U.S. Nuclear Regulatory Commission
Site-Specific RO Written Examination

Applicant Information	
Name:	
Date: 5/16/2008	Facility/Unit: Susquehanna
Region: I	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent. You have 6 hours to complete the examination.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results		
RO Total Examination Values	____	Points
Applicant's Score	____ / ____	Points
Applicant's Grade	____ / ____	Percent

PPL SUSQUEHANNA, LLC
STANDARD EXAM SHEET

Course #: ____________ Course: 2008 NRC EXAM (RO WRITTEN)

First Name: ____________ Last Name: KEY Employee #: ____________

Social Security Number

Test Form: A B C D

Test Date: 5/16/2008

Test Series: ____________

Test Number: RO Written

KEY

ANSWER KEY

Test Taking is an Individual Effort: Any test misconduct is a violation of the Academic Honesty Policy (NTP-QA-14.2) and the PPL Corp. Standards of Conduct and Integrity.

Signature: ____________

____________ # Correct ____________ % Score

(Updated 5/21/2008)

Question	Answer	Question	Answer	Question	Answer	Question	Answer	Question	Answer
1.	A	21.	A	41.	A	61.	C	81.	
2.	C	22.	B	42.	C	62.	C	82.	
3.	A	23.	C	43.	D	63.	D	83.	
4.	D	24.	A	44.	D	64.	D	84.	
5.	B	25.	B	45.	D	65.	B	85.	
6.	A	26.	A	46.	A	66.	B	86.	
7.	C	27.	C	47.	D	67.	A	87.	
8.	C	28.	D	48.	B	68.	A	88.	
9.	C	29.	A	49.	D	69.	B	89.	
10.	D	30.	D	50.	A	70.	C	90.	
11.	B	31.	A	51.	C	71.	B	91.	
12.	B	32.	C	52.	D	72.	A	92.	
13.	C	33.	D	53.	D	73.	C	93.	
14.	A	34.	A	54.	A	74.	C	94.	
15.	C	35.	C	55.	B	75.	C	95.	
16.	C	36.	C	56.	C	76.		96.	
17.	A	37.	B	57.	D	77.		97.	
18.	A	38.	C	58.	D	78.		98.	
19.	B	39.	D	59.	B	79.		99.	
20.	A	40.	A	60.	D	80.		100.	

FORM NO. O-104268-PPL M4 4000 - 154 - 10 9 8

QUESTION 1

Unit 1 is operating at rated power conditions:

- Both Reactor Building Chilled Water Loop pumps have tripped.
- Reactor Building Chilled Water is lost.
- Drywell Cooling has swapped over to RBCCW.
- No operator actions are taken.

Which one of the following RBCCW actuations occur as a result of the loss of Reactor Building Chilled Water?

A. Reactor Water Cleanup system isolates on high NRHX temp.

B. Reactor Water Cleanup pump trips on high motor winding temp.

C. Containment Inst Gas Compressors trip on low cooling pressure.

D. Reactor Building Sample Coolers isolate on low cooling water flow.

K&A # 400000 CCW
Importance Rating 2.9

QUESTION 1
K&A Statement: K3.01 - Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Justification:

A. Correct – The loss of RB Chilled Water will result in Drywell Cooling swapover to RBCCW. In order to preserve RBCCW cooling, the swapover logic automatically performs a load shed by isolating the RBCCW supply to the NRHX, HV-11315. Without operator action, the loss of cooling to the NRHX will eventually result in a RWCU isolation on NRHX high temp.

B. In-Correct – For this malfunction, RBCCW cooling is maintained to the RWCU pumps. This is plausible since the RWCU pumps are cooled by RBCCW and trip on high motor temp (135 F). Further, the RWCU pumps trip as a result of the RWCU isolation on valve interlock only. The candidate may remember that RWCU is affected by the RBCCW swapover to drywell cooling and pick this choice.

C. In-Correct – For this malfunction, RBCCW cooling is maintained to the CIG compressors. This is plausible since the CIG Compressors do trip on RBCCW low pressure or high temperature. Candidate may believe all of RBCCW is affected by the swapover.

D. In-Correct – For this malfunction, RBCCW cooling is maintained to the RB Sample Station Coolers. This is plausible since the coolers are cooled by RBCCW and the Chillers (to the coolers) which do trip on low cooling water flow or high temperature. Candidate may believe only less important loads are load shed on the swapover.

References: TM-OP-014-ST pages 11,18,19,23 Student Ref: required No

Learning Objective: TM-OP-014-OB 1694

Question source: New

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 2

Unit 1 experienced a transient that caused reactor water level to lower to minus (-) 40 inches.

- Level has been recovered, RPV level is now +25 inches.
- The AUTO initiation signals have been reset on RCIC and HPCI.
- HPCI Aux Oil Pump Handswitch 1P213 in AUTO.

How will HPCI oil system respond, if the operator places HPCI in Manual and lowers the speed to zero (0) RPM? (Assume this was the only operator action taken)

The Auxiliary Oil Pump (AOP):

A. will automatically start when control oil pressure, as sensed at the discharge of the main lube oil pump, is less than 35 psig. Control oil will be maintained and the Stop valve will close. Oil lubrication to turbine and pump bearings will be provided by the AOP.

B. will **NOT** automatically start when control oil pressure drops below 35 psig during system shutdown and the Stop valve remains open. Oil lubrication to turbine and pump bearings will be provided by the slinger rings.

C. will automatically start when control oil pressure, as sensed at the discharge of the main lube oil pump, is less than 35 psig. Control oil will be maintained and the Stop valve remains open. Pressure Reducing Valve (PCV-15616) will maintain lubricating oil to turbine and pump bearings at 35 to 40 psig.

D. will **NOT** automatically start when control oil pressure drops below 35 psig during system shutdown and the Stop valve will close. Pressure Reducing Valve (PCV-15616) will throttle open and lubricating oil to the turbine and pump bearings will be lost as oil pressure lowers to 0 psig.

K&A # 206000 HPCI
Importance Rating 3.0

QUESTION 2

K&A Statement: K4.13 - Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Turbine and pump lubrication: BWR-2,3,4

Justification:

A. In-Correct - With the HPCI auto initiation signal reset, the AOP will not receive an auto start signal. This is plausible since by procedure, the operator is required to place AOP in START when taking manual control, and candidates are accustomed to seeing the AOP Auto start. If they do not know why the pump is placed it in start, they may choose this answer.

B. In-Correct – HPCI turbine and pump bearing are not lubricated by slinger rings and the stop valve closes. This is plausible since this may be confused with RCIC valves, which open without oil pressure and RCIC has slinger rings to lubricate bearings on startup and shutdown.

C. Correct – Although the HPCI initiation signal is reset, the AOP start logic is sealed-in and the AOP will automatically start on low pressure. The seal-in is maintained provided the handswitch is maintained in AUTO.

D. In-Correct – Although the HPCI initiation signal is reset, the AOP start logic is sealed-in and the AOP will automatically start on low pressure. This is a plausible choice if the candidate believes the seal-in logic is removed when the initiation signal is reset.

References: TM-OP-052-ST pg 40 &41 Student Ref: required No

Learning Objective: TM-OP-052-OB 2032 & 2038

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

****Answer Key changed based on newly discovered technical information discovered during the post-exam review. Correct answer is C.**

QUESTION 3

Unit 1 is operating at rated power with the following Instrument Air (IA) and Service Air (SA) system lineup:

- 'A' IA Compressor, 1K107A is in Lead.
- 'B' IA Compressor, 1K107B is in Standby.
- 'C' Dryer Skid (1F116E/F) is in Service.
- 'B' Dryer Skid (1F116C/D) is in Standby.
- 'A' Dryer Skid (1F116A/B) is OOS for corrective maintenance.
- Service Air compressors are in a normal lineup.

Control Room observes Instrument Air header pressure is 90 psig, slowly lowering. FUS reports a high differential pressure across the 'C' Dryer skid and appears to be obstructed. The 'B' Dryer Skid cannot be placed in service due to a stuck closed outlet Isolation valve.

With IA Header pressure now 80 psig and lowering, which one of the following Crosstie Valves, when opened, can restore air header pressure to supply Instrument Air loads?

A. Unit 1 IA – Unit 2 IA Crosstie Valve, 025091.

B. Unit 1 SA – Unit 2 SA Crosstie Valve, 025125.

C. Unit 1 SA – Unit 1 IA Crosstie valve, PCV-12560.

D. Unit 1 SA – Unit 1 IA PCV Bypass valve, 125143.

K&A # 300000 Instrument Air
Importance Rating 3.0

QUESTION 3

K&A Statement: K4.02 - Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Cross-over to other air systems

Justification:

A. Correct – The IA Unit to Unit cross-tie will supply IA header loads in this condition since the cross-connect is downstream of the IA Dryers. This is the only crosstie that will work in the event of a Dryer malfunction, a realistic failure mode. The unit crosstie is the last option directed in the offnormal procedure due to the impact on both units and requires Shift Supervision approval.

B. In-Correct – The SA Unit to Unit cross-tie will not restore pressure and supply IA loads since it is upstream of the IA Dryers. This is plausible since without knowledge of the flowpaths, the candidate may know that this is the preferred cross-tie, directed by the offnormal procedure.

C. In-Correct – The SA to IA cross-tie will not restore pressure and supply IA loads since it is upstream of the IA Dryers. This is the normally aligned backup and the PCV auto opens on low pressure. In this case, with a dryer malfunction, pressure at the receivers is good and the valve will not open. This is plausible since the first (and preferred) action in the offnormal procedure. This is normally the success path for a loss of Instrument Air.

D. In-Correct – The SA to IA cross-tie will not restore pressure and supply IA loads since it is upstream of the IA Dryers. This is plausible since the candidate may recognize that at 80 psig header pressure, and know the PCV-12560 should have opened. The offnormal procedure directs the bypass opened around the PCV at this pressure since it assumes it failed. This is normally a success path for a loss of Instrument Air.

References: ON-118-001 rev 21 page 21
TM-OP-018-ST page 28-29

Student Ref: required No

Learning Objective: TM-OP-018-OB 10229 & 1772

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Michaels/Morgan

QUESTION 4

During a Unit 1 reactor startup, the ROD OUT BLOCK (AR-104-H03) alarm is received with the following neutron monitoring conditions:

- SRMs 'A' and 'B' are Fully Inserted, reading 5 x 10^4 cps.
- SRM 'C' is Partially withdrawn, reading 1 x 10^3 cps.
- SRM 'D' is Partially withdrawn, reading 60 cps.
- Reactor Period is reading + 50 seconds.
- IRMs are Fully inserted, reading 4 on Range 1.

Which condition below will clear the ROD OUT BLOCK, so that rod withdrawal and the startup may continue?

The ROD OUT BLOCK will clear if:

A. power is allowed to rise until the IRMs indicate above 5 on Range 1.

B. SRM Detectors A and B are driven out until the countrate indicates less than 10^3 cps.

C. a control rod is inserted until period is longer than 300 seconds.

D. SRM Detector D is driven in until the countrate indicates greater than 100 cps.

K&A # 215004 SRM
Importance Rating 2.8

QUESTION 4

K&A Statement: K5.03 - Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM : Changing detector position

Justification:

A. In-Correct - The IRM downscale is bypassed on Range 1. This is a plausible distracter since the IRM reading of 5 is a downscale alarm and a rod out block (bypassed on Range 1 in this case).

B. In-Correct - High countrate is 1 x E5. This is plausible since the candidate may confuse this with an IRM (108/120) Rod Block.

C. In-Correct - Period does not cause the rod out block. This is a plausible distracter since the alarm setpoint is 50 seconds, and by procedure, rod withdrawal is stopped and inserted to establish less that 300 second period.

D. Correct - The detector retract permit is causing the rod block. Retract Permissive interlock is not satisfied if count rate is < 100 cps and the detector is not Full In.

References: AR-104 H03 rev 24
TM-OP-78A-ST page 5, 18

Student Ref: required No

Learning Objective: TM-OP-078A-OB # 1345

Question source: Bank: #TMOP078A/10026/1

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 5

Unit 2 is at rated power with Turbine Control Valve testing in-progress. An EHC malfunction during the test resulted in a reactor pressure transient. No operator action was taken and five seconds into the transient, the following readings were observed on all APRM channels:

- APRM power (FLUX) peaked at 114%, then lowered to pre-transient levels.
- APRM power (STP) peaked at 109%, then lowered to pre-transient levels.
- Reactor Recirculation Drive flow read 90%.
- APRM UPSCALE (AR-203-B06) alarm and Rod Block were received.

Which one of the following statements is correct about the APRM function

A. APRMs are operating correctly. The alarm and rod block were provided by the APRM FLUX function.

B. APRMs are operating correctly. The alarm and rod block were provided by the APRM STP function.

C. APRMs are malfunctioning. The alarm was provided by the APRM STP function but the rod block was not expected at this power level.

D. APRMs are malfunctioning. In addition to the alarm and rod block, a full scram was expected from both APRM STP and FLUX functions.

K&A # 212000 RPS
Importance Rating 2.7

QUESTION 5

K&A Statement: K5.01 - Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM : Fuel thermal time constant

Justification:

A. In-Correct – STP function provides the alarm and rod block. This is plausible since this is often confused with the Flux function. For the stated powers levels, only an Upscale Alarm and Rod Block are expected.

B. Correct - STP filter introduces a six-second time constant to account for the fact that, during power increase, transient neutron flux leads thermal power because of fuel heat transfer time constants. STP power would lag Flux, which is instantaneous. Unit 2 EPU Setpoints (nominal TS) are:

   a) APRM Neutron Flux High Trip is 118 percent (fixed) - Alarm, vote and rod block.

   b) APRM STP High Trip, Unit 2 , 0.62(W) + 62.2% (flow biased, clamped at 113.5%) - Alarm, vote and rod block.

   c) APRM STP High Alarm, Unit 2, 0.62(W) + 57.7% (flow biased, clamped @ 108 percent) - Alarm and rod block only.

   In this question, STP @ 109% will result in an alarm and rod block. Flux @114% provides no alarms, rod block or scram. Note that the Tech Spec settings are presented and the actual instrument settings are set below these.

C. In-Correct – APRMs are functioning correctly. Candidate may believe this based on the differences in the power readings provided on the NUMAC Operator Display. Candidate may either incorrectly calculate the setpoint or not know that the APRM Upscale rod block setting is clamped at 108%.

D. In-Correct – APRMs are functioning correctly. Candidate may believe this based on the differences in the power readings provided on the NUMAC Operator Display. Candidate may believe that the power level was above the rod block and settings. This is plausible since the STP scram setting is 113.5% (clamped in this case) and confuse with Flux trip setting.

References: TM-OP-078D-FS rev 3 page 4
AR-103 A06 rev 32

Student Ref: required No

Learning Objective: TM-OP-078D-OB #15709, 15710

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 6

The 'A' Diesel Generator is in Test running in parallel with the 1A201 4KV ESS Bus. The following occurs:

- Unit 2 DBA LOCA occurs at (time) t = 0 seconds.
- A Loss of Offsite Power (LOOP) occurs at (time) t = 10 seconds.

1) The Unit 1,1A20104 Generator Output Breaker will ____(1)____
2) The Unit 2, 2A20104 Generator Output Breaker will ____(2)____

A. (1) OPEN at t =0 seconds.
(2) CLOSE at t =10 seconds, reenergizing the bus.

B. (1) OPEN at t =10 seconds.
(2) CLOSE at t =10 seconds, reenergizing the bus.

C. (1) OPEN at t =0 seconds.
(2) remain OPEN and breaker must be manually closed.

D. (1) OPEN at t =10 seconds.
(2) remain OPEN and breaker must be manually closed.

	K&A #	262001 AC Electrical Distribution
	Importance Rating	3.5

QUESTION 6

K&A Statement: K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION: Generator trip

Justification:

A. Correct - Unit 1 output breaker opens on a LOCA signal with normal or alternate breakers closed, since the Diesel is running in the Test Mode. Unit 2 output breaker would then close on ESS Bus 2A undervoltage (with normal and alt source breakers open).

B. In-Correct - Unit 1 output breaker opens immediately on LOCA signal.

C. In-Correct - Unit 2 output breaker closes on bus undervoltage (with normal and alt breakers open) as a result of the LOOP.

D. In-Correct - Unit 1 output breaker opens immediately.

References: TM-OP-024-ST pg 63-65 Student Ref: required No

Learning Objective: TM-OP-024-OB 2260

Question source: Bank: #TMOP024/2260/2

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 7

Regarding the Residual Heat Removal system piping, which system maintains the Low Pressure Core Injection System (LPCI) discharge piping filled; AND What is the impact of losing this system?

A. Demineralized Water System supplies keep-fill for LPCI;
Loss of the Demineralized Water System requires the passive Keep-fill system aligned to maintain the piping full.

B. Demineralized Water System supplies keep-fill for LPCI;
Loss of the Demineralized Water System requires starting a RHR pump in each loop IAW with the "Slow Fill" section of OP-149-001.

C. Condensate Transfer System supplies keep-fill for LPCI;
Loss of the Condensate Transfer System requires the passive Keep-fill system aligned to maintain the piping full.

D. Condensate Transfer System supplies keep-fill for LPCI;
Loss of the Condensate Transfer System requires starting a RHR pump in each loop IAW with the "Slow Fill" section of OP-149-001.

K&A # 203000 RHR/LPCI: Injection Mode
Importance Rating 3.3

QUESTION 7

K&A Statement: K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) : Keep fill system

Justification:

A. In-Correct - Demineralized Water System does not normally supply keep fill. It is an alternate source by cross-tying to Condensate Transfer System. This is a plausible distracter since Demin water is an alternate source with operator action.

B. In-Correct - Demineralized Water System does not normally supply keep fill. It is an alternate source by cross-tying to Condensate Transfer System. This is a plausible distracter since Demin water is an alternate source with operator action.

C. Correct – Condensate Transfer System is the normal source of Keep-Fill. Unit 1 and 2 Modifications 220438 (U1) and 308866 (U2) have been installed to provide a passive keepfill system that maintains a source of keepfill water in head tanks installed on Elev. 779' in both reactor buildings.

D. In-Correct – Prior to the keep-fill modification, this would have been the correct choice. This is a plausible distracter since this action is directed by the offnormal procedures if piping pressure is low, due to loss of keepfill.

References: ON-037-001 rev 7 page2, 10 Student Ref: required No

Learning Objective: AD045 #15305

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 8

During rated power operation, a trip of the 480V Load Center supply breaker to the Division 1 Swing Bus Motor-Generator (M-G) set, 1G202, occurs, resulting in a trip of the M-G set.

Complete the statement below regarding the response of the 1B219 Swing Bus.

The 1B219 Swing Bus is:

A. maintained energized since this is a make-before-break transfer to the alternate source, 1B230 480V Load Center.

B. maintained energized since this is a make-before-break transfer to the alternate source, 1B240 480V Load Center.

C. deenergized until the bus automatically transfers to the alternate source, 1B230 480V Load Center, after a short time delay.

D. deenergized until the bus is manually transferred to the alternate source, 1B240 480V Load Center by the NPO at the local panel.

K&A # 262002 UPS (AC/DC)
Importance Rating 2.5

QUESTION 8
K&A Statement: A1.02 - Ability to predict and/or monitor changes in parameters associated with operating the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) controls including: Motor generator outputs

Justification:

A. In-Correct – Swing bus does not immediately transfer to ensure selective tripping and is not make-before-break to prevent a fault on the bus from being transferred to the alternate source. Alternate source is from 1B230 in the same division. Plausible since other UPS schemes exist in the plant that are make-before-break.

B. In-Correct – Swing bus does not immediately transfer to ensure selective tripping and is not make-before-break to prevent a fault on the bus from being transferred to the alternate source. Alternate source is from 1B230 in the same division. Plausible since other UPS schemes exist in the plant that are make-before-break.

C. Correct – The Swing Bus transfer switch will automatically transfer to the alternate source after a short time delay (8-17 seconds depending on sensed trip). Alternate source is from 1B230, 480V LC supplied from 'C' 4KV ESS bus in the same division.

D. In-Correct – The Swing Bus transfer switch will automatically transfer on this condition. This is plausible since the candidate may confuse this with other transfer switches in the plant which require manual local action to transfer. Alternate source is from 1B230 in the same division.

References: TM-OP-004-ST, Pages 20-21 Student Ref: required No

Learning Objective: TM-OP-004-OB # 10121

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 9

A Unit 2 startup is in progress. The reactor is critical and power is in the source range with an infinite period.

The 1D673, +24VDC Battery Charger fails, resulting in no output from the charger.

Assuming no operator action, which describes the effect on 24VDC supplied loads?

A. The +24VDC bus is energized by either charger, in this case 1D674. All Process Rad Monitors, SRMs and IRMs remain energized since no loss of power occurs.

B. The +24VDC bus is energized by the (-) neg 24VDC charger. Half of Process Rad Monitors are de-energized, but SRMs and IRMs remain energized and unaffected.

C. The +24VDC bus will be energized by the battery. A RPS half-scram will occur in about four hours when bus power is lost to the Process Rad Monitors, SRMs and IRMs.

D. The +24VDC bus will be energized by the battery. A RPS full-scram will occur in about four hours when bus power is lost to all Process Rad Monitors, SRMs and IRMs.

K&A # 263000 DC Electrical Distribution
Importance Rating 2.5

QUESTION 9
K&A Statement: A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: Battery charging/discharging rate

Justification:

A. In-Correct - The -24VDC charger cannot carry the +24VDC bus. If the candidate believes that either of the chargers can carry the entire bus, this answer will be chosen.

B. In-Correct - Battery will carry the loads as it discharges If the candidate does not understand the electrical bus alignment, this answer will be chosen. Candidates may believe that the SRMs and IRMs will have redundant power supplies and will remain energized.

C. Correct - The battery will initially carry the load but will discharge without the charger. As the battery discharges (4 hours), the bus will eventually be lost and the Division 1 IRMs will de-energize, resulting in a RPS Half-Scram.

D. In-Correct - The battery will initially carry the load but will discharge without the charger. As the battery discharges (4 hours), the bus will eventually be lost and the Division 1 IRMs will de-energize, resulting in a RPS Half-Scram, not a Full-Scram.

References: TM-OP-075-ST, pages 4,10 Student Ref: required No

Learning Objective: AD045 # 15308 & TM-OP-075-OB # 10102

Question source: Bank: #183

Question History: LOC 21 Cert Exam

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 10

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

- PCIS LOOP A RX LEVEL 2 ISOLATION alarm (AR-111-A02) was received.
- Rx Water level is in the normal operating band.
- No other control room alarms exist.

I&C reports that LITS-B21-1N026A Wide Range Level Instrument minus (–) 38 inch switch has failed in the tripped position. All other switches associated with the level instrument are OPERABLE.

Subsequently, during troubleshooting on 1C004 rack, I&C inadvertently trips the LITS-B21-1N026B Level Transmitter downscale.

Predict the impact of the transmitter failures and the required procedure actions. (Assume I&C restores LITS-B21-1N026B Level Transmitter to normal.)

A. Inboard and Outboard MSIVs will close and reactor will scram; Perform EO-100-102 "RPV Control" and ON-184-001 "Main Steam Line Isolation And Quick Recovery" and reopen the MSIVs.

B. Inboard MSIVs will close, Outboard MSIVs will **NOT** close, and reactor will scram; Perform EO-100-102 "RPV Control" and ON-184-001 "Main Steam Line Isolation And Quick Recovery" and reopen the Inboard MSIVs.

C. RWCU Inboard and Outboard Isolation Valves, F001 and F004, will close; Implement ON-159-002 "Containment Isolation" to reset N4S Logic and restore RWCU to service IAW OP-161-001 "RWCU System".

D. RWCU Inboard Isolation Valve, F001, will close; Outboard Isolation Valve, F004, will **NOT** close; Implement ON-159-002 "Containment Isolation" to reset N4S Logic and restore RWCU to service IAW OP-161-001 "RWCU System".

K&A # 223002 PCIS/NSSS
Importance Rating 3.3

QUESTION 10

K&A Statement: A2.05 - Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abn cond or ops. Nuclear boiler instrumentation failures

Justification:

A. In-Correct – The MSIV isolation is at – 129 inches; This is a plausible distracter since with the LITS-B21-1N026A switch failed and LITS-B21-1N026B failing downscale, the students may believe that a full MSIV isolation signal is present. The isolations are from the same transmitters, but use different switches. Only a quarter isolation exists on the MSIV logic.

B. In-Correct – The MSIV isolation is at – 129 inches; This is a plausible distracter since with the LITS-B21-1N026A switch failed and LITS-B21-1N026B failing downscale, the students may believe that a full MSIV isolation signal is present. The isolations are from the same transmitters, but use different switches. Only a quarter isolation exists on the MSIV logic.

C. In-Correct - The logic de-energizes (open) to activate. With LITS-B21-1N026A, -38 inch switch failed OPEN; the INBD logic is only waiting for the 1N026B to indicate below minus (-) 38 inches and close the INBD Isolation valves. This is a plausible distracter since the candidate may believe that with the LITS-B21-1N026A failed, no isolation will occur.

D. Correct – N4S logic requires 2-out-of-2 to close (need A and B channels for INBD valves or need C and D channels for OTBD valves) Isolation valves with the exception of MSIV logic (MSIVs are 1-out-of-2 twice logic). N4S logic de-energizes (open) to activate. With LITS-B21-1N026A, minus (-) 38 inch switch failed OPEN; the INBD logic is only waiting for the 1N026B to indicate < minus (-) 38 inches to close the INBD Isolation valves. ON-159-002 is used to reset the logic and OP-161-001 is used to restore RWCU System.

References: TM-OP-061-ST, page 8
TM-OP-059B-ST, page 17

Student Ref: required Yes

ON-145-004 pages 1-15

Learning Objective: TM-OP-061-OB-1693
TM-OP-059B-OB-2142

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 11

A loss of feedwater occurs and the reactor scrams at +13 inches. RPV level is slowly lowering. The Unit Supervisor directs you to place RCIC in-service to maintain RPV level +13 to +54 inches.

Before you initiate RCIC, you notice the following two alarms:

- RCIC OUT OF SERVICE (AR-108-B05).
- RCIC INV FAILURE (BIS AR-151-B01).

Predict the impact of these alarms and the required procedure actions.

A. The Governor Valve will fail OPEN; Implement ES-150-003, "RCIC Manual Injection With Loss Of AC And DC Power" to place RCIC in-service and maintain RPV level +13 to +54 inches.

B. The Governor Valve will fail OPEN; Implement OP-150-001, "RCIC System" Section 2.5, Manual Startup Using Turbine Trip And Throttling Valve and maintain RPV level +13 to +54 inches.

C. The Governor Valve will fail CLOSED; Implement ES-150-003, "RCIC Manual Injection With Loss Of AC And DC Power" to place RCIC in-service and maintain RPV level +13 to +54 inches.

D. The Governor Valve will fail CLOSED; Implement OP-150-001, "RCIC System" Section 2.5, Manual Startup Using Turbine Trip And Throttling Valve and maintain RPV level +13 to +54 inches.

K&A # 217000 RCIC
Importance Rating 3.1

QUESTION 11

K&A Statement: A2.10 - Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine control system failures

Justification:

A. In-Correct – ES-150-003 is not required or directed to use. It is only directed during a loss of AC and DC power to the valves, not control circuit power. Plausible Distracter, since direction to use ES-150-003 is on EO-100-102, for a loss of AC 480 VAC and 250 VDC power.

B. Correct - Loss of control signal will fail governor valve full open, caused by Topaz inverter failure or loss of power. In this condition, the operating procedure OP-150-001 directs RCIC turbine operation utilizing Section 2.5 Manual Startup using Turbine Trip and Throttle valve.

C. In-Correct- The Governor Valve fails open with no control signal. Plausible Distracter since the student may confuse the Governor Valve response with HPCI. ES-150-003 is not required or directed to use. It is only directed during a loss of AC and DC power to the valves, not control circuit power. Plausible Distracter, since direction to use ES-150-003 is on EO-100-102, for a loss of AC 480 VAC and 250 VDC power.

D. In-Correct- The Governor Valve fails open with no control signal. Plausible Distracter since the student may confuse this with the HPCI Governor valve, which fails closed.

References: OP-150-001 rev 27, Section 2.5 Student Ref: required No
AR-151-B01 rev 4

Learning Objective: TM-OP-050-OB # 10441 and 2015

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 12

An unisolable leak has occurred outside of the primary containment. The following timed sequence of plant conditions is provided:

t = 0 seconds:	RPV level is minus (-) 130 inches, lowering. RPV pressure is 800 psig, lowering slowly. Drywell pressure is 1.4 psig, rising slowly.
t = 100 seconds:	RPV level is minus (-) 145 inches, lowering. RPV pressure is 775 psig, lowering slowly. Drywell pressure is 1.8 psig, rising slowly.

Assuming no change in parameter trends above and all ECCS equipment operates as designed, which one of the following is correct regarding the actuation of the Automatic Depressurization System (ADS) valves?

Based on the timeline above, the ADS valves will automatically open when time reaches:

A. t = 102 seconds.

B. t = 202 seconds.

C. t = 420 seconds.

D. t = 522 seconds.

K&A # 218000 ADS
Importance Rating 3.7

QUESTION 12
K&A Statement: A3.04 - Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: Primary containment pressure

Justification:

A. In-Correct – ADS valves do not open at t =102 seconds. With Drywell pressure < 1.72 psig, the ADS Drywell pressure bypass timer starts (7 minutes or 420 seconds) when level reaches -129 inches. The ADS Logic Initiation timer (102 seconds) is not running.

B. Correct – ADS valves will open at t =202 seconds. With RPV level below -129 inches, the ADS Logic Initiation timer (102 seconds) starts when drywell pressure reaches 1.72 psig (t= 100 seconds). Candidate must also recognize that ECCS pumps (logic permissive) will be running when level reached -129 inches.

C. In-Correct – ADS valves will open at t =202 seconds. With RPV level below -129 inches, answer is plausible if the candidate believes that the valves will not open until the Drywell pressure bypass timer is satisfied (7 minutes or 420 seconds).

D. In-Correct – ADS valves will open at t = 202 seconds. With RPV level below -129 inches, answer is plausible if the candidate believes that the valves will not open until the Drywell pressure bypass timer is satisfied (7 minutes or 420 seconds) which would then start the ADS Logic Initiation timer (102 seconds). From t=0, the time of 522 seconds is t=420 seconds + 102 second timer. This would be correct if the High Drywell pressure signal did not exist at t=100 seconds.

References: TM-OP-083E-ST, pages 9-11 Student Ref: required No

Learning Objective: TM-OP-083E-OB # 2105

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 13

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

- MAIN STEAM DIV 2 SRV OPEN (AR-110-E03) alarm received.
- All SRV Handswitches on 1C601 are in the AUTO position.
- No RED indicating lights on the SRV handswitches are lit.
- No other control room alarms exist.

Which monitoring system is providing the alarm; AND What is the status of the SRVs?

A. SRV/ADS Temperature Recorder TRS-B21-1R614;
SRV is open on the Safety feature.

B. SRV/ADS Temperature Recorder TRS-B21-1R614;
SRV is open on the Relief feature.

C. Acoustical Monitor;
SRV is open on the Safety feature.

D. Acoustical Monitor;
SRV is open on the Relief feature.

K&A # 239002 SRVs
Importance Rating 3.6

QUESTION 13
K&A Statement: A3.04 - Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: Acoustical monitor noise: Plant-Specific

Justification:

A. In-Correct - SRV/ADS Temperature Recorder TRS-B21-1R614 does not cause this alarm. If the temperature recorder senses increasing temperature, the Yokogawa recorder brings in AR-110-E01 SRV LEAKING alarm. Plausible Distracter since the candidate may assume that the temperature recorder caused this alarm.

B. In-Correct - SRV/ADS Temperature Recorder TRS-B21-1R614 does not cause this alarm. If the temperature recorder senses increasing temperature, the Yokogawa recorder brings in AR-110-E01 SRV LEAKING alarm. Plausible Distracter since the candidate may assume that the temperature recorder caused this alarm. The SRV is not open on the Relief mode, as indicated by no red indicating lights.

C. Correct – Acoustic Monitor causes this alarm, and with no red indicating light on the SRV handswitches, the SRV is not open in the Relief mode. This conclusion is reached by interpreting both alarm and light indications on Control Room panel 1C601.

D. In-Correct - With no red indicating light on the SRV handswitches, the SRV is not open in the Relief mode. Plausible Distracter since the Relief mode is one of the mode of SRV operation and the candidate may misinterpret the panel indications.

References: AR-110-001 E03 and TM-OP-83-ST page 7-9 and 12 | Student Ref: required No

Learning Objective: TM-OP-083-OB # 10240 and 10021

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 14

Unit 2 is operating at rated power when the following occurs:

- SBLC SQUIB VLVS LOSS OF CKT CONTINUITY (AR-207-A03) alarms.
- PCO observes the white light for the “A” Squib Valve is **NOT** lit.

Upon a manual initiation of the Unit 2 Standby Liquid Control (SLC) system:

A. either “A” or “B” SLC Pump can be started to inject at approx 40 gpm.

B. only the “B” SLC Pump can be started to inject at approx 40 gpm.

C. both SLC Pumps will start and inject at approx 80 gpm.

D. both SLC Pumps will start to inject at approx 40 gpm.

K&A # 211000 SLC
Importance Rating 4.1

QUESTION 14
K&A Statement: A4.05 - Ability to manually operate and/or monitor in the control room: Flow indication: Plant-Specific

Justification:

A. Correct – One pump flow is ≥40 gpm and due to recent plant mod for power uprate, only one pump will start. Plant designed such that a failure of one squib valve will not prevent full flow to the RPV.

B. In-correct – Either pump can be started. Plausible Distracter since the candidate may believe that with the "A" squib valve INOP that he must use the "B" SLC Pump and "B" squib valve. Plausible since the control circuit of the pump interfaces with the squib valve but a failure or loss of power in the squib valve circuit will not affect pump start circuit.

C. In-correct – With the recent plant mod, only one SLC Pump can be running. The new design of the handswitch only allows you to either start the "A" or "B" SLC Pump. Plausible Distracter since the Unit 1 current configuration will start both SLC Pumps. At the completion of this refueling outage, its configuration will be similar to Unit 2. The candidate may also believe that a failure of one (1) squib valve will only allow half flow.

D. In-correct – With the recent plant mod, only one SLC Pump can be running. The new design of the handswitch only allows you to either start the "A" or "B" SLC Pump. Plausible Distracter since the Unit 1 current configuration will start both SLC Pumps. At the completion of this refueling outage, its configuration will be similar to Unit 2.

References: TM-OP-053-ST page 21
OP-253-001 rev 24 page 5
Student Ref: required No

Learning Objective: TM-OP-053-OB # 1209, 1210, and 1214

Question source: Modified Bank # TMOP053/1214 3 See original attached

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 15

Unit 1 is operating at full power with APRM Channel #3 tripped INOP and previously Bypassed.

Later in the shift, an unrelated problem results in the Reactor Recirculation Drive Flow instrument, FT-B31-1N014A failed to zero (0% flow signal), an input to APRM Channel #1.

Given the above conditions, which one of the following is the expected response of the Power Range Neutron Monitoring System?

A. No Rod Block, No Scram Vote and No RPS Scram Channels tripped.

B. No Rod Block, One Scram Vote and One RPS Scram Channel tripped – Half Scram condition.

C. A Rod Block, One Scram Vote and No RPS Scram Channels tripped.

D. A Rod Block, Two Scram Votes and all Four Scram Channels tripped – Full Scram condition.

K&A # 215005 APRM / LPRM
Importance Rating 3.7

QUESTION 15
K&A Statement: A3.08 - Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Control rod block status

Justification:

A. In-Correct – At full power, the failed Recirc flow transmitter low would result in an APRM Upscale High condition, Rod Block as well as APRM Upscale Trip condition, one Vote condition, due to flow-biased setpoints exceeded. This is a possible choice if the candidate does not recognize the impact of the failed drive flow signal on the APRM.

B. In-Correct – At full power, the failed Recirc flow transmitter low would result in an APRM Upscale High condition, Rod Block as well as APRM Upscale Trip condition, one Vote condition, due to flow-biased setpoints exceeded. This is a possible choice if the candidate confuses the RPS scram channels and does not understand how the 2-Out-Of-4 Logic Module functions. Also, before the PRNMS modification, one APRM failure did result in a half-scram, similar to one IRM channel failed in ½ twice logic.

C. Correct – The flow transmitter instrument failure (0% flow) results in half of the flow signal originally input to the APRM. At full power, the power signal will exceed the flow-biased setpoint for the power signal resulting in an APRM Upscale High, Rod Block and APRM Upscale Trip, Scram Vote. With the 2-Out-Of-4 Logic module, aka Voter module, no scram channels will be tripped in RPS, since it requires two simultaneous votes. With APRM #3 Bypassed, only one vote exists.

D. In-Correct – Only one Vote exists from APRM #1, APRM #3 although tripped is bypassed at the Voter Module. With only one vote, no RPS scram channels are tripped. If the candidate believes two votes exist, then all four scram channels would be tripped, Full Scram and this choice is plausible.

References: TM-OP-078D-ST pages 23,24, 32, 42 Student Ref: required No

Learning Objective: TM-OP-078D-OB #15708, 15716

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 16

During 100 percent power operation on Unit 1, the "A" RPS MG Set fails, resulting in a loss of the "A" RPS Bus. What is the expected response of the Containment Radiation Monitoring System (CRMs) Containment Isolation Valves?

A. The Outboard and Inboard Isolation Valves close in both "A" and "B" Loops of the CRMs.

B. The Outboard Isolation Valves close, and the Inboard Valves remain as is in both "A" and "B" Loops of the CRMs.

C. The Inboard Isolation Valves close, and the Outboard Valves remain as is in both "A" and "B" Loops of the CRMs.

D. The Inboard and Outboard Isolation Valves close in the "A" Loop of the CRMs; the "B" loop Isolation Valves remain as is.

K&A # 212000 RPS
Importance Rating 3.3

QUESTION 16

K&A Statement: K1.05 - Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following: Process radiation monitoring system

Justification:

A. In-Correct – A loss of "A" RPS results in Inboard Isolation of PCIS isolation valves for the CRMs. Plausible distracter; Candidates may confuse this with a loss of "B" RPS which results in both divisions of PCIS.

B. In-Correct – A loss of "A" RPS results in Inboard Isolation of PCIS isolation valves for the CRMs. Plausible distracter; Candidates may confuse which set of valves (Inboard vs. Outboard) are Div 1 and Div 2.

C. Correct – A loss of "A" RPS results in Inboard Isolation of PCIS isolation valves for the CRMs. Both inboard isolation valves close for both CRM channels.

D. In-Correct – A loss of "A" RPS results in Inboard Isolation of PCIS isolation valves for the CRMs. Plausible distracter; Candidates routinely get this concept mixed up with the H2/O2 Analyzers.

References: ON-158-001 Attachment "A" Student Ref: required No

Learning Objective: TMOP079X obj #1901 and AD045 obj # 15300

Question source: Modified: TMOP079X/1903/2 See original attached

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 17

Unit 2 is operating at 100% power with Feedwater Level Control in a normal alignment, when Steam Flow Transmitter, 2N003B fails.

- FI-C32-2R603B on 2C652 SIP Panel indicates zero (0) lbm/hr.

Which of the following is the response of the Feedwater Water Level Control System?

A. Master FW Level CTL/Demand Signal LIC-C32-2R600 will null (vertical meter in green band). RPV level will stabilize with the RFPs controlling at a lower than original level.

B. Master FW Level CTL/Demand Signal LIC-C32-2R600 will NOT null (vertical meter out of band). RPV level will continue to lower to the scram setpoint.

C. Master FW Level CTL/Demand Signal LIC-C32-2R600 will null (vertical meter in green band). RPV level will stabilize with the RFPs controlling at a higher than original level.

D. Master FW Level CTL/Demand Signal LIC-C32-2R600 is unaffected (vertical meter remains as-is). RPV level will not change and remain at its original level.

K&A # 259002 Reactor Water Level Control
Importance Rating 3.2

QUESTION 17

K&A Statement: K1.02 - Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: Main steam flow

Justification:

A. Correct – When one steam flow transmitter fails downscale, FWLC System will immediately see an error of too much feed flow for the indicated steam flow. The 2R600 controller vertical meter will indicate upscale. The FWLC System will respond by lowering RFPT Speed until the vertical meter is again nulled at ≈35 inches conditioned signal. Actual water level will be ≈25 inches

B. In-Correct – Master FW Level CTL/Demand Signal LIC-C32-2R600 vertical meter will null. If the candidate believes that the controller will NOT null (first indication of failure would be controller vertical meter (Demand) would fail high) than he would chose this responds.

C. In-Correct – The FWLC System will respond by lowering RFPT Speed until the vertical meter is again nulled at ≈35 inches conditioned signal. Actual water level will be ≈25 inches. If the failure was a feed flow instrument level would stabilize at a high level.

D. In-Correct – When one steam flow transmitter fails downscale, FWLC System will immediately see an error of too much feed flow for the indicated steam flow. The 2R600 controller vertical meter will indicate upscale and as level recovers to a new value the controller will null. If the candidate believes this steam flow transmitter and indicator is **not** part of FWLC system (indication only or part of MSIV PCIS Logic System), he will choose this response.

References: ON-245-001 rev 21 pages 2,3 Student Ref: required No

Learning Objective: TM-OP-045-OB # 10291 & # 10297

Question source: New See original attached

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 18

Unit 1 is in Mode 4 with the 1B RHR pump in Shutdown Cooling. The following conditions exist to support outage maintenance:

- 1A201 4KV Bus de-energized.
- 'A' RPS Bus on alternate power.
- Primary Containment has been relaxed.
- Secondary Containment has been relaxed.

A leak developed on the 'B' RHR Heat Exchanger Inlet Isolation valve. The operator manually shutdown and isolated the 'B' RHR loop and RPV Level lowered to and stabilized at +35 inches.

Which of the following is required to establish reactor cooling IAW ON-149-001, "Loss of RHR Shutdown Cooling Mode"?

A. Restore RPV level to +90 inches and place the 'A' RHR loop of Shutdown Cooling in-service IAW OP-149-002. Power is available to 1C RHR pump and RHR pump suction F006C and RHR injection F015A valve.

B. Maintain RPV level and place the 'A' RHR loop of Shutdown Cooling in-service IAW OP-149-002. Power is NOT available to 'A' Loop valves, locally open the RHR pump suction F006C and RHR injection F015A valve.

C. Maintain RPV level and allow pressure to rise to 20 to 98 psig and open ADS SRVs to maintain pressure within this range, place 'A' loop of RHR in Supp Pool Cooling IAW OP-149-005.

D. Restore RPV level to +90 inches and open ADS SRVs and circulate water from the reactor to supp pool via the SRVs, place 'A' loop of RHR in Supp Pool Cooling IAW OP-149-005.

K&A # 205000 Shutdown Cooling
Importance Rating 2.5

QUESTION 18

K&A Statement: K2.02 - Knowledge of electrical power supplies to the following: Motor operated valves

Justification:

A. Correct – With the loss of 'B' Loop RHR, the 'A' Loop of RHR remains partially available for Shutdown Cooling. The candidate must evaluate that power is available to the following components:
- 1C RHR pump power from 1A203, '1C' ESS Bus.
- F006C Powered from 1B237 MCC, also powered from the 1A203, '1C' Bus.
- F015A is powered from the Swing Bus and available.

This allows the crew to place the 'A' Loop RHR in Shutdown Cooling. RPV level must be restored above +45 inches for core circulation.

B. In-Correct – Power is available from 1B237 and swing bus. Plausible detractor if the student believes all shutdown cooling valves powered from 'A' Bus. RPV level must be restored above +45 inches for core circulation.

C. In-Correct –IAW ON-149-001 Primary and Secondary containment must be established to use this method for decay heat removal. Plausible detractor, if student believes that SDC is lost and cannot be restored by the 'A' RHR loop he may chose this course of action. RPV level must be restored above +45 inches for core circulation.

D. In-Correct – RPV level must be raised to 131 inches in order to establish communication between RPV and Supp Pool. Plausible detractor, if student believes that SDC is lost and cannot be restored by the 'A' RHR loop he may chose this course of action. If this alternate decay heat removal method were required, RPV level must be raised to the Main Steam Lines.

References: TM-OP-049-ST page 15, 39 Student Ref: required No
ON-149-001 rev 20 page 7
ON-104-203 rev 9 page 19

Learning Objective: TM-OP-049-OB 196 & 10499

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 19

Unit 2 was in MODE 1 when a reactor scram occurred. Concurrently, the 4.16 kV ESS Bus 2A202 de-energized due to a lockout.

Several minutes later, the following plant conditions exist:

- Drywell pressure is 15 psig, rising slowly.
- RPV level is -100 inches, steady.
- RPV pressure is 400 psig, lowering.

Given that the above conditions have existed for the past two minutes and no operator actions have been taken, which of the following describes the status of the Core Spray Loop 'B', 2B and 2D Pumps AND the 'B' Inboard Injection Valve, HV-252-F005B?

A. CS Pump 2B is NOT running.
CS Pump 2D is NOT running.
HV-252-F005B is CLOSED.

B. CS Pump 2B is NOT running.
CS Pump 2D is running.
HV-252-F005B is CLOSED.

C. CS Pump 2B is running.
CS Pump 2D is NOT running.
HV-252-F005B is OPEN.

D. CS Pump 2B is NOT running.
CS Pump 2D is running.
HV-252-F005B is OPEN.

K&A # 209001 LPCS
Importance Rating 3.0

QUESTION 19
K&A Statement: K2.01 - Knowledge of electrical power supplies to the following: Pump power

Justification:

A. In-Correct – With 2A202 deenergized, 2D pump should be running with the injection valve closed (no power). This choice is plausible if the candidate does not recognize that a LOCA signal exists (RPV level is given to be above -129 inches).

B. Correct – With a LOCA signal, both pumps have start signals but the '2B' pump does not have bus power. Only the '2D' pump is running. RPV pressure is below the low pressure permissive (420 psig) and the valve has an open signal and should be fully open (after one minute). However, the Inboard Injection Valve, F005B is powered from 1B227, which is powered from the 2B ESS Bus, 2A202. This valve remains closed.

C. In-Correct - A LOCA signal exists along with the low pressure permissive for the injection valves to open. This choice is plausible if the candidate does not recognize that bus power to the '2B' pump motor is lost as well as power supply to the F005B. Candidate may also believe that F005 is the normally open injection valve.

D. In-Correct – With 2A202 deenergized, 2D pump should be running with the injection valve closed. This choice is plausible if the candidate does not recognize that LOCA conditions exist (RPV level is given to be above -129 inches) or if the candidate does not recognize that valve power is from the deenergized bus. Candidate may also believe that F005 is the normally open injection valve.

References: TM-OP-051-ST, pages 31, 56 Student Ref: required No
ON-204-202 rev 10 pages 15,17

Learning Objective: TM-OP-051-OB #10386

Question source: Bank: #TMOP051/2093/2

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 20

Unit 1 is at rated power and Unit 2 is shutdown in Mode 4 with the following conditions:

- Startup Bus 20 is out of service and deenergized for maintenance.
- All 4KV ESS buses are energized.
- '1A' 4KV Bus normal supply breaker, 1A20101 spuriously trips.
- The 'A' Emergency Diesel Generator (EDG) started and successfully re-energized the bus.
- Five minutes later, ENGINE LUBE OIL PRESSURE LOW (LA-0521-A01) alarm received at local Panel 0C521A and the NPO reports lube oil pressure is lowering.

Which of the below describes the design response of 'A' EDG and '1A' 4KV Bus 1A201 to these conditions?

A. The Diesel will trip and the DG output breaker to Bus 1A201 will open. The 4KV Bus 1A201 will remain de-energized.

B. The Diesel will trip and the DG output breaker to Bus 1A201 will open. The Alternate supply breaker 1A20109 will close re-energizing 1A201.

C. The Diesel will continue to run in EMERGENCY MODE with the Standby Oil pump operating, and will maintain 4KV Bus 1A201 energized.

D. The Diesel will continue to run in EMERGENCY MODE until engine seizes and trips, then the Alternate supply breaker 1A20109 will close re-energizing 1A201.

K&A # 264000 EDGs
Importance Rating 3.9

QUESTION 20

K&A Statement: K3.02 - Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: A.C. electrical distribution

Justification:

A. Correct – With the DG running in Emergency, this Low Lube Oil pressure condition will trip the diesel which in turn, trips the generator output breaker. Based on the electrical lineup and no power available from SUB 20, 1A201 will deenergize.

B. In-Correct – The bus will remain de-energized. Based on the electrical lineup and no power available from SUB 20, 1A201 will remain deenergized. This is plausible since this is the normal design response of the 4KV buses and if the candidate does not evaluate the electrical lineup, they could choose this answer.

C. In-correct – This Low Lube oil pressure will trip DG even when running in EMERGENCY. This is plausible since the candidate may believe the Standby motor driven lube oil pump would restore pressure and prevent the shutdown. This oil pressure, based on the alarm and report does, in-fact, result in an immediate diesel shutdown.

D. In-correct - This Low Lube oil pressure will trip DG even when running in EMERGENCY. This is plausible since the candidate may believe the Standby motor driven lube oil pump would restore pressure and prevent the shutdown. This oil pressure, based on the alarm and report does, in-fact, result in an immediate diesel shutdown. Also, in this electrical lineup the alternate supply breaker cannot close because SUB 20 is out of service.

References: TM-OP-024-ST pages 52,53
LA-0521-A01, C01 rev 15 pages 2,18

Student Ref: required No

Learning Objective: TM-OP-024-OB #2068 b

Question source: Modified Bank: #7 See original question attached.

Question History: None recent

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

10CFR 55.41 X

Comments: Created or Modified by: Morgan
Reviewed by: Michaels

QUESTION 21

Which one of the following will require an immediate entry into a Technical Specification LCO?

A. Loss of power to SGTS Fan 0V109A.

B. Loss of power to DW Cooler Fan 1V415A.

C. Loss of power to Unit 1 RB Chiller 1K206A.

D. Loss of power to DG BLDG Fan 0V512A.

K&A # 261000 SGTS
Importance Rating 3.9

QUESTION 21
K&A Statement: 2.2.42 - Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (SGTS)

Justification:

A. Correct – This is a 7 day LCO in accordance with TS 3.6.4.3. SGTS is a safety system required to mitigate an accident sequence.

B. In-Correct – This is not a Tech Spec LCO. This is a plausible distracter since select Drywell fans are required for Tech Specs and a loss of power to Fan 1V414A/B, 1V416A/B, or 1V418A/B is a TS entry. The stated DW fan is not a Tech Spec required fan.

C. In-Correct – This is not a Tech Spec LCO. This is a plausible distracter since the Reactor Building ventilation system is required for Secondary Containment. A loss of ventilation would require a plant shutdown and potential entry into emergency procedure EO-100-104 Secondary Containment Control.

D. In-Correct – This is not a Tech Spec LCO. This is a plausible distracter since the DG is an emergency electrical power supply and the candidate may believe that the DG Supply fan is a required Tech Spec component.

References: TS 3.6.4.3 Student Ref: required No

Learning Objective: TM-OP-070-OB # 12637

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 22

A normal plant startup is in progress and preparations are being made to begin control rod withdrawal. Condenser vacuum has been established. Level is being lowered to establish the required RPV level band of 30 to 40 inches and the following RPV level indications are observed:

- Upset Range Level Instruments read 25 inches.
- Shutdown Range Level Instrument reads 35 inches.
- Narrow Range Level Instruments read 50 inches.
- Wide Range Level Instruments read 60 inches.
- Extended Range Level instruments read 60 inches.

Determine actual RPV level using ON-145-004, "RPV Anomaly" (Attachment A) and the action required to control RPV level at this point in the GO-100-002, Plant Startup procedure.

A. Actual RPV level is approx. 35 inches. Maintain level by throttling the FW START UP BYPASS HV-10640.

B. Actual RPV level is approx. 35 inches. Maintain level by controlling RWCU letdown flow.

C. Actual RPV level is approx. 50 inches. Lower level by throttling RWCU letdown flow.

D. Actual RPV level is above 60 inches. Lower level by throttling the FW START UP BYPASS HV-10640.

K&A # 259002 Reactor Water Level Control
Importance Rating 4.2

QUESTION 22

K&A Statement: 2.2.44 - Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives effect plant and system conditions. (Reactor Water Level Control)

Justification:

A. In-Correct – Level is in band and maintained using only RWCU letdown (and CRD makeup). Although feedwater is in long-path recirculation, feedwater is not used for level control at this point in the reactor startup.

B. Correct – Using ON-145-004 Step 3.4 and Attachment 'A', actual RPV level is approximately 35 inches. Level is in band and will be maintained using RWCU letdown (and CRD makeup) at this point in the startup, prior to rod withdraw IAW GO-100-002 Step 5.10.

C. In-Correct – Actual RPV level is 35 inches. Candidate may believe level is high if the instruments are not correctly interpreted correctly using ON-145-004 Attachment 'A'. Actual level of 50 inches is plausible if the candidate believes NR level is accurate and usable.

D. In-Correct – Actual RPV level is 35 inches. Candidate may believe level is high if the instruments are not correctly interpreted using ON-145-004 Attachment 'A'. Actual level of above 60 inches is plausible if the candidate incorrectly plots extended and wide range on the curves. Further, level is maintained using only RWCU letdown (and CRD makeup). Although feedwater is in long-path recirculation, feedwater is not used for level control at this point in the reactor startup.

References: ON-145-004, rev 12, Step 3.4
ON-145-004 Attachment 'A'
GO-100-002, Step 5.10

Student Ref: required Yes

ON-145-004 Attachment 'A'

Learning Objective: AD045 #15307

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 23

Which one of the following accident conditions provides Adequate Core Cooling as defined by the Emergency Operating Procedures?

A. Reactor is shutdown; LPCI is injecting and RPV level is -205 inches and steady.

B. Reactor is shutdown and level is indeterminate; LPCI is injecting, 3 SRV are open and RPV Pressure is > 82 psig above Supp Chamber pressure.

C. Reactor is **NOT** shutdown and level is indeterminate; LPCI is injecting, 6 SRVs are open and maintaining RPV pressure above 152 psig.

D. Reactor is **NOT** shutdown; LPCI injecting and RPV level is – 205 inches and steady.

K&A # 203000 RHR/LPCI: Injection Mode
Importance Rating 3.5

QUESTION 23

K&A Statement: K5.02 - Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) : Core cooling methods

Justification:

A. In-Correct – The Minimum Zero Injection Water level, aka. steam cooling does not establish ACC in this condition since LPCI is injecting. This is a plausible distracter since steam cooling may be used if the water level is below TAF and no makeup source is available. The candidate may believe ACC exists.

B. In-Correct – In this condition, maintaining the reactor depressurized with more than 2 SRVs open does not ensure ACC. Flooding to the MSL to provide core submergence is a method of ensuring ACC. This is a plausible distractor since depressurization is a required condition stated in the EOPs but again in no way assures ACC.

C. Correct – This condition assures ACC in the given conditions RPV level indeterminate during an ATWS. Adequate Core Cooling is accomplished by maintaining minimum steam cooling pressure during an ATWS. The candidate will evaluate that the stated combination of SRVs and RPV pressure is acceptable for maintaining ACC.

D. In-Correct – The Minimum Zero Injection Water level, aka. steam cooling stated here does not establish ACC. Further, use of this strategy is not permitted in an ATWS condition. This is a plausible distracter since steam cooling with injection is a method of adequate core cooling and the candidate may believe ACC exists.

References: SC006P page 2, 11
EO-000-114 page 15

Student Ref: required Yes
EO-100-114 flowchart

Learning Objective: SC006P # 5466

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 24

The 'E' Diesel Generator is running per SO-024-014 to demonstrate operability. The 'E' diesel is NOT substituting for another diesel.

Which one of the following describes the effect a Loss of Offsite Power (LOOP) will have on 'E' Diesel Generator while running in this condition?

The 'E' Diesel Generator will:

A. automatically trip. The NPO will verify that the diesel ESW Supply/Return valves automatically close.

B. automatically trip. The NPO must manually close the diesel ESW Supply/Return valves from Panel 0C577E.

C. continue to run. The NPO must manually shutdown the diesel then manually close the ESW Supply/Return valves locally since no AC power is available to the MCC.

D. continue to run. The NPO must emergency shutdown the diesel since the Intercooler and Jacket Water Valves fail open and then close the diesel ESW Supply/Return valves from Panel 0C577E.

K&A # 264000 EDGs
Importance Rating 3.2

QUESTION 24

K&A Statement: K1.04 - Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Emergency generator cooling water system

Justification:

A. Correct – When running in Test, the 'E' DG will immediately shutdown upon receipt of a LOOP or LOCA signal. This will automatically close the ESW supply and return valves to ensure ESW supplied to the diesels and the 'E' DG will immediately shutdown (on the Emergency Service Water Interlock).

B. In-Correct – The ESW supply and return valves will automatically close and in-fact it is this signal which shuts down the diesel automatically. This is plausible since normally, during test operation, the ESW valves are manually opened prior to the DG test and closed when the test is completed, DG shutdown.

C. In-Correct - The ESW supply and return valves will automatically close and in-fact it is this signal which shuts down the diesel automatically. This is plausible since normally, during test operation, the ESW valves are manually opened prior to the DG test and closed when the test is completed, DG shutdown. The 'E' DG ESW supply and return valves are DC powered from an independent battery system in the 'E' Diesel, for this very scenario.

D. In-Correct - The ESW supply and return valves will automatically close and in-fact it is this signal which shuts down the diesel automatically. This is plausible since normally, during test operation, the ESW valves are manually opened prior to the DG test and closed when the test is completed, DG shutdown. Further, though this would not occur in this case, the intercooler ESW temperature control valves do fail open on loss of air or control power.

References: TM-OP-024A-ST page 32, 78-80 Student Ref: required No

Learning Objective: TM-OP024A-OB #2707a

Question source: Bank #TMOP024A/874/1

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 25

Unit 1 is conducting SO-151-B02 "Quarterly Core Spray Flow Verification Division 2". The following conditions exist:

- Both 1B and 1D CS pumps are running.
- Indicated Core Spray Flow is 6,350 gpm.
- HV-152-F031B CORE SPRAY LOOP 'B' MIN FLOW valve is CLOSED.
- HV-152-F015B CORE SPRAY LOOP 'B' TO SUPP POOL valve throttled OPEN.

A LOCA occurs on Unit 1, RPV level is minus (–) 140 inches, lowering slowly and RPV pressure is 730 psig, lowering at 40 psig per minute.

What is the present status of 'B' loop of Core Spray?

A.
- 1B and 1D Core Spray pumps TRIP and must be manually started.
- Core Spray Loop 'B' Min Flow valve, HV-152-F031B is OPEN.
- Core Spray Loop 'B' To Supp Pool, HV-152-F015B is throttled OPEN.

B.
- 1B and 1D Core Spray pumps are running
- Core Spray Loop 'B' Min Flow valve, HV-152-F031B is OPEN.
- Core Spray Loop 'B' To Supp Pool, HV-152-F015B is CLOSED.

C.
- 1B and 1D Core Spray pumps TRIP and must be manually started.
- Core Spray Loop 'B' Min Flow valve, HV-152-F031B is OPEN.
- Core Spray Loop 'B' To Supp Pool, HV-152-F015B is CLOSED.

D.
- 1B and 1D Core Spray pumps are running.
- Core Spray Loop 'B' Min Flow valve, HV-152-F031B is CLOSED.
- Core Spray Loop' 'B' To Supp Pool, HV-152-F015B is throttled OPEN.

K&A # 209001 LPCS
Importance Rating 2.8

QUESTION 25
K&A Statement: K4.10 - Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Testability of all operable components

Justification:

A. In-Correct – On the LOCA signal, the Core Spray pumps will not trip but the Test valve will close. Plausible Distracter; since pump trips could occur, e.g., if the LOCA had occurred on Unit 2, Div 2 is Unit 2 preferred pump and would trip 1B and 1D CS pump in test.

B. Correct – The LOCA signal will cause the HV-152-F015B Core Spray Loop B to Supp Pool (Full Flow Test Valve) to close. The pumps continue to run and HV-152-F031B Core Spray Loop B Min Flow valve, would open. This is the design response of the Core Spray system to automatically align for RPV injection from the Test condition.

C. In-Correct – On the LOCA signal, the Core Spray pumps will not trip but the Test valve will close. Plausible Distracter; since pump trips could occur, e.g., if the LOCA had occurred on Unit 2, Div 2 is Unit 2 preferred pump and would trip 1B and 1D CS pump in test.

D. In-Correct - The LOCA signal will cause the HV-152-F015B Core Spray Loop B to Supp Pool (Full Flow Test Valve) to close. The pumps continue to run and HV-152-F031B Core Spray Loop B Min Flow valve, would open. This is the design response of the Core Spray system to automatically align for RPV injection from the Test condition. Plausible Distracter; since the candidate may not know that the F015B get a closed signal upon initiation and may believe that operator action is required to align the system.

References: TM-OP-051-ST pages 13,14, 23, 42 Student Ref: required No

Learning Objective: TM-OP-051-OB # 2080

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 26

Unit 1 is in Mode 4 with the 'A' RHR pump in the Shutdown Cooling mode of operation. The following occurs:

A leak develops on the valve body of the RHR Shutdown Cooling Suction Outboard Isolation valve, HV-151-F008.

What is the response of RPV level; AND the status of RHR for LPCI Injection Mode? (Assume no operator action and the RHR system operates as designed)

A.
- RPV level lowers and the leak is isolated at +13 inches.
- HV-151-F015A and HV-151-F015B, RHR Injection Outboard Isolation valves are CLOSED.
- No RHR pumps are running and operator action is required to align for LPCI injection.

B.
- RPV level lowers and the leak is isolated at – 38 inches.
- HV-151-F015A, RHR Injection Outboard Isolation valve remains OPEN and HV-151-F015B is CLOSED.
- No RHR pumps are running and operator action is required to align for LPCI injection.

C.
- RPV level lowers and the leak is isolated at – 129 inches.
- HV-151-F015A and HV-151-F015B, RHR Injection Outboard Isolation valves are CLOSED.
- 'A' and 'C' RHR pumps are tripped.
- 'B' and 'D' RHR pumps are running but not injecting and operator action is required to align for LPCI injection.

D.
- RPV level lowers and the leak CANNOT be isolated.
- HV-151-F015A and HV-151-F015B, RHR Injection Outboard Isolation valves are OPEN.
- The "A" RHR pump is tripped.
- The "B", "C" and "D" RHR pumps are injecting into the reactor vessel.

K&A # 205000 Shutdown Cooling System

Importance Rating 3.2

QUESTION 26

K&A Statement: K3.01 - Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor water level: Plant Specific

Justification:

A. Correct – With the leak on the F008, the F009 will close to isolate the leak at +13 inches and the 'A' loop pumps get a trip signal (A RHR running) on interlock. In addition to the F008 and F009, both F015A and F015B will isolate at +13 inches in the SDC mode and RPV level will stabilize. Since level is above -129 inches, the RHR pumps will not receive a start signal. RHR remains in the Shutdown Cooling Mode until the operator resets both RHR Loop A/B Shutdown Cooling Reset pushbuttons to convert to LPCI mode.

B. In-Correct – The leak is isolated at +13 inches. Both F015A and F015B valves are automatically isolated in the Shutdown Cooling Mode. This is plausible since – 38 inches is an isolation and RHRSW pump trip. The candidate may also not believe both the F015A and F015B (normally associated with LPCI injection) get closed signals.

C. In-Correct – The leak is isolated at +13 inches. Since level is above -129 inches, the RHR pumps will not receive a start signal. This is plausible if the candidate believes the isolation does not occur until -129 inches when RHR will convert from Shutdown Cooling to LPCI.

D. In-Correct – The leak is isolated at +13 inches. Since level is above -129 inches, the RHR pumps will not receive a start signal. This is plausible if the candidate believes the leak is not isolable (confuses the leak location since the F008 is actually the Outboard isolation valve) and then level would lower to -129 inches.

References:	TM-OP-049-ST pages 12,13, 26	Student Ref: required	No
Learning Objective:	TM-OP-049-OB #181		
Question source:	Modified Bank: #TMOP049/192/4	See original attached	
Question History:	None		
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	 X	
10CFR	55.41 X		
Comments:	Created/Modified by: Bank Reviewed by: Michaels/Morgan		

QUESTION 27

Unit 1 was in a normal electrical lineup at 85% power and Unit 2 is in Mode 3 lineup with a cooldown is in progress.

A Loss of the Montour-Mountain Line occurred and Startup Transformer 10 (T-10) deenergized. All plant equipment operated as designed.

Which one of the following isolations occurred due to the electrical transient, AND is affecting Reactor Recirculation Pump (RRP) cooling?

A. RB Chilled Water Isolation only to the 'A' RRP motor windings cooler.

B. RB Closed Cooling Water Isolation only to the 'A' RRP pump seal and bearing coolers.

C. RB Chilled Water Isolation to both the 'A' and 'B' RRP motor windings coolers.

D. RB Closed Cooling Water Isolation to both the 'A' and 'B' RRP pump seal and bearing coolers.

K&A # 202001 Recirculation
Importance Rating 3.1

QUESTION 27

K&A Statement: K6.02 - Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION SYSTEM : Component cooling water systems

Justification:

A. In-Correct – The loss of power and N4S isolation will isolate chilled water to both Recirc pumps (inboard valves of one pump and outboard valves of the other). Plausible choice if the candidate believes the isolation is an "A" isolation instead of a Div 1 isolation.

B. In-Correct – The loss of power and N4S isolation will isolate chilled water to both Recirc pumps (inboard valves of one pump and outboard valves of the other). Plausible since the difference between these cooling loop supplies, difference between N4S and DWCIS logics and valve isolations is a common misconception. Also, candidate may believe the isolation is an "A" isolation instead of a Div 1 isolation, affecting only the "A" pump. Also plausible since a momentary loss of power to RBCCW containment isolation valves does occur, but is not an isolation and the containment isolation valves immediately reopen when buses reenergize.

C. Correct – Loss of SUB 10 will cause 1A201 to swap to alternate power, 1A RPS EPA Breakers will trip, de-energizing 'A' RPS bus causing DIV 1 N4S Isolation. Loss of Reactor Building Chilled water to both Rx Recirc Pumps will occur (inboard valves of one pump and outboard valves of the other). Valve logics must be manually reset for each pump.

D. In-Correct – The loss of power and N4S isolation will isolate chilled water to both Recirc pumps (inboard valves of one pump and outboard valves of the other). This isolation will not result in a sustained loss of RBCCW since the bus loss is momentary and the containment isolation valves will reopen when buses reenergize. Plausible since the difference between these cooling loop supplies, difference between N4S and DWCIS logics and valve isolations is a common misconception. Also, candidate may believe the isolation is an "A" isolation instead of a Div 1 isolation, affecting only the "A" pump.

References: TM-OP-064-FS page 3 Student Ref: required No
TM-OP-064-ST Figure 12
TM-OP-059B-ST Figure 19
ON-158-001 rev 12 Attach 'A' page 6

Learning Objective: TM-OP-064C-OB # 2558 and AD045 # 15300

Question source: Modified AD045/15300/1 (See original attached)

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 28

A catastrophic failure of Startup Transformer T-10 results in a fire, a loss of offsite power (LOOP) and dual unit scram. All 4KV Buses are energized by the emergency diesel generators.

The transformer fire protection deluge system actuated to extinguish the fire.

What fire protection pumps below are running (and why)?

A.
- Motor Driven Fire Pump 0P512 Running (started automatically at 95 psig)
- Diesel Driven Fire Pump 0P511 Running (started automatically at 85 psig)
- Jockey Fire Pump 0P543 Initially Running now stopped, (started at 105 psig, stopped at 125 psig)

B.
- Motor Driven Fire Pump 0P512 Running (started automatically at 95 psig)
- Diesel Driven Fire Pump 0P511 **NOT** Running (fire protection header pressure > 85 psig)
- Jockey Fire Pump 0P543 **NOT** Running (no power)

C.
- Diesel Driven Fire Pump 0P511 Running (started automatically at 85 psig)
- Motor Driven Fire Pump 0P512 **NOT** Running (no power)
- Jockey Fire Pump 0P543 Initially Running now stopped, (started at 105 psig, stopped at 125 psig)

D.
- Diesel Driven Fire Pump 0P511 Running (started automatically at 85 psig)
- Motor Driven Fire Pump 0P512 **NOT** Running (no power)
- Jockey Fire Pump 0P543 **NOT** Running (no power)

K&A # 286000 Fire Protection
Importance Rating 2.9

QUESTION 28

K&A Statement: K2.02 - Knowledge of electrical power supplies to the following: Pumps

Justification:

A. In-Correct – With this loss of power condition, the Motor Driven Fire Pump and Jockey pumps will not have power. Plausible Distracter since the student may believe the electric pumps are powered from 4KV ESS buses.

B. In-Correct – With this loss of power condition, the Motor Driven Fire Pump and Jockey pumps will not have power. Plausible Distracter since the student may believe only the Motor Driven Fire Pump is powered from 4KV ESS buses.

C. In-Correct – With this loss of power condition, the Motor Driven Fire Pump and Jockey pumps will not have power. Plausible Distracter since the candidate may believe the smaller Jockey pump is still powered, but not needed with the Diesel Fire Pump running.

D. Correct – Only the Diesel Driven Fire Pump is running (auto started at 85 psig header pressure) in this condition. The Motor Driven Fire Pump is deenergized; normally powered from 1B180, from the 13KV Startup Bus 10. The Jockey Pump is also deenergized, normally powered from 0B512, from the 13 KV Auxiliary Bus 11B. All Aux Buses deenergized when the Units tripped.

References: TM-OP-013-FS pages 2 Student Ref: required No

Learning Objective: TM-OP-013-OB # 2299, 5081, and 2301

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 29

A Unit 1 Reactor Startup is in progress with reactor power at 5%.

- Control Rod 30-35 was withdrawn to notch position '12'.
- A failed reed switch was identified for Control Rod 30-35 at position '12'.
- A substitute rod position was entered into the Rod Worth Minimizer (RWM) in accordance with the off-normal procedure, ON-155-004, RPIS Failure.
- On a subsequent rod group, Control Rod 30-35 is withdrawn to notch position '14' which has a good reed switch.

For Control Rod 30-35 rod position, which one of the following is the response of the Rod Worth Minimizer (RWM) and PICSY displays (including OD-7) when the control rod is withdrawn to Notch Position '14'?

The RMW and PICSY displays for 30-35 rod position will:

A. continue to display the substitute value of '12'.

B. update and show the valid rod position of '14'.

C. show '__' BLANK rod position with a new scan.

D. show 'UNK' since rod position is now Unknown.

K&A # 201006 RWM
Importance Rating 3.4

QUESTION 29

K&A Statement: K1.02 - Knowledge of the physical connections and/or cause-effect relationships between ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) and the following: Rod position indication system: P-Spec(Not-BWR6)

Justification:

A. Correct – Once the substituted rod position is entered, it stays in the program until manually deleted. All RWM and PICSY displays will continue to show the substitute value.

B. In-Correct – The RWM and PICSY displays will continue to show the substituted value. However, the four-rod display on 1C652 was blank and will update with the new reed switch, showing the position '14'.

C. In-Correct – This RWM and PICSY displays will continue to show the substituted rod position until manually deleted. Codes of UNK or Blank are likely initially when initially withdrawing the rod to the bad reed switch. UNK and Blank are possible PICSY values.

D. In-Correct – This RWM and PICSY displays will continue to show the substituted rod position until manually deleted. Codes of UNK or Blank are likely initially when initially withdrawing the rod to the bad reed switch. UNK and Blank are possible PICSY values.

References: TM-OP-031-ST pages 17-19 Student Ref: required No

Learning Objective: TM-OP-031D-OB #1575

Question source: Bank: #LOR question

Question History: None in Initial Training

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 30

Unit 2 is operating at 90% power with all Condensate and Feedwater Pumps in service. The following conditions exist for the Reactor Recirculation Pumps (RRPs):

RRP 'A'
Speed control is in Manual with Scoop Tube locked.
Speed is 85 percent.

RRP 'B'
Speed control is in Manual.
Speed is 89 percent.

SELECT the final status of the Unit 2 RRPs if the '2A' Reactor Feedwater Pump Turbine (RFPT) trips and RPV level lowers and stabilizes at 32 inches.

A. Pump A speed is 85 percent. Pump B speed is 89 percent.

B. Pump A speed is 85 percent. Pump B speed is 30 percent.

C. Pump A speed is 48 percent. Pump B speed is 48 percent.

D. Pump A speed is 85 percent. Pump B speed is 48 percent.

K&A # 259001 Reactor Feedwater
Importance Rating 3.5

QUESTION 30

K&A Statement: K4.11 - Knowledge of REACTOR FEEDWATER SYSTEM design feature(s) and/or interlocks which provide for the following: Recirculation runbacks: Plant-Specific

Justification:

A. In-Correct - On Unit 2, a RFPT trip results in an immediate Recirc Runback to Limiter #2 (48%). This would be correct for Unit 1, since pre-modification, a RFPT trip requires a concurrent confirmatory 30 inch low RPV level signal to initiate the runback to Limiter #2.

B. In-Correct – On Unit 2, a RFPT trip results in an immediate Recirc Runback to Limiter #2 (48%) not Limiter #1 (30%). A low level signal by itself can result in a Limiter #1 and is a possible choice if the candidate confuses the initiating runback signals.

C. In-Correct - On Unit 2, a RFPT trip results in an immediate Recirc Runback to Limiter #2 (48%). Since the 'A' RRP is locked, the 'A' RRP will not automatically occur and the pump speed must be lowered locally at the Jordan positioner. A runback does not automatically unlock a scoop tube. This would be correct if both RRP scoop tubes were in normal.

D. Correct – On Unit 2, a RFPT trip results in an immediate Recirc Runback to Limiter #2 (48%). Since the 'A' RRP is locked, the 'A' RRP will not automatically occur and the pump speed must be lowered locally at the Jordan positioner. NOTE: This is a Unit difference until the completion of U1-15 RIO modification. For power uprate, as presently installed on Unit 2, the Recirc Runback modification removes the 30 inch level confirmatory signal to provide an immediate runback.

References: TM-OP-064A-FS page 2 Student Ref: required No

Learning Objective: TM-OP-064A-OB #2581

Question source: Modified from TMOP064A/2309/6 Original question is attached.

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 31

Unit 2 is in MODE 1 at 100% rated power.

The following alarms are received:

- FW LOOP B PANEL 2C102 TRBL (AR-202-H07)
- 5B LEVEL HI (LA-202-A02)
- 5B LEVEL DUMP VALVE NOT 100% CLOSED (LA-202-C02)

Upon investigation it is discovered that a level control malfunction has occurred and the PCO observes 5B Feedwater Heater level is 12 inches, rising.

As level rises, predict the effect when the Hi-Hi Level Trip setpoint is reached; AND What procedure action is required to mitigate this event?

A. The 5B Feedwater Extraction Steam Isolation Valve will ISOLATE and rising power will raise the potential for fuel failure due to Pellet Clad Interaction. Perform ON-247-001 "Loss of Feedwater Heating Extraction Steam" and reduce power to ≤85% power.

B. The 5B Feedwater Extraction Steam Isolation Valve will ISOLATE and ensure the Reactor Recirculation pumps runback. Perform ON-247-001 "Loss of Feedwater Heating Extraction Steam" and ON-264-002 "Loss of Reactor Recirculation Flow" to reduce power to ≤85% power.

C. The 'B' Feedwater Heater String will ISOLATE and ensure the Reactor Recirculation pumps runback. Perform ON-247-002 "Loss of Feedwater Heating String" and ON-264-002 "Loss of Reactor Recirculation Flow" to reduce power to ≤85% power.

D. The 'B' Feedwater Heater String will ISOLATE and when feedwater temperatures lower, reactor power will rise above 100% power. Perform ON-200-004 "Reactor Power Greater Than License Limit" and initiate prompt action to reduce power at ≤100% power.

K&A # 259001 Reactor Feedwater
Importance Rating 3.3

QUESTION 31

K&A Statement: A2.04 - Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of extraction steam

Justification:

A. Correct – Hi-Hi level in 5B FW Heater will cause the extraction steam isolation valve to close. FW temperature will lower causing reactor power to increase, action must be taken to lower reactor power. ON-147-001 Loss of Extraction Steam directs the operator to immediately lower reactor power to ≤85% power due to an immediate concern for fuel damage due to PCI.

B. In-Correct – Extraction Steam Isolation does not cause a Reactor Recirc Runback. Plausible Distracter; FW Heater 1 or 2 will cause a Heater String Isolation on Hi-Hi level and a Recirc runback.

C. In-Correct – Feedwater heater 5B is not a string isolation on Hi-Hi level. Plausible Distracter; Feedwater Heaters 1 or 2 will cause a Heater String Isolation on Hi-Hi level and a Recirc runback.

D. In-Correct – Feedwater heater 5B is not a string isolation on Hi-Hi level. Plausible Distracter; Feedwater Heaters 1 or 2 will cause a Heater String Isolation on Hi-Hi level and because 5B would cause power to rise > 100%.

References: ON-147-001 rev 21 pages 2, 7 Student Ref: required No
TM-OP-047-ST page 16

Learning Objective: TM-OP-047-OB # 1852

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 32

During rated power operation, a scaffold crew inadvertently trips both reactor pressure switches PS-B21-1N023A and PS-B21-1N023B on 1C004 rack, resulting in several alarms including:

- REACTOR PRESS HIGH TRIP (AR-103-B02)
- REACTOR PRESS HIGH TRIP (AR-104-B02)

Determine the effect on the Reactor Protection System (RPS); AND the effect on the Alternate Rod Insertion (ARI) System.

A. NO RPS channels tripped and NO RPS actuation.
Two ARI channels tripped and FULL ARI actuation.

B. Two RPS channels tripped and Div 1 Half-Scram.
NO ARI channels tripped and NO ARI actuation.

C. Two RPS channels tripped and Full-Scram.
NO ARI channels tripped and NO ARI actuation.

D. Two RPS channels tripped and Full-Scram.
Two ARI channels tripped and Full ARI actuation.

K&A # 216000 Nuclear Boiler Inst.
Importance Rating 3.9

QUESTION 32
K&A Statement: K1.01 - Knowledge of the physical connections and/or cause-effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: Reactor protection system

Justification:

A. In-Correct – This combination of switches and logic will result in a RPS Full Scram. This is not an ARI actuation since ARI uses different switches and logic. Plausible if candidate believes these are the ARI pressure switches, which are on the same rack and use 2-out-of-2 logic.

B. In-Correct – The 'A' and 'B' channels are on the same rack (same reference leg) and feed into A1 and B1 RPS channels respectively for a Full-Scram. Half-Scram is plausible if the candidate believes the pressure switches both feed into A1 /A2 logic for a Half-Scram.

C. Correct – The 'A' and 'B' channels are on the same rack (same reference leg) and feed into A1 and B1 RPS channels respectively for a Full-Scram. ARI uses separate pressure switches and logic.

D. In-Correct – This combination of switches and logic will result in a RPS Full Scram. This is not an ARI actuation since ARI uses different switches and logic. Plausible if candidate believes these are the ARI pressure switches, which are on the same rack and use 2-out-of-2 logic.

References: TM-OP-058-ST pages 22, 46, 47 Student Ref: required No
AR-103-B02 rev 33 page 10
AR-104-B02 rev 25 page 10

Learning Objective: TM-OP-058-OB # 2487, 2504
TM-OP-080-OB #10561

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 33

During a Unit 2 plant startup, the following condition exists:

- Two (2) Condensate pumps, 2A and 2B are running.

Evaluate the system parameters below and determine which Unit 2 system lineup provides adequate Condensate Pump Min Flow protection.

Condensate system lineup is in:

A.
- Condensate pump 2A discharge pressure reads 620 psig.
- Condensate pump 2B discharge pressure reads 625 psig.
- Short-path recirc with Cond Recirc Flow controller, FIC-20508 in AUTO set at 1600 gpm with 3200 gpm indicated on FI-20508.

B.
- Condensate pump 2A discharge pressure reads 700 psig.
- Condensate pump 2B discharge pressure reads 720 psig.
- Short-path recirc with Cond Recirc Flow controller, FIC-20508 in MANUAL at 30% with 7200 gpm indicated on FI-20508.

C.
- Condensate pump 2A discharge pressure reads 700 psig.
- Condensate pump 2B discharge pressure reads 720 psig.
- Long-path recirc with the Cond Recirc Flow controller, FIC-20508 in MANUAL at 30% with 4000 gpm indicated on FI-20508 and Feedwater Flow is 3200 gpm indicated on computer point CDF09.

D.
- Condensate pump 2A discharge pressure reads 620 psig.
- Condensate pump 2B discharge pressure reads 625 psig.
- Long-path recirc with the Cond Recirc Flow controller, FIC-20508 in AUTO set at 3200 gpm with 0 gpm indicated on FI-20508 and Feedwater Flow is 7200 gpm indicated on computer point CDF09.

K&A # 256000 Reactor Condensate
Importance Rating 3.1

QUESTION 33
K&A Statement: A1.07 - Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: System lineup

Justification:

A. In-Correct – Adequate min flow protection is not assured. 3200 gpm is required for each pump

B. In-Correct – Adequate min flow protection is not assured. Though flow is above 3200 gpm per pump, discharge pressure indicates that one or both pumps are deadheaded, challenging pump operation.

C. In-Correct – Adequate min flow protection is not assured. Though flow is above 3200 gpm per pump, discharge pressure indicates that one or both pumps are deadheaded, challenging pump operation.

D. Correct – Adequate min flow exists. Sufficient flow of > 3200 gpm per pump exists with a discharge pressure below 700 psig on both pumps.

References: OP-244-001 rev 41 pages 6,9, 13,15 Student Ref: required No
PICSY Cond System format

Learning Objective: TM-OP-044-OB #10726

Question source: New

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 34

Unit 1 is at 100% power in a normal lineup when I&C inadvertently equalizes the 'A' Narrow Range Level Transmitter, PDT-C32-1N004A (0 psid across DP Cell).

Which of the following will be the indicated level on the 1C652 SIP Panel indicator, LI-C32-1R606A; AND What is the actual RPV level response with Feedwater Level Control in Master Auto? (Assume no operator action)

A. Indicated level at 1C652, LI-C32-1R606A will read upscale.
Actual RPV level will lower to approx. 23 inches and stabilize.

B. Indicated level at 1C652, LI-C32-1R606A will read downscale.
Actual RPV level will rise to approx. 47 inches and stabilize.

C. Indicated level at 1C652, LI-C32-1R606A will read upscale.
Actual RPV level will continue to lower to the scram setpoint.

D. Indicated level at 1C652, LI-C32-1R606A will read downscale.
Actual RPV level will continue to rise to the turbine trip setpoint.

K&A # 216000 Nuclear Boiler Inst.
Importance Rating 2.9

QUESTION 34
K&A Statement: A1.02 - Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including: Removing or returning a sensor (transmitter) to service

Justification:

A. Correct – Indicated level for equalized transmitter will be pegged upscale. FWLC will respond to this level input in Auto and will stabilize at + 23 inches with no immediate operator action required.

B. In-Correct – Indicated level is upscale, not downscale. Actual RPV level will lower and stabilize at + 23 inches. This is the plausible response if the indicated NR level failed downscale with FWLC in Auto.

C. In-Correct – Indicated level is upscale. Actual RPV level will stabilize at + 23 inches with no immediate operator action required. This response is plausible with multiple level instrument failures.

D. In-Correct – Indicated level is upscale, not downscale and actual RPV level will lower and stabilize at + 23 inches with no immediate operator action required. This response is plausible with multiple level instrument failures.

References: ON-145-001 rev 21 page 2
TM-OP-045-ST page 55
TM-OP-080-ST page 56

Student Ref: required No

Learning Objective: TM-OP-045-OB # 10297

Question source: Modified Bank: #63 See original attached

Question History: LOC 21 Cert Exam

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 35

While operating at rated power, an instrument air line break results in a loss of air pressure (0 psig) supplied to the SJAE Dilution Steam Flow Control valves, FV-10702A and FV-10702B.

Which one of the following is the response of the Offgas System to this failure? (Assume no other components are affected by the instrument air leak.)

SJAE Dilution Steam Flow Control valves, FV-10702A and FV-10702B will fail:

A. OPEN and cause a SJAE Isolation.

B. OPEN and cause a Recombiner Shutdown.

C. CLOSED and cause a SJAE Isolation.

D. CLOSED and cause a Recombiner Shutdown.

K&A #	271000 Off-gas
Importance Rating	2.9

QUESTION 35

K&A Statement: A3.02 - Ability to monitor automatic operations of the OFFGAS SYSTEM including: System flows

Justification:

A. In-Correct – FV-10702A/B fail closed on a loss of IA and result in a SJAE Isolation on Low Dilution steam flow. If they failed open, it is plausible that a SJAE isolation could eventually occur on High Recombiner Discharge temperature.

B. In-Correct – FV-10702A/B fail closed on a loss of IA and result in a SJAE Isolation on Low Dilution Steam Flow. If they failed open, it is plausible that a Recombiner shutdown could eventually occur on High Condenser temperature.

C. Correct – The FV-10702A/B control valves automatically maintain dilution steam flow based on two flow controllers at 9350 lbm/hr. The valves fail closed on a loss of IA (or power), resulting in a low dilution steam flow and SJAE isolation.

The Offgas system is an integrated system that includes the Air Removal system and the Recombiner system. As seen below, the signals that cause an isolation versus a shutdown are often confused. The following conditions result in a SJAE Isolation (automatically closes the SJAE suction valves HV-10716, 10717, 10718, and 10719):

- Low Dilution Steam Flow Trip at 8000 lbs/hr.
- Low Recombiner Inlet Temperature at 225°F,
- High Recombiner Discharge Temperature at 925°F,
- Low Motive Steam Jet Condenser Cooling Water Flow at 110 gpm,
- Low Recombiner Condenser Cooling Water Flow at 275 gpm,
- Air Removal Emergency Shutdown (ARESD)

A Recombiner shutdown (automatically closes the Recombiner process and steam supply valves) is the direct result of a Recombiner Condenser Outlet Temperature High-High at 250 °F.

D. In-Correct – FV-10702A/B fail closed on a loss of IA and result in a SJAE Isolation on Low Dilution Steam Flow. As stated above, SJAE Isolation and Recombiner Shutdowns are often confused.

References: TM-OP-072B-ST pages 28, 29 Student Ref: required No

Learning Objective: TM-OP-072B-OB #1300

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 36

Unit 1 is in a refueling outage. The SRMs indicate:

SRM A	90 cps
SRM B	.9 cps
SRM C	.4 cps
SRM D	Inoperable and bypassed

Signal to noise ratio for all SRMs is 6:1.

In which core quadrants can core alterations be performed?

A. All quadrants

B. A, B, and C

C. A and B

D. A only

K&A # 234000 Fuel Handling Equipment
Importance Rating 3.7

QUESTION 36
K&A Statement: A4.01 - Ability to manually operate and/or monitor in the control room: Neutron monitoring system

Justification:

A. In-Correct – Could not load fuel around the "C" or "D" detectors. Plausible Distracter; Two SRM required to be OPERABLE in MODE 5. "A" and "B" are OPERABLE, but you need to have the detector OPERABLE in the quadrant you are moving fuel.

B. In-Correct - Could not load fuel around the "C" detector. Plausible Distracter; Two SRM required to be OPERABLE in MODE 5. "A" and "B" are OPERABLE, but you need to have the detector OPERABLE in the quadrant you are moving fuel and the adjacent quadrant.

C. Correct - For core alterations, must have an operable detector in the quadrant and in an adjacent quadrant (SR 3.3.1.2.2). Per SR 3.3.1.2.4, a SRM is operable if it is 3 cps with a S/N ratio of 2.1 or is within Figure 3.3.1.2-1. SRMS "A" and "B" are operable. SRM "C" is outside acceptable range of the Figure and is inoperable. SRMS "A" and "B" are in adjacent quadrants, therefore fuel may be loaded in A and B quadrants.

D. In-Correct – SRM "A" and "B" are OPERABLE, you need to have SRM Detector in the quadrant you are moving fuel and the adjacent. "A" and "B" are adjacent; therefore you can also load fuel in the "B" quadrant. Plausible Distracter; candidates may not understand figure1, a common error.

References:	TS 3.3.1.2	Student Ref: required TS 3.3.1.2	Yes
Learning Objective:	TM-OP-078-OB # 12661 and TM-OP-081A-OB # 12693		
Question source:	Bank TMOP078A/12663 Modified	See Attached	
Question History:	No recent history		
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	X	
10CFR	55.41 X		
Comments:	Created/Modified by: Bank Modified Reviewed by: Michaels/Morgan		

QUESTION 37

An accident is in progress on Unit 1 with the following parameters:

- HPCI and RCIC are controlling level +13 to +54 inches.
- Reactor pressure is 940 psig, slowly lowering.
- SPOTMOS temperature is 92°F, slowly rising.
- Supp Chamber pressure is 5 psig, slowly rising.
- Drywell pressure is 10 psig and slowly rising.

Five minutes later, the following information is reported:

- 'A' Loop RHR is in Suppression Chamber Sprays.
- Supp Chamber pressure is 6 psig, slowly rising.
- Drywell pressure is 11 psig, slowly rising.
- SPOTMOS temperature is 115° F, slowly rising.
- Supp Chamber airspace temperature is 109° F, slowly rising.

Which of the following describes the Suppression Chamber response; AND What is the next required containment control action?

A. The Suppression Chamber airspace contained mostly steam prior to initiating sprays. Place a second RHR loop in Suppression Chamber Spray mode before Suppression Chamber pressure reaches 13 psig.

B. The Suppression Chamber airspace contained no steam prior to initiating sprays. When Suppression Chamber pressure exceeds 13 psig, spray the Drywell.

C. Suppression Pool water temperature is too high to reduce airspace pressure. Place 'B' loop RHR in Suppression Chamber Spray mode using RHRSW before Suppression Chamber pressure reaches 13 psig.

D. Leaking Suppression Chamber vacuum breakers have bypassed the pressure suppression function. When Suppression Chamber pressure exceeds 13 psig, spray the Drywell.

K&A # 230000 RHR/LPCI: Torus/Pool Spray Mode
Importance Rating 4.2

QUESTION 37

K&A Statement: 2.4.31 - Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures. (RHR/LPCI: Torus/Pool Spray Mode)

Justification:

A. In-Correct - If the vapor space contained steam following initiation of sprays, a reduction in Drywell pressure should occur. Use of a second loop of Suppression Chamber sprays is not permitted since a single spray header exists by design with either RHR loop supplying that header at a maximum of 500 gpm.

B. Correct - Vapor space pressure is caused by accumulation of non-condensibles (nitrogen in this case), and the use of sprays in the Supp Chamber vapor space will have little or no effect on SC pressure. Drywell spray is not permitted in this condition until Suppression Chamber pressure exceeds 13 psig.

C. In-Correct – Supp Chamber sprays with a 115 °F water temperature is not too high to reduce vapor space pressure. At this point, Supp Chamber pressure is more a function of Supp Pool level than temperature. Supp Chamber airspace temperature will reach an equilibrium with the spray water temperature due to convective heat transfer. Further, using sprays from RHRSW is not warranted in this condition.

D. In-Correct - If vacuum breaker valves were leaking, the differential pressure between the Drywell and Suppression Chamber vapor space would be a value less than 5 psid.

References: EO-000-103 rev 3 page 24 Student Ref: required No

Learning Objective: PP002 14586 & 14594

Question source: Bank: # 352

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 38

Unit 2 conditions are as follows:

- Reactor power is 100 percent.
- Main Generator load is 1,200 MWe.
- Main Generator output current is 30,000 amperes on each phase.

A rapid reduction in grid load causes Main Generator output current on each phase to drop from 30,000 to 15,000 amperes within 25 milliseconds.

Which of the following describes the EHC and plant response to this transient; AND What procedure actions are required?

A. The Turbine Control Valves throttle close, AND the reactor remains at power. Perform ON-203-001, “Grid Instabilities”.

B. The Turbine Intercept Valves throttle close, AND the reactor remains at power. Perform ON-203-001, “Grid Instabilities”.

C. The Turbine Control Valves rapidly close, causing an immediate RPS scram. Perform ON-200-101, “Scram, Scram Imminent and ON-293-001 “Main Turbine Trip”.

D. The Turbine Intercept Valves rapidly close, causing an immediate RPS scram. Perform ON-200-101, “Scram, Scram Imminent and ON-293-001 “Main Turbine Trip”.

K&A # 241000 Reactor/Turbine Pressure Regulator

Importance Rating 3.4

QUESTION 38

K&A Statement: A2.09 - Ability to (a) predict the impacts of the following 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: Loss of generator load

Justification:

A. In-Correct – The TCV will not throttle close but fast-close (fast acting solenoid valves actuate) on Power Load Unbalance (PLU) signal. This is plausible since this is the CV will throttle to control pressure if the PLU was not armed or load reject was below the threshold (40% power to load unbalance).

B. In-Correct – PLU senses the load reject and will cause an immediate TCV Fast Closure and scram. The IV will not throttle close until an overspeed condition is sensed. Plausible since the purpose of the Intercept Valves is to close on overspeed (105-107%) and fast-close in the event of rapid overspeed (5% trigger step change), which is possible during a load reject transient. The Unit will not stay online.

C. Correct - The Power Load Unbalance (PLU) circuitry monitors generator output current (load) and compares it to Turbine power as sensed by Cross-around steam pressure (power). If a rate-sensitive load unbalance of >40 percent in < 35 msec is detected, the circuit actuates to protect the Turbine from load reject and avoid turbine generator damage as well as a potential overspeed condition. Scram occurs due to TCV fast closure with power > 30%. A turbine trip follows on either reverse power or sync breaker opening and these offnormal procedures are entered.

D. In-Correct – PLU sense the load reject and cause an immediate TCV Fast Closure and scram. The IV will not throttle close until an overspeed condition is sensed. IVs also do not cause an immediate RPS scram. Plausible since the purpose of the Intercept Valves is to close on overspeed (105-107%) and fast-close in the event of rapid overspeed (5% trigger step change), possible during a load reject transient.

References: TM-OP-093L page 17, 31
ON-293-002 rev 11 page 7

Student Ref: required No

Learning Objective: TM-OP-093L-OB # 10322 and 1641

Question source: Bank: #TMOP093L/1641/4

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 39

The following conditions exist on Unit 1:

- A large leak has occurred in the Drywell.
- Suppression Pool level is 19 feet, steady.
- Suppression Chamber Sprays are in-service.
- Drywell sprays are being initiated.

When the Drywell Spray Outboard Isolation Valve, HV-151-F016A was throttled open to establish the required spray flow, the valve continued stroking to the full open position, instead of stopping when the handswitch was released.

In accordance with EO-000-103, "PC Control" bases, which one of the following describes the concern of initiating the maximum Drywell Spray flow due to the valve stroking full open?

A. RHR Pump motor overcurrent trip due to excessive flow and runout.

B. Drywell Spray Header and piping damage due to water hammer.

C. RHR pump damage due to exceeding RHR Vortex and NPSH Limits.

D. Containment damage due to exceeding design differential pressure.

K&A # 295024 High Drywell Pressure
Importance Rating 4.1

QUESTION 39
K&A Statement: EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : Drywell integrity: Plant-Specific

Justification:

A. In-Correct - This is not a concern since RHR pumps are rated for 12,200gpm @ 204 psig and are provided with an orifice to restrict flow to 13,500 gpm to prevent pump runout.

B. In-Correct – This is not a concern since the piping is normally maintained filled and vented and drywell spray header is the high point inside the containment.

C. In-Correct - This is not a concern since Suppression Pool level is above the level specified in the EOPs for Vortex and NPSH concerns.

D. Correct – In accordance with the EOP bases, the concern is for evaporative cooling that occurs when water is sprayed into a superheated atmosphere. The water at the surface of each droplet is heated and flashed to steam until the surrounding atmosphere saturates, absorbing heat energy from the atmosphere. In the drywell, this cooling process results in an immediate, rapid and large reduction in pressure at a rate much faster than can be handled by the vacuum breakers (even if they were full open at time of spray initiation). Analytical results indicate drywell pressure drops of up to 12 psig can occur in less than 0.5 seconds. Unrestricted operation of drywell sprays could thus cause an excessive negative differential pressure to occur between the drywell and suppression chamber.

References: EO-000-103 rev 3 page 27,28 Student Ref: required No

Learning Objective: PP002 #14613

Question source: Bank: #PP002/2598/18

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 40

ON-149-001, "Loss of RHR Shutdown Cooling Mode" directs the operator to maintain RPV water level ≥+45 inches when RHR Shutdown Cooling is lost.

Which one of the following is the basis for this action?

A. To ensure the water in the annulus is high enough to provide a flowpath from core shroud to the downcomer for circulation.

B. To ensure the feedwater spargers effectively mix the feedwater added to prevent stratification in the vessel downcomer area.

C. To ensure Narrow Range Level instruments remain on-scale for continuous level and temperature monitoring.

D. To ensure the water from the annulus area of the Reactor flows into the core shroud as level decreases for circulation.

K&A # 295021 Loss of Shutdown Cooling / 4
Importance Rating 3.6

QUESTION 40

K&A Statement: AK1.04 - Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : Natural circulation

Justification:

A. Correct - Level is raised to ensure communication between the shroud and the annulus via the moisture separator skirt. Raising level promotes natural circulation by allowing the level in the heat source (shroud) to rise and flow to the heat sink (annulus or downcomer).

B. In-Correct – This is not the basis. Raising level does not prevent stratification in the downcomer area and has nothing to do with the FW spargers. Plausible since preventing stratification and good mixing is a desirable condition within the reactor core area.

C. In-Correct – This is not the basis. Though it may be on-scale, the NR level instrument is not used for RPV level while in shutdown cooling. Further, temperature monitoring is not assured regardless of RPV level. The idea of level and temperature is plausible since there is correlating discussion of monitoring temperature using RPV flange and metal temperatures.

D. In-Correct – This is not the basis. This is a reverse of 'B' if the candidate does not understand the mechanism by which natural circulation occurs. Coolant does not flow from the annulus into the shroud.

References: ON-149-001 rev 20 page 20 Student Ref: required No

Learning Objective: AD045 #15304 & TM-OP-062-OB # 10555

Question source: Bank: # 37

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 41

While operating at full power on Unit 1, a blown fuse results in a loss of all 125 VDC control power to the 'A' Control Rod Drive (CRD) pump. What effect will this have on the 'A' CRD pump?

The Unit 1 'A' CRD pump will:

A. continue to run. Pump motor protective features are disabled.

B. continue to run. Pump motor protective features are functional.

C. immediately trip. Pump motor protective features are disabled.

D. immediately trip. Pump motor protective features are functional.

K&A # 295004 Partial or Total Loss of DC Pwr / 6

Importance Rating 3.3

QUESTION 41

K&A Statement: AK1.05 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Loss of breaker protection

Justification:

A. Correct – On the loss of control power to the breaker, all automatic protective trips are disabled, therefore pump will continue to operate. The pump can only be tripped locally at the breaker.

B. In-Correct – All automatic protective trips are disabled at the breaker. This is plausible since the pump will continue to run and the candidate may believe protective features are powered separately.

C. In-Correct – The operating CRD pump will not trip. This is plausible since there are pump logics in the plant that will trip on a loss of control power, e.g., Reactor Recirc Pumps that automatically trip on loss of 125 VDC control power. In the case of the Control Rod Drive pumps, the auto trips are disabled with a loss of control power. This is a plausible choice if the candidate believes this is a fail-safe feature.

D. In-Correct – The operating CRD pump will not trip. This is plausible since there are pump logics in the plant that will trip on a loss of control power, e.g., Reactor Recirc Pumps that automatically trip on loss of 125 VDC control power. In the case of the Control Rod Drive pumps, the auto trips are disabled with a loss of control power. This is a plausible choice if the candidate believes this is a fail-safe feature.

References: TM-OP-055-ST page 7
E-169 rev 17 sheet 6

Student Ref: required No

Learning Objective: TM-OP-004-OB # 2239
TM-OP-055-OB # 2419

Question source: Bank: # 477

Question History: 2005 LOC 21 NRC Exam

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 42

A steam leak has occurred on Unit 1, resulting in the following conditions:

- Reactor Building Elevation 749' temperatures were 217 °F rising and the building is no longer accessible.
- Wide range level readings lowered to minus (-) 150 inches and are now indicating minus (-) 132 inches, slowly rising.
- All Fuel Zone Level indication is unavailable.

Based on the Wide Range level readings, which of the following is the RPV Level and Trend?

A. Level is ABOVE TAF. Trend is **NOT** valid.

B. Level is BELOW TAF. Trend **IS** valid.

C. Level is INDETERMINATE. Trend is **NOT** valid.

D. Level is INDETERMINATE. Trend **IS** valid.

K&A # 295031 Reactor Low Water Level / 2
Importance Rating 4.6

QUESTION 42

K&A Statement: EA2.01 - Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : Reactor water level

Justification:

A. In-Correct – RPV level cannot be determined since Wide Range level is below MIL. However, the candidate may interpret the status of wide range recovering as good level. The off-calibration condition makes the trend capability plausible.

B. In-Correct – RPV level cannot be determined since Wide Range level is below MIL. The candidate may recognize that level went off-scale low and recovered and interpret the level trending to also be available.

C. Correct – With the Reactor Building 749' elev. high temperature (above 212 F) and the building inaccessible, ON-145-004 and EOP Caution 1 identifies Wide Range level MIL from -145 inches to -125 inches. Since the level was below MIL and is still below MIL, Wide Range level cannot be determined. Trend cannot be used in this condition.

D. In-Correct – Since the level was below MIL and is still below MIL, Wide Range level cannot be determined. Trend cannot be used in this condition. Candidate may recognize that level is indeterminate but believe that it can still be used for trending purposes, a common misconception for candidates.

References: ON-145-004 rev 12 pages 16-17
EO-000-100 rev 3 pages 2-4

Student Ref: required No

Learning Objective: AD045 #1365

Question source: Bank: #AD045/1365/10

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 43

Units 1 & 2 are operating at 100% power with the following electrical lineup:

- T-201 ESS Transformer 0X203 is out of service for scheduled maintenance
- ESS Busses 1A204 and 2A204 are on alternate power.
- A loss of Startup Bus 10 subsequently occurs.

Assuming no Operator actions and all equipment functions as designed. What will be the status of the Unit 1 'A' ESS bus breakers following this event?

A. Bus 1A - (1A20101) Breaker Open
Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Open

B. Bus 1A - (1A20101) Breaker Closed
Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Open

C. Bus 1A - (1A20101) Breaker Open
Bus 1A - (1A20109) Breaker Closed
D/G A to Bus 1A - (1A20104) Breaker Open

D. Bus 1A - (1A20101) Breaker Open
Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Closed

K&A # 295003 Partial or Complete Loss of AC / 6

Importance Rating 4.1

QUESTION 43

K&A Statement: AK2.02 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: Emergency generators

Justification:

A. In-Correct – The bus will not remain deenergized. ESS Transformer T-201 (0X203) is the alternate supply for Unit '1A' ESS Bus, therefore 1A20109 Breaker will not close. The DG will start and energize the bus via the 1A20104 DG output breaker. This is a plausible choice if the candidate does not recognize the bus condition and breaker alignment.

B. In-Correct – On the loss of T-10, the normal supply breaker 1A20101 will trip open on undervoltage. Since the alternate source is not available, the diesel will start and energize the bus via the 1A20104 DG output breaker. This is a plausible choice if the candidate does not recognize the bus condition and breaker alignment.

C. In-Correct – On the transient, with the ESS Transformer T-201 (0X203) out of service, the bus alternate supply is not available and the alternate breaker, 1A20109 will not close. This is plausible since this is the response in the normal lineup.

D. Correct – The ESS Transformer T-201 (0X203) is the alternate supply for Unit '1A' ESS Bus, and since it is out of service, the 1A20109 Breaker will not close. With the normal and alternate source not available, the diesel will start and energize the bus via the 1A20104 DG output breaker.

References: TM-OP-004-ST page 23
ON-003-001 rev 17 pages 2,3
ON-104-201 rev 9 page 2

Student Ref: required No

Learning Objective: TM-OP-004-OB #2239

Question source: Modified Bank: #32 See original attached.

Question History: LOC 22 Cert

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 44

Unit 1 was operating at 100% power when a High Vibration condition caused a Main Turbine trip. During the transient, EOPs were entered.

EO-100-102, "RPV Control", directs the operator to reset the Main Generator Lockouts if RPV level can be maintained > minus (-)129 inches.

SELECT the correct basis for this EOP step.

A. To allow the transfer of House Loads to the Tie Bus.

B. To allow the Reactor Recirculation Pumps to be restarted.

C. To prevent a trip of the Reactor Recirculation Pumps.

D. To prevent a Plant Auxiliary Bus Load Shed signal.

K&A #	295031 Reactor Low Water Level
Importance Rating	3.2

QUESTION 44

K&A Statement: EK2.15 - Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: A.C. distribution: Plant-Specific

Justification:

A. In-Correct – This is not the basis for the step. This is a plausible choice since the turbine trip and generator lockout would result in the fast transfer of Aux Buses. The operators are required to ensure the transfer of house loads following the generator trip.

B. In-Correct – This is not the basis for the step. This is plausible since the candidate may recognize the RRPs would be tripped on either the EOC-RPT trip or the low RPV level at -38 inches. Since restoring core circulation is preferred post-scram, the candidates may believe that this is the reason.

C. In-Correct – This is not the basis for the step. This is plausible since the candidate may confuse the fact that RRPs trip on low RPV level (albeit -38 inches) and maintaining core circulation is preferred post-scram as the reason.

D. Correct – The EOPs direct this step to avoid an unnecessary Plant Auxiliary Load Shed signal. In anticipation of ECCS loads, either a Low RPV Level -129 inches (or Hi DW pressure 1.72 psig) with a main generator lockout will initiate Plant Aux Bus Load Shed. This logic trips balance of plant loads to include:
   - Condensate Pumps
   - Service Water Pumps
   - Circulation Water Pumps

   Balance of plant loads are maintained, if possible, to support the RPV pressure and level control strategy.

References: EO-000-102 rev 2 RC/L-8 page 17 Student Ref: required No

Learning Objective: TM-OP-0030-OB # 10059
PP002 # 14613

Question source: Bank: #319

Question History: LOC 19 NRC Exam 2003

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 45

Unit 1 operating in Mode 1 when a Service Water leak develops on the TBCCW Heat Exchanger (HX). The following conditions exist:

- TBCCW HX AREA FLOODED (AR-123-H01) alarm.
- NPO reports a leak on the upstream flange of the Unit 1 'A' TBCCW HX Outlet Isolation Valve, HV-10943A3.
- The 'A' TBCCW HX is in-service.

Which of the following is required to isolate the leak and restore cooling to TBCCW? (Assume the TBCCW HX area is accessible)

A. Maintain 'A' TBCCW HX in-service; Transfer 'A' TBCCW HX to ESW for cooling. Isolate the leak by closing the 'A' TBCCW SW Inlet Isolation Valve (109086).

B. Maintain 'A' TBCCW HX in-service; Transfer 'A' TBCCW HX to ESW for cooling. Isolate the leak by closing the 'A' TBCCW SW Inlet (109086) and Outlet isolation valves (109087).

C. Swap to the 'B' TBCCW HX in-service and align to Service Water. Isolate the leak by transferring the 'A' TBCCW HX to ESW which will isolate the outlet valve (HV10943A3).

D. Swap to the 'B' TBCCW HX in-service and align to Service Water. The leak is isolated by closing the 'A' TBCCW SW Inlet (109086) and Outlet Isolation Valve (109087).

K&A # 295018 Partial or Total Loss of CCW

Importance Rating 3.2

QUESTION 45

K&A Statement: AK3.05 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Placing standby heat exchanger in service

Justification:

A. In-Correct – Isolating the 1A TBCCW SW inlet and outlet isolation valves (109086 and 109087) will not isolate the leak, since ESW transfer valves, including HV10943A3, are inside the manual isolation valves. Placing ESW in-service on the 1A TBCCW heat exchanger will simply result in an ESW leak. This is plausible since on a loss of SW, swapping to ESW is a means to restore cooling and the candidate must understand the flowpath to know this is not a viable solution.

B. In-Correct – Closing the 1A TBCCW SW inlet isolation valve (109086) will not isolate the SW leak, since transferring to ESW will cause an ESW leak, as discussed above.

C. In-Correct – Swapping to 1B TBCCW HX will restore cooling. However, the leak cannot be isolated by transferring to ESW, as discussed above.

D. Correct – Isolating the SW inlet and outlet isolation valve on 1A TBCCW Heat exchange is the only way to isolate the SW leak. Swapping to the 'B' HX is required to restore TBCCW cooling.

References: TM-OP-115-ST pages 6-7
P&ID M-109 rev 13 sheet 2

Student Ref: required No

Learning Objective: TM-OP-011-OB # 1740f
AD045 # 15304

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 46

Both Units are operating in MODE 1. A fire has been detected and confirmed at Unit 1 'A' Reactor Feed Pump Turbine area. The following action has been taken:

- ON-013-001, "Response to Fire" was entered.
- Fire Brigade has been dispatched.

Subsequently, FIRE DET 106_Z4 ALM (Control Structure Outside Air Intake) actuates at Simplex Panel.

The Unit Supervisor directs you to place CREOASS in Recirculation mode per OP-030-002.

Placing CREOASS in-service is required to:

A. ensure control room habitability due to smoke from the fire.

B. ensure a Safe Shutdown Path is established for cold shutdown.

C. avoid the need to manually align CS HVAC from 0C879.

D. minimize the potential for the fire spreading to the control structure.

K&A # 600000 Plant Fire On-site
Importance Rating 2.8

QUESTION 46

K&A Statement: AK3.04 - Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site

Justification:

A. Correct – A fire outside of the Control Structure can introduce smoke into the Control Room via the outside air plenum. This is the purpose of the smoke detector in the plenum, and the control room simplex alarm. Placing CREOASS in the recirculation mode will ensure Control Room habitability.

B. In-Correct – This is not the reason. This is plausible since one of the purposes of the attachments in the offnormal procedure is to ensure that the plant can reach cold shutdown (safe shutdown).

C. In-Correct – This is not the reason. This is plausible distracter; for a loss of CS HVAC, ON-030-002 would be entered and the operator may have to reestablish CS HVAC at remote panel 0C879.

D. In-Correct – This is not the reason. This is plausible since the candidate may believe that the simplex alarm indicates that the fire is spreading to the control structure.

References: ON-013-001 rev 22 page 7 Student Ref: required No

Learning Objective: AD045 # 15310 and 15320

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 47

Unit 1 is at 18% power and shutting down in accordance with GO-100-004, "Plant Shutdown To Minimum Power" for a refueling outage following a 2 year operating cycle.

The turbine has just been tripped per step 5.21.2 of GO-100-004. With no other operator action, what is the expected plant response over the next 15 minutes?

A. Reactor power lowers to a new stable value due to BPVs opening to control RPV pressure.

B. Reactor power lowers to a new stable value, due to decay of xenon.

C. Reactor power rises to new stable value due to RPV level stabilizing at a new higher value.

D. Reactor power rises to new stable value due to feedwater temperature lowering.

K&A # 295005 Main Turbine Generator Trip
Importance Rating 2.8

QUESTION 47
K&A Statement: AK3.03 - Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: Feedwater temperature decrease

Justification:

A. In-Correct – When the turbine is tripped the BPVs open to maintain RPV pressure at EHC pressure set. Therefore BPV opening should not have any affect on reactor power.

B. In-Correct – xenon would be building in due to the power decrease adding negative reactivity. Plausible Distracter; xenon is a poison while will cause reactor power to change depending on if building in or burning out.

C. In-Correct –Expect RPV level to change slightly when removing the turbine from service. Several minutes after turbine is tripped level should return to normal based on FWLC system.

D. Correct – Following turbine trip, a loss of extraction steam to the feedwater heaters and FW temperature would lower adding positive reactivity to the core.

References: GO-100-004 rev 40 page 13 Student Ref: required No
TM-OP-047-ST page 31

Learning Objective: TM-OP-047-OB # 1852

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by:

QUESTION 48

Following a turbine trip from rated power, more than half of all control rods failed to insert. The PCOs initiated a manual scram and initiated ARI. The SCRAM AIR HEADER LOW PRESSURE (AR-107-G01) alarm was received and the following plant conditions exist:

- Reactor power is ~15% power.
- Scram Valves are open.
- ARI Valves repositioned.
- CRD is maximized.

Which one of the following actions is NEXT directed to insert control rods in accordance with the EOPs?

A. Manually vent the Scram Air Header in accordance with EO-100-113 Sheet 2, "Posted Hardcard".

B. Reset the SCRAM, and insert a manual SCRAM in accordance with ES-158-002, "RPS and ARI Trip Bypass".

C. Manually vent the Control Rod Drives in accordance with ES-155-001, "Venting of CRD to Insert Control Rods".

D. Remove RPS fuses to HCU scram solenoids in accordance with ES-158-001, "De-energizing Scram Pilot Solenoids".

K&A # 295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1

Importance Rating 3.9

QUESTION 48

K&A Statement: EA1.05 - Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : CRD hydraulics systems

Justification:

A. In-Correct – Not the correct action. This is plausible since the action is listed but will not be directed from EO-100-113 Sheet 2. Conditions provided are that of a hydraulic not electrical ATWS. Scram valves are open and venting the scram air header will be ineffective.

B. Correct – Action to bypass the scram and ARI is directed next IAW ES-158-002. With CRD available, this method is a success path to insert control rods in a hydraulic ATWS. Other action to manually drive control rods will also be directed concurrently.

C. In-Correct – Not the correct action. This is plausible since action is listed but will not be directed from EO-100-113 Sheet 2. Though this action should physically work in this plant condition, EO-100-113 Step CR-29 explicitly states that this is directed only after all other methods have failed. This is reserved due to the radiological hazards expected when this procedure is performed.

D. In-Correct – Not the correct action. This is plausible since the action is listed but will not be directed from EO-100-113 Sheet 2. Conditions provided are that of a hydraulic not electrical ATWS. Scram valves are open and venting the scram air header will be ineffective.

References: EO-000-113 rev 2 page 47-59 Student Ref: required No

Learning Objective: PP002 #14594

Question source: Bank #PP002/2638/2

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 49

With Unit 1 at full power, Instrument Air header pressure is slowly degrading.

Which one of the following is the Immediate Operator Action in accordance with ON-118-001 "Loss of Instrument Air" to initiate manual reactor scram AND the correct reason for the action?

As Instrument Air header pressure lowers, a manual reactor scram is required when Instrument Air header pressure reaches:

A. ≤75 psig since MSIVs will begin to drift closed, causing a pressure transient and scram.

B. ≤75 psig since Scram Inlet and Outlet Valves will drift open; causing control rods to insert.

C. ≤65 psig since MSIVs will begin to drift closed, causing a pressure transient and scram.

D. ≤65 psig since Scram Inlet and Outlet Valves will drift open, causing control rods to insert.

K&A #	295019 Partial or Total Loss of Inst. Air / 8
Importance Rating	3.3

QUESTION 49

K&A Statement: AA1.02 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Instrument air system valves: Plant-Specific
At Susquehanna, the Scram Valves are indirectly supplied by the the Instrument Air Header via the Scram Air Header. Since there is no direct monitoring of the Scram Air Header in the Control Room, the Scram Valves are considered IA system supplied valves, like e.g.,MSIVs, in the Loss of Instrument Air procedure.

Justification:

A. In-Correct – Reactor scram is not required until IA pressure reaches ≤65 psig. MSIV closure is not the reason, since this is an analyzed transient and scram. This is plausible since MSIVs will drift closed (<50 psig) and cause a significant transient.

B. In-Correct - Reactor scram is not required until IA pressure reaches ≤65 psig.

C. In-Correct - MSIV closure is not the reason, since this is an analyzed transient and scram. This is plausible since MSIVs will drift closed (<50 psig) and cause a significant transient.

D. Correct – Manual reactor scram is performed when Instrument Air Header pressure reaches ≤65 psig. From ON-118-001 bases: "It is of the utmost importance that the reactor be scrammed prior to the scram inlet and outlet valves drifting open (slow loss of instrument air pressure). Scram valves drifting open can result in random Control Rod insertion with the potential for scram discharge volume in-leakage and abnormal Core flux patterns." A manual scram is required since this can create an unanalyzed core condition.

References: ON-118-001 rev 21 pages 3, 21
TM-OP-018-ST page 4

Student Ref. required: No

Learning Objective: TM-OP-018-OB # 10245 &
AD045 # 15308

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 50

A Dual-Unit Shutdown is in progress.

Unit 1 is operating in Mode 3 with '1A' and '1B' RHR Pumps are running for Supp Pool Cooling. SPOTMOS average temperature reads 125°F.

Unit 2 is operating in Mode 4 when a loss of drywell cooling results in a high drywell pressure and valid LOCA signal. All LOCA actuations have occurred.

What is the status of the Unit 1 RHR pumps AND what is the availability of Unit 1 RHR to cool the Supp Pool?

A. All Unit 1 RHR pumps receive a trip signal. Any Unit 1 RHR pump can be started for Supp Pool Cooling when the associated Unit 2 RHR pump is stopped.

B. Only non-preferred RHR pumps receive a trip signal. One preferred RHR pump can be started for Supp Pool Cooling when any Unit 2 RHR pump is stopped.

C. Only the '1C' and '1D' RHR pumps receive a trip signal. '1A' and '1B' RHR pumps continue to run in Supp Pool Cooling.

D. All Unit 1 RHR pumps receive a trip signal. '1A' and '1B' RHR pumps can be immediately restarted for Supp Pool Cooling.

K&A # 295026 Suppression Pool High Water Temp. / 5
Importance Rating 4.1

QUESTION 50
K&A Statement: EA1.01 - Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling

Justification:

A. Correct – All Unit 2 RHR pumps start on the LOCA. All Unit 1 RHR pumps receive a trip signal. The RHR pumps are interlocked (breaker to breaker) such that a Unit 1 RHR pump can only be restarted when the associated Unit 2 RHR pump is overridden, i.e., manually stopped on Unit 2. Then the associated Unit 1 pump can be manually started for Supp Pool Cooling (or any other mode of RHR).

B. In-Correct – All Unit 1 RHR pumps receive a trip signal. Any running RHR pumps will be tripped on the Unit 2 LOCA signal. This is plausible since the LOCA/false-LOCA logic deals with preferred and non-preferred pumps. Preferred pump logic is A, B on Unit 1 and C,D on Unit 2.

C. In-Correct – All Unit 1 RHR pumps receive a trip signal. Any running RHR pumps will be tripped on the Unit 2 LOCA signal. This is plausible since the LOCA/false-LOCA logic deals with preferred and non-preferred pumps. Preferred pump logic is A, B on Unit 1 and C,D on Unit 2.

D. In-Correct – All Unit 1 RHR pumps receive a trip signal. The RHR pumps are interlocked (breaker to breaker) such that a Unit 1 RHR pump can only be restarted when the associated Unit 2 RHR pump is overridden, i.e., manually stopped on Unit 2. Then the associated Unit 1 pump can be manually started for Supp Pool Cooling (or any other mode of RHR). This is plausible since the LOCA/false-LOCA logic deals with preferred and non-preferred pumps. Preferred pump logic is A, B on Unit 1 and C,D on Unit 2.

References: EO-000-103 rev 3 page 6
TM-OP-049-ST page 15, 17

Student Ref: required No

Learning Objective: TM-OP-049-OB #192

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 51

Unit 1 was operating at 100% power when the following transient occurred;

- Loss of Off-Site Power to BOTH Units.
- The Main Turbine is tripped.
- PCOM reports an ATWS and only a few rods inserted.
- REACTOR HI PRESS (AR-101-C17) alarm is in.
- RX VESSEL HI PRESS TRIP (AR-103,104-B02) alarms are in solid.
- 4 SRVs automatically OPENED and RPV pressure is stable.
- ALL APRMs are de-energized.

The Unit Supervisor requests PCO to report "Initial ATWS Power" in accordance with EO-100-113.

Based on the above plant conditions, "Initial ATWS Power":

A. is approximately 65%.

B. is approximately 55%.

C. is approximately 25%.

D. cannot be determined.

K&A # 295025 High Reactor Pressure

Importance Rating 4.2

QUESTION 51

K&A Statement: EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor power

Justification:

A. In-Correct – Initial ATWS power is ~25%. This is a plausible distracter since the candidate may assume since no APRM power given, power level must be at 65%, the power level associated with natural circulation at the 100% rod line.

B. In-Correct – Initial ATWS power is ~25%. This is a plausible distracter since the candidate may assume since no APRM power given, power level must be at 55% assuming steam flow to the bypass valves. MSIVs are closed in this case.

C. Correct – Initial ATWS power is ~25% since each SRV approximately 850,000 lbm.hr steam flow or ≈ 6% power. Emergency operating procedures requires the use of alternate indications of power to determine Initial ATWS Power.

D. In-Correct – Power level can be determined by the number of SRVs open. This is a plausible distracter since the candidate may recognize that APRMs are deenergized and the bases state that if you have no alternate methods, then you can assume >5% and cannot be determined. Power can be determined in this case.

References: EO-000-113 rev 3 page 6
TM-OP-083-FS page 1
TM-OP-083E-FS page 1

Student Ref: required No

Learning Objective: PP002 # 14586

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 52

Unit 1 is in MODE 5. Refueling operations and movement of irradiated fuel assemblies are in progress. Fuel Pool level lowered and stabilized at 20.2 feet. Plant personnel are investigating and refueling has been stopped.

Regarding the Technical Specification (TS) LCO for the Spent Fuel Storage Pool Level, determine if the current fuel pool level is ABOVE or BELOW the MINIMUM level over irradiated fuel assemblies and the technical basis for this level?

A. Fuel pool is **ABOVE** the minimum level for the analysis since adequate inventory is available to maintain the spent fuel pool <115°F over the next 25 hours following a loss of fuel pool cooling event.

B. Fuel pool is **BELOW** the minimum level for the analysis since adequate inventory (lbm) is NOT available for anticipated pool boiling over the next 8 hours following a loss of fuel pool cooling event.

C. Fuel pool is **ABOVE** the minimum level for the analysis to ensure that refuel floor radiation levels for a postulated fuel handling accident will be maintained within the SPING operability and 10CFR20 limits.

D. Fuel pool is **BELOW** the minimum level for the analysis to ensure that the iodine release for a postulated fuel handling accident is adequately captured by the water to maintain dose within allowable limits.

K&A #	295023 Refueling Acc Cooling Mode
Importance Rating	3.4

QUESTION 52

K&A Statement: AA2.02 - Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : Fuel pool level

Justification:

A. In-Correct - The level is below the Tech Spec Limit (LCO) ≥22 feet and 21 feet, the actual analysis limit. This is not the technical basis for maintaining level over irradiated fuel assemblies. This is a plausible distractor since the fuel pool analysis do require fuel pool temperature be maintained below 115 °F.

B. In-Correct – This is not the technical basis for maintaining level over irradiated fuel assemblies. This is a plausible distractor since adequate inventory for fuel pool boiling is analyzed based on the decay heat load and administratively maintained by cross-connecting the fuel pools.

C. In-Correct –This is not the technical basis for maintaining level over irradiated fuel assemblies. This is plausible since the level in the pool does provide shielding and minimum levels are administratively controlled for personnel exposure during operations. SPING operability is affected by refuel floor background radiation levels.

D. Correct - The level is below the Tech Spec Limit (LCO) ≥22 feet (21 feet is the actual analysis limit stated in the bases). This level is based on a Fuel Handling Accident. With an assumed minimum water level of 21 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite and control room doses are maintained within allowable limits.

References: TS 3.7.7 and Bases Student Ref: required No

Learning Objective: TM-OP-035-OB #12574

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 53

Unit 2 has experienced a LOCA with ECCS systems injecting to the reactor.

- Suppression Pool level has lowered to 19 feet.

Which of the following is the concern of the emergency operating procedures with regard to the low Suppression Pool level?

A. The RCIC Turbine Exhaust has been uncovered.

B. The HPCI Turbine Exhaust has been uncovered.

C. Vortex / NPSH Limits for RHR pumps are violated.

D. Indicated SPOTMOS AVG temperatures are invalid.

K&A #	295030 Low Suppression Pool Water Level / 5
Importance Rating	3.9

QUESTION 53

K&A Statement: EA2.02 - Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Suppression pool temperature

Justification:

A. In-Correct – This is not the concern. This is plausible since the RCIC Turbine exhaust is a concern when SP level is below 17 feet.

B. In-Correct – This is not the concern. This is plausible since the HPCI Turbine exhaust is a concern when SP level is below 17 feet.

C. In-Correct – This is not the concern. This is plausible since the NPSH and Vortex Limit for RHR is a concern when SP level is below 18 feet.

D. Correct – This is the concern as stated in the EOPs. The upper temperature elements in the Supp Pool become uncovered at 20.5 feet, which are used in Supp Pool average temperature, making SPOTMOS and PICSY average invalid. The operator must recognize the limitation of the instrument and report Supp Pool temperature using only the lower SPOTMOS sensors.

References: EO-000-103 rev 4 pages 4, 5 Student Ref: required No

Learning Objective: PP002 #2602

Question source: Bank # 43

Question History: LOC 21 Certification Exam

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

10CFR 55.41 X

Comments Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 54

A seismic event has occurred and the following conditions exist on Unit 1:

- Loss of offsite power (LOOP) occurred.
- Steam leak occurred inside the Drywell.
- RPV Level is minus (-) 91 inches, lowering 2 inches/min.
- RPV Pressure is 720 psig, lowering 10 psig/min.
- DW temperature is 285°F, rising at 5°F/min.
- Drywell pressure is 18.2 psig, rising.
- Suppression Pool Temperature is 104°F, rising.
- Suppression Pool Pressure is 13.1 psig, rising.

Which one of the following EOP actions is required to mitigate this accident?

A. Align Drywell Sprays.

B. Initiate RPV Flooding.

C. Align all RHR Pumps for LPCI.

D. Initiate Rapid Depressurization.

K&A # 295028 High Drywell Temperature / 5
Importance Rating 3.7

QUESTION 54
K&A Statement: 2.4.6 - Emergency Procedures / Plan: Knowledge of EOP mitigation strategies as it applies to: High DW Temperature

Justification:

A. Correct – Given the above conditions, EO-100-103 Primary Containment Control, PC/P-7 and/or DW/T-5 requires spraying the drywell to lower containment pressure and/or pressure. Since ACC is assured and RPV pressure is high, this action is required to control primary containment parameters.

B. In-Correct – RPV Flooding IAW EO-100-114 is not required. This is a plausible distractor since the candidate must evaluate the plant parameters and determine if RPV level is usable.

C. In-Correct – Aligning all RHR pumps for LPCI injection is not required. EOP bases address this exact situation where it is appropriate to address the containment based on current RPV pressure and level. Core Cooling is assured by core submergence. This is plausible since with level at -91 inches, lowering at 2 inches/min, it will take 35 minutes to reach TAF and the EOP provides guidance for aligning RHR to LPCI.

D. In-Correct – Rapid Depressurization IAW EO-100-112 is not required. This is plausible since rapid depress would be required if HCTL limits of Figure 2 or PSL limits of Figure 4 were threatened. Also plausible since Drywell temperature is 285°F rising and a rapid depress is directed at 340°F. Candidate may also believe that a rapid depress is needed to inject with low pressure ECCS.

References: EO-000-103 rev 4 page 27, 45
EO-000-102 rev 3 page 19

Student Ref: Yes required
EO-100-103 flowchart (modified)

Learning Objective: PP002 # 14594

Question source: Bank: # 513

Question History: 2005 SSES NRC Exam

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 55

Unit 1 operating with reactor power at 100% when the Reactor Recirculation Pump Discharge Valve, HV-143-F031A, spuriously closes.

Following the runback of 1A Reactor Recirculation Pump, which set of indications at Panel 1C651 would be correct?

A.
- Gen 1A Speed SI-14032A 48%
- Gen 1A Demand XI-14032A 48%
- M/A Station output SY-B31-1R621A 79%

B.
- Gen 1A Speed SI-14032A 30%
- Gen 1A Demand XI-14032A 30%
- M/A Station output SY-B31-1R621A 79%

C.
- Gen 1A Speed SI-14032A 48%
- Gen 1A Demand XI-14032A 48%
- M/A Station output SY-B31-1R621A 48%

D.
- Gen 1A Speed SI-14032A 30%
- Gen 1A Demand XI-14032A 30%
- M/A Station output SY-B31-1R621A 30%

K&A # 295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4 (2.1.31)

Importance Rating 4.6

QUESTION 55

K&A Statement: 2.1.31 - Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Justification:

A. In-Correct - Discharge valve not full open causes a Limiter #1 runback. Both MG Set Speed and Demand meters would read 30%. This is plausible if the candidate confuses the runback signals and does not know which runback occurs.

B. Correct - Discharge valve not full open causes a Limiter #1 runback. Speed would lower to 30%, demand would lower to 30%, but the M/A station output on the horizontal meter would still indicate previous output (no operator action).

C. In-Correct - Discharge valve not full open causes a Limiter #1 runback. Both MG Set Speed and Demand meters would read 30%. This is plausible if the candidate confuses the runback signals and does not know which runback occurs or believes the M/A station output would match, a common misconception related to "controller output".

D. In-Correct - Discharge valve not full open causes a Limiter #1 runback. Both MG Set Speed and Demand meters would read 30%. This is plausible if the candidate confuses the runback signals and does not know which runback occurs or believes the M/A station output would match, a common misconception related to "controller output".

References:	TM-OP-064A-ST pages 8, 11, 12 Simulator Panel Diagram	Student Ref: required	No

Learning Objective: TM-OP-064-OB #2581

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 56

Given the following conditions:

- Unit 1 fire has resulted in the closure of all Outboard Main Steam Isolation Valves from 100 percent power.
- High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) both automatically initiated and are injecting.
- The Immediate Operator Actions of ON-100-009, "CONTROL ROOM EVACUATION", were completed.
- All Remote Shutdown Panel (RSP) Control Transfer Switches have been placed in "Emergency."
- RPV level is + 45 inches and rising slowly.

As RPV level continues to rise, what actions is required regarding RCIC operations IAW ON-100-009?

A. At +54 inches, ensure RCIC automatically trips on high level and AUTO starts at (minus) – 30 inches.

B. At +54 inches, RCIC must be manually tripped and ensure RCIC AUTO starts at (minus) – 30 inches.

C. At +54 inches, RCIC must be manually tripped and RCIC must be manually re-started.

D. At +54 inches, ensure RCIC trips on high level and RCIC must be manually re-started.

K&A #	295016 Control Room Abandonment / 7
Importance Rating	4.3

QUESTION 56

K&A Statement: 2.1.23 - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation. (Control Room Abandonment)

Justification:

A. In-Correct – RCIC will not automatically trip at + 54 inches. Plausible distracter; candidate may get this confused with HPCI

B. In-Correct – RCIC will not automatically initiate on minus -30 inches. Plausible distracter; the candidate may confuse this with HPCI

C. Correct – When operated from the RSDP all RCIC trips and initiations are defeated except for overspeed. Therefore the operator must trip RCIC on high water level of + 54 inches and must manually start RCIC when level lowers.

D. In-Correct – RCIC will not automatically trip at + 54 inches. Plausible distracter; candidate may get this confused with HPCI

References: OP-150-001 section 2.15 page 55 Student Ref: No required

Learning Objective: TM-OP-050-OB # 2018

Question source: Modified AD045/1361/008 See Attached

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 57

Unit 1 is operating at rated power. PJM has observed Grid Instabilities and TCC is taking emergency actions. Susquehanna is implementing actions in accordance with ON-103-001, 'Grid Instabilities' procedure when the following conditions occur:

Time (t) = 0 seconds

- GEN FLD OVERVOLTAGE ALARM (AR-106-A06)

Time (t) = 5 seconds

- GEN FLD OVERCURRENT ALARM (AR-106-B07)
- PCO reports Generator Field Current is 6150 amps, rising slowly

With no operator action, the Main Generator Voltage Regulator will:

A. provide voltage control in Automatic to maintain field between 4400 and 6000 amps.

B. initiate the Field Forcing feature in Automatic to maintain field below 5876 amps.

C. transfer to the Backup regulator and automatically control field amps between upper and lower amp limits.

D. transfer to the Manual regulator and operator action is required to control field amps within the machine limits.

K&A # 700000 Generator Voltage and Electric Grid Disturbances

Importance Rating 3.8

QUESTION 57

K&A Statement: AA1.02 - Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Turbine/generator controls.

Justification:

A. In-Correct – Given the above conditions, the voltage regulator will transfer to the Manual Voltage Regulator.

B. In-Correct – Given the above conditions, the voltage regulator will transfer to the Manual Voltage Regulator. This feature would have already occurred since the automatic regulator has control circuits to allow field forcing to assist in supporting the power system during emergencies (can last for thirty seconds or more depending on the magnitude of the excitation level). The generator field current limit is adjusted to 5876 amps DC if the Exciter Stator Overvoltage protection actuates. However, after 5 seconds, the regulator will automatically transfer to Manual, action is guaranteed once the overcurrent alarm is reached.

C. In-Correct - Given the above conditions, the voltage regulator will transfer to the Manual Voltage Regulator. This is plausible since other generators, e.g., Emergency diesels have a backup voltage regulator, however the Main Generator is not provided with a backup auto regulator.

D. Correct – After 5 seconds with the Overvoltage and Overcurrent conditions, the voltage regulator will swap to the Manual Regulator. Operator action is required to maintain field amps and the Generator output within procedure limits.

References: ON-103-001 rev 2 pages 5,6,10
TM-OP-0098D-ST page 11

Student Ref: required No

Learning Objective: TM-OP-098D-OB #1175

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 58

With station 125 VDC in a normal lineup, a system fault results in the loss of 1D634, 125 VDC Bus, which is now deenergized.

Following the DC power loss, an Emergency Diesel Generator LOCA start signal is received.

Which of the following describes the impact of this loss of DC power on the 'C' Diesel Generator?

The 'C' Diesel Generator:

A. Will NOT automatically start but can be manually started from the Control Room Panel, 0C653, 'C' Start pushbutton in Droop Mode.

B. Will NOT automatically start but can be manually started from the Engine Control Panel, 0C521, 'C' Start pushbutton in Local Mode.

C. Will automatically start when the 125V DC Transfer Switches are manually placed to 1D614 alternate supply.

D. Will automatically start when the 125V DC Transfer Switches are manually placed to 2D634 alternate supply.

K&A # 295004 Partial or Total Loss of DC Pwr / 6
Importance Rating 3.2

QUESTION 58

K&A Statement: AA2.04 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : System lineups

Justification:

A. In-Correct – The 'C' Diesel Generator will not automatically start and cannot be manually started (remotely or locally) since all 'C' DG logic power is from 1D634 and is deenergized. Alternate power must be aligned to the 'C' Diesel Generator.

B. In-Correct – The 'C' Diesel Generator will not automatically start and cannot be manually started (remotely or locally) since all 'C' DG logic power is from 1D634 and is deenergized. Alternate power must be aligned to the 'C' Diesel Generator.

C. In-Correct - The diesel generator control power transfer switches are normally aligned to their Unit 1 125VDC supply. Upon a loss of normal source, the offnormal procedure (in this case ON-102-630, Loss of 125V DC Source 1D630) directs the transfer switches aligned to the Unit 2 125V DC alternate source. Note that 1D614 is the Unit 1 Div 1 'A' Channel power supply and is the alternate source for other equipment transfer switches.

D. Correct – The diesel generator control power transfer switches are normally aligned to their Unit 1 125VDC supply. Upon a loss of normal source, the offnormal procedure (in this case ON-102-630, Loss of 125V DC Source 1D630) directs the transfer switches aligned to the Unit 2 125V DC alternate source.

References:	TM-OP-024-ST page 67 ON-102-630 rev 6 page 12	Student Ref: required	No
Learning Objective:	TM-OP-024-OB #2259, 2268, 2260g		
Question source:	Modified Bank: # SY017G3/105/105	See Original question attached	
Question History:	2001 Certification		
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	 X	
10CFR	55.41 X		
Comments:	Created/Modified by: Michaels Reviewed by: Morgan		

QUESTION 59

Following a LOCA, the following conditions exist:

• Drywell Hydrogen concentration	4.03 %
• Drywell Oxygen concentration	5.40 %
• Supp Chamber Hydrogen concentration	6.70 %
• Supp Chamber Oxygen concentration	3.00 %
• Supp Pool Level	23.9 Feet
• Reactor Pressure	0 psig

What actions are required to control containment gas IAW EO-000-103 “Primary Containment Control” AND why?

A. IF **NOT** in-service, start all Recombiners to ensure mitigation actions are in progress to avoid the need to vent the drywell.

B. IF in-service; shutdown all Recombiners to eliminate ignition sources to prevent combustion and exceeding structural capability of the drywell.

C. If in-service, shutdown Drywell Sprays to eliminate ignition sources to prevent combustion and exceeding structural capability of the drywell.

D. IF **NOT** in-service, start all DW Coolers and Fans to ensure mitigation actions are in progress to mix the drywell atmosphere and prevent hydrogen pockets.

K&A #	500000 High CTMT Hydrogen Conc. / 5
Importance Rating	3.3

QUESTION 59

K&A Statement: EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS: Containment integrity

Justification:

A. In-Correct – The intent is to eliminate possible ignition sources in the presence of a combustible atmosphere. Plausible Distracter; with H2 conc detectable in the PC Recombiners are started until COMBUSTIBLE LIMITS are reached.

B. Correct - COMBUSTIBLE LIMITS exist:

- DW OR SUPP CHMBR H2 Conc 6%
- DW OR SUPP CHMBR O2 Conc 5%

The intent of this step is to eliminate possible ignition sources in the presence of a combustible atmosphere If hydrogen and oxygen are present in sufficient concentrations, combustion may occur. High hydrogen and oxygen concentrations ignited in the confined space of the primary containment can generate peak pressures which may exceed the structural capability of the drywell, wetwell or drywell-wetwell boundary.

C. In-Correct - Drywell spray would only be initiated from EP-DS procedures in this condition. Plausible Distracter since drywell sprays are a mitigation strategy but not for the reason given here. These directions are given in EP-DS-001 not EO-000-103

D. In-Correct –, Once combustible limits are reached, EOPs require the operator to shutdown all DW Coolers and Fans. Plausible Distracter; since before H2 concentration reaches 2% Start all DW Coolers and Fans if DW Spray not initiated.

References: EO-000-103 rev 4 page 37, 38
Step PC/G-4

Student Ref: required None

EO-000-103 Flow Chart with Table 6, Combustible Gas Limits blanked out (flowchart edited)

Learning Objective: PP002 # 14584 and #14613

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 60

Given the following conditions:

- Unit 1 is operating at 100% power
- A short-circuit in the 1C668 pushbutton resulted in all Condenser Vacuum Breakers opening.
- All plant systems responded as designed.
- No operator actions were taken.
- Reactor pressure is fluctuating between a maximum pressure of 1127 psig and a minimum pressure of 995 psig.

Which one of the following is automatically controlling reactor pressure?

A. The Main Turbine Bypass Valves are controlling with the FAST Acting solenoids

B. The Main Turbine Bypass Valves are controlling in the NORMAL pressure control Mode

C. The Safety Relief Valves are controlling in the Safety Mode.

D. The Safety Relief Valves are controlling in the Relief Mode.

K&A # 295002 Loss of Main Condenser Vac / 3

Importance Rating 3.2

QUESTION 60

K&A Statement: AK2.04 - Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: Reactor/turbine pressure regulating system

Justification:

A. In-Correct – SRVs are controlling pressure. This is plausible since the Bypass valves would control pressure based on its pressure setpoint if bypass valve were available. Bypass valves not available due to MSIV isolation on low vacuum.

B. In-Correct – SRVs are controlling pressure. This is plausible since the Bypass valves would control pressure based on its pressure setpoint if bypass valve were available. Bypass valves are not available due to MSIV isolation and 7" Hg logic interlock on low vacuum.

C. In-Correct - In the Overpressure Safety mode, the SRVs function as Safety Valves and open by self-actuation, the first being at 1175 psig.

D. Correct –MSIVs and BPV close on low condenser vacuum. The SRV operate in the Relief mode, the valves are opened automatically by electrical signals from pressure switches, the first being set at 1106 psig.

References: TM-OP-083-ST pages 7,8 Student Ref: required No

Learning Objective: TM-OP-083-OB # 1651 & 10239

Question source: Bank # 201

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Bank # 201
Reviewed by: Michaels/Morgan

QUESTION 61

Regarding EO-100-103 "Primary Containment Control", maintaining plant parameters within the SAFE region of the Heat Capacity Temperature Limit protects the plant from:

A. steam being present in the Suppression Chamber Air Space during MSL Rupture.

B. excessive dynamic loading on the Suppression Chamber structures during Rapid Depressurization.

C. exceeding the opening pressure of the Primary Containment Vent Valves during Rapid Depressurization.

D. losing NPSH and Low Pressure ECCS Pump injection capability during a LOCA.

K&A # 295013 High Suppression Pool Temperature / 5
Importance Rating 3.6

QUESTION 61
K&A Statement: AK3.02 - Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE : Limiting heat additions

Justification:

A. In-Correct – This is not the basis. This is plausible since this statement is the basis for the Pressure Suppression Limit.

B. In-Correct – This is not the basis. This is plausible since this is the basis for maintaining Suppression Pool level < 38 feet.

C. Correct - The shape of the HCTL is determined by several competing factors related to the suppression chamber's ability to absorb all energy from an RPV depressurization without exceeding 65 psig in the suppression chamber (operating limit of the suppression chamber vent).

D. In-Correct – This is not the basis. This is plausible since high pool temperature affects NPSH and is bounded by the Vortex Limit Curve (SPL vs. Pump Flow) aides in maintaining ECCS capability.

References: EO-000-103 rev 4 page 11 Student Ref: required No

Learning Objective: PP002 #14616

Question source: Bank: # PP002/2602/4

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 62

Following a transient on Unit 1, HPCI is running (started component by component) in CST to CST for pressure control with the following conditions:

- RPV pressure 1,040 psig steady
- HPCI discharge pressure 625 psig steady
- HPCI speed 3,350 rpm
- HPCI flow 3,500 gpm in AUTO

Which one of the following actions is required to initiate and maximize a plant cooldown (<100°F/hr) in accordance with OP-152-001, "HPCI System Operation".

A. Place the FC-E41-1R600, HPCI Turbine Flow Controller in Manual and throttle the HV-155-F008, HPCI Test Line to CST Isolation to maximize discharge pressure.

B. Place the FC-E41-1R600, HPCI Turbine Flow Controller in Manual and throttle the HV-155-F011, HPCI Test Line to CST Isolation to minimize discharge pressure.

C. RAISE the FC-E41-1R600, HPCI Turbine Flow Controller setpoint and throttle the HV-155-F008, HPCI Test Line to CST Isolation to maximize discharge pressure.

D. RAISE the FC-E41-1R600, HPCI Turbine Flow Controller setpoint and throttle the HV-155-F011, HPCI Test Line to CST Isolation to minimize discharge pressure.

K&A # 295007 High Reactor Pressure / 3
Importance Rating 3.5

QUESTION 62
K&A Statement: AA1.02 - Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE : HPCI: Plant-Specific

Justification:

A. In-Correct – This action will not maximize the cooldown. This is plausible since throttling the HV-155-F008 valve to maximize HPCI discharge pressure will only increase turbine work slightly for the given flowrate. If anything, with HPCI in Manual, the flow controller will maintain turbine speed and flowrate will lower. This action will not maximize turbine work.

B. In-Correct – This action will not maximize the cooldown. First, the HV-155-F011 valve is not a throttleable valve and is the fully open isolation valve in the flowpath. This is plausible since the HV-155-F008 is the valve throttled and with HPCI in Manual, opening the valve would result in a slightly higher flow HPCI and lower pressure. This action will not maximize turbine work.

C. Correct – Raising the HPCI Flow Controller setpoint will raise flow, moving the most mass through a distance to maximize the work with the system (and initiate a cooldown). The HV-155-F008 is throttled to achieve the maximum discharge pressure for the highest flow to maximize turbine work.

D. In-Correct - The HV-155-F011 valve is not a throttleable valve and is the fully open isolation valve in the flowpath. This is plausible since the HV-155-F008 is the valve throttled and with HPCI , opening the valve would result in a slightly higher flow HPCI and lower pressure. This action will not maximize turbine work.

References: OP-152-001 rev 38 pages 24-27
TM-OP-052-ST page 46

Student Ref: required No

Learning Objective: TM-OP-052-OB #2031, 2032

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 63

Following a manual reactor scram on Unit 1 from 3%, you observe that not all control rods have fully inserted. The following conditions exist:

- RPV level lowered to +25 inches and stable
- RPV pressure remained stable at 965 psig on BPVs.
- Reactor power is Mid-Range 7 on IRMs, lowering.
- One control rod is at position 24.
- Five control rods are at position 04.
- Twenty-six control rods are at position 02.

Which of the following is the status of the reactor and procedure(s) required to be entered?

A. ATWS condition does **NOT** exist. Enter ON-100-101, Scram - Scram Imminent procedure only.

B. ATWS condition does **NOT** exist. Enter ON-100-101, Scram - Scram Imminent procedure and EO-100-102, RPV Control.

C. ATWS condition exists. Enter EO-100-102, RPV Control and EO-100-113, Level Power Control.

D. ATWS condition exists. Enter ON-100-101, Scram - Scram Imminent procedure and EO-100-113, Level Power Control.

	K&A #	295015 Incomplete SCRAM / 1
	Importance Rating	4.1

QUESTION 63

K&A Statement: AA2.02 - Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : Control rod position

Justification:

A. In-Correct – Rod positions are beyond the current Maximum Subcritical Bank Withdraw Position (MSBWP) of Position 02. Plausible choice since MSBWP is cycle specific and was Position 04 last cycle. ON-100-101 directs entry into EO-100-113 at LQ/Q-2.

B. In-Correct - Rod positions are beyond the current Maximum Subcritical Bank Withdraw Position (MSBWP) of Position 02. Plausible choice since MSBWP is cycle specific and MSBWP was Position 04 last cycle. Also In-Correct since no EO-102 entry conditions exist and ON-100-101 directs entry into EO-100-113 at LQ/Q-2.

C. In-Correct - No EO-102 entry conditions exist and ON-100-001 is required to direct entry into EO-113. Plausible since if EO-102 entry was required, EO-102 would be entered first. Follow-up actions in ON-100-101 would be prioritized low until EO-113 is entered and the plant is stabilized.

D. Correct - Rod positions are beyond the current Maximum Subcritical Bank Withdraw Position (MSBWP) of Position 02. Entry into ON-100-101 is required and subsequently directs entry into EO-100-113 LQ/Q-2 that provides guidance for control rod insertion. (Refer to EO-000-113 Bases attached.)

References: ON-100-101 rev 17 page 3
EO-000-113 rev3 pages 4,5 8
(PCAF 2005-1512)

Student Ref: required No

Learning Objective: PP002 #14590

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 64

An unisolable ESW leak develops in the Unit 1 Reactor Building Sump Room. The following alarms are received:

- REACTOR BLDG SUMP LEVEL HI-HI (AR-125-B01).
- PANEL 1C692 SYSTEM TROUBLE (AR-107-H14)
- CORE SPRAY ROOM A FLOODED (AR-109-H01)

The NPO reports that there is at least 2-1/2 feet of water in the 'A' Core Spray Room and the Sump Room.

Which of the following is correct regarding the Max Safe Water Level and what action is required by the EOPs?

A. Max Safe Water Level has NOT been exceeded in the Reactor Building, and continue actions to isolate the leak.

B. Max Safe Water Level has been exceeded in the Reactor Building, and commence a normal reactor shutdown IAW GO-100-004.

C. Max Safe Water Level has been exceeded in the Reactor Building, and scram the reactor and enter EO-100-102 "RPV Control".

D. Max Safe Water Level has been exceeded in the Reactor Building, and continue actions to isolate the leak.

K&A # 295036 Secondary Containment High Sump/Area Water Level / 5

Importance Rating 3.8

QUESTION 64

K&A Statement: 2.4.20 - Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes. (Secondary Containment High Sump/Area Water Level)

Justification:

A. In-Correct - Core Spray Room 'A' and Rx Bldg sump room is not separated by a water tight door, the two rooms will equalize, Max Safe for 'A' Core Spray Room is 24 inches. This is a plausible distractor since the candidate may incorrectly evaluate the level conditions in the rooms.

B. In-Correct – If the students believes that there are TWO AREA'S > Max Safe i.e. 'A' Core Spray and Sump Room, than EO-100-104 requires a reactor SD IAW GO-100-004. This is a plausible distractor since the candidate may believe that there are two area > Max Safe.

C. In-Correct – Primary system is not discharging in the secondary containment; therefore reactor scram not required. This is a plausible choice since the candidate may answer question SC/L3 In-Correctly and with one area > Max Safe Reactor Scram would be required.

D. Correct – Core Spray Room 'A' and Rx Bldg sump room is not separated by a water tight door, the two rooms will equalize; therefore Core Spray Room 'A' will be at max safe and the sump room is not a separate area there fore we only have one area > Max Safe and no further actions are required by the EOP. ON-169-002 directs securing the source of the leak.

References: EO-000-104 rev 2 pages 26, 27

Student Ref: **Yes** required
EO-100-104 with Table 10 (flowchart edited)

Learning Objective: PP002 # 14586 & 14594

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 65

Both Units are operating at rated power with the following conditions:

- Reactor Building ventilation systems are in a normal alignment.
- SGTS system is in a normal alignment.
- CREOASS is in a normal lineup with 'A' train in Lead.
- PCO reports REFUELING FLOOR HI EXHAUST HI-HI RADIATION alarm (AR-106-E03)
- RISHH-D12-1K615B, 'B' Refuel Floor High Exhaust Duct Rad Monitor is determined to be failed UPSCALE.
- No other Refuel Floor radiation monitors are alarming.

What plant response, if any, results from this single instrument failure?

A. Zone III isolation only.
Both SGTS trains and 'A' CREOASS fans start.

B. Zone III isolation only.
'B' SGTS and 'B' CREOASS fans start.

C. Both Zone I and Zone III isolations.
Both SGTS trains and Both CREOASS fans start.

D. No Zone Lockouts are tripped and No Isolations occur.
No SGTS and No CREOASS fans receive start signals.

K&A # 295034 Secondary Containment Ventilation High Radiation / 9

Importance Rating 3.9

QUESTION 65

K&A Statement: EK2.06 - Knowledge of the interrelations between SECONDARY CONTAINMENT VENTILATION HIGH RADIATION and the following: PCIS/NSSSS: Plant-Specific

Justification:

A. In-Correct – 'A' SGTS and 'A' CREOASS will not receive start signals. This instrument will trip the Zone 3B lockout only. This is plausible since SGTS and CREOASS are fully divisionalized and only 'B' SGTS and 'B' CREOSS fans start.

B. Correct - A trip of the 1B Refuel Floor High Rad Monitor will trip the Zone 3B Lockout directly. SGTS and CREOASS are fully divisionalized. 'B' SGTS will start immediately and the 'B' CREOASS will start after a time delay (since 'A' is in Lead).

C. In-Correct – This will result in a Zone 3 Isolation only (not a Zone 1 in this case since only the rad monitor tripped). 'B' SGTS and 'B' CREOSS fans start since a trip of the 1B Refuel Floor High Rad Monitor will trip the Zone 3B Lockout. This is plausible since SGTS and CREOASS are fully divisionalized. 'A' SGTS and 'A' CREOASS will not receive start signals.

D. In-Correct – This will result in a Zone 3 Isolation only. 'B' SGTS and 'B' CREOSS fans start since a trip of the 1B Refuel Floor High Rad Monitor will trip the Zone 3B Lockout. SGTS and CREOASS are fully divisionalized. 'A' SGTS and 'A' CREOASS will not receive start signals. This is a plausible choice if the candidate believes that 2/2 logic exists here and no actuations occur on a single failure.

References: TM-OP-079E-ST pages 13, 14
TM-OP-070-ST pages 34, 35
TM-OP-030-ST pages 18, 19

Student Ref: required No

Learning Objective: TM-OP-079E-OB # 1194

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 66

Due to staffing issues, the on-coming shift complement will be at MINIMUM staffing levels required by Tech Specs.

Prior to turnover, a PCO on the on-coming shift has called off due to a personal emergency and cannot report to work.

Given that this individual is necessary to support Tech Spec minimum shift complement, what actions must be implemented by Shift Supervision in accordance with NDAP-QA-0300, CONDUCT OF OPERATIONS?

A. The Shift is permitted to drop below the minimum manning for a maximum of two hours in order to replace the PCO position.

B. The Shift must holdover a PCO from the off-going shift to maintain minimum shift manning until an active licensed operator arrives.

C. The Shift is permitted to drop below minimum manning for an unspecified time period, provided a replacement is immediately called to fill the position.

D. The Shift must holdover an individual qualified as Emergency Plan Communicator until a replacement licensed operator arrives.

	K&A #	2.1.9 Conduct of Operations
	Importance Rating	2.9

QUESTION 66

K&A Statement: 2.1.9 Ability to direct personnel activities inside the control room.

Justification:

A. In-Correct – It is not permissible to drop below the minimum manning for this condition. This is a plausible distractor since on-shift manning may be less than the minimum **Emergency Plan requirement** for a period of time not to exceed 2 hours in order to accommodate unexpected unavailability of **on-duty shift individuals** (provided immediate action is taken to restore shift compliment to within the minimum requirements).

B. Correct - If an individual scheduled to fill a required position calls off or does not report to fill that position, Shift Supervision will hold over a qualified individual to fill the position until a qualified replacement can take the position.

C. In-Correct – It is not permissible to drop below the minimum manning for this condition. This is a plausible distractor since on-shift manning may be less than the minimum **Emergency Plan requirement** for a period of time not to exceed 2 hours in order to accommodate unexpected unavailability of **on-duty shift individuals** (provided immediate action is taken to restore shift compliment to within the minimum requirements).

D. In-Correct – The PCO is needed as a PCO for Tech Spec Minimum staffing, not the E-Plan which is above the minimum staffing. This is plausible since this is often confused and is the reason for NDAP-QA-0300 Attachment B.

References: NDAP-QA-0300 Conduct of Operations Section 6.2.1 Pages 8-10, 15 Student Ref: required No

Learning Objective: AD044 #14553, 14554

Question source: Bank # AD044/14553/3

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 67

Unit 1.is in MODE 1. The following conditions exist:

- A break occurred inside the RWCU Room and **cannot** be isolated.
- The Unit 1 Reactor Building SPING is Out of Service.
- Security reported steam exiting U1 Reactor Building, North-end.
- TSC is waiting on HP Dose Calculations.
- Emergency Director/TSC is evaluating an upgrade to a Site Area Emergency. (EAL RS1)
- Containment Radiation indicates 15 R/hr.

Which one of the following is required in accordance with the Emergency Operating Procedures?

A. EO-100-105 “Radioactivity Release Control” requires the Mode Switch placed in SHUTDOWN. Enter EO-100-102 “RPV Control” to commence a plant cooldown at <100°F/hr.

B. EO-100-104 “Secondary Containment Control” requires the isolation of ZONE 1 and 3 HVAC by performing ES-070-001. Commence a controlled reactor shutdown in accordance with GO-100-004.

C. EO-100-105 “Radioactivity Release Control” requires the Mode Switch placed in SHUTDOWN. Enter EO-100-102 “RPV Control” AND perform a rapid depress per EO-100-112.

D. EO-100-104 “Secondary Containment Control” requires the Mode Switch placed in SHUTDOWN. Enter EO-100-102 to maintain RPV pressure 800 to 1087 psig due to the fuel failure.

K&A #	2.1.20 Conduct of Operations
Importance Rating	4.6

QUESTION 67

K&A Statement: 2.1.20 Ability to interpret and execute procedure steps.

Justification:

A. Correct – IAW EO-000-105 Table 11 and Step RR-5 Scram the reactor and enter EO-100-102. The purpose of entering EO-100-102 is to commence a cool down to limit the energy that is released.

B. In-Correct – EO-100-104 direct the operator to scram the reactor and enter EO-100-102. This is a plausible distracter since step SC-2 states that if RB or SGTS SPING RAD Release exceeds Hi-Hi and ZONES not isolated perform ES-070-001. Also, with RB SPING out of service, the candidates may believe this is the correct action.

C. In-Correct – A rapid depressurization is not required. This is plausible since EO-100-105 requires a rapid depressurization when unmonitored release rates are unknown or Containment Rad ≥50R/hr

D. In-Correct – EO-100-105 requires entering EO-100-102 to commence cool down to limit release. This is plausible since there are other overriding bases in the other conditions to delay the depressurization to keep fuel isotope in the fuel rods.

References: EO-000-105 rev 2 pages 3, 6 Student Ref: required EO-100-104 and 105 Yes

Learning Objective: PP002 #14585 & 14594

Question source: Modified Bank # PP002/2622/9 See original attached

Question History: No recent history

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 68

Units 1 & 2 are at 100% power with 'A' Diesel Generator SOW in-progress.

Three (3) days into the diesel overhaul, Mechanical Maintenance has requested permission to perform a feed and bleed of the jacket water cooling system as approved by the SO Rep. The feed and bleed will be done by opening the Demin water supply, filling the Stand Pipe to the high level, close the Demin water supply and opening the Standpipe drain to the low level alarm and closing the drain.

Which of the following describes whether or not Maintenance is permitted to independently perform the above activity?

A. Maintenance may perform the requested activities provided Ops Supervision releases the work and mechanics document all manipulations on a status form.

B. Maintenance may perform the requested activities provided Ops Supervision releases the work and the mechanics status all manipulations with caution tags.

C. Maintenance may **NOT** perform the requested activities since the activity is on safety related equipment that must be status control tagged by Operations.

D. Maintenance may **NOT** perform the requested activities under any circumstances since only Ops personnel are allowed to manipulate plant equipment.

K&A # 2.2.17 Equipment Control
Importance Rating 2.6

QUESTION 68

K&A Statement: 2.2.17 Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, coordination with the transmission system operator.

Justification:

A. Correct - Per NDAP-QA-0302 Section 6.6 with operations permission and an appropriate instruction. Component manipulations are documented on the NDAP-QA-0502-5 Form.

B. In-Correct - The procedure states that operations will determine the need for status control tags. This is plausible if the candidate confuses the tagging requirements.

C. In-Correct – Maintenance may independently perform the activity IAW NDAP-QA-0502. This is plausible since generally, only Operations is allowed to operate plant components. If the candidate does not recognize that this is a permitted exception, this answer may be chosen.

D. In-Correct – Maintenance may independently perform the activity IAW NDAP-QA-0502. This is plausible since generally, only Operations is allowed to operate plant components. If the candidate does not recognize that this is a permitted exception, this answer may be chosen.

References: NDAP-QA-0302 rev 16 page 24 Student Ref: required No

Learning Objective: AD044 # 14562

Question source: Bank: #123

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 69

During Unit 1 refueling operations, a problem occurs during the placement of a component.

- The Refuel Platform remains mobile and operational.
- ON-081-002, "Refuel Platform Anomaly" has been entered.

Complete the statement that describes the criteria that determines the placement of the grappled component.

The placement of the grappled component is based on:

A. the type of component that is being raised (fuel bundle, blade guide) that is currently grappled.

B. the bottom of the component being raised above the core top guide or top of the fuel rack.

C. the direction in which the component is being transferred (to/from the fuel pool or to/from the core).

D. the refuel platform location and its proximity to its departure or destination points for the current move.

K&A # 2.1.41 Conduct of Operations
Importance Rating 2.8

QUESTION 69
K&A Statement: 2.1.41 Knowledge of the refueling process.
Justification:

A. In-Correct – The type of component does not determine the location of placement. This is plausible since these components have different requirements based on the refueling conditions, but this is not the criterion.

B. Correct - If the fuel bundle/blade guide has not been raised completely out of its original location (i.e., if the bottom of the fuel bundle or blade guide has not been raised above the top of the Top Guide, Control Blade, or top of Fuel Pool Rack), it is returned to its original location. If the fuel bundle/blade guide has been raised completely out if its original location, it is transferred to a setdown location in the fuel pool.

C. In-Correct – The direction of the component does not determine the location of placement. This is plausible since location requirements exist based on the refueling conditions, but this is not the criterion.

D. In-Correct – The location of the platform and its proximity does not determine the location of placement. This is plausible since certain platform requirements exist based on the refueling conditions, but this is not the criterion.

References: ON-081-002 rev 16 page 24 Student Ref: required No

Learning Objective: AD045 #15308

Question source: Bank: #AD045/1360/21

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 70

Unit 1 is at power with the RWCU system in service. Due to a broken reach-rod on 166004, RWCU BACKWASH RCV TANK DRAIN VLV, you have been directed to enter the RWCU Backwash Receiving Tank (BWRT) Room to perform the valve position verification.

A pre-job brief is being provided to support valve lineup checks in preparation for the reach-rod repair. The HP ALARA briefing identifies the following:

- Your total dose for the year is 400 mRem TEDE. The SSES Administrative Dose Limits shall not be exceeded and no dose extensions have been authorized.
- HP dose allowance for checking the position of all valves, other than 166004 is 600 mRem.
- An additional dose is required to check 166004, RWCU BACKWASH RCV TANK DRAIN VLV, closed. (Refer to the attached Area Survey map)

For this entry, determine the System Blocking Requirement and calculate the Maximum Stay Time to complete the valve position verifications.

A. System blocking is **NOT** required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 24 minutes.

B. System blocking is **NOT** required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 60 minutes.

C. System blocking **IS** required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 12 minutes.

D. System blocking **IS** required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 36 minutes.

K&A # 2.3.12 Radiation Control
Importance Rating 3.2

QUESTION 70

K&A Statement: 2.3.12 Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Justification:

A. In-Correct - Inside the shield wall requires ALARA blocking. 24 minutes is calculated using the administrative limit of 2000 and not accounting for the present dose and dose for checking the other valves and exit time.

B. In-Correct - 60 minutes is calculated using the administrative limit of 4000 which requires a dose extension and subtracting the present dose and dose for checking the other valves and exiting.

C. Correct - Candidate will need to use attached figure to determine the location of the valve is inside the shield wall. Entrance inside the shield wall requires ALARA blocking to be initiated. Candidate must then calculate max stay time not to exceed SSES Administrative limit of 2000 mrem (w/o a dose extension) (2000 limit minus 400 present dose, minus 600 for checking other valves and exit, leaves 1000 to check 166004 position. Then, 1000 mrem divided by 5rem/hr is 12 minutes.

D. In-Correct - 36 minutes is calculated using the administrative limit of 4000 which requires a dose extension and subtracting the present dose and dose for checking the other valves and exiting.

References: NDAP-QA-0625 rev 9 page 25
NDAP-QA-0626 rev 9 pages 25,28

Student Ref: required Yes

Provide Area Survey Map attached

Learning Objective: AD044 # 15106 & 15107

Question source: Bank: # 401

Question History: None

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

10CFR 55.41 X

Comments: Created/Modified by: Bank
Reviewed by: Michaels/Morgan

QUESTION 71

An emergency condition exists on site involving both units. The Emergency Plan has been entered. It is determined that normal station exposure limits will be exceeded to perform a search and rescue operation in the Reactor Building for a disabled worker.

Whose final approval is required to allow an individual to exceed 10CFR20 exposure limits and what is the Emergency Plan limit (EPA-400) for this Lifesaving operation?

A. Emergency Director approval not to exceed 10 Rem.

B. Emergency Director approval not to exceed 25 Rem.

C. Radiation Protection Coordinator approval not to exceed 10 Rem.

D. Radiation Protection Coordinator approval not to exceed 25 Rem.

K&A # 2.3.4 Radiation Control
Importance Rating 3.2

QUESTION 71

K&A Statement: 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

Justification:

A. In-Correct – Limit for lifesaving activity is 25 Rem. This is plausible since 10 R is the limit for equipment protection in accordance with EP-PS procedure.

B. Correct – ED may authorize up to 25 Rem emergency limit for lifesaving activity.

C. In-Correct – Rad Protection Coordinator may not independently authorize this emergency dose. Plausible since the Rad Protection Coordinator should be consulted during the authorization. Limit for lifesaving activity is 25 Rem. This is plausible since 10 R is the limit for equipment protection in accordance with EP-PS procedure.

D. In-Correct - Rad Protection Coordinator may not independently authorize this emergency dose. Plausible since the Rad Protection Coordinator should be consulted during the authorization.

References: EP-PS-101 Tab 8
E-Plan rev 49, page 61-62, 95

Student Ref: required No

Learning Objective: AD044 #15106

Question source: Modified Bank: #EPPS1006/19 19 see original attached

Question History: 2002 NRC SRO Exam

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 72

While operating at rated power, a spurious isolation of Unit 1 Reactor Building (RB) ventilation system occurred with the following conditions:

- RB Zone 1 isolation will NOT reset.
- RB Zone 1 differential pressure reads 0.00 inches WG.
- RB Zone 3 isolation was reset and automatically restarted.
- RB Zone 3 differential pressure reads 0.30 inches WG.
- RB Area temperatures are rising, but none are in alarm.

ON-134-002, High/Low Reactor Building Differential Pressure was entered. Engineering and FIN report the isolation logic relay is failed and must be replaced.

Given these conditions have existed for the past one hour, which set of procedure actions below is required?

A.
1) Continue actions in ON-134-002, "High/Low Reactor Building Differential Pressure".
2) Entry into EO-100-104, "Secondary Containment Control" is immediately required, execute procedures concurrently.
3) Start ESW and all Unit Coolers with Cooling Source.

B.
1) Continue actions in ON-134-002, "High/Low Reactor Building Differential Pressure".
2) Entry into EO-100-104, "Secondary Containment Control" is NOT required for the next three hours.
3) Commence a normal reactor shutdown IAW GO-100-004.

C.
1) Entry into EO-100-104, "Secondary Containment Control" is immediately required.
2) Exit ON-134-002, "High/Low Reactor Building Differential Pressure" to avoid conflicting with EOP guidance.
3) Start ESW and all Unit Coolers with Cooling Source.

D.
1) Entry into EO-100-104, "Secondary Containment Control" is immediately required.
2) Exit ON-134-002, "High/Low Reactor Building Differential Pressure" to avoid conflicting with EOP guidance.
3) Commence a normal reactor shutdown IAW GO-100-004.

K&A # 2.4.16 Emergency Procedures / Plan
Importance Rating 3.5

QUESTION 72

K&A Statement: 2.4.16 Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, AOP's and SAMG's.

Justification:

A. Correct – Zone 1 (2) or Zone 3 differential pressure < 0.25' WG Vacuum is indicative of a potential loss of reactor building structural integrity and could result in uncontrolled release of radioactivity to the environment. Consistent with Technical Specification LCO, a four (4) hour qualifier is added (to the EO entry condition) to allow off-normal (ON) and system operating (OP) procedures to be utilized in attempting to restore secondary containment HVAC. If, however, a SC HVAC isolation cannot be reset and differential pressure cannot be maintained > 0 inches WG vacuum (e.g., SGTS does not initiate), entry into this procedure at SC-1 is directed by ON-134-002 (ON-234-001), Loss of Reactor Building Differential Pressure.

Note that both procedures are executed concurrently. ESW and Unit Coolers are started to provide ECCS Room Cooling.

B. In-Correct – Although the EO-104 entry condition states 'ZONE 1 OR 3 < 0.25'WG VACUUM FOR 4 HRS), entry into EO-104 is required in this condition as stated in both EO-100-104 and ON-134-002 procedures. Also, a reactor shutdown is not required until Zone 1 ventilation is lost for 4 hours.

C. In-Correct – Entry into EO-100-104 is immediately required in the conditions provided in the stem. Both EO-100-104 and ON-134-002 procedures are executed concurrently.

D. In-Correct – Entry into EO-100-104 is immediately required in the conditions provided in the stem. Both EO-100-104 and ON-134-002 procedures are executed concurrently. Also, a reactor shutdown is not required until Zone 1 ventilation is lost for 4 hours.

References: EO-000-104 rev 2 page 5
ON-134-002 rev 12 pages 4, 14

Student Ref: required No

Learning Objective: PP002 #2621,2622

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan

QUESTION 73

A call is received from the FBI at 2310 that a CREDIBLE imminent security threat exists for the Susquehanna Power Station.

The Shift Manager assumes the role of Emergency Director and directs the NRC Communicator to activate ALTERNATE NERO.

Which location below is the required facility for Technical Support Center personnel to report?

A. Information and Visitor Center on US Route 11.

B. State Highway Patrol Barracks on US Route 11.

C. Emergency Operations Facility in Wilkes-Barre.

D. Operations Support Center in the South Building.

	K&A #	2.4.42 Emergency Procedures / Plan
	Importance Rating	2.6

QUESTION 73

K&A Statement: 2.4.42 Knowledge of emergency response facilities.

Justification:

A. In-Correct – This is not the alternate location. This is plausible since this location can be used as the Media Operations Center.

B. In-Correct – This is not the alternate location. This is plausible since the Incident Command Team reports here.

C. Correct – During a security event after hours, TSC personnel are required to report to the Alternate TSC, which is the EOF at the East Mountain Business Center in Wilkes-Barre.

D. In-Correct – This is not the alternate location. This is plausible since this is the location NPOs and FUS report.

References: EP-PS-100-G rev 14 page 4
Emergency Plan rev 49 page 15

Student Ref: required No

Learning Objective: EP001 #3

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

QUESTION 74

Unit 1 is operating at 90% power while RWCU was isolated to repair a flange leak on the Regenerative Heat Exchanger. Shortly after RWCU is returned to service, RWCU INLET HI-LO CNVTY (AR-101-E02) alarms. The following reactor coolant chemistry values are confirmed:

- RWCU Inlet conductivity: 3.20 μmho/cm
- RWCU Outlet conductivity: 0.05 μmho/cm
- Cond Demin Inlet conductivity: 1.60 μmho/cm
- Feedwater Inlet conductivity: 0.15 μmho/cm

Based on the above conditions, which one of following is the required action?

A. Power operation may continue at 90%. Coordinate with Chemistry and remove RWCU filter-demin from service.

B. Commence a power reduction to < 75%. Coordinate with Chemistry and remove COND filter-demin from service.

C. Commence a power reduction to <75%. Coordinate with Chemistry and remove affected condenser waterbox from service.

D. Commence an orderly normal shutdown. Coordinate with Chemistry and remove affected condenser waterbox from service.

K&A #	2.1.34 Conduct of Operations
Importance Rating	2.7

QUESTION 74

K&A Statement: 2.1.34 Knowledge of primary and secondary plant chemistry limits.

Justification:

A. In-Correct – This action is not correct. This is plausible if the candidate does not recognize that although RWCU coolant conductivity is high, the RWCU Filter-demins are performing and removing impurities.

B. In-Correct – This action is not correct. This is plausible if the candidate misinterprets the High Feedwater inlet conductivity reading being caused by a failing Condensate Demineralizer. In fact, the CDI is higher than the Feedwater indicating that Cond Demins are still performing and removing impurities.

C. Correct – Based on the CDI and FW conductivity readings, the problem is in a waterbox and power must be reduced to <75%to limit the concentration in the vessel. A subsequent power reduction to <60% is not required until the affected waterbox is removed from service for ALARA.

D. In-Correct – This action is not correct. This is plausible if the candidate misinterprets the chemistry levels are above the Action 3 or Tech Spec levels, either of which directs a plant shutdown and cooldown.

References:	ON-100-003 rev 18 page 6	Student Ref: required **Yes**
		ON-100-003 pages 1-10 only
Learning Objective:	AD045 #15304	
Question source:	Modified Bank: #AD045/1358/24	See original attached.
Question History:	None	
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis: X	
10CFR	55.41 X	
Comments:	Created/Modified by: Michaels Reviewed by: Morgan	

QUESTION 75

In accordance with NDAP-QA-0320, Special, Infrequent Or Complex Test / Evolutions (SICTE), Which one of the following outage activities meets the definition of a SICTE?

A. Performance of the Div 1 LOCA/LOOP and DG test in accordance with SE-124-107.

B. Performance of the Comprehensive ESW Flow Verification Loop A test in accordance with SO-054-A08.

C. Performance of the Primary Containment Integrated Leak Rate test in accordance with SE-100-003.

D. Performance of the post-installation test of temporary Shutdown Decay Heat Removal (SDHR) system in accordance with TP-135-011.

K&A # 2.2.7 Equipment Control
Importance Rating 2.9

QUESTION 75

K&A Statement: 2.2.7 Knowledge of the process for conducting special or infrequent tests.

Justification:

A. In-Correct – This SE is not considered a SICTE. This is plausible distractor since the 24 month surveillance for LOCA/LOOP testing is performed every refueling outage in accordance with an established System Engineering procedure.

B. In-Correct – This SO is not considered a SICTE. This is plausible distractor since the ESW flow surveillance is periodically performed during outages using an established Operations Surveillance procedure.

C. Correct – This is considered a SICTE. By definition, this test shall be a SICTE since it occurs at a specified frequency greater than one refuel cycle. Further, the containment integrated leak test is an involved complex evolution that requires the coordination of the entire station to complete.

D. In-Correct – This TP is not considered a SICTE. This is plausible distractor since the temporary SDHR system is installed and tested every refueling outage using this established Technical Procedure.

References: NDAP-QA-0320 rev 8 page 6 Student Ref: required No

Learning Objective: AD044 #15315

Question source: New

Question History: N/A

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

10CFR 55.41 X

Comments: Created/Modified by: Michaels
Reviewed by: Morgan