct – The 'C' level detectors provide inputs to both RBM averaging circuits.

References: LGSOPS0074B Rev001 Student Ref. required No

Learning Objective: EO07

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 59

Unit 1 has experienced a Design Basis LOCA approximately one minute ago. You have been directed to monitor and report containment parameters.

- The crew is implementing T-100 and T-102
- No operator action has been taken for T-102
- All ECCS systems functioned as designed

WHICH ONE of the following, by itself, would indicate there has been a malfunction in the suppression chamber-drywell vacuum breakers?

A. Peak drywell pressure at 50 psig

B. Peak suppression chamber pressure at 30 psig

C. Peak drywell temperature at 340 degrees F

D. Peak suppression chamber temperature at 212 degrees F

K&A # 223001 Primary CTMT and Aux.

Importance Rating 2.8

QUESTION 59

K&A Statement: K5.03 - Knowledge of the operational implications of Down comer operation as it applies to PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES

Justification:

A. Correct – Failed open vacuum breakers under these conditions would bypass suppression function and cause drywell pressure to exceed maximum design value (44 psig).

B. In-Correct – This is the expected accident value for suppression chamber pressure. Plausible if applicants think this is too low.

C. In-Correct – This is the expected post-accident value. Plausible since this is also the max design value.

D. In-Correct – 212.5°F is the expected post-accident value. Plausible if applicant thinks the chamber should never exceed the boiling point.

References: Tech spec 3.6.1.6, 3.6.2.1, 6.2.1.1.3.1 Student Ref: required No
Tech spec Bases 3/4.6.1.5, 3/4.6.2
UFSAR 6.2.1.1.3.1

Learning Objective: N/A

Question source: New

Question History:

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

10CFR 55.41(9) X

Comments: Created/Modified by: Johnson
Reviewed by: Tomlinson

QUESTION 64

Unit 2 is at 100% power with all systems in normal lineup.

The following alarm is received "1 UNIT RECOMBINER OUTLET HI TEMP" IN THE CONTROL ROOM.

Operators are monitoring recombiner outlet temperatures.

WHICH ONE (1) describes the condition that is causing the alarm AND
(2) What major action is required to mitigate the consequences of the alarm?

	(1)	(2)
A.	SJAE flow too high	Shift to standby SJAE
B.	Recombiner is saturated with Moisture	Ensure 1st stage SJAE suction valves close at 900°F
C.	Excessive hydrogen recombination	Ensure 1st stage SJAE suction valves close at 900°F
D.	Dilution steam flow high	Shift to standby SJAE

K&A # 271000 Offgas
Importance Rating 2.7

QUESTION 64
K&A Statement: A2.12 – Ability to (a) predict the impacts of Recombiner high temperature on the OFFGAS SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations

Justification:

A. In-Correct but plausible if applicant believes the increased temperatures are due to high SJAE flow. High SJAE flow indicates a failure of the SJAE. Operator action to swap to standby SJAE would be appropriate for a failed SJAE.

B. In-Correct. The conditions are not indicative of moisture in the recombiner. Plausible since the action to ensure SJAE suction valves closed is correct.

C. Correct – High recombiner temperature indicates excessive hydrogen recombination. ARC 127 B-1 directs the operators to monitor recombiner outlet temperature and ensure that SJAE suction valves close when temperature reaches 900°F.

D. In-Correct. Dilution flow too low would generate the alarm. Plausible if applicant does not understand the effect of the dilution flow and believes that higher flow allows for more recombination. More recombination generates higher temperatures.

References: LGSOPS0069, ARC 127-B-1, ON-103 Student Ref: required No

Learning Objective: LGSOPS0069 IL2, IL4

Question source: New

Question History:

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

10CFR 55.41(4) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 66

Plant conditions are as follows:

- Unit 1 is at 100% power with
- Unit 2 is in OPCON 5
- Unit 2 Core Shuffle is in progress

You are the RO in the control room. An unexpected alarm is received for New Fuel Storage Area HI RADIATON. All other indications are normal.

WHICH ONE of the following identifies the required action(s) per ON-120, Fuel Handling Problems?

A. Raise grappled fuel assembly

B. Evacuate the area surrounding Fuel Pool Storage

C. Verify SBGT system has initiated

D. Notify Reactor Engineering to verify shutdown margin

K&A #	2.1.44 Conduct of Operations
Importance Rating	3.9

QUESTION 66

K&A Statement: K4.05 – Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

Justification:

A. In-Correct but plausible if the applicant does not distinguish between area radiation monitor alarm and observing SRM doublings as indicating inadvertent criticality (ON-120 step 2.1) Raising grappled assembly is response to indicated criticality.

B. Correct – If any Fuel Floor Radiation Monitor alarms, and is not due to object handling near water surface which is immediately re-submersible, THEN Evacuate affected area(s) of the Fuel Floor. (ON-12- step 2.2.1)

C. In-Correct but plausible if the applicant does not know that SBGT initiation is not an action dedicated to an ARM alarming.

D. In-Correct but plausible if the applicant does not recognize that ARM alarming is not indicative of inadvertent criticality.

References:	ON-120, LLOT760 pg. 37, ARC 109 C-5, D-5	Student Ref: required	No

Learning Objective: N/A

Question source: Limerick Bank

Question History: None

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

10CFR 55.41(10) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 71

Repairs on a piece of equipment are planned in a high radiation area. Repairs are expected to take an hour and a half. Two actions are being considered to reduce the dose:

- Install temporary shielding. The temporary shielding will take one individual 15 minutes to install and 15 minutes to remove the shielding around the piece of equipment but will not increase the time it takes to complete the repairs.
- Use a special tool that will allow the job to be performed at a greater distance from the equipment. Using the tool will double the time required to complete the job.

The following information has been obtained / estimated during the job planning phase.

- Dose rate at the piece of equipment without shielding is 1.2 R/hr.
- Dose rate with the shielding installed is estimated to be 300 mr/hr.
- Dose rate using the tool without installing the shielding is estimated to be 500 mr/hr.
- Dose rate using both the tool with the shielding installed is estimated to be 125 mr/hr.

Your current exposure is 900 mrem. You would be expected to install and remove the shielding as part of completing the job. Assume the decision is to perform the job using both the special tool and the temporary shielding.

Which one of the following contains the correct dose that would be received during performance of the job AND correctly identifies whether this is within or exceeds the annual dose control limit (ADCL)?

A. Total dose for this job will be 975 mrem AND you will not exceed the ADCL.

B. Total dose for this job will be 975 mrem AND you will exceed the ADCL.

C. Total dose for this job will be 1050 mrem AND you will exceed the ADCL.

D. Total dose for this job will be 1050 mrem AND you will not exceed the ADCL.

K&A # Plant-wide Generic
Importance Rating 3.2

QUESTION 71

K&A Statement: G2.3.4 – Knowledge of radiation exposure limits under normal or emergency conditions.

Justification:

A. Correct – Given the ADCL = 2000 mrem:

Total dose = **(2 x 0.25 x1200) + (2 x 1.5 x 125)** + 900 mrem = **975** mrem + 900 = 1,825 mrem

B. In-Correct but plausible if applicant believed the ADCL was 1,500 mrem. Using an ADCL = 1500 mrem:

Total dose = **(2 x 0.25 x1200)** + **(2 x 1.5 x 125)** + 900 mrem = **975** mrem + 900 = 1,825 mrem >1,500)

C. In-Correct but plausible if applicant only took into account the dose associated with the shielding case and the applicant believed ADCL was 1,500 mrem. Using an ADCL = 1,500 mrem:

Total dose = **1.5(300) + (2 x 0.25 x1200)** + 900 mrem = **1050** mrem + 900 mrem = 1950 mrem(>1,500)

D. In-Correct but plausible if applicant only took into account the dose associated with the shielding case. Using the correct ADCL = 2,000 mrem:

Total dose = **1.5(300) + (2 x 0.25 x1200)** + 900 mrem = **1050** mrem + 900 mrem = 1950 mrem (<2,000)

References: LLOT1760, Rev. 010 Student Ref: required Yes

Learning Objective: Obj 5

Question source: Modified Bank (Pilgrim) Changed from an ALARA question to a dose rate question.

Question History: Pilgrim

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 76

Unit 1 plant conditions are as follows:

- "1B" Recirc Pump tripped
- "1A" Recirc Pump speed is 52%
- Core Plate delta P is 1.0 psid
- Reactor Power is 57%
- One OPRM is out of service for testing, all other equipment is in service

The following alarms are observed:

- "1B" RECIRC M-G DRIVE MOTOR TRIP
- OPRM TRIPS ENABLED

WHICH ONE of the following actions is required?

A. Manually SCRAM the reactor

B. Restart the "1B" Recirc Pump

C. Lower "1A" Recirc Pump speed

D. Insert control rods per RMSI

K&A # 295001 Partial or Complete Loss of Core flow

Importance Rating 2.5

QUESTION 76

K&A Statement: K1.04 –Knowledge of the operational implications of Limiting cycle oscillation as it applies to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION

Justification:

A. In-Correct but plausible if applicant does not differentiate between the actions to take with OPRMs operable. OT-112 directs operator to SCRAM if OPRMs are inoperable.

B. In-Correct but plausible if the applicant know that restarting the tripped recirc pump is not allowed prior to exiting the restricted region.

C. In-Correct but plausible if applicant does not understand power to flow map.

D. Correct – OT-112 directs either increasing core flow or inserting control rods to exit the restricted region.

References: OT-112 pg.4, Attachment 1, LLOT275 Student Ref: required Yes, power to flow map without labeling

Learning Objective: N/A

Question source: Limerick Bank

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Johnson
Reviewed by: Tomlinson

QUESTION 85

Unit 1 experienced a LOCA and LOOP from 100% power 30 minutes ago. The crew is implementing T-100 and T-102.

Plant conditions are as follows:

- Drywell pressure is 5 psig and slowly increasing
- Reactor Pressure is 1000 psig; using SRVs for pressure control
- Reactor level is +15" and steady
- Suppression Pool level is 24'
- Suppression Pool temperature is 94 degrees F.
- CST level has decreased by 21,000 gallons in the last 30 minutes
- HPCI tripped on overspeed during initial start and is NOT running
- RCIC is the only source of makeup to RCS and is at full rated flow
- The radwaste operator reports he has been receiving approximately 100 gpm from the floor drains for the last 30 minutes
- The following alarm just came in: REACTOR ENCL FLOOR DRAIN SUMP PUMP on MCR 127
- There are NO alarms on Panel 116 or 117

Considering the above conditions:

Select from the following the source of 100 gpm input to radwaste

A. HPCI (water side)

B. HPCI (steam side)

C. RCIC (water side)

D. RCIC (steam side)

K&A #	295036 Secondary Containment High Sump/Area water level
Importance Rating	3.2

QUESTION 85
K&A Statement: EA2.03 Cause of the high water level
Justification:

A. Correct - With CST decreasing 3,000 gallons more than expected due to full RCIC flow (RCIC 600 gpm * 30 minutes = 18,000 gals) and Suppression Pool level in the normal band there has to be a significant loss from a system taking suction from the CST. If RCIC was leaking 100 gpm at the discharge it could NOT provide full flow to the RCS. A suction leak would have resulted in RCIC SUCT LO PRESS. HPCI is the only remaining choice. Also with the HPCI turbine tripped and no other alarms on panel 117 there are no indications of HPCI steam leak.

B. In-Correct – With the turbine tripped and no other alarms on panel 117 there are no indications of HPCI steam leak. In addition, if there were a 100 gpm steam leak RCIC could not maintain RCS level steady 30 minutes after a SCRAM. Plausible if applicant associated turbine trip with a steam leak.

C. In-Correct - If RCIC were leaking 100 gpm from the water side it could NOT provide full flow to the RCS. Plausible since RCIC is running and takes suction from CST.

D. In-Correct - With NO alarms on panel 116 there are NO indications of a steam leak. In addition, if there were a 100 gpm steam leak RCIC could not maintain RCS level steady 30 minutes after a SCRAM. Plausible since RCIC is running.

References: E-1AY160 page 1
OT-117 page 3
LGSOPS0071 pages 40 and 41

Student Ref: required No

Learning Objective: N/A

Question source: New

Question History: None

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

10CFR 55.41(5) X

Comments: Created/Modified by: Johnson
Reviewed by: Presby

QUESTION 92

The following events have occurred on Unit 1:

- A steam line break has occurred in Primary Containment.
- Several control rods failed to fully insert

The current plant conditions are as follows:

- RPV level is +22 inches.
- RPV pressure is 430 psig.
- Drywell pressure is 22 psig.
- Drywell temperature is 260 °F.
- Suppression Pool pressure is 8.2 psig.
- Suppression Pool temperature is 112°F.

Which one of the following Residual Heat Removal system lineups is needed based on the above current conditions?

A. Loop 'A' in Low Pressure Core Injection mode, Loop 'B' in Suppression Pool Spray mode.

B. Loop 'A' in Low Pressure Core Injection mode, Loop 'B' in Suppression Pool Cooling mode.

C. Loop 'A' in Suppression Pool Spray mode, Loop 'B' in Suppression Pool Cooling mode.

D. Loop 'A' in Suppression Pool Spray mode, Loop 'B' in Drywell Spray mode.

K&A #	230000 RHR/LPCI: Tours/Pool Spray Mode
Importance Rating	3.7

QUESTION 92

K&A Statement: G.2.4.6 – Knowledge of EOP mitigation strategies.

Justification:

A. In-Correct but plausible since LPCI injection would be the normal alignment following initiation due to the High Drywell Pressure signal (1.68 psig). However, since RPV pressure is currently greater than LPCI injection pressure (~370 psig) and RPV level is within the required band of +12" to +54" (T-101, Step RC/L-4), then adequate core cooling is met without flow from LPCI mode. Suppression Pool Sprays would be correct in accordance with step PC/P-5 which directs initiation before pressure in the Suppression Pool reaches 7.5 psig.

B. In-Correct but plausible since Suppression Pool cooling is needed once Suppression Pool temperature exceeds 95°F. However, LCPI alignment would not be appropriate since RPV level is within the normal control band which indicates adequate core cooling is met without any contribution from LPCI flow.

C. Correct – Step PC/P-5 requires Suppression Pool sprays prior to Suppression Pool pressure reaching 7.5 psig, if not needed for adequate core cooling. RPV level is within the normal control band so adequate core cooling is met. Suppression Pool Cooling is needed since Suppression Pool temperature is well above where T-102 directs operating (i.e., 95°F) in this mode.

D. In-Correct but plausible since Suppression Pool sprays are required before Suppression Pool pressure 7.5 psig (Step PC/P-5). With Suppression Pool pressure greater than 7.5 psig step PC/P-9 directs spraying the Drywell if within the "Safe" region of Curve PC/P-2 (DW/T-3) "Drywell Spray Initiation Limit". However, Drywell Sprays are not permitted since Drywell temperature and pressure are outside the "Safe" region of Curve PC/P-2.

References:	LLOT1560, Rev. 12 T-102 Bases, Rev. 022	Student Ref. required	Yes T-101 T-102
Learning Objective:	IL5		
Question source:	Modified Bank (Susquehanna)	Changed to require SP Spray vs SP Cooling.	
Question History:	SSES-07 (NRC)		
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	 X	
10CFR	55.43 (5) X		
Comments:	Created/Modified by: M. Riches Reviewed by: Johnson		

"DRAFT"
2nd SET 50 Q's

Facility: **Limerick Units 1 & 2** Date of Exam: **10/31/2008** Exam Level: RO **X** SRO **X**

***Note: This checklist covers the complete written exam (100 Qs).** Item Description				Initial a	Initial b*	Initial c#
1. Questions and answers are technically accurate and applicable to the facility.				Ro	N/A	JC
2. a. NRC K/As are referenced for all questions. b. Facility learning objectives are referenced as available.				Ro	N/A	JC
3. SRO questions are appropriate in accordance with Section D.2.d of ES-401				Ro	N/A	JC
4. The sampling process was random and systematic (If more than 4 RO or 2 SRO questions were repeated from the last 2 NRC licensing exams, consult the NRR OL program office).				✓		JC
5. Question duplication from the license screening/audit exam was controlled as indicated below (check the item that applies) and appears appropriate: __ the audit exam was systematically and randomly developed; or __ the audit exam was completed before the license exam was started; or X the examinations were developed independently; or __ the licensee certifies that there is no duplication; or __ other (explain)				Ro	N/A	JC
6. Bank use meets limits (no more than 75 percent from the bank, at least 10 percent new, and the rest new or modified); enter the actual RO / SRO-only question distribution(s) at right.	Bank: 22/ 5	Modified: 21 / 5	New: 32/ 15	Ro	N/A	JC
7. Between 50 and 60 percent of the questions on the RO exam are written at the comprehension/ analysis level; the SRO exam may exceed 60 percent if the randomly selected K/As support the higher cognitive levels; enter the actual RO / SRO question distribution(s) at right.	Memory: 36/ 10	C/A: 39 / 15		Ro	N/A	JC
8. References/handouts provided do not give away answers or aid in the elimination of distractors.				Ro	N/A	JC
9. Question content conforms with specific K/A statements in the previously approved examination outline and is appropriate for the tier to which they are assigned; deviations are justified.				Ro	N/A	JC
10. Question psychometric quality and format meet the guidelines in ES Appendix B.				Ro	N/A	JC
11. The exam contains the required number of one-point, multiple choice items; the total is correct and agrees with the value on the cover sheet.				Ro	N/A	JC

	Printed Name / Signature	Date
a. Author	P. Presby / [signature]	9/13/08
b. Facility Reviewer (*)	Not Applicable – NRC Developed Exam	
c. NRC Chief Examiner (#)	J. Caruso / CE [signature]	9/13/08
d. NRC Regional Supervisor	S. Hansell / BC [signature]	9/15/08

Note: * The facility reviewer's initials/signature are not applicable for NRC-developed examinations.
Independent NRC reviewer initial items in Column "c"; chief examiner concurrence required.

*** This 401-6 accompanies submittal of 2nd set of 50 Q #s: 2, 3, 6, 11, 18, 19, 20, 27, 32, 36, 41, 47, 49, 53, 55, 60, 61, 62, 63, 65, 67, 68, 69, 70, 72, 73, 74, 75, 77, 78, 79, 80, 81, 82, 83, 84, 86, 87, 88, 89, 90, 91, 93, 94, 95, 96, 97 98, 99, 100 Questions 1-75 are RO Questions 76-100 are SRO ONLY

QUESTION 2

Unit 2 has experienced a loss of offsite power with failure of all diesel generators.

The CRS has directed the PRO to determine RPV level and pressure.

WHICH ONE of the following describes the locations to obtain the readings based on the above conditions?

A. Div 1 PAMS Level, Narrow Range Pressure on 20C603

B. Wide Range Level on 20C603, HPCI Steam Line Pressure on 20C647

C. Wide Range Level on 20C603, Narrow Range Pressure on 20C603

D. Div 1 PAMS Level, HPCI Steam Line Pressure on 20C647

K&A # 295001 Partial or Complete Loss of AC

Importance Rating 4.2

QUESTION 2

K&A Statement: A2.02– Ability to determine and/or interpret Reactor power/ pressure/ and level as it applies to PARTIAL OR TOTAL LOSS OF AC

Justification:

A. In-Correct but plausible if applicant does not recall that the PAMS is safety related but not DC powered. Narrow range pressure not available.

B. Correct – Wide range level instrumentation on 10C603 is available during a station blackout with loss of all diesels. HPCI steam line pressure indication is available during a station blackout with loss of all diesels.

C. In-Correct but plausible if applicant does recall that narrow range instrumentation is not available.

D. In-Correct but plausible if applicant does not recall that the PAMS is safety related but not DC powered.

References: LGSOPS0042, E-1, rev.33 Att. 1 Student Ref: required N

Learning Objective: LGSOPS0042 IL10

Question source: Bank (Limerick)

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 03

Bus D11 is aligned normally with the 101-D11 supply breaker closed. D11 Diesel Generator is running unloaded for monthly surveillance test. The control room operator synchronizes the diesel generator with Bus D11 and closes the generator output breaker.

Immediately after breaker closure (less than 1 second), one of the DC control power fuses blows for the diesel generator output breaker control circuit.

Electricians determine the fuse is blown. While they are obtaining a new fuse, a large LOCA occurs on Unit 1, concurrent with the loss of Transformer 101 due to a fault.

Operators respond in accordance with trip procedures. 20 minutes later, the shift manager directs electricians to replace the blown DC control power fuse in the D11 Diesel Generator output breaker control circuit.

Which of the following describes the response of the diesel output breaker when the fuse is replaced? Assume no further operator action has been taken during the event related to Bus D11 or its associated diesel generator.

A. The diesel generator output breaker will remain closed and charging spring will recharge.

B. The diesel generator output breaker will remain closed and charging spring will not recharge unless the breaker handswitch is taken to TRIP and returned to NORMAL.

C. The diesel generator output breaker will trip open and will not auto close unless 201 transformer is lost.

D. The diesel generator output breaker will trip open and will not auto close unless the breaker handswitch is taken to TRIP and returned to NORMAL.

K&A #	295004 Partial or Complete Loss of DC Power
Importance Rating	3.2

QUESTION 03

K&A Statement: A2.04 – Ability to determine and/or interpret **system lineups** as it applies to PARTIAL OR TOTAL LOSS of DC POWER.

Justification:

A. Correct. DC control power needed for trip or close. Breaker was closed when DC control power lost. Therefore breaker remains closed until fuse is replaced. The LOCA trip signal for the DG breaker will no longer be in effect. It is only active for 0.5 seconds after the LOCA signal initiates. 101-D11 will have opened on the fault. 201-D11 will not have closed because the DG will have maintained bus voltage. When the fuse is installed the charging spring will recharge in preparation for a future breaker close demand signal.

B. Incorrect. Breaker closing spring will recharge when power restored to the control circuit. Plausible because there is a sequence of breaker operation related to a LOCA/LOOP with the DG breaker closed where the breaker will not operate because of anti-pumping lockout of its charging spring circuit.

C. Incorrect. Breaker will remain closed. Plausible because LOCA/LOOP initiates a 0.5 second duration trip signal and, if breaker tripped, applicant may think available201 power source will pick up the bus.

D. Incorrect. Breaker will remain closed. Plausible because there is a sequence of breaker operation related to a LOCA/LOOP with the DG breaker closed where the breaker will not operate because of anti-pumping lockout of its charging spring circuit. However, the 0.5 second LOCA trip signal will have cleared before the fuse is installed.

References: LLOT0670 Rev 12
LGSOPS0092A Rev 00
Breaker schematic

Student Ref. required No

Learning Objective: IL3 (LGSOPS0092A)
Obj 10, (LLOT0670)

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 6

WHICH of the following is the reason why the reactor is scrammed prior to evacuating the main control room in accordance with SE-1, "Plant Shutdown from the Remote Shutdown Panel"?

A. Ensures that inventory makeup requirements will be within HPCI capability.

B. Ensures that inventory makeup requirements will be within RCIC capability.

C. Scramming from outside of the control Room would require access to plant areas that may be inaccessible due to post-accident high rad levels.

D. Scramming from outside of the control room would require RPS bus power to be tripped causing concurrent isolations of PCIS valve groups.

K&A #	295016 Control Room Abandonment
Importance Rating	4.1

QUESTION 6

K&A Statement: A3.01– Knowledge of the reasons for Reactor SCRAM as it applies to CONTROL ROOM ABANDONMENT

Justification:

A. In-Correct – HPCI is only used with SE-10 and is not applicable for this condition.

B. Correct – RCIC is used for inventory makeup per SE-1. Scramming the reactor reduces the inventory loss by reducing the heat load.

C. In-Correct – Accidents are not within the scope of SE-1. Plausible if applicant thinks that accident conditions should be considered as part of remote shutdown.

D. In-Correct – MSIVs are manually closed prior to evacuation of the control room and all Group isolations are expected during SE-1. Plausible if applicant is considering the actions to take for Alternate Remote Shutdown, SE-6, which directs opening of breakers to initiate the SCRAM.

References: SE-1, SE-6, Student Ref: required N

Learning Objective: N/A

Question source: Modified Bank (Peach Bottom)

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 11

Unit 1 has remained at 100% power for 100 days with all systems normal. The following alarm annunciates at PNL 107:

- DRYWELL HI/LO PRESS

You observe the following containment parameters:

- Drywell pressure is 1.1 psig and slowly increasing.
- Drywell temperature is 130 degrees F and slowly increasing.
- FI-87-120 "AIR COOLER FL" is reading 0.4 gpm.

The CRS directs you to implement OT-101, High Drywell Pressure. You determine the following:

- Drywell nitrogen supply has been isolated
- Recirc pump seal parameters indicate 500 psid across each seal
- Suppression Pool nitrogen mass is 8000 pounds

What action is required by OT-101 to address the HI/LO drywell pressure alarm?

A. Vent the drywell to REECE by opening HV-57-111 and HV-57-117.

B. Place RECW in service to reduce drywell temperature and allow drywell venting.

C. Maximize drywell cooling to reduce drywell temperature and allow drywell venting.

D. Ensure Main Steam Line and Recirc sample valves are closed.

K&A # 295024 High Drywell Pressure

Importance Rating 3.9

QUESTION 11

K&A Statement: G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation

Justification:

A. In-Correct Plausible since the drywell venting conditions, per Attachment 2 are satisfied.

B. In-Correct Plausible if the applicant mis-reads Attachment 2 and believes drywell temperature needs to be lowered in order to vent and drywell chillers are not effective

C. In-Correct Plausible since the first immediate action in OT-101 is to maximize drywell cooling

D. Correct – With FI-87-120 indicating 0.4 gpm (plus other stem conditions) the applicant should conclude there is a steam leak in the drywell. The CAUTION before Step 3.10.5 states "IF Primary Containment steam leak is detected ...THEN do not open HV-57-111 and HV-57-117", so A, B and C are incorrect. Step 3.11 specifies "IF high Drywell pressure persists then ISOLATE the following...".

References:	ARP MCR 107 F2; OT-101, page 6, 7 & 8	Student Ref: Figure 2 of OT-101	Yes

Learning Objective:

Question source: New

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Johnson
Reviewed by:Presby

QUESTION 18

Which of the choices below correctly completes the following statement to describe the analyzed basis for the primary coolant activity limit?

The Technical Specification primary coolant activity limit ensures that whole body dose limits at the site boundary are not exceeded in the event of a main steam line rupture ____(1)____ and are based on ____(2)____.

A. (1) inside primary containment, (2) 10 CFR 50 limits

B. (1) inside primary containment, (2) 10 CFR 100 limits

C. (1) outside primary containment, (2) 10 CFR 100 limits

D. (1) outside primary containment, (2) 10 CFR 50 limits

K&A # 295038 High Off-site Release Rate

Importance Rating 4.2

QUESTION 18

K&A Statement: K1.02 – Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Protection of the general public

Justification:

A. In-Correct – Plausible if applicant assumes an unisolable break inside containment is the higher risk for extended offsite dose due to the potential for core damage and/or breach of containment.

B. In- Correct – Plausible that TS analysis considered site specific conditions rather than a typical site evaluation performed by the NRC.

C. Correct – Tech spec 3/4 4.5 limit is based on ensuring the 2 hour thyroid and whole body doses resulting from a main steam line break will not exceed the dose guidelines on 10 CFR Part 100.

D. In-Correct – Plausible that TS analysis considered site specific conditions rather than a typical site evaluation performed by the NRC.

References: Tech spec 3/4 4.5 Student Ref: required N

Learning Objective:

Question source: Modified Bank (Pilgrim)

Question History: None

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

10CFR 55.41(12) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 19

Unit 1 is operating at 100% power with all systems operating normally.

You observe the following conditions in the Offgas system:

- Sudden rise in pressure
- Sudden change in system flows
- Sudden rise in system temperatures
- Sudden drop in hydrogen concentration

WHICH ONE of the following describes
(1) the cause of the transient and
(2) the major action required to address the transient?

	(1)	(2)
A.	A fire is occurring in the Offgas system.	Purge the system with nitrogen.
B.	The in-service SJAE has failed.	Place the idle SJAE in service.
C.	The charcoal beds are saturated with water.	Bypass the charcoal beds.
D.	There is excess oxygen into the recombiner.	Hold Hydrogen Water Chemistry Switch (HS-06-54) in SHUTDOWN for 15 seconds.

K&A # 600000 Plant Fire Onsite
Importance Rating 3.8

QUESTION 19

K&A Statement: G2.4.11– Knowledge of the abnormal condition procedures as it relates to PLANT FIRE ON SITE

Justification:

A. Correct – Action is as specified in ON-103, Control of Sustained Combustion in the Offgas system, steps 2.5 - 2.10. Fire in the Offgas system is indicated by a sudden rise in Offgas system pressures and temperatures, change in system flows and drop in hydrogen concentration.

B. In-Correct – Symptoms present are indicative of a sustained fire in the Offgas system, not a failure of a SJAE. Plausible since placing the idle SJAE is specified in ON-103.

C. In-Correct – Symptoms present are indicative of a sustained fire in the Offgas system. Plausible since the applicant may associate clogged charcoal with the system pressure and flow changes. Saturated charcoal could restrict system flow and cause an increase in system pressure.

D. In-Correct – Symptoms present are indicative of a sustained fire in the Offgas system. Plausible if the applicant believes that excess oxygen in the recombiner is causing the decrease in hydrogen levels.

References: ON-103, LLOT1550 Student Ref: required N

Learning Objective: LLOT1150 Obj. 1

Question source: New

Question History: None

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

10CFR 55.41(10) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 20

Unit 1 is at 100% power with the following plant conditions:

- HPCI is operating in the full flow test mode on the condensate storage tank for post maintenance testing
- RHR Loop 'A' is operating in the Suppression Pool Cooling Mode to support the HPCI test
- Average Suppression Pool temperature is 100°F and steady
- D11 Diesel Generator has been started for ST-6-092-111-1, D11 Diesel Generator 24 Hour Endurance Test
- The PPO is adjusting D11 Diesel Generator speed so that the sync scope indicator will rotate slowly in the fast direction

A grid disturbance causes a ground overcurrent relay to trip on 101 Safeguard Transformer and the following alarm is received:

- 120 F1 101 SAFEGUARD XFMR DIFF GRD LOCKOUT

Which of the choices below completes the following statements, describing conditions 3 minutes after receiving the alarm? Assume no operator action is taken.

Suppression pool temperature will be __(1)__. Bus D11 is energized from __(2)__.

	(1)	(2)
A.	stable	101 Safeguards Transformer
B.	stable	D11 Diesel Generator
C.	increasing	D11 Diesel Generator
D.	Increasing	201 Safeguards Transformer

K&A # 700000 Generator Voltage and Electric Grid Disturbances

Importance Rating 3.9

QUESTION 20

K&A Statement: A1.05 – Ability to operate and/or monitor Engineered Safety Features as it applies to GENERATOR AND ELECTRIC GRID DISTURBANCES

Justification:

A. In-Correct. High side and low side breakers on 101 Safeguards Transformer will open on actuation of the ground overcurrent relay. The bus will be re-energized from another source. Plausible if applicant does not think ground overcurrent causes transformer isolation.

B. In-Correct. RHR Pump 1A will not restart on restoration of bus voltage. Pool temperature will increase due to loss of cooling flow. Plausible if applicant thinks diesel will re-energize the bus before loads are shed on undervoltage.

C. Correct. The 101 Safeguards Transformer ground overcurrent fault will trip the D1x-101 breakers, de-energizing D11 thru D34 Buses. D11 Diesel Generator is ready to load so its output breaker will automatically close onto the D11 Bus after bus voltage has decayed to < 40%. D11-201 will not close because D11 Bus voltage will be restored before the 1 second transfer closure interlock time delay elapses. RHR Pump 1A will shed on the initial bus undervoltage and will not auto start upon bus voltage restoration.

D. Incorrect. D11-201 has a 1 second time delay following bus undervoltage before it will close. D11 Diesel Generator output breaker has a shorter 0.5 second time delay. The diesel will re-energize the bus before the closure interlock is satisfied on D11-201 Breaker.

References: LLOT0370 pages 12&25, S51.8.A, LGSOPS0092A   Student Ref: required No

Learning Objective: LLOT0370 #14, LGSOPS0092A IL4

Question source: New

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Presby

QUESTION 27

Unit 1 is in a refueling outage with fuel movement in progress. Unit 2 is starting up following a forced outage. Unit 2 Standby Gas Treatment system (SGTS) is running for containment venting.

The following annunciator windows are in alarm on panel 10C800:

- REACTOR ENCL AREA HI RADIATION
- 1 REAC ENCL REFUEL FLR VENT EXHAUST RAD MON A/B HI - HI / DOWNSCALE

The following annunciator window is NOT currently in alarm on Panel 10C800:

- 1 REAC ENCL REFUEL FLR VENT EXHAUST RAD MON C/D HI - HI / DOWNSCALE

Given the above conditions:

(1) What is the current status of Unit 1 reactor enclosure ventilation
(2) What is the current status of Unit 2 reactor enclosure ventilation

	Unit 1	Unit 2
A.	Isolated	Not Isolated
B.	Not isolated	Not isolated
C.	Not isolated	Isolated
D.	Isolated	Isolated

K&A # 295034 Secondary Containment Ventilation High Radiation

Importance Rating 3.9

QUESTION 27

K&A Statement: K2.04- Knowledge of the interrelations between SECONDARY CONTAINMENT VENTILATION HIGH RADIATION and Secondary Containment ventilation

Justification:

A. Correct – It only takes one HI-HI signal to isolate reactor enclosure HVAC. Reactor enclosure isolation is divided into two divisions. Each division controls one set of dampers. An isolation signal on either division will result in a full isolation of the reactor enclosure ventilation. Unit 2 reactor enclosure ventilation is separate from Unit 1. The isolation signal present is on Unit 1; therefore, Unit 2 ventilation is not affected.

B. In- Correct – It only takes one HI-HI signal to isolate reactor enclosure HVAC, so Unit 1 reactor enclosure HVAC is isolated. Plausible if the applicant thinks that two (2) Hi-HI inputs are required for an isolation on Unit 1.

C. In-Correct - Plausible if applicant thinks that Unit 1 A/B monitors are downscale based on the alarm, which would not cause an isolation on Unit 1. Applicant may believe that Unit 2 would be affected due to venting condition.

D. In-Correct – Plausible if applicant thinks both units HVAC would respond to the HI-HI rad condition on a Unit 1 instrument. The refuel floor HVAC is common and an isolation signal on one unit would also trip the same division on the other unit's logic. Possible that applicant would believe that a similar response would occur for reactor enclosure HVAC.

References: LLOT0200 pg. 6, 7, 28, 29; ARC 10800 E-1, 10C800B-4 — Student Ref: required No

Learning Objective: LLOT0200 Obj. 3

Question source: Modified Bank (Susquehanna)

Question History: None

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

10CFR 55.41(9) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 32

Unit 1 conditions are as follows:

- Reactor is in OPCON 1
- Core spray pumps “B” and “D” are operating in the full flow test mode

Core spray suction valve HV-52-1F001D position indication limit switch fails, producing a signal corresponding to an intermediate position.

WHICH of the following describes the resulting system conditions?

A. “D” core spray pump will continue running. Automatic and manual starts of “D” pump are not affected.

B. “D” core spray pump will continue running. Automatic start of pump is inhibited but manual start is not.

C. “D” core spray pump will trip. “CORE SPRAY OUT OF SERVICE” annunciator will actuate. Automatic start is inhibited, manual start is not.

D. “D” core spray pump will trip. “CORE SPRAY OUT OF SERVICE” annunciator will actuate. Automatic and manual starts are inhibited.

K&A # 209001 LPCS
Importance Rating 2.5

QUESTION 32

K&A Statement: A1.08 –Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including system lineup.

Justification:

A. Correct. The pump will continue running. Automatic and manual pump starts are not affected based on suction valve position.

B. In-Correct but plausible. The pump will continue to run. Plausible if the applicant believes that an interlock exits to protect the pump if the suction valve is not fully open.

C. In-Correct – The pump continue to run. The annunciator does not alarm. Plausible if the applicant believes that the pump would trip based on valve position to protect the pump if the suction valve is not fully open and that an interlock exits on the automatic pump start.

D. In-Correct – The pump continues to run. The annunciator does not alarm. Automatic and manual starts are not affected. Plausible if the applicant believes that the pump trips based on suction valve position and that interlocks exist on pump start.

References: LLOT0350 pg. 17, 18, S52.5.B, rev. 26 "Core Spray Header Flush and Vessel Floodup", Dwg. 052-01 rev.2; Dwg. 8031-M-52 rev. 48 Student Ref: required N

Learning Objective: LLOT0350 Obj. #6, 7, 8

Question source: Modified Bank (Hope Creek)

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 36

A Unit 2 turbine trip from full power has caused a reactor scram. RPV water level lowered to 10" during the initial transient, level has been restored to the normal operating band for two minutes. The scram has NOT BEEN RESET.

Select the answer which CORRECTLY describes the status of the RPS Backup Scram valves in this plant condition.

Both Backup Scram valves should be...

A. De-energized and venting the scram air header to the reactor building.

B. Energized and venting the scram air header to the reactor building.

C. De-energized and not venting the scram air header to the reactor building.

D. Energized and not venting the scram air header to the reactor building.

K&A # 212000 RPS
Importance Rating 3.4

QUESTION 36

K&A Statement: A1.08 –Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM including valve position.

Justification:

A. In-Correct Back up SCRAM valves energize to close. Plausible if the applicant believes that the backup SCRAM valves are de-energize to reposition, similar to the SCRAM valves.

B. Correct – Backup SCRAM valves are normally deenergized. Backup SCRAM valves energize to operate and reposition to vent the SCRAM air header. Since the SCRAM has not been reset, even though the SCRAM condition cleared, the valves would still be energized and venting.

C. In-Correct – The backup SCRAM valves have received an initiation signal and would have energized and closed. Plausible if applicant believes that the valves are de-energize to operate.

D. Correct - Backup SCRAM valves are energize to operate. The backup SCRAM valves receive an initiation signal on the SCRAM and would reposition. Since the SCRAM has not been reset, even though the SCRAM condition cleared, the valves would still be energized and venting. Plausible if the applicant believes that once the scram condition has cleared the valves would return to their de-energized state.

References: LGSOPS 0071 pg. 22 Student Ref: required N

Learning Objective: LGSOPS EO4, IL7

Question source: Modified Bank (Duane Arnold)

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Presby

QUESTION 41

WHICH ONE of the following describes the interlock associated with the RCIC pump suction valves?

A. The suction source will automatically swap from the CST to the Suppression Pool upon a low level in the CST.

B. The suction source must be manually swapped from the CST to the Suppression Pool upon a low level in the CST.

C. The suction source will automatically swap from the Suppression Pool to the CST upon a low level in the Suppression Pool.

D. The suction source will automatically swap from the Suppression Pool to the CST upon a high level in the Suppression Pool.

K&A # 217000 RCIC
Importance Rating 3.6

QUESTION 41

K&A Statement: K1.03 – Knowledge of the physical connections and/or cause-effect relationship between REACTOR CORE ISOLATION COOLING and Suppression pool

Justification:

A. Correct – The Suppression Pool suction valves automatically open on a CST low level after a 20 second time delay. The CST suction valve receives a close signal when the suppression pool suction valves are open.

B. In-Correct – The Suppression Pool suction valves automatically open on a CST low level. Plausible if the applicant recalls that the suppression pool suction valves are manual open and close but does not recall the automatic function.

C. In-Correct – There is not an automatic swap on suppression pool low level. Plausible if applicant thinks that on low suppression pool level the correct action would be to swap to the CST. If the pool is empty, then no suction source is aligned. The automatic swap from the CST to the Suppression Pool can be overridden by manual action.

D. In Correct - There is not an automatic swap on suppression pool low level. Plausible if applicant thinks that on high suppression pool level the correct action would be to swap to the CST.

References: LLOT0380 pg. 11, 13 Student Ref: required N

Learning Objective: LLOT0380 Obj. 7a, 7b

Question source: Modified Bank (Perry)

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 47

Given the following conditions:

- A valid initiation signal for Standby Gas Treatment is received
- SGTS Fan "A" is in AUTO
- SGTS Fan "B" is in STANDBY

WHICH ONE of the following will initiate the "B" fan of the Standby Gas Treatment System?

A. Refueling floor exhaust radiation level of 2 mR/hr

B. Reactor Enclosure exhaust radiation level of 10mR/hr

C. Low flow condition on Standby Gas Treatment System for 20 seconds

D. Low flow condition on Reactor Enclosure Recirc System for 20 seconds

K&A # 261000 SBGTS
Importance Rating 3.2

QUESTION 47

K&A Statement: A3.01- Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including System flow

Justification:

A. In-Correct – Since the "B" fan is in STBY, it will only start following a low flow condition on the system. Plausible if the applicant believes that in STBY the fan will start on a second initiation signal. Refuel Floor radiation of 2mR/hr is an initiation signal.

B. In-Correct - Since the "B" fan is in STBY, it will only start following a low flow condition on the system. Plausible if the applicant believes that in STBY the fan will start on a second initiation signal.

C. Correct – Since the "B" fan is in STBY, it will only start following a low flow condition on the system.

D. In-Correct - Since the "B" fan is in STBY, it will only start following a low flow condition on the SBGT system. Plausible if the applicant believes that the fan will start on a low flow condition of RERS.

References: LLOT0200 pg. 21, 28, 29 Student Ref: required No

Learning Objective: LLOT0200 3, 10b

Question source: Modified Bank (Dresden)

Question History:

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 49

During Unit 1 startup at 20% power the following alarms on PNL122 annunciate:

- 1B RPS & UPS STATIC INVERTER TROUBLE
- 1B RPS & UPS DIST PNL TROUBLE

The Reactor Operator announces there has been a half scram on Channel B.

What action must be taken to correct the RPS UPS problems?

A. Transfer the "B" RPS AC supply to the TSC UPS per S94.2.I "Placing the Technical Support Center Uninterruptable Power Supply (UPS) Modules in Service".

B. Transfer the "B" RPS AC supply to the TSC UPS per S94.2.B "Bypassing and Removing the *B RPS UPS Static Inverter from Service".

C. Transfer the "B" RPS AC supply to the Non-Safeguard 480 V AC per S94.2.B "Bypassing and Removing the *B RPS UPS Static Inverter from Service".

D. Transfer the "B" RPS DC supply to the ALTERNATE SOURCE per S94.2.B "Bypassing and Removing the *B RPS UPS Static Inverter from Service".

K&A # 262002
Importance Rating 2.7

QUESTION 49

K&A Statement: K6.03-Knowledge of the effect that a loss or malfunction of **Static Inverter** will have on the UNINTERRUPTABLE POWER SUPPLY System

Justification:

A. In-Correct. Right supply; wrong procedure. Plausible since TSC UPS is the preferred ALTERNATE supply to RPS.

B. Correct. With both alarms in, and a Half Scram the static switch did NOT transfer to the alternate source, so manual transfer is required (S94.9.A). The preferred AC source during a startup is the TSC UPS since the secondary source can experience a voltage dip whenever large motors are started.

C. In-Correct. Right procedure; wrong supply. During a startup should NOT use the non-safeguard due to potential voltage dips (LLOT 0650 pg 9). Plausiible since the applicant may assume the preferred AC supply was lost.

D. In-Correct. There is no manual transfer of DC. This is done automatically. Also the alarms indicate there has been a failure in the inverter so a second DC supply would not help. Plausible since there are two DC supplies to the UPS inverter (LLOT 0650, pg 9).

References: ARC 122 D-12, F-4, & A-5; LLOT 0650 pg 9; S94.2.B pgs 2 and 3, S94.9.A Student Ref: required NONE

Learning Objective: N/A

Question source: New

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Johnson
Reviewed by: Presby

QUESTION 53

Both units are operating at 100% power. The “1A” RWCU and “2B” and “2C” RWCU pumps are in service. In the event of an RECW High Radiation condition, procedure S13.0B directs operators to isolate suspected contamination sources sequentially in order to locate the source and terminate the leak. Which of the following is the LEAST likely potential source of radioactive water in leakage into the RECW system?

A. 1A RWCU Pump

B. 2B RWCU Pump

C. 1A RWCU Non-regenerative Heat Exchanger

D. 2B RWCU Non-regenerative Heat Exchanger

K&A # 400000 Component Cooling Water
Importance Rating 4.6

QUESTION 53

K&A Statement: G2.1.20 – Ability to interpret and execute procedure steps as it relates to COMPONENT COOLING WATER

Justification:

A. Correct. The A RWCU pump has a canned rotor designed pump motor. This is a modification from the original plant design to reduce pump seal leakage. Per note 1 on page 4 of S13.0B, "OF all suspected radiation sources identified in this procedure, the 1A RWCU pump is the least likely suspect.

B. In-Correct. The "B" pump is the original plant design. The "A" pump is a different design to reduce the likelihood of a pump seal leak. Per note 1 on page 4 of S13.0B, "OF all suspected radiation sources identified in this procedure, the 1A RWCU pump is the least likely suspect. Plausible if applicant does not remember the difference between the "A" and "B" RWCU pump designs.

C. In-Correct. Per note 1 on page 4 of S13.0B, "OF all suspected radiation sources identified in this procedure, the 1A RWCU pump is the least likely suspect. Plausible if applicant believes the heat exchanger is a less likely source of radiation than the "A" pump.

D. In-Correct. Per note 1 on page 4 of S13.0B, "OF all suspected radiation sources identified in this procedure, the 1A RWCU pump is the least likely suspect. Plausible if applicant believes the heat exchanger is a less likely source of radiation than the "A" pump.

References: LGOPS0013, LLOT 1570, LLOT 0110 Student Ref: required none

Learning Objective: LLOT 1570 Obj. 11e, LLOT 0110 Obj. 3

Question source: New

Question History: none

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 55

Unit 1 is initially at steady state 100% power

The hold down bolt on #14 Jet Pump failed due to intergranular stress corrosion cracking.

WHICH ONE of the following identifies the system response to the failure?

A. Indicated total core flow increases, reactor power increases

B. Loop "A" drive flow increases, reactor power increases

C. Indicated dp on Jet Pump #13 decreases, reactor power decreases

D. Loop "B" drive flow increases, reactor power decreases

K&A # 202001 Recirculation
Importance Rating 3.5

QUESTION 55

K&A Statement: K6.01 - Knowledge of the effect that a loss or malfunction of Jet Pumps will have on the Recirculation System

Justification:

A. In-Correct. Core power decreases on a jet pump failure. Indicated core flow will rise due to the addition of reverse flow through the failed jet pump. Plausible if applicant uses the fact that indicated flow increases and an increase in total core flow would cause power to increase.

B. In-Correct. Core power decreases on a jet pump failure. The recirc drive flow in the loop containing the failed jet pump will increase due to the decrease of flow resistance; however, total core flow decreases causing a decrease in core power.

C. Correct – Indicated dp on the jet pump sharing the riser with the failed jet pump decreases. The dp will drop due to the preferential flow of drive water out the failed jet pump. Actual total core flow decreases, causing a decrease in reactor power.

D. In-Correct. Loop "B" drive flow will not change. Actual power increases. Plausible if applicant does not remember that loop a discharges through jet pump 14 or believes that drive flow will increase in the unaffected loop.

References: ON-100; LGSOPS0043A, LLOT1550 Student Ref: required No

Learning Objective: LLOT1550 Obj. 1

Question source: New

Question History:

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

10CFR 55.41(2) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 60

Following a scram on High Drywell pressure, Unit 2 plant conditions are as follows:

- Reactor level is currently -140" inches and stable.
- All loops of RHR are unavailable.
- Both loops of Core Spray are injecting into the reactor vessel.
- Containment conditions require initiation of Drywell Sprays.
- RHRSW Loop 'A' is aligned for Drywell Spray up to the point of transitioning RHRSW flow from the 2A RHR Heat Exchanger to the Drywell Spray headers.
- Reactor pressure is 180 psig.
- RHRSW discharge pressure is 125 psig.
- Operators are preparing to perform step 4.6.13 of T-225, "Startup and Shutdown of Suppression Pool and Drywell Spray Operation" (step shown below)

NOTE

Step 4.6.13 will require coordination between an Operator at 00C667 **AND** a second Operator at 20C681

4.6.13 Simultaneously **PERFORM** the following to maintain RHR Service Water discharge pressure 75 to 120 psig as indicated on PI-12-001A-1, "Pump A/C Disch" (Px), at 00C667 Main Control Room):

- Throttle Fully **CLOSED** HV-51-2F068A, "2A RHR Htx SW Outlet Vlv" (2A) at 00C667 (Main Control Room). ________

CAUTION

Slowly throttling open Outboard Drywell Spray valve will prevent rapid pressure drop.

- Throttle Fully **OPEN** HV-51-2F016A, "2A RHR Cntmt Spray Line Outboard PCIV" (OUTBOARD) to initiate spray **AND MAXIMIZE** flowrate as indicated on FI-51-2R603A, FL. ________

Based on the above plant conditions, which one of the following describes an additional limit that must be observed while performing this step?

A. Ensure 2B RHR SW flow does not exceed 11,000 gpm AND a minimum Drywell Spray flow of 9,250 gpm.

B. Ensure RHR Service Water Pressure does not exceed 105 psig during flow transition.

C. Ensure Drywell Spray flow does not exceed 10,500 gpm.

D. Ensure RHR Service Water Pressure does not drop below 90 psig during flow transition.

K&A # 226001 RHR/LPCI: Containment Spray

Importance Rating 4.3

QUESTION 60

K&A Statement: G.2.1.23 – Ability to perform specific system and integrated plant procedures during all modes of plant operation, as it relates to RHR/LPCI: CONTAINMENT SPRAY.

Justification:

A. In-Correct but plausible since T-225 does limit flow through the RHR Heat Exchanger to 11,000 gpm when spraying the drywell with the RHR system.

B. Correct – With reactor pressure at 180 psig allowing the pressure to exceed 105 psig will result in LPCI injection valves getting an auto open signal due to RHR pressure to reactor pressure being less than 74 psid.

C. In-Correct but plausible since T-225 does provide a range of 9,500 to 10,500 gpm Drywell Spray flow when spraying the drywell with the RHR system.

D. In-Correct but plausible since T-225 does require that RHRSW pressure be maintained between 75 psig and 120 psig.

References: LLOT1561, Rev.007
T-225, Rev. 20

Student Ref. required No

Learning Objective: IL2, IL3

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: Johnson

QUESTION 61

Unit 2 Refueling operations are in progress with the following conditions:

- All rods are fully inserted in the core.
- The following functions/conditions have been verified for all SRMs:
  - Indicate fully inserted in the core.
  - Provide visual indication in the control room.
  - Capable of providing an audible alarm in the control room.
- The "Detector Not Full In" control rod block function associated with SRM 'B' is faulty.
- A new fuel assembly is positioned over the core and ready to be lowered into position 27-50.
- SRM 'C' fails downscale and is declared inoperable.

The list below provides the fuel move plan steps for placement of fuel assembly 27-50 and subsequent fuel moves in the core.

Step #1:	Place new fuel assembly in position 27-50
Step #2:	Place new fuel assembly in position 13-30.
Step #3:	Place new fuel assembly in position 47-14.
Step #4:	Remove spent fuel assembly from position 21-38.

Which one of the following describes the steps that may be performed and remain in compliance with T. S. 3.9.2 Refueling Operations - Instrumentation?

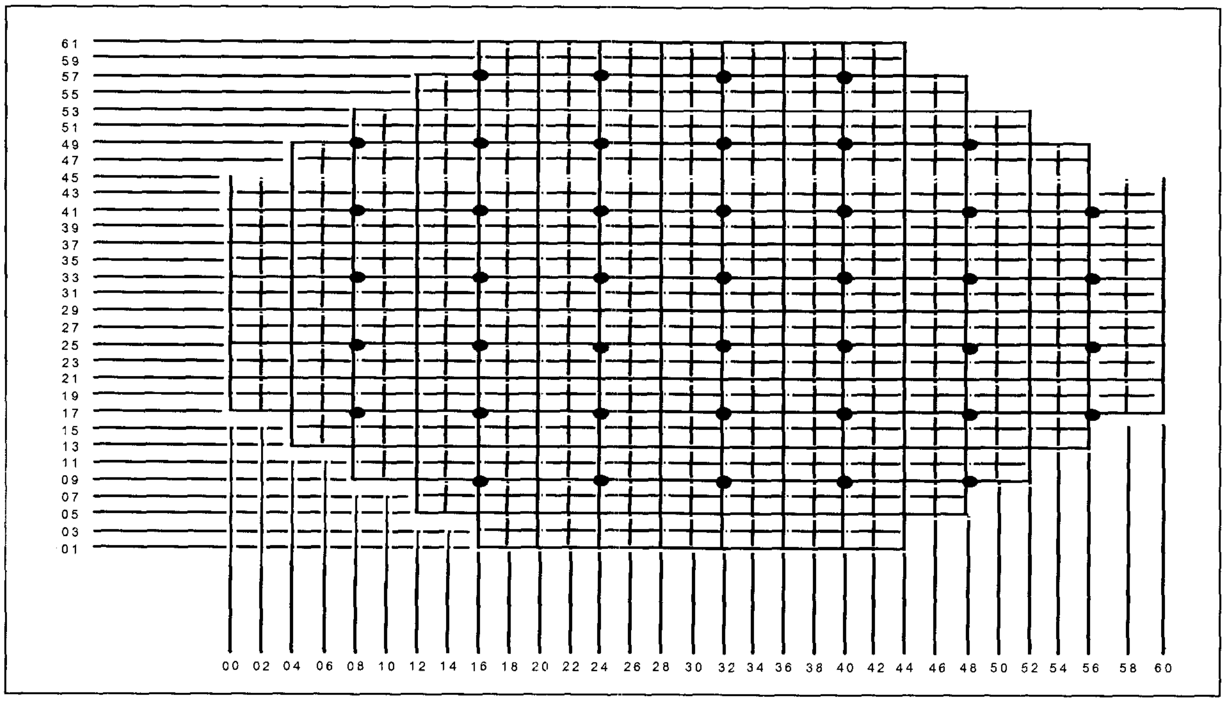
Core Map provided to locate positions of planned fuel moves.

A. Immediately stop all core alterations. Do not perform Step #1.

B. Fuel moves can continue up through Step #2. However, core alterations must be terminated prior to Step #3.

C. Fuel moves can continue up through Step #3. However, core alterations must be terminated prior to Step #4.

D. Fuel moves can continue up through Step #4.



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K&A #	234000 Fuel Handling Equipment
Importance Rating	G.2.2.39

QUESTION 61

K&A Statement: G.2.2.39 – Knowledge of less than or equal to one hour Technical Specification action statements for systems as they relate to Fuel Handling Equipment.

Justification:

A. In-Correct SRMs "A" and "B" are operable. Fuel assembly 27-50 is located in the same quadrant as SRM "A" and is adjacent to SRM "B". Plausible if the applicant did not understand that operability of the control rod block function has no effect on the SRM 'B' meeting the requirements of T.S. 3.9.2. As stated in the stem of the question, all SRMs are fully inserted into the core.

B. Correct -T.S. 3.9.2 requires two SRMs be operable; the SRM in the quadrant that the assembly is located and the adjacent SRM. Fuel assembly location 13-30 is in the same quadrant as SRM "D" and adjacent to SRM "A".; therefore, this assembly can be lowered into the core. Fuel assembly location 47-14 is located in the same quadrant as SRM "C"; therefore this assembly cannot be lowered into the core and all core alterations must be immediately suspended prior to step #3.

C. In-Correct – T.S. 3.9.2 requires two SRMs be operable for core alterations. All core alterations must be immediately suspended if "...One of the required SRM detectors located in the **quadrant where CORE ALTERATIONS** are being performed **and** the **other required SRM detector located in an adjacent quadrant**" is not operable. In this instance, fuel position 47-14 is in the same quadrant SRM 'C' and would require suspension of core alterations prior to performing this fuel movement, since SRM "C" is inoperable.

D. In-Correct but plausible if the applicant believed that as long as two SRMs were operable adequate coverage was provided. It would also seem reasonable, since removing fuel from the core would increase shutdown margin decreasing the likelihood of inadvertent criticality.

References: LLOT0760, Rev. 014
LLOT0240, Rev. 009
Student Ref. required No

Learning Objective: Obj 12 (LLOT0760), Obj. 3 (LLOT0240)

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 62

Unit 2 is at 23% power when steam is inadvertently isolated to the auxiliary steam loads supplied from MSL 'B'.

Assuming no operator action, which one of the following describes the effect this will have on plant operation?

A. "B" reactor feedpump turbine will swap to the low pressure steam supply

B. Turbine trip on low condenser vacuum.

C. Main Condenser vacuum will remain constant

D. Reactor scram on low reactor water level.

K&A # 239001 Main Steam and Reheat System

Importance Rating 2.9

QUESTION 62

K&A Statement: K1.08 – Knowledge of the physical connections and/or cause-effect relationships between MAIN AND REHEAT STEAM SYSTEM and Condenser Air Removal.

Justification:

A. In-Correct. On unit 2, the reactor feed pump turbines are supplied by the "C" main steam line. Plausible since on unit 1, the reactor feed pump turbines are supplied byt the "B" main steam line.

B. Correct – On Unit 2, the condenser air removal system is supplied from the Main Steam system off of 'B' MSL which would affect the Steam Jet Air Ejectors (SJAE) and Steam Seal Evaporator (SSE). Loss of steam flow to the SJAEs would result in non-condensable gases building up in the condenser. The Turbine will trip when condenser vacuum degrades to 21" Hg Vac.

C. In-Correct. Loss of steam flow to the SJAEs would result in non-condensable gases building up in the condenser.

D. In-Correct but plausible since on Unit 1 steam to the RFPTs is supplied from 'B' MSL. Loss of high pressure steam to the RFPTs could result in a low reactor level condition because of inadequate low pressure steam supply pressure to the feed pump turbines at low main turbine power level. However, Unit 2 high pressure steam supply to the RFPTs is from 'C' MSL.

References: LGSOPS0007, Rev. 000   Student Ref. required   No
LGSOPS0069, Rev. 000
LLOT1870, Rev.001

Learning Objective: IL3 (LGSOPS0007), IL3 (LGSOPS0069), Obj 1 (LLOT1870)

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 63

Unit 2 plant conditions are as follows:

- Rx power is 24%
- Generator load is 245 MWe
- Turbine 1st stage pressure is 178 psig
- Three bypass valves are isolated and tagged because of leak-by (BPVs 1, 2 & 3)

A grid problem results in a full load rejection.

Which of the following describes (1) the expected response of the turbine control valves (TCV) and the reactor protection system (RPS) and (2) reactor pressure response?

A. (1) TCVs close on fast closure signal; Rx scrams on TCV closure
(2) Reactor pressure will stabilize post-scram on SRVs

B. (1) TCVs close on speed error signal; Rx scrams on TCV closure
(2) Reactor pressure will stabilize post-scram on bypass valves

C. (1) TCVs close on fast closure signal; Rx scrams on high reactor pressure
(2) Reactor pressure will stabilize post-scram on SRVs

D. (1) TCVs close on speed error signal; Rx scrams on high reactor pressure
(2) Reactor pressure will stabilize post-scram on bypass valves

K&A #	245000 Main Turbine Gen./Aux
Importance Rating	3.5

QUESTION 63

K&A Statement: A1.05 – Ability to predict and/or monitor changes in parameters associated with operating the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEM controls including Reactor pressure

Justification:

A. In-correct. The TCV fast closure signal is generated on load reject through a power-load imbalance circuit, which requires power greater than setpoint. The fast closure signal will not be generated because reactor power is less than 40% equivalent as measured by turbine 1st stage pressure. At the given power level, TCVs will close on a speed error signal, not a fast closure signal. Plausible if applicant thinks load reject fast closure enabled at current conditions.

B. In-correct. Reactor does not scram on TCV closure. Scram enabled at 1st stage pressure >190 psig. Plausible if applicant thinks scram is enabled.

C. In-correct. Reactor pressure does not stabilize post-scram on SRVs. Plausible if applicant thinks reactor pressure will continue to increase post-scram to SRV setpoint and control on SRVs. SRV lift setpoints are 1170 psig, 1180 psig and 1190 psig.

D. Correct. Speed error signal will develop to close TCVs as speed approaches 105% of rated speed. The bypass valves will open in response to rising steam throttle pressure. However, the capacity of remaining valves (6 of 9 BPVs) is insufficient at 2/3 of 25% rated power capacity, or 16.7% rated power capacity. Reactor pressure will increase to 1096 psig trip setpoint. Upon reactor scram, reactor power will drop to less than the capacity of the available bypass valves. Reactor pressure will return to normal and control on the bypass valves at approximately 960 psig.

References:
- LGSOPS0032 pg. 12, LGSOPS0001A pg. 7, 10, Dwg. 071-02a
- RPS LP LGSOPS0071, page 13 of 48 (states 1st stg >190 equivalent to 25% nominal rx pwr
- TS 3.3.1, Table entry j, states equiv to > 30% rated thermal power
- GP-3, page 22 of 68, Rev 124, states setpoint is rx power = 25.5%
- Main turbine LP, EHC LP, Main Steam LP
- Dwg. 001-004, SRV setpoints
- Dwg. 034-04, EHC logic

Student Ref Required: No

Learning Objective: LGSOPS0001A EO2, EO4, IL2

Question source: Modified Bank (Hope Creek) Changes: Init conditions, removed failure of PLU relay, added RPV pressure response

Question History: HC NRC 1998

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

10CFR	55.41(7) X
Comments:	Created/Modified by: Tomlinson Reviewed by:

QUESTION 65

Unit 1 is initially at steady state 100% power. Reactor Building Ventilation System lineup is as follows:

- "A" and "C" supply fans in run, "B" supply fan in auto
- "A" and "B" exhaust fans in run, "C" exhaust fan in auto

All running exhaust fan discharge dampers close on a spurious signal.

Which ONE of the following states the automatic response of the Reactor Building Ventilation System?

A. Exhaust Fan "C" will automatically start to reduce reactor building pressure.

B. Exhaust Fans "A" and "B" will automatically restart to reduce reactor building pressure.

C. Supply Fans "A" and "C" will trip after a time delay to stop the increase in reactor building pressure.

D. Supply Fans "A" and "C" will trip immediately to stop the increase in reactor building pressure.

K&A # 290001 Secondary Containment

Importance Rating 3.4

QUESTION 65

K&A Statement: K4.02 - Knowledge of SECONDARY CONTAINMENT SYSTEM design feature(s) and/or interlocks which provide protection against over pressurization Plant System

Justification:

A. In-Correct- The "C" pump will automatically start but will not be able to reduce pressure with two supply fans running.

B. In-Correct – There is no auto restart signal for the exhaust fans after they trip.

C. Correct – The supply fans will trip after a time delay due to less than two exhaust fans running.

D. In-Correct – The supply fans will trip after a time delay to allow the exhaust fan in auto to attempt to start.

References: LLOT0200 pg. 6, 9 Student Ref: required No

Learning Objective: LLOT0200 Obj. 6, 8

Question source: Modified Bank (Dresden)

Question History:

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

10CFR 55.41(9) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 67

The following sequence of events occur on Unit 2:

- MCR Annunciator 107 C-1, SCRAM DISCHARGE VOLUME HIGH LEVEL TRIP is lit.
- RPS failed to de-energize.
- Four (4) SRVs opened automatically to control reactor pressure.

Subsequently, all but five (5) control rods fully insert.

Current indications are as follows:

- Blue lights on the Full Core display for these five rods are not lit.
- CRD system flow indicates 36 gpm.
- CRD Cooling Water flow indicates 33 gpm.
- CRD Drive Water flow indicates 3 gpm.

Which one of the procedures listed in the choices below provides the preferred method to insert the control rods that did not fully insert?

A. T-215, De-energization of Scram Solenoids.

B. T-217, RPS/ARI Reset And Backup Method Of Draining Scram Discharge Volume.

C. T-218, Control Rod Insertion by Withdrawal Line Venting.

D. T-219, Maximizing CRD Cooling Water Header Flow During ATWS Conditions.

K&A # Plant Generic
Importance Rating 4.3

QUESTION 67

K&A Statement: G.2.1.23 – Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Justification:

A. In-Correct but plausible since this is one of the procedures listed in T-101 (step RC/Q-13) to manually insert rods with scram valves closed (i.e., blue light on Full Core Display ON). However, this method is only recommended if available SDV. Typically, CRD flow would be full scale immediately following a scram and CRD flow control valves would close to direct flow to the Charging Header to recharge the HCUs. The system flow readings indicate low CRD flow with all the flow either being directed to the Drive Header or the Cooling Water header, which would be an indication that the Charging Header and downstream are hydraulically locked.

B. In-Correct but plausible since this is one of the procedures listed in T-101 (step RC/Q-13) to manually insert rods. However, this method is directed if the scram valves on the individual rod are open (i.e., blue light on Full Core Display ON).

C. In-Correct but plausible since this is one of the procedures listed in T-101 (step RC/Q-13) to manually insert rods and would be effective regardless of the available capacity in the SDV. However, this method is directed if the scram valves on the individual rod are open (i.e., blue light on Full Core Display ON).

D. Correct – RPS failed to scram the reactor. However, RRCS energized the ARI valves based on the high reactor pressure (>1149 psig) and depressurized the Scram Air Header and isolated the Scram Discharge Volume. With the scram valves closed (i.e., blue lights on Full Core Display ON) and the system hydraulically locked, T-219 is the recommended method for inserting the rods.

References: LGSOPS0046 Rev001
LLOT1560, Rev 007

Student Ref. required Yes
T-101

Learning Objective: IL 5(LLOT1560)

Question source: Modified Bank (Limerick)

Added CRD flow and scram valve ind. Changed scram. Ans now driven by hydraulically locked SDV vs determining scram valve position

Question History: LGS NRC-05, OYS CERT-04

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 68

Unit 2 plant conditions are as follows:

- Unit 2 is at 100% power
- "2A" CRD pump tripped 10 minutes ago
- Annunciator CRD HYDRAULIC HI TEMP (108 G-5) is lit
- Plant operator reports high temperatures greater than 250 degrees on 30 CRDs

A plant operator is placing CRD Pump 2B in service using procedure S46.1.A, Control Rod Drive Hydraulic System Startup.

> **2A PUMP ONLY**
>
> **4.16 Slowly THROTTLE 046-2014A, "A CRD Pump Min-flow Stop Valve," approximately 5 - 53/4 turns in close direction from full open position.**
>
> **2B PUMP ONLY**
>
> **4.17 slowly THROTTLE 046-1014B, "B CRD Pump Min-Flow Stop Valve," approximately 5 ¼ - 5 ½ turns in close direction from full open position.**

WHICH ONE of the following describes the proper actions based on the conditions above?

A. CRS initiates a Procedure Problem Identification System issue (PPIS), then plant operator repositions 046-2014B.

B. Plant operator repositions 046-2014B to correctly lineup the system to immediately restore cooling flow to reactor recirc pumps and CRDs.

C. CRS processes a Temporary Change (TC), then plant operator repositions 046-2014B.

D. Plant operator repositions valve 046-2014B to correctly lineup the system to immediately restore cooling flow to reactor recirc pumps and CRDs, then CRS processes a Temporary Change (TC) to document system alignment.

K&A # Equipment Control
Importance Rating 3.0

QUESTION 68

K&A Statement: G.2.2.6 – Knowledge of the process for making changes to procedures.

Justification:

A. In-Correct – Errors in equipment numbers shall be considered TC's. The change is beyond a PPI.

B. In-Correct – Procedure must be corrected prior to manipulating the valve.

C. Correct. Per HU-AA-104, the procedure must be addressed first. Errors in equipment numbers shall be considered TC's.

D. In-Correct. Per HU-AA-104, the procedure change must be addressed before the valve can be manipulated.

References: HU-AA-104 Student Ref: required No

Learning Objective: None

Question source: Bank (Limerick)

Question History:

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

10CFR 55.41(10) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Hansell

QUESTION 69

Unit 1 initial plant conditions are as follows:

- Reactor power is 72%
- Reactor level is +35 inches
- RPV steam dome pressure is 1035 psig

WHICH ONE of the following describes a condition that will violate a Unit 1 Technical Specification Safety Limit? Assume automatic features perform as designed and initial operator actions are performed as appropriate.

A. Drywell pressure rises to 60 psig

B. Reactor level drops to -170 inches

C. Reactor pressure rises to 1280 psig

D. Minimum Critical Power Ratio (MCPR) lowers to 1.08

K&A # Equipment Control
Importance Rating 4.0

QUESTION 69

K&A Statement: 2.2.22– Knowledge of limiting conditions for operations and safety limits

Justification:

A. In-Correct- Drywell pressure is not a safety limit but plausible since it exceeds the maximum design pressure for the drywell.

B. Correct – Although the limit does not apply in OPCON 1, the plant will scram at +12.5 inches (decreasing) and be in an applicable OPCON at that point.

C. In- Correct – Reactor pressure rising to 1280 psig does not exceed a Safety Limit but plausible since it is above the Safety valve setpoint.

D. In-Correct Plausible since MCPR must be greater than 1.09 with one recirc loop in operation. Based on plant conditions, 2 two loops of recirc are running and safety limit is 1.07.

References: Tech Spec. 2.1, LGSOPS1800

Student Ref: required No

Learning Objective: N/A

Chg: init pwr level, MCPR distractor.

Question source: Modified (Limerick)

Question History:

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

10CFR 55.41(5) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 70

During Unit 2 operation at 100% power, an Equipment Operator (EO) discovers Trip Unit B21-2N693B to be tripped high (HPCI level 8 trip).

Which ONE of the following gives the status of HPCI operability?

A. HPCI is tripped and therefore inoperable.

B. HPCI is operable, the inoperable trip unit must be repaired within 24 hours or HPCI declared inoperable.

C. HPCI will not restart on a subsequent low level signal after tripping on a high level signal and is therefore inoperable.

D. HPCI is operable, the inoperable trip unit is in a tripped condition, therefore power operation can continue indefinitely.

K&A #	Equipment Control
Importance Rating	3.6

QUESTION 70

K&A Statement: G2.2.37- Ability to determine operability and/or availability of safety related equipment

Justification:

A. In-Correct. Four level 8 trip units are required to initiate a HPCI turbine shutdown. Plausible if applicant believes that the HPCI turbine will not start with a level 8 trip unit failed upscale.

B. Correct - HPCI is operable but the trip unit must be repaired within 24 hours or then HPCI must be declared inoperable.

C. In-Correct – Level 8 trip units supply indication for a HPCI turbine trip and are not part of the initiation logic. Plausible if applicant believes the upscale level will prevent a low level signal from initiating a HPCI start.

D. In-Correct Per Tech Spec Table 3.3.3-1 Action 30, HPCI must be declared inoperable with in 24 hours if the trip unit can not be made operable.

References: TS 3/4 3-35, 3/4 3-33, 3/4 3-36, LLOT0340 pg. 32

Student Ref: required TS 3/4 3-35, 3/4 3-33, 3/4 3-36

Learning Objective: N/A

Question source: Bank (Limerick)

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 72

A startup is in progress on Unit 2. A Drywell Entry at Power is planned to inspect for signs of leakage following replacement of RPV Instrumentation Condensing Pot XY 1D004A.

Using supplied references, determine the maximum allowed reactor power level and the radiological restriction(s) that must be met per RP-AA-460-105, Drywell Entries at Power.

A. Maintain Reactor Power no more than 3%.
AND
Identify a low-dose zone for workers during performance of radiological surveys.

B. Maintain Reactor Power no more than 7%.
AND
Identify a low-dose zone for workers during performance of radiological surveys.

C. Maintain Reactor Power no more than 3%. Identification of a low-dose zone for workers during performance of radiological surveys is NOT required.

D. Maintain Reactor Power no more than 7%. Identification of a low-dose zone for workers during performance of radiological surveys is NOT required.

K&A # Plant-wide Generics
Importance Rating 3.2

QUESTION 72

K&A Statement: G.2.3.12 – Knowledge of radiological safety practices pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Justification:

A. In-Correct but plausible if the applicant determines the instrument to be located at the 296' Elevation. Step 4.1.2.1 of RP-LG-460-105 states that Reactor Power Levels shall be maintained 3% or less for access to 296'elevation. Also, step 4.9.4 states "***If** access is required on **296'** elevation **or** within 6 feet of bioshield penetrations on 277' (N1, N2, or N8), **then USE** the following special precautions and radiological survey techniques:*". One of the special precautions states "***IDENTIFY** low dose area **and ALLOW** worker to stand in low dose area (i.e., stairwell or outside of drywell) while radiation surveys are performed.*".

B. In-Correct but plausible if the applicant determines the instrument to be located at the 277' Elevation. Step 4.1.2.2 of RP-LG-460-105 states that Reactor Power Levels shall be maintained 7% or less for access to **277'**, 286', 303' or 313' elevations. Also, step 4.9.4 states "***If** access is required on 296' elevation **or** within 6 feet of bioshield penetrations on **277'** (N1, N2, or N8), **then USE** the following special precautions and radiological survey techniques:*". One of the special precautions states "***IDENTIFY** low dose area **and ALLOW** worker to stand in low dose area (i.e., stairwell or outside of drywell) while radiation surveys are performed.*".

C. In-Correct but plausible if the applicant determines the instrument to be located at the 296' Elevation and doesn't read the other restriction concerning the establishment of a low dose area. *See explanation in choice 'A'.*

D. Correct – The applicant must first determine the elevation Condensing Pot XY 1D004A is located. Using M-42 Sheet 1, it can be determined that Condensing Pot XY 1D004A comes off RPV penetration N12 and that Vessel Zero is 266' 3". Using Figure 1: Elevation Correlation Chart on M-42 Sheet 2, it can be determined that N12 is 599.0" above Vessel Zero, which would put the Condensing Pot on 313' Elev.

(266' 3" + 599.0 = 266' 3" + 49' 11 = 316' 2" or ~ 313').

Step 4.1.2.2 of RP-LG-460-105 states that Reactor Power Levels shall be maintained 7% or less for access to 277', 286', 303' or 313' elevations.

References: LLOT1760, Rev. 010
ON-122, Rev.017

Student Ref. required Yes
RP-LG-460-105, Drywell Entries at Power.
M-42, Sheets 1&2

Learning Objective: Obj. 11 (LLOT1760)

Question source: New

Question History:

Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	 X
10CFR	55.41	X
Comments:	Created/Modified by: M. Riches Reviewed by: P. Presby	

QUESTION 73

Unit 1 experienced a LOCA/LOOP approximately 10 minutes ago. The crew is implementing T-101, RPV CONTROL and T-102, PRIMARY CONTAINMENT CONTROL. Plant parameters are as follows:

- Reactor pressure is 450 psig
- Reactor power is < 4%
- Three control rods are at position 04
- Suppression Pool temperature is 110°F
- Suppression pool level is 23 feet
- Drywell pressure is 10 psig
- Drywell temperature is 351°F
- RPV Level Indicators LI-42-1R606 A & B are upscale; R606C is reading 0".
- All other RPV level indications are downscale or at 0"

Given the above conditions, what action is required to ensure the integrity of the fuel cladding is maintained?

A. Raise injection into the RPV to establish 5 ADS/SRVs open and RPV pressure not going down and at 60 psig.

B. Slowly raise injection into the RPV to establish 5 ADS/SRVs open and RPV pressure at 260 psig.

C. Restore and maintain RPV level between +12.5" and +54".

D. Exit RC/L of T-101, RPV CONTROL and enter T-117 LEVEL/ POWER CONTROL.

K&A #	Equipment Control
Importance Rating	4.0

QUESTION 73

K&A Statement: G2.4.21- Knowledge of parameters and logic used to assess the status of safety functions such as reactivity control, core cooling, reactor coolant system integrity, containment conditions, radioactivity release control, etc

Justification:

A. In-Correct. This action is taken ONLY if there is no ATWS. With three rods at 04, ATWS conditions apply. Plausible if applicant does not follow path from T-101 to ATWS leg of T-116.

B. Correct. This is the action required since the containment conditions and RPV level indication are indicative of a loss of level indication. Level is unknown because level instrumentation is providing opposing readings. Entry into T-116 is required and the ATWS path is required.

C. In-Correct. This action requires knowledge of RPV level, which is not satisfied by stem conditions. Plausible since this is an early action in T-101.

D. In-Correct. This action requires knowledge of RPV level, which is not satisfied by stem conditions. Plausible since this is an action in T-101 if an ATWS condition is present (which it is).

References:	T-101, T-102, T-291, T-116, Step RF-15	yes	T-101 and T102

Learning Objective: N/A

Question source: Bank (Limerick)

Question History: None

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

10CFR 55.41(7) X

Comments: Created/Modified by: Johnson
Reviewed by: Johnson

QUESTION 74

Unit 2 initial plant conditions are as follows:

- Reactor power is 40%.
- RPV level is 32".
- The following MCR alarm windows annunciate on Panel 212
    - D-2, 2B RECIRC PUMP MOTOR TRIP
    - E-3, 2B RECIRC M-G GENERATOR LOCKOUT TRIP
- Reactor Recirc flows are:
    - RRP 'A' Drive flow = 44,000 gpm
    - RRP 'B' Drive flow = 0 gpm

Based on the above conditions, which one of the following choices contains the correct "applicable" immediate operator actions including the proper sequence in accordance with OT-112, Recirculation Pump Trip?

A. Immediately manually scram the reactor.
Manually control RPV level as necessary until RPV level is normal.

B. Manually control RPV level as necessary until RPV level is normal.
If a scram condition occurs then enter T-100 Scram / Scram Recovery.

C. Manually adjust RRP drive flow to ensure plant is outside the OPRM Trips Enabled region.
If a scram condition occurs then enter T-100 Scram / Scram Recovery.

D. Manually adjust RRP 'A' flow to ensure plant is outside the OPRM Trips Enabled region.
Manually control RPV level as necessary until RPV level is normal.

K&A # Plant-wide Generic
Importance Rating 4.6

QUESTION 74

K&A Statement: G2.4.49 – Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Justification:

A. In-Correct but plausible since these would be the actions if both recirculation pumps were running.

B. Correct. With only one RRP trip, the first "applicable" immediate actions are controlling RPV level and scramming if a scram condition occurs.

C. In-Correct but plausible since the tripping of an RRP can cause the plant to enter the area where OPRM trips are enables (i.e., THI possible). The second action (Scram if a scram condition occurs) is a correct immediate action.

D. In-Correct but plausible since the tripping of an RRP can cause the plant to enter the area where OPRM trips are enables (i.e., THI possible). The second action (RPV level control) is a correct immediate action.

References: LLOT1540, Rev. 009
OT-112

Student Ref: required No

Learning Objective: Obj 2, (LLOT1540)

Question source: New

Question History:

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 75

Unit 2 is operating at 100% power. The following alarm windows annunciate on MCR annunciator panel 211:

- A-1, A RECIRC PUMP SEAL STAGE HI/LO FLOW
- A-2, 2A RECIRC PUMP SEAL LEAKAGE HI FLOW

The 2A RRP seal indications are as follows:

- Seal #1 pressure 480 psig
- Seal #2 pressure 460 psig

Which of the following resulted in the above alarms and indications on the 2A recirculation pump?

A. Only Seal #1 has failed.

B. Breakdown orifice between the seals has plugged.

C. Only Seal #2 has failed.

D. Both Seal #1 and Seal #2 have failed.

K&A #	Plant-wide Generic
Importance Rating	4.2

QUESTION 75

K&A Statement: G2.4.31 – Knowledge of annunciator alarms, indications, or response procedures.

Justification:

A. In-Correct but plausible since the pressures would be about equal in both seal #1 and seal #2. However, the pressure would be much higher (~1,000 psig) if seal #1 failed. Also, only alarm A-1 would be received if seal #1 failed.

B. In-Correct but plausible since both of the listed alarms (A-1 and A-2) would be received if the breakdown orifice plugged. However, pressure on #2 seal would decrease while pressure on #2 seal would remain at approximately 1,000 psig.

C. In-Correct but plausible since both of the listed alarms (A-1 and A-2) would be received if #2 seal failed. However, pressure on #1 seal would remain approximately 1,000 psig and #2 seal pressure would drop below 500 psig.

D. Correct. Normal pressure on the #1 and #2 seals would be 1,000 psig and 500 psig, respectively. Approximately equally low pressure on both seals is an indication of failure of both pump seals.

References: LGSOPS0043A, Rev. 001 — Student Ref: required No

Learning Objective: IL 6

Question source: Modified Bank (Brunswick) — Modified conditions to change answer from plugging #1 seal breakdown orifice to failure of both seals.

Question History: BRWK NRC-07 Exam

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

10CFR 55.41 X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 77

Unit 1 plant conditions are as follows:

- Unit startup is in progress.
- All APRMs indicate 28%.
- Reactor pressure is 980 psig.
- Pressure control is on Pressure Set.
- Turbine first stage pressure is 186 psig.
- Main Generator output is 276 MWe.

Subsequently, the following MCR alarm windows annunciate.

- 105 J-1, EHC EMERGENCY TRIP PRESS LO PRESS TRIP
- 105 H-1, EHC HYD FLUID LO PRESS TRIP

Which one of the following describes the expected plant response to the above alarms AND the procedure to respond to the event?

A. Main Turbine trips and reactor power increases. Enter OT-104, Unexpected/Unexplained Positive or Negative Reactivity Insertion.

B. Main Turbine trips, Reactor Recirculation Pumps trip, and reactor power decreases. Enter OT-112, Recirculation Pump Trip.

C. Main Turbine trips and reactor scrams. Enter T-100, Scram / Scram Recovery.

D. Main Turbine trips, Reactor Recirculation Pumps trip and reactor scrams. Enter T-100, Scram / Scram Recovery.

K&A #	295005 Main Turbine Generator Trip
Importance Rating	3.9

QUESTION 77

K&A Statement: A2.05 – Ability to determine and/or interpret **Reactor power** as it applies to MAIN TURBINE GENERATOR TRIP.

Justification:

A. Correct. A turbine trip occurs on low EHC pressure and results in a reactor scram and Reactor Recirculation Pump (RRP) trip **if** first stage pressure $\geq$ 190 psig power which correlates to a power level of 25%. Under the current conditions (i.e first stage pressure < 190 psig) the turbine will still trip. But instead of a reactor scram, reactor power will increase due to increased feedwater subcooling since feedwater heating is lost when extraction steam is isolated by the turbine trip.

B. In-Correct. Plausible since the RRPs trip on a turbine trip when power is above 25%. In addition, reactor power would be expected to decrease due to reduced subcooling margin on the reactor coolant inlet as a result of the RRPs tripping. However, the 25% power setting is based on the corresponding turbine first stage pressure of 190 psig. At a turbine first stage pressure of 186 psig, a reactor recirculation trip will not occur.

C. In-Correct. Plausible since a reactor scram is the expected response when power is above 25%. However, the 25% reactor power setting is based on a corresponding turbine first stage pressure of 190 psig. At a turbine first stage pressure of 186 psig, a reactor scram will not occur.

D. In-Correct. Plausible since a RRP trip and reactor scram are expected responses when power is above 25%. However, the power level setting is based on a corresponding turbine first stage pressure of 190 psig. At a turbine first stage pressure of 186 psig, neither a reactor scram nor a RRP trip will occur.

References: LGSOPS0001A, Rev. 000 Student Ref: required No

Learning Objective: IL3, IL4

Question source: New

Question History:

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 78

Given the following. conditions:

- Unit 2 is operating at 80% power.
- Unit 1 is shutdown for a maintenance outage.
- Plant operator notifies control room of smoke outside the Remote Shutdown panel cage
- Smoke and noxious odor is coming from the back of Unit 2 main control room panels.
- Smoke and odor quickly engulf the control room.
- Shift Supervisor orders IMMEDIATE evacuation of the control room.

Which of the statements below describes Unit 2 reactor operator IMMEDIATE actions prior to leaving the control room and the proper procedure for shifting control of equipment outside the main control room?

A. Manually scram the reactor, place mode switch to SHUTDOWN, close MSIVs, obtain Remote Shutdown key ring from the Shift Manager Key Locker and enter SE-1-2.

B. Manually scram the reactor, place mode switch to SHUTDOWN, trip the turbine, close MSIVs and enter SE-1-2.

C. Manually scram the reactor, place mode switch in shutdown, trip main turbine, close MSIVs and enter SE-6.

D. Manually scram the reactor, place the mode switch to SHUTDOWN, close MSIVs, initiate RCIC and enter SE-6.

K&A #	295016 Control Room Abandonment
Importance Rating	4.2

QUESTION 78

K&A Statement: 2.4.11 - Knowledge of Abnormal Condition procedures as it applies to CONTROL ROOM ABANDONMENT

Justification:

A. In-Correct –Obtaining remote shutdown keys is not an immediate operator action. Plausible since SE-1 lists obtaining keys as a follow-up action, prior to leaving the control room. Plant conditions require entry into SE-6 since Remote Shutdown panel is not accessible due to fire in area. Entry into SE-1-2 is required on loss of Offsite Power.

B. In-Correct –Plant conditions require entry into SE-6 since Remote Shutdown panel is not accessible due to fire in area. Entry into SE-1-2 is required on loss of Offsite Power.

C. Correct – SE-1 immediate operator actions are listed. Entry into SE-6, Alternate Remote Shutdown, is required because the Remote Shutdown Panel is not accessible due to fire in the area.

D. In-Correct – Initiating RCIC is not an immediate operator action per SE-1, Remote Shutdown. Plausible since initiating RCIC is a follow-up action in SE-1.

References: SE-1, Remote Shutdown, LLOT0735 Student Ref: required No

Learning Objective: None

Question source: Modified Bank (LaSalle)

Question History: none

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

10CFR 55.43(5) X

Comments: Created/Modified by: Tomlinson
Reviewed by: Hansell

QUESTION 79

Unit 1 is shutdown with the “A” RHR loop in Shutdown Cooling with the following plant conditions:

Reactor pressure 25 psig
Reactor Level is +60 inches
MSIVs are closed
Main Turbine is tripped

The following alarm annunciates 1A RPS & UPS DIST PNL TROUBLE

Considering the above conditions, select the procedure that should be implemented first.

A. S44.0.A Operating RWCU to Meet Plant Conditions

B. ON-113 Loss of RECW.

C. OT-112 Recirculation Pump Trip

D. ON-121 Loss of Shutdown Cooling.

K&A # 295021 Loss of Shutdown Cooling

Importance Rating 3.9

QUESTION 79

K&A Statement: G2.1.20– Ability to interpret and execute procedure steps as it relates to LOSS OF SHUTDOWN COOLING

Justification:

A. In-Correct. This is not a high priority with the plant shutdown. Plausible since a loss of RWCU will occur and recovery will be needed to assist with RCS level control.

B. In-Correct. This is not a high priority with the plant shutdown. Plausible since a loss of RECW will occur.

C. In-Correct. This is not a high priority with the plant shutdown. Plausible since both reactor recirculation pumps will receive a tripped signal (turbine is tripped).

D. Correct. The stem conditions provide that Shutdown Cooling suction isolation valves have isolated (Loss of 1A RPS UPS POWER). Since there is no other viable decay heat removal method available then the first thing to address is the loss of Shutdown Cooling.

References: On-113, OT-112, OT-121, S44.0.A, ARC -120 F5, LLOT0370, LLOT1550 Student Ref: required No

Learning Objective: LLOT1550 Obj.1

Question source: INPO bank

Question History:

Cognitive level: Memory/Fundamental knowledge:

Comprehensive/Analysis: X

10CFR 55.43(5) X

Comments: Created/Modified by: Johnson
Reviewed by: Hansell

QUESTION 80

Operators on Unit 2 initiated a manual scram from full power due to lowering water level in the Suppression Pool. Subsequent to the scram, the Turbine Bypass Valves failed to open.

Upon completion of immediate scram actions, the crew has entered the applicable TRIP procedures and observes the following post-scram initial indications:

- Reactor level is +22 inches and slowly rising
- Suppression pool level is at 17' 6" and slowly dropping.
- Reactor pressure is cycling between 1170 and 1180 psig.
- Suppression Pool temperature is 168°F.
- All RHR pumps are available, but are not currently operating.

Based on these conditions, which one of the following actions should be taken?

A. Enter T-112 "Emergency Blowdown" and perform concurrently to T-102, "Primary Containment Control".

B. Obtain RHR 'A' suction temperature and determine Heat Capacity Temperature limit.

C. Start RHR 'A' pump and obtain RHR 'A' suction temperature.

D. Open additional SRVs until RPV pressure is below the Heat Capacity Temperature Limit.

K&A # 295025
Importance Rating 4.1

QUESTION 80

K&A Statement: A2.03 – Ability to determine and/or interpret **Suppression Pool Temperature** as it applies to HIGH REACTOR PRESSURE.

Justification:

A. In-Correct but plausible since the T-102 Bases for step SP/T-8 states, "...*if during the initial evaluation of the HCTL curve, the operating point is on the unsafe side of the HCTL, no action may be taken to restore and maintain the safe side of the HCTL. The heat capacity of the suppression pool has been lost and emergency RPV depressurization is required.*" Subsequently, step SP/T-10 directs entering T-112 and executing concurrently with T-102. However, with Suppression Pool level below 17.8', the Suppression Pool temperature probes are uncovered and are not providing valid readings.

B. InCorrect but plausible with Suppression Pool level below 17.8', Note #2 of T-102 directs use of the suction temperature of an operating RHR pump to determine a valid temperature for the Suppression Pool. However, RHR 'A' pump is not operating so this would not provide a valid suppression pool temperature.

C. Correct - With Suppression Pool level below 17.8', Note #2 of T-102 directs use of the suction temperature of an operating RHR pump to determine a valid temperature for the Suppression Pool. An RHR pump must be started to obtain a valid suppression pool temperature based on the pump's suction temperature.

D. In-Correct but plausible if the applicant was unaware of the above bases discussion for step SP/T-8. Since this step directs maintaining RPV pressure on the safe side of HCTL the opening of additional SRVs would be consistent with the direction provided by the step.

References: LLOT1560, Rev. 012
T-102 Bases, Rev 022

Student Ref: required Yes
T-101
T-102

Learning Objective: IL 6

Question source: New

Question History:

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 81

A reactor shutdown was in progress on Unit 2 due to lowering Suppression Pool level. At ~40% power, a MSL isolation occurred on High Steam Flow due to an unisolable break in MSL 'A' upstream of the inboard MSIV. Current conditions are as follows:

- MSL radiation monitors are reading above Hi-Hi alarm setpoint.
- RPV Pressure is 325 psig.
- Suppression Pool parameters:
  - Level = 4' 2"   Pressure = 15 psig   Temperature = 150°F
  - H_2 conc = 4.5% O_2 conc = 6.4%
- Drywell parameters:
  - Pressure = 15 psig   Temperature = 250°F
  - H_2 conc = 4.2% O_2 conc = 6.2%
- Attempts to align water makeup to the Suppression Pool have been unsuccessful.
- Attempts to establish flow through Post-LOCA recombiners have been unsuccessful.
- Wide Range Accident Monitors (WRAM) have been reset and it has been determined that current containment conditions would exceed WRAM Hi-Hi alarm setpoint.

Based on the above conditions, which one of the following describes the course of action required AND the basis for the action?

A. Re-establish flow to Turbine Bypass Valves to depressurize the RPV. Basis: To prevent exceeding the pressure capability of primary containment.

B. Align nitrogen inerting system to inert/purge the Drywell per T-228 regardless of offsite release rate. Basis: To displace hydrogen and oxygen from the Drywell while maintaining the Drywell atmosphere inert

C. Bypass PCIG isolation logic to depressurize RPV using all ADS valves. Basis: To use available Suppression Pool Heat Capacity and ensure primary containment integrity.

D. Align the nitrogen inerting system to inert/purge the Suppression Pool per T-228 regardless of offsite release rate. Basis: To displace hydrogen and oxygen from the Drywell while maintaining the Drywell atmosphere

K&A # 295030
Importance Rating 4.1

QUESTION 81

K&A Statement: G.2.4.18 - Knowledge of the specific bases for EOPs as it relates to LOW SUPPRESSION POOL LEVEL.

Justification:

A. Correct – With SP level below 4.2' (4' 2 ½"), decision step EB-9 directs the user to step EB-16 of T-112, Emergency Blowdown, directs using one of several flowpaths to depressurize the RPV. The Turbine Bypass Valves are one of the listed flowpaths. Below a depth of 4.2', the SRV tailpipes will become uncovered which will negate the quenching function of the SP and result in the direct pressurization of primary containment.

B. In-Correct but plausible. H_2 / O_2 concentrations meet the criteria for performance of T-102 DW/G-1 leg. Conditions have been met for inerting the Drywell with N_2. However, the actions to purge/inert the Drywell **regardless** of offsite release rate are associated with performance of T-102 DW/G-3 leg and H_2 concentration isn't high enough (i.e., 5.99%) to warrant entry into DW/G-3. The basis is correct for the action to push the H_2 / O_2 out of the Drywell space while maintaining an inert atmosphere.

C. In-Correct but plausible. If SP level is above 4.2' (4' 2 ½"), step EB-9 directs the user to step to perform step EB-10 thru EB-12 to open ADS valves and quench the reactor discharge within the volume of the SP. However, the SP level is just below the limit where ADS valves are used to depressurize the reactor.

D. In-Correct but plausible since the H_2 / O_2 concentrations meet the criteria for performance of leg SP/G-1 of T-102. All conditions have been met for inerting the Suppression Pool with nitrogen. However, the actions to purge/inert the SP **regardless** of offsite release rate are associated with the performance of leg SP/G-3 of T-102 and H_2 concentration isn't high enough (i.e., 5.99%) to warrant entry into the SP/G-3. The basis for the action to displace the H_2 / O_2 from the SP vapor space with the inert nitrogen gas is correct.

References: LLOT1560, Rev. 012 Student Ref: required Yes T-102, 112

Learning Objective: IL5, IL6

Question source: New

Question History:

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 82

Unit 1 scrammed from 100% on a loss of condenser vacuum and a subsequent Main Steam isolation. An emergency blowdown was initiated when the Heat Capacity Temperature Limit could not be met due to lowering Suppression Pool level.

Currently, the following plant conditions exist on Unit 1:

- An RPV depressurization is in progress using the HPCI steam supply lines and MSL Drains per T-260, "Reactor Pressure Vessel Venting".
- Drywell and Suppression Pool venting is in progress per T-200, "Primary Containment Emergency Vent Procedure".
- RPV parameters are:
  - Level = 22" Pressure = 45 psig Temperature = 180°F
- Drywell parameters are:
  - Pressure = 18 psig Temperature = 255°F
  - H_2 conc. = 1.5% O_2 conc. = 3.5%
- Suppression Pool parameters are:
  - Level = 3' 6" Pressure = 18 psig Temperature = 155°F
  - H_2 conc. = 0.8% O_2 conc. = 4.5%

Conditions have been met requiring entry into T-104, "Radioactivity Release Control". Which one of the following describes the flowpaths that could be isolated in accordance with step RR-8 of T-104?

A. RPV depressurization via HPCI steam supply line only.

B. Suppression Pool vent path only.

C. RPV depressurization via HPCI steam supply line and MSL Drains.

D. Drywell and Suppression Pool vent paths.

K&A # 295038 High Off-site Release Rate

Importance Rating 4.7

QUESTION 82

K&A Statement: G.2.4.6 – Knowledge of EOP mitigation strategies.

Justification:

A. In-Correct but plausible since T-112 "Emergency Blowdown" directs that RPV depressurization may end when the reactor is in Cold Shutdown (EB-22). However, the HPCI steam line path is not the only primary system discharging outside of Secondary Containment.

B. In-Correct but plausible since the Suppression Pool concentrations are low enough to terminate venting (H_2 limit < 1.0%, O_2 limit < 5.0%). Although this flowpath discharges outside Secondary Containment (discharges to either the North Stack or South Stack), it is not a primary system as defined by the T-104 basis document which states, "'Primary Systems' consist of the **pipes, valves and other equipment which connect directly to the RPV**, such that a reduction in RPV pressure will cause a drop in the flowrate of steam or water being discharged through a un-isolated break in the system." The Suppression Pool vent path is not directly connected to the RPV.

C. Correct – RPV depressurization has proceeded to the point where Cold Shutdown (i.e., less than 200°F) has been reached. Therefore, per step EB-22, both RPV depressurization flowpaths (HPCI steam supply line and MSL Drains) can be isolated since they are flowpaths directly connected to the RPV and are no longer needed by other TRIP procedures.

The purpose (mitigating strategy) of T-104 is "...to isolate primary system discharges and to control RPV pressure as necessary to minimize offsite radioactivity release during emergency response conditions". While other systems / flowpaths may be discharging radioactive materials, T-104 assumes these will be addressed by T-103, Secondary Containment Control.

D. In-Correct but plausible if the applicant believes that the H_2 / O_2 concentrations in both the Drywell and Suppression Pool are low enough to terminate venting. Also, both flowpaths discharge outside Secondary Containment. However, neither vent flowpath are "Primary Systems" as defined in the T-104 basis document.

References: LLOT1560, Rev. 12
T-104 Bases, Rev. 013

Student Ref. required Yes
T-104
T-102
T-112

Learning Objective: IL5

Question source: New

Question History:

Cognitive level: Memory/Fundamental knowledge:

	Comprehensive/Analysis:	X
10CFR	55.43 (4) X	
Comments:	Created/Modified by: M. Riches Reviewed by: P. Presby	

QUESTION 83

The following plant conditions exist on Unit 2:

- A reactor scram has occurred on low reactor water level.
- All rods have inserted.
- RPV level is -142" and slowly rising.
- RPV pressure is 435 psig and stable.
- ADS timers have initiated.
- Actions are underway to maintain RPV level above TAF and the crew has progressed to step 8 on RC/L leg of T-101 RPV Control.

Which one of the following explains why automatic ADS initiation would be inhibited based on these conditions?

A. Subsequent actions will require exiting RC/L leg of T-101 and enter T-111, Level Restoration / Steam Cooling, which provides specific instructions on when and how ADS initiation should occur.

B. The rapid depressurization would result in an excessive cooldown of the RPV and could aggravate efforts to establish RPV level control.

C. The pressure and thermal transient imposed by opening all ADS SRVs at this point could result in flashing on the RPV level reference legs resulting in loss of RPV level indication.

D. The opening of the ADS SRVs could result in exceeding the heat capacity of the Suppression Pool before adequate cooling to the core can be assured.

K&A #	295009 Low Reactor Water Level
Importance Rating	4.7

QUESTION 83

K&A Statement: G.2.4.6 – Knowledge of EOP mitigation strategies.

Justification:

A. In-Correct but plausible since step RC/L-9 of T-101, "RPV Control" transitions the user to T-111, "Level Restoration / Steam Cooling" if RPV level cannot be maintained above -161. However, the initial conditions of the questions stated that RPV level was at -142" and slowly increasing.

B. Correct – The bases discussion for step RC/L-8 of T-101 lists four reasons why automatic ADS initiation may be undesirable. The first bullet states, "*ADS Actuation can impose a severe thermal transient on the RPV and may complicate efforts to control RPV level.*"

C. In-Correct but plausible since rapid changes in RPV pressure and temperature are conditions that could cause flashing within the reference legs. However, this has nothing to do with the reasons for inhibiting ADS auto initiation.

D. In-Correct but plausible since exceeding the Heat Capacity Temperature Limit of the Suppression Pool is a concern during plant cooldown under emergency conditions. However, this is not the basis for inhibiting ADS automatic initiation.

References:	LLOT1560, Rev. 12 T-101 Bases, Rev. 019	Student Ref. required	Yes T-101 T-117

Learning Objective: IL5

Question source: New

Question History:

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 84

Unit 2 is operating at 100% power.

I&C Technicians cause an inadvertent LOW LOW LOW LEVEL signal on Channel C of Group VIIIB isolation signal while investigating an instrumentation problem. Valve HV-13-106, "Recirc Pump Clg Wtr Isol" closes on the isolation signal. GP8.4, ISOLATION BYPASS, states:

1.	**IF** one **OR** more of the following conditions are met, **THEN** bypassing of Containment Isolation Interlocks may be performed: - As directed by the TRIP/SAMP procedures. - To protect the health **AND** safety of the public as per 10CFR50.54X. - Per OP-AA-101-111, OP-AA-101-112 **AND** OP-AA-103-104 in the event of an emergency **not** covered by approved procedures. - Actions shall be taken to minimize personnel injury **AND** damage to the facility **AND** to protect the health **AND** safety of the public. - **IF** an inadvertent isolation signal exists, **THEN** per Tech Spec 3.6.3 action statement since the automatic valve isolation capability is INOPERABLE.

Assume the isolation signal will not clear and 6 minutes have elapsed since the valve closed. WHICH of the following is correct regarding operation of HV-13-106?

A. The isolation signal may be bypassed and the valve re-opened for up to 4 hours because another isolation valve in series is operable.

B. The isolation signal may be bypassed and the valve re-opened because less than 10 minutes have elapsed since the isolation.

C. The valve may NOT be re-opened because the administrative controls for bypassing the isolation signal are not adequate.

D. The valve may NOT be re-opened because the isolation signal has not cleared.

	K&A #	295020 Inadvertent Cont. Isolation
	Importance Rating	4.6

QUESTION 84

K&A Statement: G2.1.20 – Ability to interpret and execute procedure steps, as it relates to INADVERTENT CONTAINMENT ISOLATION

Justification:

A. Correct. GP8.4 allows a containment isolation signal to be bypassed as governed by TS 3.6.3. The TS requires maintaining an operable isolation in the line and isolation of the line within 4 hours if the inoperable valve cannot be restored. Once the signal is bypassed, the valve may be opened for up to 4 hours from time of the inadvertent isolation.

B. In-Correct. The valve may be opened but not because of the short time duration. Plausible if applicant thinks the 10 minute maximum isolation time for trip of the recirc pumps and the reactor is the reason for why the valves may be re-opened.

C. In-Correct. The isolation signal can be bypassed since HV-13-108 remains operable. HV-13-108 is an isolation valve in series with HV-13-106. Tech Spec 3.6.3 allows HV-13-106 to be opened provided a second isolation valve in series remains operable. Plausible because TS 3.6.3 basis describes specific administrative controls for re-opening a valve that has been closed to meet TS 3.6.3. These controls are to station a person locally at the isolation valve that is used to isolate the line for the TS LCO Action Statement. The individual must be in constant communication with the control room. These administrative controls do not apply to HV-13-106 at this time because the valve is not yet being used to isolate the penetration to comply with action requirements.

D. In-Correct. Under the specified conditions, the isolation signal can be bypassed to allow the valve to be opened under restrictions of TS 3.6.3. Plausible if applicant thinks signal bypass procedure requires signal reset.

References:	GP8, GP8.1, GP8.4, M-13, TS 3/4.6.3, TS Bases	Student Ref: required	TS 3/4.6.3

Learning Objective: N/A

Question source: New

Question History: None

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

10CFR 55.43 (2) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 86

A seismic event resulted in a RPV leak in the Drywell and a rupture of the CST.

Plant conditions are currently as follows:

- MSIVs are closed and two (2) SRVs are currently open.
- Reactor Pressure is 540 psig and stable.
- Reactor Water level is -131" and increasing.
- CST level is off-scale low.
- Suppression Pool pressure is 4.2 psig and slowly increasing.
- Suppression Pool temperature is 173°F and slowly increasing.
- Suppression Pool level is 20' 8".
- Drywell temperature is 250°F and increasing.
- Drywell pressure is 15 psig and increasing.
- All low pressure ECCS pumps are running
- "A" loop of RHR is in Suppression Pool Spray mode.
- HPCI is injecting to the vessel at 3,200 gpm.
- RCIC failed to auto start, but is available.

Based on the above conditions and trends, which one of the following is an immediate concern AND which mitigating action should be taken?

A. Heat capacity of the Suppression Pool has been exceeded. Initiate Emergency Blowdown per T-112, Emergency Blowdown.

B. HPCI Pump damage is possible. Establish RPV level control using the Condensate System.

C. Heat capacity of the Suppression Pool has been exceeded. Initiate RPV depressurization per T-101 RPV Control.

D. HPCI Pump damage is possible. Establish RPV level control using RCIC.

K&A # 206000 High Pressure Coolant Injection

Importance Rating 4.2

QUESTION 86

K&A Statement: A2.08 – Ability to (a) predict the impacts of **High Suppression Pool Temperature** on the HIGH PRESSURE COOLANT INJECTION System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations. [Had the wrong K/A statement from the test outline listed.]

Justification:

A. In-Correct. Plausible since for higher RPV pressures (i.e., > 700 psig), Suppression Pool heat capacity is exceeded. However, Suppression Pool temperature is just below the Heat Capacity Temperature Limit (~180°F) at current SP temperature.

B. Correct –The bases for step RC/L-4 "...that operating the HPCI and RCIC Systems with high suction temperatures (above 170°F) could result in equipment damage. With Suppression Pool temperature above 170°F, HPCI pump damage is a concern. At the current RPV pressure, the Condensate system could be used to feed the RPV.

C. In-Correct but plausible since for higher RPV pressures (i.e., > 700 psig), the heat capacity of the Suppression Pool is exceeded. However, at the current RPV pressure, the Suppression Pool temperature is just below the Heat Capacity Temperature Limit (~180°F).

D. In-Correct but plausible since Suppression Pool temperatures above 170°F could result in damage to the HPCI pump. However, the same caution also applies to the RCIC pump. High suction temperature to RCIC can result in pump damage.

References: LLOT1560, Rev. 012
T-102 Bases, Rev. 022
T-101 Bases, Rev. 019

Student Ref: required Yes
T-101
T-102

Learning Objective: IL4

Question source: New

Question History:

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 87

Unit 1 is at 100% power with all systems normal when the following alarm is received at 120 D11: 1A RPS & UPS DISTR PNL TROUBLE

Plant conditions are as follows:

- The [BOP] states that all Confirming Indications in E-1AY160, Loss of 1A RPS UPS Power, are present
- The Reactor Operator reports RWCU isolation has occurred
- As control room supervisor, you observe various process Rad Monitor Trips and Alarms

Considering the above conditions, as control room supervisor:
(1) Select the appropriate procedure to implement, and
(2) Select the appropriate basis.

	(1)	(2)
A.	ON-113 Loss of RECW	Loss of 1AY160 caused RECW Isolation
B.	GP-4 Rapid Plant Shutdown	Loss of 1AY160 caused loss of CRD Flow Controller power
C.	OT-112 Recirculation Pump Trip	Loss of 1AY160 caused Recirc Pump trip
D.	ON-109 Total Loss of SRM, IRM or APRM Systems	Loss of 1AY160 caused loss of power to all IRMs

K&A # 212000 RPS

Importance Rating 4.2

QUESTION 87

K&A Statement: G2.4.11 Knowledge of Abnormal Condition Procedures as they relate to Reactor Protection System

Justification:

A. Correct Per stem conditions indicate there has been a loss of 1A RPS UPS power. Per E-1AY160 this will result in RECW isolation. Step 2.1 (Initial Actions) of E-1AY160 specifies "Enter ON-113, Loss of RECW".

B. In-Correct – Loss of 1AY160 will not affect CRD Flow Controllers (E-1AY160 Section 1.0). Plausible if some other power supply is assumed lost or if associates with RWCU isolation.

C. In-Correct but plausible. Loss of 1AY160 will not on its own cause a recir pump trip. If Recirc Pumps trip, E-1AY160 directs entering OT-112.

D. In-Correct but plausible since half of IRMs (A, C, E, and G) will lose power.

References: E-1AY160 page 1
OT-117 page 3
LGSOPS0071 pages 40 and 41

Student Ref: required No

Learning Objective: N/A

Question source: Limerick Bank

Question History: None

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

10CFR 55.43(5) X

Comments: Created/Modified by: Johnson
Reviewed by: Presby

QUESTION 88

The following plant conditions exist on Unit 2:

- The Unit is at 100% power.
- Annunciators:The following MCR alarm windows annunciate on panel 208:
    - A-4, OPRM TRIPS ENABLED
    - B-3, APRM UPSCALE TRIP/INOP
    - B-4, APRM UPSCALE
    - E-3, APRM/RBM FLOW REF OFF NORMAL
- Recirc Loop "A" Drive Flow indicates 42,000 gpm on FI-043-2R613.
- Recirc Loop "B" Drive Flow indicates 44,200 gpm on FI-043-2R617.
- Total Recirc Loop Drive Flow Recorder indicates 86,200 gpm on FR-043-2R614.

The following information is displayed on the ODA for the APRM channels:

	APRM '1'	APRM '2'	APRM '3'	APRM '4'
% STP	100	100	100	100
Upscale Alarm Setpoint	109%	86.7%	109%	109%
Upscale Trip Setpoint	116.6%	94.3%	116.6%	116.6%
Total Recirc Drive Flow	98.0%	98.0%	98.0%	98.0%

Which one of the following identifies the failure indicated by the above conditions AND the appropriate procedure that should be performed?

A. An upscale LPRM failure associated with APRM '2' has occurred resulting in fewer than the required number of LPRM inputs. Perform S.74.7.A, Bypassing an LPRM.

B. Flow transmitter FT-043-2N024B for 'B' RRP has failed to zero for APRM '2'. Perform ARC 208 E-3, APRM/RBM FLOW REF OFF NORMAL.

C. An upscale LPRM failure associated with APRM '2' has occurred. Perform S.74.7.A, Bypassing an LPRM.

D. Flow transmitter FT-043-2N014B for 'A' RRP has failed to zero for APRM '2'. Perform ARC 208 E-3, APRM/RBM FLOW REF OFF NORMAL.

K&A # 215005 APRM/LPRM
Importance Rating 3.4

QUESTION 88

K&A Statement: A2.07 Ability to predict (a) the impacts of **Recirculation flow channels flow mismatch** on the APRM/LPRM System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations.

Justification:

A. In-Correct but plausible since an upscale failure of an LPRM could result in Simulated Thermal Power increasing and would cause an APRM upscale/inop alarm. The LPRM failure would effect APRM, RBM, and OPRM indications. However, the condition would result in an APRM Upscale/Inop alarm only. Also, an upscale failure of an LPRM would not have any effect on triggering the "OPRM enabled" alarm. In addition, the LPRM circuit does not count the number of LPRMs that have invalid readings, but tracks how many detectors associated with an APRM are bypassed.

B. Correct – Failure of flow transmitter FT-043-2N024B would result in 1) indicated flow for APRM channel '2" falling below the point (<60%) where OPRM '2' would be enabled (ARC 108 A-4), 2) APRM flow-biased alarm and trip setpoints would fall to 86.7% and 94.3%, respectively (ARC 108 B-4), and 3) flow transmitter input signal would be less than 1 mA (ARC 108 E-3). Flow indications for loop and total drive flow would remain unchanged since they receive their flow signals from APRM Channels '1' and '4'.

C. In-Correct but plausible since an upscale failure of an LPRM could result in Simulated Thermal Power increasing and would cause an APRM upscale/inop alarm. The LPRM failure would also have some effect APRM, RBM, and OPRM indications. However, the condition would result in an APRM Upscale/Inop alarm only. Also, an upscale failure of an LPRM would not have any effect on triggering the "OPRM enabled" alarm.

D. In-Correct but plausible since failure of flow transmitter FT-043-2N014B would result in 1) indicated flow for APRM channel '2' falling below the point (<60%) where OPRM '2' would be enabled (ARC 108 A-4), and 2) flow transmitter input signal would be less than 1 mA (ARC 108 E-3). However, the APRM flow-biased alarm and trip setpoints would indicate 88.2% and 95.8%, respectively.

References: LLOT0275, Rev. 012 Student Ref. required No

Learning Objective: Obj. 18

Question source: New

Question History:

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

10CFR	55.43 (5) X
Comments:	Created/Modified by: M. Riches Reviewed by: P. Presby

QUESTION 89

The following plant conditions exist on Unit 2:

- The Unit is at 100% power
- The following MCR alarm windows annunciate:
  - 216 A-1, RCIC OUT OF SERVICE
  - 221 G-2, 2PPC1/2PPC2 125 VDC DIST PANELS UNDERVOLTAGE

Which one of the following describes the effect this will have on RCIC AND the procedure to be performed once power is restored?

A. All RCIC inboard isolation valves will fail to respond to a valid isolation signal. Perform 1S49.1.A (COL), Valve Alignment to Assure Availability of the RCIC System".

B. All RCIC inboard isolation valves close. Perform S49.1.B, "Recovery from RCIC Steam Line Isolation and Resultant Turbine Trip".

C. All RCIC outboard isolation valves will fail to respond to a valid isolation signal. Perform 1S49.1.A (COL), Valve Alignment to Assure Availability of the RCIC System".

D. All RCIC outboard isolation valves close. Perform S49.1.B, "Recovery from RCIC Steam Line Isolation and Resultant Turbine Trip".

K&A # 223002 PCIS/Nuclear Steam Supply Shutoff System

Importance Rating 3.2

QUESTION 89

K&A Statement: A2.02 Ability to predict (a) the impacts of **D.C. electrical distribution failures** on the PCIS / NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM; and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations.

Justification:

A. Correct –Div 3 125 VDC distribution panel 2PPC1 supplies power to the isolation logic for the inboard RCIC isolation valves. Loss of Div 3 125VDC power to the RCIC isolation logic will prevent the isolation valves from closing when a valid Grp VA isolation signal is received.

B. In-Correct but plausible since most NSSSS isolation groups will isolate on a loss of power to the initiation logic. However, the initiation logic for most NSSSS isolation groups are powered from AC not DC.

C. In-Correct but plausible since loss of power to Div 1 125 VDC distribution panels would effect the ability of the RCIC outboard isolation valves to close on a valid Grp 5A isolation signal.

D. In-Correct but plausible since most NSSSS isolation groups will isolate on a loss of power to the initiation logic. However, the initiation logic for most NSSSS isolation groups are powered from AC not DC.

References: LLOT0380, Rev. 024 — Student Ref. required: No

Learning Objective: IL1, IL5

Question source: Modified Bank (Limerick) — Added different conditions and added identification of the correct procedure to study guide (LLOT0380) question

Question History:

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 90

Initial plant conditions:

- Unit 2 is in OPCON 1 with no equipment out of service and no activities in progress.
- Unit 1 is in OPCON 5.
- Reactor Enclosure Secondary Containment Integrity is established.
- Maintenance is replacing the drive mechanism on Control Rod 14-55.
- LPRM String 40-33 is being removed for replacement.
- RWCU is providing decay heat removal in accordance with S44.7.B, Using Reactor Water Cleanup as an Alternate Method of Decay Heat Removal to support testing of the SDC isolation signals.

Subsequently, laboratory test results are received, indicating that SBGT charcoal adsorber samples for Unit 1 and Unit 2 SBGT systems have failed the methyl iodide penetration test.

Which of the following describes the action to take to comply with T.S. 3.6.5.3, "Standby Gas Treatment System – Common System"?

A. Suspend maintenance activities associated with replacing the drive mechanism on Control Rod 14-55.

B. Suspend activities associated with the removal of LPRM String 40-33 for replacement.

C. Suspend testing on the SDC isolation signals. Re-align a loop of RHR to SDC and suspend alternate decay heat removal using RWCU.

D. Enter data into LCO tracking system. Condition must be addressed before placing the plant in an applicable OPCON mode.

K&A # 261000 Standby Gas Treatment System
Importance Rating 4.7

QUESTION 90

K&A Statement: G.2.2.39 – Knowledge of less than or equal to one hour Technical Specification action statements for systems, as it relates to STANDBY GAS TREATMENT SYSTEM.

Justification:

A. Correct – T.S. 3.6.5.3, action b. states "*With **both** standby gas treatment subsystems inoperable, if in progress, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS or **operations with a potential for draining the reactor vessel**.*". Section 3.3. of GP-6.2, Shutdown Operations – Shutdown Condition Tech Spec Actions addresses "Suspending Operations with a potential for Draining the Reactor Vessel". Specifically, step 3.3.2.3 identifies "**CRD maintenance (i.e. drive removal and drive replacement)** except for HCU hydraulic line maintenance" as a specific activity that falls in this category.

B. In-Correct but plausible since action b. of T.S. 3.6.5.3. requires suspension of CORE ALTERATIONS. However, the movement of LPRM detectors are specifically excepted in the definition of CORE ALTERATION.

C. In-Correct but plausible since RWCU operations "other than beyond normal makeup or letdown" are specifically mentioned in GP-6.2 as operations to be considered as having a potential to drain the reactor vessel. However, only if the system's isolation capability on either Reactor Low Water Level or RWCU High Differential Flow is not maintained.

D. In-Correct but plausible since the Applicability statement for this LCO specifically lists OPCON 1, 2, and 3. However, it does not specifically mention OPCON 5, but rather the specific operational conditions (See discussion above in choice A.)

References: T.S. 3.6.5.3 Student Ref. required No
GP-6.2, Rev.042
LGSOPS0069

Learning Objective: IL10 (LGSOPS0069)

Question source: New

Question History:

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

10CFR 55.43(2) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

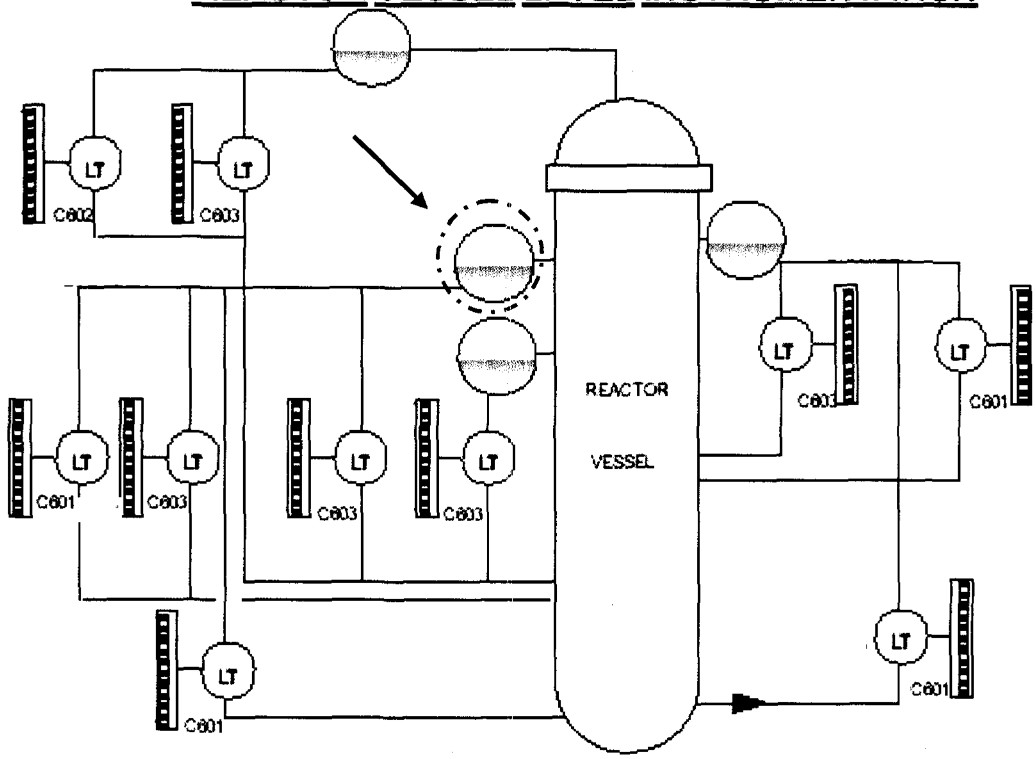
QUESTION 91

Unit 1 is in OPCON 1. Blockage is suspected on Condensing Pot XY1D004A identified in the simplified diagram provided below. Based on the apparent blockage the affected instruments have been declared inoperable.

Using the supplied references, which one of the following identifies how this condition affects the status of the Instrumentation LCOs (TS 3.3) listed below?

	3.3.1	3.3.2	3.3.3	3.3.5	3.3.7.5
A.	Met	Met	Not Met	Not Met	Not Met
B.	Not Met	Met	Met	Not Met	Met
C.	Met	Not Met	Not Met	Met	Not Met
D.	Met	Met	Met	Not Met	Not Met

REACTOR VESSEL LEVEL INSTRUMENTATION



K&A #	216000 Nuclear Boiler Instrumentation
Importance Rating	4.7

QUESTION 91

K&A Statement: G.2.2.22 – Knowledge of limiting conditions for operations and safety limits, as it relates to NUCLEAR BOILER INSTRUMENTATION.

Justification:

A. Correct – From M-42 sheet 1, the affected instruments would be:

- LT-1N095A (ECCS – ADS Level 3 Permissive)
- LT-1N080A (RPS/NSSS Level 3 input)
- LT-1N081A (NSSSS Level 2, 1)
- LT-1N091E (RCIC Level 2, 8)
- LT-1N402A (RRCS Level 2 RRP Trip)
- LT-1N097E (RCIC Level 2, 8)
- LT-1N097A (RCIC Level 2, 8)
- LT-115A (Post-Accident Monitoring)
- LT-1N085A (Fuel Zone Indication)

- PT-1N403A (RRCS)
- PT-1N090J (Core Spray 'A')
- PT-1N090N (Core Spray 'A')
- PT-1N090A (Core Spray 'A' / RHR 'A')
- PT-1N090E (Core Spray 'A' / RHR 'A')

From Sheet 2, Table 1 "Water Level Instrumentation Utilization" the function and level trip point for each level transmitter can be found in the last two columns of the table. Based on the information, LCO 3.3.3 is not met since the ADS Level 3 Permissive requires a minimum of 1 operable channel (1/trip system; two trip systems with 1 channel per trip system). LCO 3.3.5 is not met since RCIC Level 2 and Level 8 trips require a minimum of 4 operable channels (4 per trip function). Currently, three of the level channels associated with RCIC are inoperable. LCO 3.3.7.5 is not met since the required number of operable Post Accident Monitoring (PAM) channels (2) is not met.

B. In-Correct but plausible if the applicant doesn't know how many level channels are actually available for the RPS and PAM circuits.

C. In-Correct but plausible since the Level 3 NSSSS input is used to isolate RHR SDC which would only be in service during OPCON 4 or 5.

D. In-Correct but plausible since only two channels (one / subsystem) provide signals for the ADS Level 3 permissive. If the applicant doesn't notice the (***) note at the bottom of the page concerning that the "Minimum Operable Channels per Trip Function" is per subsystem then this choice appears to be a correct answer.

References: LGSOPS0042, Rev. 000 Student Ref. Yes

M-42 Nuclear Boiler Instrumentation T.S Section 3.3 Instrumentation	required	T.S. Sec 3.3, Inst & M-42, Sheets 1 & 2

Learning Objective: IL11

Question source: New

Question History:

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

10CFR 55.43 (2) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 93

Unit 1 plant conditions are as follows:

- Reactor Power is 22% power.
- RT-6-031-321-1, "Mechanical Trip Valve and Master Trip Solenoid Valves Operability Test" is in progress.
- Bypass Valves #2 and #3 have been manually closed and de-energized due to problems with mechanical linkage between the valves and their actuators.

When the Oil Trip Pushbutton is depressed, all main turbine stop valves fail shut.

Which one of the following identifies the expected plant response, AND the procedure to be entered?

A. Reactor Recirculation Pumps will NOT trip. The Bypass Valves will NOT control pressure below scram setpoint. Enter T-101, RPV Control.

B. Reactor Recirculation Pumps will NOT trip. The Bypass Valves will control pressure below scram setpoint. Enter OT-102, High Reactor Pressure.

C. Reactor Recirculation Pumps will trip. The Bypass Valves will control pressure below scram setpoint. Enter OT-102, High Reactor Pressure.

D. Reactor Recirculation Pumps will trip. The Bypass Valves will NOT control pressure below scram setpoint. Enter T-101, RPV Control.

K&A # 241000 Reactor/Turbine Pressure Regulator
Importance Rating 3.9

QUESTION 93

K&A Statement: A2.05 – Ability to (a) predict the impacts of **Failed open/closed main stop valve(s)** on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations.

Justification:

A. In-Correct but plausible if the applicant believes the steam flow would exceed the flow capacity of the available bypass valves, then the reactor would scram on high pressure.

B. Correct – Bypass capacity with all nine BPV can handle up to 30% of full power steam flow, with each bypass valve capable of handling 3.33% of the flow. Two inoperable BPVs would reduce the system's bypass capacity to 23.7% (30% - 2(3.33%)). Since reactor power is less than the bypass capacity, the operable BPVs can control reactor pressure. One entry condition for OT-102 is the "Unexpected/unexplained opening of a Main Turbine Bypass Valve."

C. In-Correct but plausible if the applicant misunderstands what will cause the RRPs to trip. While the RRPs will trip when pressure is greater than 1049 psig, the trip is not enabled unless power is above 25%.

D. In-Correct but plausible if the applicant misunderstands what will cause the RRPs to trip and miscalculates the flow capacity of the available bypass valves. The RRPs will trip when pressure is greater than 1049 psig, but the trip is not enabled unless power is above 25%. The available bypass capacity (23.7%) is greater than the current power level (22%).

References: LGSOPS0031B, Rev. 000
OT-102, Rev.017

Student Ref. required No

Learning Objective: IL2

Question source: Modified Bank (Clinton) Changed power level, word choice. Added inoperable BPVs.

Question History: Clinton Bank

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 94

The following events occurred on Unit 1:

- A reactor startup is in progress with reactor power at 12%.
- While conducting the startup, a single control rod has become mispositioned from its group of rods and is eight positions further inserted into the core then was required by the Rod Pull sheets.
    - Current position single rod: Position 24
    - Target position of rod group: Position 30
- ON-123, Mispositioned Control Rod, has been entered to address the problem.
- A P-1 Edit run has been performed and the thermal limits have been verified to be less than 0.98.
- Reactor Engineering has been contacted for recovery direction on the mispositioned rod incident.
- Reactor Engineering has evaluated the situation and directed the withdrawal of the rod to the Target group position.

In accordance with OP-AB-300-1001, BWR Control Rod Movement Requirements, which one of the following is correct concerning administrative restrictions on the proper method for realigning the rod?

A. The mispositioned rod must be withdrawn one notch at a time until the target rod position is reached.

B. The mispositioned rod can be continuously withdrawn, but movement must stop such that the control rod settles at least one notch away from the target notch. Subsequently, single notch movement must be used to reach the target rod position.

C. The mispositioned rod may be withdrawn up to three notches at a time until the target rod position is reached.

D. The mispositioned rod may be withdrawn up to three notches at a time, but movement must stop such that the control rod settles at least one notch away from the target notch. Subsequently, single notch movement must be used to reach the target rod position.

K&A # Plant-wide Generics
Importance Rating 4.6

QUESTION 94

K&A Statement: G.2.1.37 – Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Justification:

A. In-Correct but plausible since this would be the correct method if the difference between the current position and the target position is three notches or less.

B. Correct – In accordance with step 4.6 of OP-AB-300-001, "BWR Control Rod Movement Requirements" state "When using continuous rod withdrawal it must stop and allow the control rod to settle one notch from the target position. Then use single notch position to reach the final target rod position."

C. In-Correct but plausible since the procedure uses a limit of three notches or less to apply a restriction on the type of rod movement. However, the limit is used to distinguish between when continuous or single notch method should be used. However there is no limit of three notches per individual rod movements.

D. In-Correct but plausible since the procedure uses a limit of three notches or less to apply a restriction on the type of rod movement. However, the limit is used to distinguish between when continuous or single notch method should be used. However there is no limit of three notches per individual rod movements.

References: LLOT0085, Rev. 000 Student Ref. required No
ON-123, Rev.019

Learning Objective: Obj. 1 (LLOT0085)

Question source: New

Question History:

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 95

If the Reactor Mode switch is in STARTUP/HOT STANDBY, which of the following instruments is NOT required to be operable?

A. Reactor Vessel Pressure High for ARI

B. Reactor Vessel Level 1 for ADS

C. Reactor Vessel Pressure for High Pressure Scram

D. Reactor Vessel Level 2 RWCU System Isolation

K&A #	Equipment Control
Importance Rating	4.7

QUESTION 95

K&A Statement: G2.2.22 – Knowledge of limiting conditions for operations and safety limits

Justification:

A. Correct – ARI is not required in per Tech Specs.

B. In-Correct – Per Tech spec 3/4 3.3.3-1, pg. 3/4 3-34 instrumentation is required in OPCON 2.

C. In-Correct – Per Tech spec 3/4 3.3.3-1, pg. 3/4 3-2 instrumentation is required in OPCON 2.

D. In-Correct- Per Tech spec 3/4 3.3.3-1, pg. 3/4 3-12 instrumentation is required in OPCON 2.

References: Tech. Spec. 3/4 3.3.3-1 Student Ref: required No

Learning Objective: LLOT Obj. 3, 6a, 7

Question source: Bank (Fermi)

Question History: None

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

10CFR 55.41(10) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 96

Unit 1 is at 100% power. It is August 5th at 0800.

The operator completing the channel checks for HPCI Drywell Pressure instrumentation, per ST-6-107-590-1, reports to you that two channels are reading 0.5 psig and two channels are reading 0.1 psig. On the previous day all channels were reading 0.4 psig.

You call I&C to investigate and they inform you that “the signals to the trip units are consistent with the readings obtained by the operators”.

At 0900, I&C Technicians completing the ADS Surveillance Procedure report that the ADS timer “as left” value cannot be adjusted below 120 seconds. When asked, the technicians report that the Manual actuation as well as ADS Timer Override are unaffected by the “as left” condition.

All other Unit 1 equipment is functioning normally.

Considering the above conditions:

(1) Is there any LCO action statement in effect with an action of less than 7 days, and

(2) When, if ever, will a change in OPERATIONAL CONDITION be required? Start time from 0900.

	(1)	(2)
A.	Yes	7 hours
B.	Yes	12 hours
C.	No	7 days
D.	No	OPCON change not required

K&A #	Equipment Control
Importance Rating	4.6

QUESTION 96

K&A Statement: G2.2.23 – Ability to track Tech Spec limiting conditions for operation

Justification:

A. Correct – Conditions describe both HPCI and ADS INOPERABLE. SRO must enter Tech Spec 3.0.3 which requires OPCON change within 7 hours (1 hour to initiate action and then place the unit in at least STARTUP within the next 6 hours). HPCI is INOPERABLE based on more than one channel of drywell pressure instrumentation being inoperable. Two of the four channels are reading almost 50% different and "normal" drywell pressure is around 0.4 psig. This means that the two reading 0.1 psig are non-conservative. ADS timer is inoperable (>117 seconds), making ADS INOPERABLE as well.

B. In-Correct – Plausible if applicant "zeros in" on 3.3.3.c.2.

C. In-Correct – Plausible if applicant "zeros in" on the "otherwise" statement in 3.3.3.

D. In-Correct- Plausible if applicant thinks ACTION 30 (a) applies. That is, the only action necessary is to place affected channel in Tripped. This is, typically, what happens with 3.3.3 but is NOT applicable in this instance since ADS timer is also INOP and there is only one channel provided.

References: ON-120, LLOT760 pg. 37
ST-6-107-590-1

Student Ref: required — TS pgs 3/4 3-32 thru 3-37 and 5-3

Learning Objective: N/A

Question source: New

Question History: None

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

10CFR 55.43(2) X

Comments: Created/Modified by: Johnson
Reviewed by: Presby

QUESTION 97

Unit 2 plant conditions are as follows:

- Reactor power is 100%
- There is a steam leak from the "2B" RWCU pump
- Dose rates in the "2B" RWCU pump room are 2 R/hour

Entry is required into the "2B" RWCU pump room.

WHICH ONE of the following describes the type of Locked High Radiation Area and required authorization for release of the key?

	Type of Locked High Radiation Area for "2B" RWCU Pump	Required Authorization For Release of Key
A.	Level 1	RP Supervision
B.	Level 1	Operations Shift Manager
C.	Level 2	RP Supervision
D.	Level 2	Operations Shift Manager

K&A # Radiation Control
Importance Rating 4.7

QUESTION 97

K&A Statement: G2.3.13 – Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Justification:

A. In-Correct – Based on dose rates, the "2B" RWCU pump room is a Level 2 Locked High Radiation Area. Per RP-AA-460-001 entry into a locked High Rad area requires RP Supervision authorization except in emergencies. Plausible if applicant does not remember the dose rate for level 1 locked high radiation area.

B. In-Correct – Plausible if applicant identifies the room as a Level 1 Locked High Rad area, but does not remember that entry requires RP supervision approval except in emergencies.

C. Correct – Based on dose rates, the "2B" RWCU pump room is a Level 2 Locked High Radiation Area. Level 2 dose rate are greater than 15000 mR/hr. Per RP-AA-460-001 entry into a locked High Rad area requires RP Supervision authorization except in emergencies.

D. In-Correct- Plausible if applicant classifies the area as Level 2 and does not remember that entry into any locked high rad area requires RP Supervision approval except in emergencies.

References: RP-LG-460-1010, RP-LG-460-1016, LLOT1760 Student Ref: required

Learning Objective: LLOT1760 Obj. 1

Question source: Modified Bank (Limerick) Chgs: location, dose, correct ans.

Question History: ILT NRC Exam 2005

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

10CFR 55.41(12) X

Comments: Created/Modified by: Tomlinson
Reviewed by:

QUESTION 98

Unit 1 is at 100% power and has been at steady state operations for the last 22 hours. An Offgas release rate calculation is in progress per GP-5, Steady State Operations.

The following data has been obtained for A and B channels.

- Previous hourly Off-gas release rate was 99.80 µCi/sec.
- Radiation levels as read from RR-026-1R601, point 1and 2, respectively:
    - Channel A = 37.7 Mrem/hr   Channel B = 42.4 Mrem/hr
- System Flow as read from FR-069-115, "SJAE Discharge Flow from Recombiner" is 53 scfm.
- The U1 Off gas placard contains the following information:

Sum of six values	139
K_a	0.082
K_b	0.080

Using the provided portion of GP-5, determine which of the following describes the status of the calculated offgas release rate AND the action to take concerning the calculated offgas release rate.

A. The calculated offgas release rate is invalid. Recalculate using flow readings from FIT-070-150, "HEPA Filter 10F371 Dsh. Flow, Local Indication, Low Range".

B. The calculated offgas release rate has risen. Request Chemistry to perform ST-5-041-885-1, "Dose Equivalent I-131 Determination".

C. The calculated offgas release rate has risen. Request Chemistry to perform ST-5-070-885-1, "Isotopic Off-gas Analysis".

D. The calculated offgas release rate has risen. Request Chemistry to perform ST-5-041-885-1, "Dose Equivalent I-131 Determination" AND ST-5-070-885-1, "Isotopic Off-gas Analysis".

K&A # Plant-wide Generics
Importance Rating 4.3

QUESTION 98
K&A Statement: G.2.3.11 – Ability to control radiation releases.
Justification:

A. In-Correct but plausible since a flow rate below 50 scfm would invalidate the calculation and require use of FIT-070-150.

B. In-Correct but plausible since an increase of more than 15% from the last hourly calculation in the offgas release rate requires performance of ST-5-041-885-1, "Dose Equivalent I-131 Determination". However, this is only applicable if offgas release rates are above 75,000µCi/sec.

C. Correct – The applicant must first determine the average offgas release rate using the following formula for each channel.

RR = rad level x offgas flow x K constant

Ave RR = $(RR_{chA} + RR_{ch\,B}) / 2$ = 171.81 µCi/sec

This results in an increase of 72% increase over the last hourly increase. Performance of ST-5-070-885-1, "Isotopic Off-gas Analysis" is required if the offgas release rate increases by more than 50% from the last hourly calculation regardless of the offgas release rate level.

D. In-Correct but plausible since an increase of more than 15% in the offgas release rate calculation requires performance of ST-5-041-885-1, "Dose Equivalent I-131 Determination". In addition, an increase of more than 50% in the offgas release rate calculation requires performance of ST-5-070-885-1, "Isotopic Off-gas Analysis". However, ST-5-041-885-1, "Dose Equivalent I-131 Determination" would only be required if offgas release rates are above 75,000µCi/sec.

References:	LLOT1790, Rev. 006 GP-5, Rev. 135	Student Ref. required GP-5, pp 15-16 *Spec. step 3.1.18*	Yes

Learning Objective:

Question source: New

Question History:

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

10CFR 55.43 (5) X

Comments: Created/Modified by: M. Riches
Reviewed by: P. Presby

QUESTION 99

The following conditions exist on Unit 2.

- A reactor scram from full power has occurred due to a loss of offsite power.
- Shortly after the scram, lit alarm windows on MCR annunciator panels 201 through 212, 215 through 217, 220 and 222 go out.
- Many MCR instruments are indicating downscale.
- The crew enters ON-122, Loss of Main Control Room Annunciators.
- Twenty minutes later the screens on the Plant Monitoring System go blank.

Which one of the following identifies the required action for the annunciator failure per ON-122 <u>AND</u> the Emergency Action Level per EP-AA-1008 for the given conditions?

	Required Action per ON-122	Emergency Action Level
A.	Dispatch operator to investigate conditions on electrical distribution panels 2PP01 and 2PP02.	Alert, MA6
B.	Dispatch operator to investigate conditions on electrical distribution panels 2-PPB-1 AND 2-PPB-2.	Alert, MA6
C.	Dispatch operator to investigate conditions on electrical distribution panels 2PP01 and 2PP02.	Site Area Emergency, MS6
D.	Dispatch operator to investigate conditions on electrical distribution panels 2-PPB-1 AND 2-PPB-2.	Site Area Emergency, MS6

K&A # Plant-wide Generics
Importance Rating 4.0

QUESTION 99

K&A Statement: G.2.4.32 – Knowledge of operator response to loss of all annunciators.

Justification:

A. In-Correct but plausible since MCR Annunciator power is provided by Non-safeguard 125 VDC. The electrical distribution panels 2PP01 and 2PP02 are those that supply annunciators that have lost power. However, the conditions exceed the "Alert Emergency Action Level.

B. In-Correct but plausible if the applicant believes that the MCR annunciators are supplied from the Safeguard 125V DC system and that the conditions meet the threshold values for an "Alert" classification.

C. Correct – MCR Annunciator power is provided by Non-safeguard 125 VDC. The electrical distribution panels 2PP01 and 2PP02 are those that supply annunciators that have lost power. A Site Area Emergency is declared since a majority of the annunciators have been lost for more than 15 minutes, a plant transient is in progress (reactor scram) and Compensatory Non-alarming indications are unavailable (PMS).

D. In-Correct but plausible if the applicant believes that the MCR annunciators are supplied from the Safeguard 125V DC system.

References:	LLOT0690, Rev. 12 ON-122, Rev.017	Student Ref. required Yes EP-AA-1008 EAL HOT Matrix
Learning Objective:	Obj. 4 (LLOT0690)	
Question source:	Modified Bank (Limerick)	Changed conditions 1) correct ans now Site Area Emergency and 2) requiring examinee to know nomenclature distinguishing non-safeguard and safeguard DC buses.
Question History:	Limerick	
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	 X
10CFR	55.43 (5) X	
Comments:	Created/Modified by: M. Riches Reviewed by: P. Presby	

QUESTION 100

While conducting a locked valve routine surveillance test per RT-6-000-360-1, the equipment operator in the field reports that the locking mechanisms on the following are missing and the valves are closed.

- 48-1F001A, B, and C, SBLC TK OUTLET Valves
- 48-1F003A, B, and C, SBLC PP DISCH Valves
- 47-1F089 A and B, SCRAM DISCH VOL AIR HDR SUPPLY Valves

The correct locked position for these valves is open. The operator also reports damage to the piping around the valves.

Per procedure, what direction should you provide to the equipment operator in the field and what additional action is required?

A. Immediately open the valves, contact security

B. Immediately open the valves, reinstall the locking devices

C. Leave the valves in the closed position, contact security

D. Leave the valves in the closed position, take locking devices to security

K&A #	Emergency Procedures
Importance Rating	4.1

QUESTION 100

K&A Statement: G2.4.28 – Knowledge of procedures relating to a security event (non-safeguards information)

Justification:

A. In-Correct – Mispositioned valves need to be investigated as to why they are in their current position, system lineup should be verified prior to repositioning valves. Plausible if applicant believes that repositioning valves to restore system operability is the correct action.

B. In-Correct – Mispositioned valves need to be investigated as to why they are in their current position, system lineup should be verified prior to repositioning valves. Plausible if applicant believes that repositioning valves to restore system operability is the correct action. Multiple mispositioned valves is indicative of sabatoge. SE-3 should be entered and security should be notified.

C. Correct – Multiple mispositioned locked valves and indication of piping damage could be indication of sabatoge. Per SE-3, security should be notified. Valves should not be repositioned until the system lineup is properly understood. Per RT-6-000-360-1, mispositioned valves should be immediately brought to shift supervision and noted in comments section of the test

D. In-Correct – Taking the locking devices to security is not in accordance with SE-3. SE-3 directs notifying security by phone, public address or radio.

References: SE-3, LGSOPS2000 Student Ref: required N

Learning Objective: LGSOPS2000 Obj. 1

Question source: New

Question History: None

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

10CFR 55.41(10) X

Comments: Created/Modified by: Tomlinson
Reviewed by: