U.S. Nuclear Regulatory Commission
Site-Specific
SRO Written Examination
Applicant Information

Applicant Information

Name: **ANSWER KEY**

Date: **12/15/03**

Facility/Unit: **SUSQUEHANNA**

Region: [✓ I] / II / III / IV

Reactor Type: W / CE / BW / [✓ 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 overall, with a 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require an 80.00 percent to pass. You have eight hours to complete the combined examination, and three hours if you are only taking the SRO portion.

Applicant Certification

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

Applicant's Signature

Results

RO / SRO-Only / Total Examination Values	N/A / 25 / 25 Points
Applicant's Scores	N/A / ___ / ___ Points
Applicant's Grade	N/A / ___ / ___ Percent

PPL SUSQUEHANNA, LLC
STANDARD EXAM SHEET

Course #: RO17 Course: LOC-19 NRC LICENSE RE-EXAM

First Name: KEY. Last Name: ______ Employee #: ______

Social Security Number

0	○	○	○	○	○	○	○	○	○
1	○	○	○	○	○	○	○	○	○
2	○	○	○	○	○	○	○	○	○
3	○	○	○	○	○	○	○	○	○
4	○	○	○	○	○	○	○	○	○
5	○	○	○	○	○	○	○	○	○
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Test Form

○ A ○ B ○ C ○ D

Test Date: 12/15/03

Test Series: N/A

Test Number: N/A

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

*ANSWER CHANGED FROM "B" TO "C" BASED ON RESOLUTION OF POST EXAM COMMENT

1/7/04

	A	B	C	D	E	
1.	○	●	○	○	○	B
2.	○	○	●	○	○	C
3.	○	○	●	○	○	C
4.	○	●	○	○	○	B
5.	○	☒	●	○	○	B *
6.	○	○	○	●	○	D
7.	●	○	○	○	○	A
8.	○	○	○	●	○	D
9.	○	○	○	●	○	D
10.	○	○	●	○	○	C
11.	●	○	○	○	○	A
12.	○	●	○	○	○	B
13.	○	○	○	●	○	D
14.	○	○	○	●	○	D
15.	○	●	○	○	○	B
16.	○	○	○	●	○	D
17.	○	○	●	○	○	C
18.	○	●	○	○	○	B
19.	●	○	○	○	○	A
20.	○	○	●	○	○	C
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22.	○	○	○	●	○	D
23.	●	○	○	○	○	A
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SCANTRON FORM NO. O-104268-PPL MA 4000 -154-5 4 3

SSES SRO NRC Re-Exam

1 Unit 1 is at 100% power with total core flow of 102 mlbm/hr.

A scoop tube lock occurred during a pressure transient on Recirc MG A lube oil system.

Shortly after the scoop tube lock, a Limiter #2 runback is initiated.

What will be the final plot of reactor power and core flow on the attached power to flow map and what actions if any will need to be initiated?

A. Position #2. Immediately insert control rods to exit Region II.

B. Position #1. Verify recirculation loop flow mismatch is less than or equal to 5 million lbm/hr for given core flow conditions.

C. Position #2. Immediately increase total core flow to exit Region II.

D. Position #1. Declare the recirculation loop with lower flow to be "not in operation" within 12 hours.

Question Data

B Position #1. Verify recirculation loop flow mismatch is less than or equal to 5 million lbm/hr for given core flow conditions.

Explanation/Justification:

A. Position #2 is not the plot of a single pump runback caused by Limiter #2. The position is indicative of a runback of two pumps with a Limiter #1 signal. The action is correct for a plot in region #2.

B. is correct. Position #1 is correct for a runback of one pump with the other pump scoop tube locked. Technical Specifications require the stated conditions be met to declare the system operable.

C. Position #2 is not the plot of a single pump runback caused by Limiter #2. The position is indicative of a runback of two pumps with a Limiter #1 signal. The action is correct for a plot in Region II.

D. Position #1 is correct for a runback of one pump with the other pump scoop tube locked. The recirculation loop with the lower flow is required to be not in operation within 2 hours not 12 hours.

Sys #	System	Category	KA Statement
295001	**Partial or Complete Loss of Forced Core Flow Circulation**	**Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:**	**Power/flow map**

K/A# **295001.AA2.01** K/A Importance **3.8** Exam Level **SRO**

References provided to Candidate **Tech Spec 3.4.1.** Technical References: **ON-164-002, Tech Spec 3.4.1.**

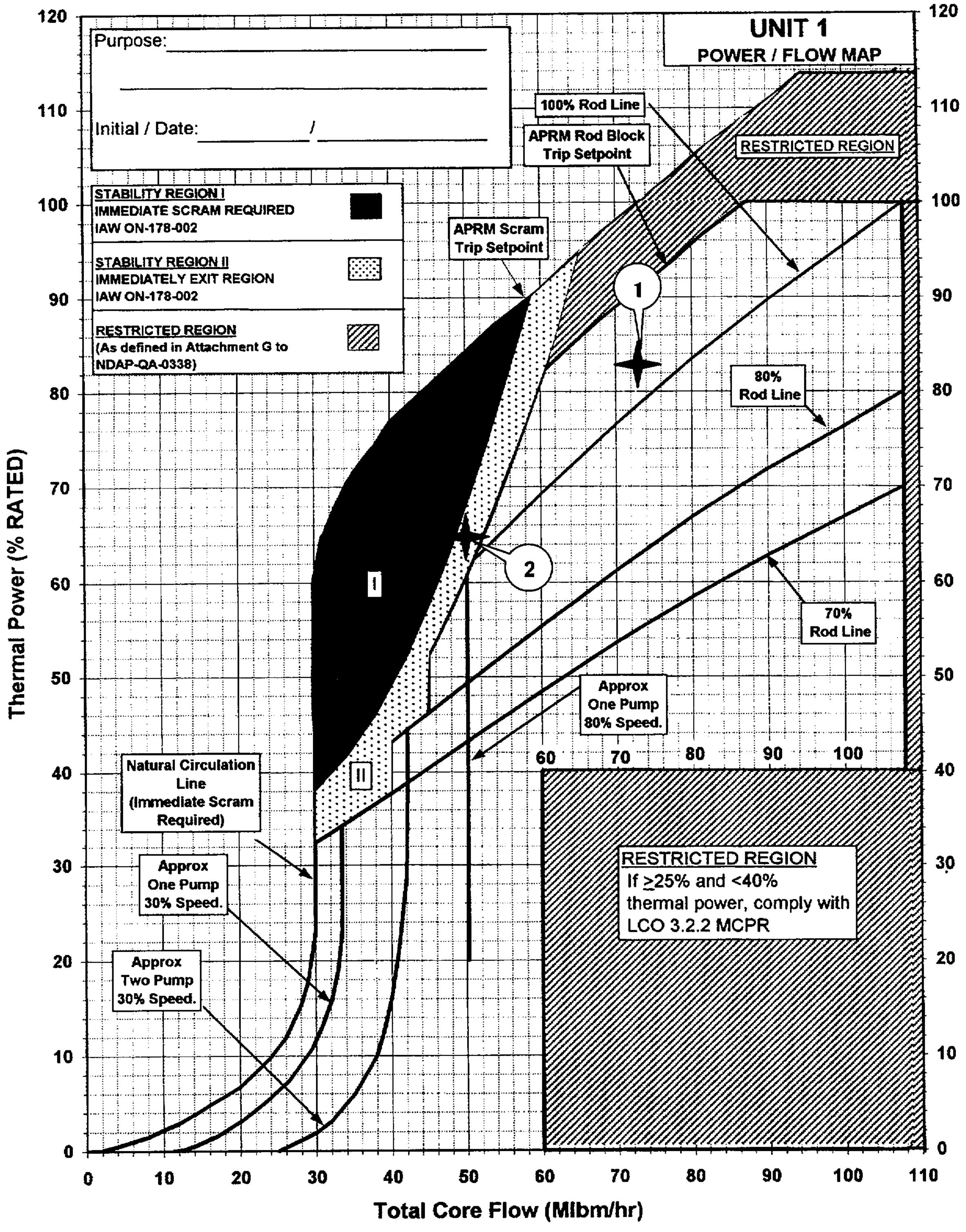
Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **1357** **Determine if the Plant responded correctly to an off-normal situation.**

Training Task: **64ON010** **Implement Loss Of Reactor Recirculation Flow**

SSES SRO NRC Re-Exam



NFE-B-NA-068 Rev. 12

SSES SRO NRC Re-Exam

2 Both Units were at 100% initial power when a station blackout occurs.

Which of the Emergency Diesel Generators should you select to substitute with E Emergency Diesel Generator and why?

A. Substitute for A or B Emergency Diesel Generators to eliminate the need for hooking up 'Blue Max'.

B. Substitute for D Emergency Diesel Generator to supply class 1E 250 VDC loads with their chargers.

C. Substitute for A or B Emergency Diesel Generators because 125 VDC control power availability is maximized.

D. Substitute for C Emergency Diesel Generator to make one loop of RHR suppression pool cooling operable.

Question Data

C Substitute for A or B Emergency Diesel Generators because 125 VDC control power availability is maximized.

Explanation/Justification:

A. the portable diesel generator is required for 125 VDC channel B.

B. all class 1E 250 VDC loads are not supplied from bus 1D.

C. is correct. Core Damage is four times less likely if A and B Emergency Diesel Generator s are aligned compared to C and D Emergency Diesel Generators.

D. an operable loop of RHR SPC is not available with bus 1C energized.

Sys #	System	Category	KA Statement
295003	**Partial or Complete Loss of A.C. Power**	**Emergency Procedures and Plan**	**Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.**

K/A# **295003.2.4.22** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **None** Technical References: **EO-100-030**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.43**

Training Objective: **2645** **Prioritize the order in which multiple Emergency Support procedures are to be performed. {SRO only}**

Training Task: **00EO024** **Implement Unit 1(2) Response To Station Blackout**

3 Unit 1 is in MODE 2.
During performance of SO-150-002, "Quarterly RCIC Flow Verification" annunciator, 250V DC PANEL 1L650 SYSTEM TROUBLE (AR-106-A11) is received.

The 1L650 reflash panel alarm 250 VDC System Low Voltage is present and battery terminal voltage is 218 VDC.

What actions are required?

A. Direct performance of SM-188-002, "250 VDC Station Batteries Quarterly Electrical Parameter Checks". Verify pilot cell parameters meet Table 3.8.6-1 Category C limits within 1 hour or declare 1D650 inoperable.

B. Direct performance of SM-188-002, "250 VDC Station Batteries Quarterly Electrical Parameter Checks". Enter the E-plan under an ALERT classification.

C. Declare 250 VDC battery 1D650 inoperable. Ensure MODE 1 is not entered until the battery is operable.

D. Declare 250 VDC battery 1D650 inoperable. Entry into MODE 1 is allowed with this battery inoperable.

Question Data

C Declare 250 VDC battery 1D650 inoperable. Ensure MODE 1 is not entered until the battery is operable.

Explanation/Justification:

A. SM-188-002 verifies battery cell parameters for category B limits. TS required actions verify category C limits are being met.
B. E-Plan entry is not required based on a single DC system being inoperable.
C. is correct. The AR addresses compliance with numerous TS, since SR 3.8.4.1 requires 250 VDC battery voltage equal to or greater than 258 VDC for battery operability.
D. entry into Mode 1 is not allowed since the action for an inop battery will require going to mode 3.

Sys #	System	Category	KA Statement
295004	**Partial or Complete Loss of D.C. Power**	**Emergency Procedures and Plan**	**Knowledge of annunciator response procedures.**

K/A# **295004.2.4.10** K/A Importance **3.1** Exam Level **SRO**

References provided to Candidate **Tech Specs, AR-106-A11** Technical References: **AR-106-A11, TS 3.8.4 & 3.8.6**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **1397** **Predict how each supported system will be affected by any of the following 250 Volt DC System failures.**

- **a. Blown Fuse**
- **b. Ground**
- **c. Battery Charger Trip (Effect on Tech Spec LCO)**
- **d. Loss of 250 VDC System**

Training Task: **88ON003** **Implement Loss Of 250V DC Bus**

SSES SRO NRC Re-Exam

4

Unit 1 was at 100% power when grid instabilities result in activation of EHC power load unbalance protection circuitry. Subsequent to the reactor scram the unit is stabilized with the following conditions:

- Reactor water level + 5 inches
- Reactor pressure ~955 psig controlled with bypass valves
- RPV bottom head drain temperature is 380 deg F

Which of the following subsequent operator actions for reactor water level control should you direct based on these conditions?

A. Maintain level -30 to +5 inches.

B. Restore level +13 to +30 inches.

C. Restore and maintain level +45 to +54 inches.

D. Restore level +13 to +54 inches.

Question Data

B Restore level +13 to +30 inches.

Explanation/Justification:

A. no bases for level band, -30 inches is initiation setpoint for RCIC

B. correct level band with stratification and no recirc pump in service. Must determine the delta T between bottom head temp and saturation temp using steam tables, recognize delta T is greater than 145 deg F and from memory know the proper level band for control of +13 to +30 inches.

C. level band when delta T is equal to or less than 145 deg F

D. level band requires at least one recirc pump in service

Sys #	System	Category	KA Statement
295006	**SCRAM**	**Conduct of Operations**	**Ability to direct personnel activities inside the control room.**

K/A# **295006.2.1.9** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **Steam Tables** Technical References: **ON-100-101**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **1364** **Explain the reasons for steps contained in an off-normal procedure.**

Training Task: **00ON018** **Implement Scram**

5 A control room evacuation was required. A transfer switch malfunction caused loss of RHR pump control at Unit 1 Remote Shutdown Panel 1C201.

As Unit Supervisor what direction will be given to start an RHR pump when using the MANUAL method for suppression pool cooling?

A. At 1A20102 RHR PUMP 1A breaker, transfer DC Trip and Control power to alternate, pull the lateral control switch to HANDLE OUT position and place the lateral control switch to CLOSE for RHR pump 1A.

B. At 1A20102 RHR PUMP 1A breaker, pull the lateral control switch to HANDLE OUT position and place the lateral control switch to CLOSE for RHR pump 1A.

C. At 1A20402 RHR PUMP 1D breaker, pull the lateral control switch to HANDLE OUT position and place the lateral control switch to CLOSE for RHR pump 1D.

D. At 1A20202 RHR PUMP 1B breaker, OPEN the DC Trip and Control knife switch and place the lateral control switch to CLOSE for RHR pump 1B.

Question Data

C ~~B~~ M At 1A20102 RHR PUMP 1A breaker, pull the lateral control switch to HANDLE OUT position and place the lateral control switch to CLOSE for RHR pump 1A.

Changed Answer to "C" See Post Exam Comment Resolution

Explanation/Justification:

A. DC trip and control power is not transferred to alternate for this evolution.

B. is correct. The B loop of RHR is controlled from the Remote Shutdown Panel, the manual backup method uses the RHR loop A. In addition to manual valve alignment required to use RHR loop A, pump operation is accomplished by pulling the lateral control switch to the out position thereby transferring control locally at the breaker. Placing the control switch to CLOSE will start the RHR pump A providing 125 VDC control power is available. SRO responsible to implement an off normal situation

C. 1D RHR pump is not the manual method for Unit 1.

D. DC trip and control power is not opened for this evolution.

Sys #	System	Category	KA Statement
295016	**Control Room Abandonment**	**Conduct of Operations**	**Ability to locate and operate components, including local controls.**

K/A# **295016.2.1.30** K/A Importance **3.4** Exam Level **SRO**

References provided to Candidate **None** Technical References: **OP-149-005**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.43**

Training Objective: **1489** **List the RPV Instrumentation functions and components that can be operated from the Remote Shutdown Panel.**

Training Task: **00ON025** **Implement Plant Shutdown From Outside Control Room**

6 Unit 1 is at 100% power.

A RWCU pump trip and system isolation was preceded by the following two alarms:
- AR-101-B01, RWCU FILTER INLET HI TEMP
- AR-101-A01, RWCU FILTER INLET HI TEMP ISO

Which of the following events is responsible for the RWCU response and what administrative action is required?

A. Insufficient RWCU blowdown flow to main condenser or Liquid Radwaste. Notify chemistry to align reactor coolant sampling to Reactor Recirc Loop B.

B. High Reactor Building Chilled Water temperature caused isolation of RBCCW to RWCU non-regenerative heat exchanger. Perform an eight hour ENS notification due to an unplanned actuation of systems that mitigate the consequences of significant events.

C. TCV 11028 SW Temperature Control valve for RBCCW failed closed. Obtain a grab sample conductivity measurement every 24 hours.

D. Low flow in Reactor Building Chilled Water caused isolation of RBCCW to RWCU non-regenerative heat exchanger. Obtain an in-line conductivity measurement once per 4 hours.

Question Data

D Low flow in Reactor Building Chilled Water caused isolation of RBCCW to RWCU non-regenerative heat exchanger. Obtain an in-line conductivity measurement once per 4 hours.

Explanation/Justification:

A. excessive blowdown flow can lead to high temperature isolation, not insufficient flow. Continuous conductivity monitoring is continued by aligning sampling to reactor recirc loop B.

B. high RBCW temperature will not cause isolation of RBCCW to RWCU non-regenerative heat exchanger. High temperature isolation of RWCU is not reportable IAW NDAP-QA-720.

C. loss of service water cooling would lead to high temperature isolation of RWCU. If continuous conductivity recording is not available from RWCU grab sample conductivity measurement is required one per 4 hours per TRO 3.4.1

D. is correct. A low RBCW flow signal with a 13 second time delay will cause isolation of RBCCW to RWCU non-regenerative heat exchanger. If continuous conductivity recording is not available, in-line conductivity measurement is required one per 4 hours per TRO 3.4.1.
Continuous conductivity monitoring is continued by aligning sampling to reactor recirc loop B.

Sys #	System	Category	KA Statement
295018	**Partial or Complete Loss of Component Cooling Water**	**Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:**	**Cause for partial or complete loss**

K/A# **295018.AA2.03** K/A Importance **3.5** Exam Level **SRO**

References provided to Candidate **TRO 3.4.1** Technical References: **ON-134-001**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **1358** **Determine a course of action to mitigate or correct an off-normal situation.**

Training Task: **34ON005** **Implement Loss Of Reactor Building Chilled Water**

7 Unit 1 was at 100% power when a reactor scram occurred with the following conditions:

- No control rod movement.
- Equipment failures require boron injection with RCIC.
- The RCIC suction hose connection uncouples and drains ~600 gallon to the RCIC room floor before it is isolated.
- Initial SBLC tank level was 4900 gallons.
- Suction line repairs have allowed boron injection with RCIC.

For these conditions what is the maximum SBLC tank level before you direct RPV water level restored and maintained +13 to +54 inches?

A. 2500 gallons

B. 2800 gallons

C. 1800 gallons

D. 200 gallons

Question Data

A 2500 gallons

Explanation/Justification:

A. correct based upon injection started at 4300 gallons minus 1800 gallons injection required for hot shutdown boron weight.

B. value directed from EOP assuming Tech Spec minimum SBLC tank volume or 4587 gallons.

C. no bases for this tank volume, this value is the amount required to be injected as listed in the procedure.

D. tank volume required to trip SBLC pumps, per EOP direction. Also, at this volume cold shutdown boron weight should have been injected.

Sys #	System	Category	KA Statement
295037	**SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown**	**Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:**	**SBLC tank level**

K/A# **295037.EA2.03** K/A Importance **4.4** Exam Level **SRO**

References provided to Candidate **None** Technical References: **EO-000-113 step LQ/L**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.43**

Training Objective: **1215** **Define and/or discuss the operational implications of the following terms for the Standby Liquid Control System:**

a. Hot Shutdown boron weight

b. Cold Shutdown boron weight

Training Task: **00EO031** **Implement Level/Power Control**

8 Both Units are in MODE 1 at 100% power.

- E Emergency Diesel Generator is not aligned for Standby Automatic Operation.
- The fire protection system auto initiated and suppressed a fire in C Emergency Diesel panel 0C519C.
- Emergency Diesel Generator C was transferred to Local while a damage assessment is completed.
- B SGTS is out of service for replacement of 'B' SBGT FAN INLET DAMPER HD-07553B.

Given the following time line:
10/5/03 0500 B SGTS is inoperable.
10/6/03 0920 Emergency Diesel Generator C transferred to Local.

The maximum time permitted for both units to enter MODE 3 is:

A. 0220 on 10/7/03

B. 2120 on 10/12/03

C. 2120 on 10/9/03

D. 0520 on 10/7/03

Question Data

D 0520 on 10/7/03

Explanation/Justification:

A. Time permitted to enter MODE 3 if LCO 3.03 was entered after SBGTS A was declared inoperable in accordance with TS 3.8.1.B.2.

B. time permitted to enter MODE 3 for failure of D/G C only per TS 3.8.1. Action B.4 requires D/G restoration in 6 days from discovery of failure to meet LCO (0920 10/12/03); enter TS 3.8.1 Condition E, be in MODE 3 in 12 hours (2120 10/12/03).

C. time to enter MODE 3 for failure of D/G C only per TS 3.8.1. Action B.4 requires D/G restoration in 72 hours (0920 10/9/03); enter TS 3.8.1 Condition E, be in MODE 3 in 12 hours (2120 10/9/03).

D. is correct D/G C being inoperable per TS 3.8.1 Condition B at 0920 on 10/6/03; TS 3.8.1 required action B.2 declares A SBGTS inoperable after 4 hours since B SBGTS has been inoperable (1320 on 10/6/03); enter TS 3.6.4.3 Condition D for 2 SBGTS subsytems inoperable with completion time of 4 hours (1320 on 10/6/03); Enter TS 3.6.4.3 Condition E after 4 hours (1720 10/6/03); TS 3.6.4.3 required action be in MODE 3 in 12 hours (0520 on 10/7/03)

Sys #	System	Category	KA Statement
600000	**Plant Fire On Site**	**Conduct of Operations**	**Knowledge of conditions and limitations in the facility license.**

K/A# **600000.2.1.10** K/A Importance **3.9** Exam Level **SRO**

References provided to Candidate **Tech Spec 3.8.1, 3.6.4.3 & SO-024-013** Technical References: **TS 3.8.1, TS 3.6.4, 3.6.4.3, SO-024-013**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **2263** **Given a set of Unit 1(2) Technical Specifications and a set of plant conditions, determine if a Diesel Generator is required to be operable per Technical Specifications.**

Training Task: **00TS001** **Ensure Plant Operates In Accordance With The Operating License, Technical Specifications (TS), and Technical Requirements Manual (TRM)**

9 Unit 1 has scrammed from 100% power when MSIVs closed and the following conditions exist:

- EO-100-102 has been entered.
- HPCI and RCIC injected for level control
- HPCI now in service CST to CST for pressure control.
- RCIC shutdown
- Suppression Pool Cooling has not been placed in service.
- Suppression Pool bulk water temperature, point MAT37 on the PICSY format, CONTAINMENT ATMOSPHERIC CONTROL, indicates Red at 90 deg F.
- Alarms on 1C601, SUPP POOL DIV 1 AVERAGE TEMP HI (AR-111-F04) and SUPP POOL DIV 2 AVERAGE TEMP HI (AR-112-F04) have been received.
- SPOTMOS Div I & II are in alarm indicating 101 deg F and 103 deg F respectively.

Assess these Suppression Pool water temperature indications and determine what actions are required.

A. The SRV used for pressure control should have caused the MAT 37 point to indicate the same as SPOTMOS, contact I&C to investigate failed temperature input. Direct RHR Suppression Pool Cooling to be placed in service.

B. The MAT 37 point on the PICSY format has failed, contact I&C to investigate failed temperature input. Direct RHR Suppression Pool Cooling to be placed in service.

C. HPCI exhaust steam has heated the bulk of the Suppression Pool. Direct 'A' Loop of RHR Suppression Pool Cooling to be placed in service.

D. HPCI exhaust steam is heating a local area of the Suppression Pool. Direct 'B' Loop of RHR Suppression Pool Cooling to be placed in service.

Question Data

D HPCI exhaust steam is heating a local area of the Suppression Pool. Direct 'B' Loop of RHR Suppression Pool Cooling to be placed in service.

Explanation/Justification:

A. Use of one SRV for pressure control could cause local high temperatures. The temperature elements used for MAT 37 do not indict in the local area of the HPCI exhaust into the Suppression Pool. MAT 37 uses 6 temp elements at the pool surface plus 4 temp elements located at the bottom of the pool. The 4 temps at the bottom of the pool will be much cooler than the 6 surface elements used for SPOTMOS. Since MAT 37 uses the 4 lower temp elements with the 6 upper elements and SPOTMOS uses only the 6 upper elements it is expected that SPOTMOS will indicate higher than the mat 37 POINT. With no RHR Suppression Pool Cooling in service there is no mixing of the bulk sup pool water allowing a local area and the surface of the pool water to indicate high temperature which will be seen by the averaging circuit.

B. The temperature elements used for MAT 37 do not indicate in the local area of the HPCI exhaust into the Suppression Pool. MAT 37 uses 6 temp elements at the pool surface plus 4 temp elements located at the bottom of the pool. The 4 temps at the bottom of the pool will be much cooler than the 6 surface elements used for SPOTMOS. Since MAT 37 uses the 4 lower temp elements with the 6 upper elements and SPOTMOS uses only the 6 upper elements it is expected that SPOTMOS will indicate higher than the mat 37 POINT. With no RHR Suppression Pool Cooling in service there is no mixing of the bulk sup pool water allowing a local area and the surface of the pool water to indicate high temperature which will be seen by the averaging circuit.

C. indications provided are a result of local heating of the suppression pool not an over all heatup of the water.

D. Correct answer, With HPCI in service and no Suppression Pool mixing, a local area of the Suppression Pool will heat up and the procedure directs that 'B' loop of RHR be placed in Suppression Pool Cooling. 'B' Loop is the preferred loop due to the location of the RHR suction and discharge in relation to the HPCI exhaust. SRO responsible to determine if preferred loop is the appropriate loop to place in service based on overall conditions.

Sys #	System	Category	KA Statement

SSES SRO NRC Re-Exam

295013	**High Suppression Pool Temperature**	**Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE:**	**Localized heating/stratification**

K/A# **295013.AA2.02** K/A Importance **3.5** Exam Level **SRO**

References provided to Candidate **None** Technical References: **AR-112-001**

Question Source: **New** **Susquehanna, 10/2/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **337** **Determine if SPOTMOS readings are appropriate for stated Suppression Pool level.**

Training Task: **59ON006** **Implement Containment Isolation**

10 A large primary system line break has occurred with blowout panel actuation and the following conditions are indicated:

- Noble gas activity has been detected by OSCAR at the site boundary.
- Area Radiation Monitors in Unit 1 Reactor Building unchanged.
- Area Radiation Monitors on 818' Elevation for Unit 1 and Unit 2 unchanged.
- Unit 1 Reactor Building SPING Noble Gas channel unchanged.
- Area Radiation Monitors in Unit 1 Turbine Building trending upward.
- Unit 1 Turbine Building SPING Noble Gas channel trending upward.
- Security reports vapor plumes above the Unit 1 CST, and from the upper west side of Unit 1 Reactor Building.

The TSC Dose calculator calls the Control Room Emergency Director to find out the source of the release. What response should the Control Room Emergency Director provide to the Dose Calculator?

The source of the release is:

A. Reactor Water Cleanup primary coolant line break in the RWCU room.

B. HPCI Room Steam Line break.

C. Main Steam Line Break in the Reactor Building Steam Tunnel

D. Main Steam Line Break in the Turbine Building at the Reactor Feed Pumps

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Question Data

C Main Steam Line Break in the Reactor Building Steam Tunnel

Explanation/Justification:

A. For a break in the RWCU room there would be no release into the Reactor Building because the Reactor Water Cleanup Room has BDIDs isolating the building ventilation. A major line break in the Reactor Water Cleanup room will cause the rupture disk for the room to actuate releasing energy/radioactive material to the environment in the Unit 1 CST area.

B. HPCI Room does not have Back Draft Isolation Dampers (BDIDs) and would cause radiation levels in the Reactor Building to increase and only one vapor plume from the CST berm area

C. Correct answer, steam line break in the Reactor Building Steam Tunnel will not be seen on any radiation monitors in the Reactor Building due to back draft isolation dampers going closed. A major steam leak in the RB Steam Tunnel will lift the blow out panels in the steam tunnel to the atmosphere and to the Turbine Building which provides a vapor plume in two places.

D. There will be no release into the Reactor Building and no vapor plume into the CST berm area.

Sys #	System	Category	KA Statement
295017	**High Off-Site Release Rate**	**Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE:**	**Source of off-site release**

K/A# **295017.AA2.04** K/A Importance **4.3** Exam Level **SRO**

References provided to Candidate **None** Technical References: **FSAR APPENDIX 3.6A**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **EP010-3** **List common pathways for radioactive source terms to exit the plant and the impact of the offsite exposure based on the pathway.**

Training Task: **00EO028** **Implement Secondary Containment Control**

11 Which combination of Reactor Building Area temperatures would indicate equipment operability has degraded to the point where procedure EO-100-104, "Secondary Containment Control" would require a reactor shutdown?

A. RWCU Pump Room 165 deg F AND RCIC Equipment Area 175 deg F.

B. HPCI Equipment Area 175 deg F AND HPCI Emerg Area Cooler 173 deg F.

C. CS Pump Room B 148 deg F AND RHR Equipment Area 1 128 deg F.

D. CS Pump Room A 116 deg F AND CS Pump Room B 120 deg F.

Question Data

A RWCU Pump Room 165 deg F AND RCIC Equipment Area 175 deg F.

Explanation/Justification:

A. correct, Two areas are addressed and both are above max safe values.

B. Both temperatures are greater than max safe values but only affect one area.

C. Two areas are addressed. CS PUMP ROOM B is greater than max safe area for one area, but RHR is below max safe value.

D. Two areas are addressed but neither room is above max safe value.

Sys #	System	Category	KA Statement
295032	**High Secondary Containment Area Temperature**	**Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE:**	**Equipment operability**

K/A# **295032.EA2.02** K/A Importance **3.5** Exam Level **SRO**

References provided to Candidate **EO-100-104** Technical References: **EO-100-104**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **2597** **Explain the basis for each caution and note in EO-100-100.**

Training Task: **00EO028** **Implement Secondary Containment Control**

12 Unit 1 is operating normally at 100 % power, when a leak into the 'B' Core Spray Pump Room occurs. The leak is isolated and water level stops rising when it reaches "Max Safe Level" for the room.

In addition to the Core Spray equipment, the 'B' Core Spray Room contains the HPCI SYSTEM INSTR RACK, 1C014, which has one division of HPCI Turbine Trip pressure switches located on it.

Which of the following Technical Specification actions if any, are required as a result of reaching "Max Safe Level" water level for the room?

A. Immediately verify by administrative means that RCIC is operable and restore HPCI to OPERABLE status within 14 days and restore low pressure ECCS injection/spray subsystem to OPERABLE status within 7 days.

B. Place the affected HPCI instrument channels to trip within 24 hours then immediately verify by administrative means that RCIC is operable and restore HPCI to operable status within 14 days.

C. 'B' Core Spray and HPCI are both operable. No Technical Specification actions are required.

D. Restore low pressure ECCS injection/spray subsystem to OPERABLE status and be in MODE 3 within 12 hours.

Question Data

B Place the affected HPCI instrument channels to trip within 24 hours then immediately verify by administrative means that RCIC is operable and restore HPCI to operable status within 14 days.

Explanation/Justification:

A. Distracter fails to recognize the that the Max Safe Level for the 'B' Core Spray room is the HPCI high pressure exhaust pressure switches. The max safe level does not impact any equipment associated with Core Spray.

B. Correct answer. To correctly answer, the definition of max safe water level must be understood. Definition: Max Safe operating water levels are ECCS room water levels at or above the elevation that would submerge equipment necessary to assure safe shutdown of the plant. The 'B' Core Spray room contains the HPCI high pressure exhaust switches which are noted in the EOP basis. Given the definition of max safe water level, it is assumed that the pressure switches would be inop. Inop pressure switches require that the channel be placed in trip in 24 hours which would inop HPCI.

C. 'B' Core Spray is operable and not affected by the water level at Max Safe, HPCI is inoperable due to instrumentation on the instrument rack being under water at the Max Safe level.

D. The Core Spray equipment is not affected with the water level in the room at the Max Safe Level.

Sys #	System	Category	KA Statement
295036	**Secondary Containment High Sump/Area Water Level**	**Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL:**	**Operability of components within the affected area**

K/A# **295036.EA2.01** K/A Importance **3.2** Exam Level **SRO**

References provided to Candidate **Tech Spec** Technical References: **TS 3.3.6.1, 3.5.1**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **2598** **For each Symptom Based EOP: Explain the basis for each step.**

Training Task: **00EO028** **Implement Secondary Containment Control**

13 Unit 1 at ~7 % power, Reactor pressure 955 psig, 2 Bypass Valves open, starting up from refueling. Plant startup on hold to perform HPCI Post Maintenance Testing, and quarterly flow verification surveillance before transferring the mode switch to RUN.

- At 0100 hours the Reactor pressure and flow conditions necessary to perform the test were met.
- At 0130 hours SO-152-002, "Quarterly HPCI Flow Verification" test was begun.
- At 0145 hours with HPCI turbine at 2500 rpm, valve HPCI L-O CLG WTR HV-156-F059 unexpectedly closes, and cannot be re-opened.

If the HPCI turbine continues to run, what alarms will actuate?

What, if any, actions will be required?

A. Only HPCI TURBINE OIL COOLER DSCH HI TEMP (AR-114-D03) alarm will actuate.

Monitor lube oil discharge temperature, HPCI remains OPERABLE.

B. HPCI TURBINE OIL COOLER DSCH HI TEMP (AR-114-D03) and HPCI BARO CDSR VACUUM TANK HI PRESSURE (AR-114-G01) alarms will actuate.

After 1300 hours, declare HPCI INOPERABLE.

C. Only the HPCI BARO CDSR VACUUM TANK HI PRESSURE (AR-114-G01) alarm will actuate.

Monitor Barometric Condenser vacuum, prior to reaching atmospheric pressure, shutdown HPCI.

D. HPCI TURBINE OIL COOLER DSCH HI TEMP (AR-114-D03) and HPCI BARO CDSR VACUUM TANK HI PRESSURE (AR-114-G01) alarms will actuate.

Immediately declare HPCI INOPERABLE.

Question Data

D HPCI TURBINE OIL COOLER DSCH HI TEMP (AR-114-D03) and HPCI BARO CDSR VACUUM TANK HI PRESSURE (AR-114-G01) alarms will actuate.

Immediately declare HPCI INOPERABLE.

Explanation/Justification:

A. The HPCI BARO CDSR VACUUM TANK HI PRESSURE (AR-114-G01) alarm will also actuate. Cooling water for the RCIC turbine is a parallel flowpath, candidate may confuse HPCI and RCIC cooling water flowpaths. IF candidate does not recognize that this is a necessary support system for HPCI to perform its function, then the candidate would select HPCI remains OPERABLE. No Technical Specification actions required.

B. The alarms listed are correct, however the Technical Specification 12 hour allowance to complete the surveillance does not allow for waiting until the original 12 hour period expires before declaring the system INOPERABLE. Since the cooling subsystem is a necessary support system for HPCI to perform its function, HPCI should be declared INOPERABLE and Technical Specification Action 3.5.1 for Condition D is applied.

C. The HPCI TURBINE OIL COOLER DSCH HI TEMP (AR-114-D03) alarm will also actuate. IF candidate does not recognize that this is a necessary support system for HPCI to perform its function, then the candidate would select HPCI remains OPERABLE. No Technical Specification actions required.

D. Correct Answer, With F059 closed cooling water to the lube oil cooler and the barometric condenser is isolated. Lube oil will heat up and the barometric condenser pressure will increase causing both alarms listed. The cooling subsystem is a necessary support system for HPCI to perform its function, HPCI should be declared INOPERABLE and Technical Specification action 3.5.1 for Condition D is applied.

Sys #	System	Category	KA Statement
206000	**High Pressure Coolant Injection System**	**Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:**	**Valve closures: BWR-2, 3, 4**

K/A# **206000.A2.02** K/A Importance **3.5** Exam Level **SRO**

References provided to Candidate **Tech Specs 3.5** Technical References: **AR-114-D03 & G01, TS 3.5.1, (178)**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **2030** **Describe the flowpaths for any mode of operation of the High Pressure Coolant Injection System, including the following components in the description as appropriate.**

a. Main Steam Line **b. Containment Isol**

Training Task: **52EO008** **Implement HPCI Turbine Isolation, Trip And Initiation Bypass**

SSES SRO NRC Re-Exam

14 Which of the following proposed changes will require a complete 10 CFR 50.59 "EVALUATION" prior to implementing the change?

A. Moving the TSC emergency response facility from the Control Structure to the West Building located outside the fence.

B. Permanently raising the S&A Building channel ARM Hi Alarm setpoint.

C. Moving the Security perimeter fence to include the entire 500kV yard as part of the onsite facilities.

D. Permanently removing the motor operator and check valve internals for the FW INLET LINE A & B STOP CKV (HV-141F032A&B)

Question Data

D Permanently removing the motor operator and check valve internals for the FW INLET LINE A & B STOP CKV (HV-141F032A&B)

Explanation/Justification:

A. Emergency Plan facilities are regulated by 10 CFR 50.47.
B. ARM setpoints have no automatic function and apply to 10 CFR 20 not to any SCC.
C. Security systems and designs are regulated by 10 CFR 73.
D. Correct answer, these check valves are on the main feedwater header and are part of the containment boundary. As part of the containment boundary a Technical Specification change would be required since these valves are part of containment design.

Sys #	System	Category	KA Statement
259002	**Reactor Water Level Control System**	**Equipment Control**	**Knowledge of the process for determining if the proposed change, test or experiment increases the probability of occurrence or consequences of an accident during the change, test or experiment.**

K/A# **259002.2.2.9** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **None** Technical References: **NDAP-QA-0726**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **3925** **Be able to DEFINE:**

a. Commitment Document
b. Expedited Review Revision
c. Intent Change
d. Interim Approval
e. Quality Assurance Document Review (QADR)
f. Safety-Related
g. Technical Review
h. 50.59 Evaluation

Training Task: **00AD028** **Implement Nuclear Department Procedure Program**

15 A loss of coolant is in progress on Unit 1 with the following plant conditions:

- No offsite power, all Emergency Diesel Generators running and loaded.
- EO-100-114 "RPV Flooding" implemented to step RF-13.
- Division I & II Core Spray and RHR LPCI at rated flow with Reactor Pressure at 95 psig.

Predict the response of the 'A' loop of LPCI if the 'C' Emergency Diesel Generator trips and describe directions provided to control the situation.

A. 'C' RHR Pump will coast down, 'A' RHR pump trips on over current. Place RHRSW X-Tie in service to regain RPV level, monitor RPV to Suppression Chamber pressure differential and reset the time RPV Flooding conditions were met, as required.

B. 'C' RHR Pump will trip, 'A' RHR pump not in runout condition. Monitor RPV to Suppression Chamber pressure differential and reset the time RPV Flooding conditions were met, as required.

C. 'C' RHR Pump will trip, 'A' RHR pump in runout condition. Throttle LPCI injection flow, Monitor RPV to Suppression Chamber pressure differential and reset the time RPV Flooding conditions were met, as required.

D. 'C' RHR Pump will coast down, 'A' RHR pump trips on over current. Monitor 'B' loop LPCI not in runout and contact TSC to enter EP-DS-003 "RPV Level Determination".

Question Data

B 'C' RHR Pump will trip, 'A' RHR pump not in runout condition. Monitor RPV to Suppression Chamber pressure differential and reset the time RPV Flooding conditions were met, as required.

Explanation/Justification:

A. 'C' pump will trip and not coast down, the 'A' pump will not trip on over current due to the orifice in the discharge line.

B. correct answer, 'C' RHR pump will trip on loss of voltage due to D/G tripping. Flow limited to 13,500 gpm due to orifice in discharge of each RHR pump to prevent pump trip on over current. Loss of water source will require monitoring level and flooded criteria. If delta p between RPV and Suppression Chamber drop less than 81 psid the flooded time will have to be reset.

C. 'A' RHR pump will not trip on over current for given conditions due to the flow orifice in the pump discharge.

D. 'C' pump will trip and not coast down, the 'A' pump will not trip on over current due to the orifice in the discharge line. The 'B' loop will not be in a runout condition due to flow orifices in the pump discharge lines even though the total head the system is pumping against is reduced.

Sys #	System	Category	KA Statement
203000	**RHR/LPCI: Injection Mode (Plant Specific)**	**Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:**	**Emergency generator failure**

K/A# **203000.A2.06** K/A Importance **3.9** Exam Level **SRO**

References provided to Candidate **None** Technical References: **TM-OP-049**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Training Objective: **2680** **Determine the correct course of action when given plant conditions.**

Training Task: **00EO032** **Implement RPV Flooding**

SSES SRO NRC Re-Exam

16 Unit 1 is operating at 50% power.

- The static inverter for Vital UPS 1D666 is tagged out for maintenance.
- Vital Distribution Panel 1Y629 is being powered by the alternate power supply MCC 1B246 through the manual bypass switch.

MCC 1B246 voltage begins to drop. When voltage drops to zero volts and alarm VITAL AC UPS PANEL 1L666 TROUBLE/ABNORMAL (AR-106-E11) is received:

How will Vital UPS 1D666 respond to this zero voltage condition on MCC 1B246, and what operator actions will you direct, in response to these conditions?

Vital UPS 1D666 will:

A. Automatically swap to the preferred source. Direct the PCOM to perform Scram imminent actions, Scram the reactor and trip all feedpumps IF RPV water level approaches either the low or the high alarm points.

B. Automatically swap to the preferred source. Direct the PCOM to reset the runbacks and the scoop tube positioners on the A and B MG sets.

C. NOT automatically swap to the preferred source. Direct the NPO to place the static switch to "Alternate load".

D. NOT automatically swap to the preferred source. Direct the PCOM to perform Scram imminent actions, Scram the reactor and trip all feed pumps.

Question Data

D NOT automatically swap to the preferred source. Direct the PCOM to perform Scram imminent actions, Scram the reactor and trip all feed pumps.

Explanation/Justification:

A. Is incorrect. Static switch will only automatically transfer if bypass switch is in "Normal Mode" to begin the transient. The actions are addressed as necessary in ON-117-001.

B. Is incorrect. Static switch will only automatically transfer if bypass switch is in "Normal Mode". If candidate believes MCC 1B246 provides the preferred power to 1Y128 distribution panel in addition to 1D666, then these actions would be necessary IAW ON-117-001

C. Is incorrect. The static switch will not automatically transfer, however placing the static switch to Alternate load position will not restore the bus since the power feed to the distribution panel is the same power source that has just degraded to zero volts.

D. Correct answer. To arrive at this answer the candidate must know from memory that this Vital UPS has only one AC source and not 2 like most of the other Vital UPS, and must know from memory that the static switch will not Automatically transfer when it is in the Manual bypass position. Candidate must then conclude that the distribution panel would be de-energized, and follow the AR procedure. The AR then references the ON and the ON must be followed correctly to apply the appropriate directed actions. 50% power was chosen as a starting point to avoid an immediate automatic scram on low level when the feed pump recirc valves go open on the loss of power to the panel.

Sys #	System	Category	KA Statement
262002	**Uninterruptable Power Supply (A.C./D.C.)**	**Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:**	**Under voltage**

SSES SRO NRC Re-Exam

K/A# **262002.A2.01** K/A Importance **2.8** Exam Level **SRO**

References provided to Candidate **AR-106-E11, and ON-117-001** Technical References: **AR-106-E11,** and ON-117-001

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: 55.43

Training Objective: **1467** **Predict how the failure of the following support systems may impact the Instrument AC System.**

a. Loss of preferred source to the Non-Class 1E UPS Panel

b. Loss of the alternate source to the Non-Class 1E UPS Panel

Training Task: **17ON003** **Implement Loss Of Instrument Bus**

17 Unit 1 is at 100% power.

Annunciator alarm AR-107-D04 ARI DIV 1 INOP/BYPASS was received. Electrical Maintenance investigated cause of the alarm and reports a loss of power to the ARI DIV I logic.

What is the impact of this failure and what actions are required?

A. Div 2 ATWS-ARI remains operable with rod scram times extended to 25 seconds. Restore Div 1 ATWS-ARI to operable within 14 days and evaluate for potential violation of 10 CFR 50.62.

B. CRD backup scram protection for Div 1 is unavailable. Restore backup scram protection within 1 hour.

C. Manual and automatic actuation of ARI are inoperable. Restore ATWS-ARI trip capability within 14 days.

D. ATWS-ARI trip input signals to Division 1 RPS logic are inoperable. Place channel in trip condition within 12 hours.

Question Data

C Manual and automatic actuation of ARI are inoperable. Restore ATWS-ARI trip capability within 14 days.

Explanation/Justification:

A. Div 2 ATWS-ARI remains operable, however, both divisions must energize to cause scram air header isolation and venting. Rods scram times are not extended to 25 seconds in this condition.

B. A separate 125 VDC source provides power to Div 1 backup scram valves, therefore, the function remains operable.

C. Is correct. Both divisions must energize to cause scram air header isolation and venting. Since a power loss is involved neither manual nor automatic actuation is operable. TRO 3.1.1 requires trip capability restored within 14 days.

D. ATWS-ARI does not provide trip signals to RPS, it is an independent function to RPS.

Sys #	System	Category	KA Statement
201001	**Control Rod Drive Hydraulic System**	**Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:**	**Power supply failures**

K/A# **201001.A2.03** K/A Importance **3.1** Exam Level **SRO**

References provided to Candidate **TRO 3.1.1** Technical References: **TRO 3.1.1**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 3

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: 55.43

Training Objective: **2328** **Given various instrumentation and computer indications, determine if RPS and supported system(s) response(s) are correct for each of the following conditions:**

a. Normal operation

b. Loss of offsite power

c. Loss of RPS bus power to one RPS Division

Training Task: **58ON004** **Implement Loss Of RPS**

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

- All rods fully inserted.
- Drywell pressure is 10 psig and slowly increasing.
- HPCI and RCIC are controlling RWL +13 to +54 inches.
- Reactor pressure is 940 psig and slowly lowering.
- SPOTMOS temperature is 89 deg F and slowly increasing.
- Suppression Chamber pressure is 5 psig and slowly increasing.

Five minutes after Suppression Chamber Sprays were initiated on RHR loop A, the following containment data was reported:

- Drywell pressure is 11 psig and slowly increasing.
- SPOTMOS temperature is 91 deg F and slowly increasing.
- Suppression Chamber pressure is 6 psig and slowly increasing.
- Suppression Chamber vapor space temperature is 91 deg F.

Explain the Suppression Pool response and the proper containment pressure control action you will direct?

A. The Suppression Chamber vapor space 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 vapor space 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 vapor space 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.

Question Data

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

Explanation/Justification:

A. 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 directed since a single spray header exists by design with either RHR loop supplying that header.

B. is correct. Vapor space pressure is caused by accumulation of nitrogen, use of sprays in the vapor space will have little affect on pressure. Drywell spray is not permitted until Suppression Chamber pressure exceeds 13 psig.

C. 91 deg F water temperature is not too high to reduce vapor space pressure. Using sprays from RHRSW is not warranted in this condition.

D. If vacuum breaker valves were leaking the d/p between the Drywell and Suppression Chamber vapor space would be less than 5 psig.

SSES SRO NRC Re-Exam

Sys #	System	Category	KA Statement
230000	**RHR/LPCI: Torus/Suppression Pool Spray Mode**	**Conduct of Operations**	**Ability to execute procedure steps.**

K/A# **230000.2.1.20** K/A Importance **4.2** Exam Level **SRO**

References provided to Candidate **None** Technical References: **EO-100-103 step PC/P-4**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 3

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: 55.43

Training Objective: **266** **Explain the sequence of events and flowpaths that occur within the Primary Containment during a DBA LOCA. In your discussion include:**

- **Drywell**
- **Suppression Pool**

Training Task: **00.EO.027** **Implement Primary Containment Control**

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19 Unit 1 is at 100% reactor power.
Operations has been notified a calibration error for D MSL Flow-High isolation instrumentation has resulted in the following trip setpoint data:

D MSL Flow-High Instrument Number	Trip Setpoint
FIS-B21-1N009A	135 psid
FIS-B21-1N009B	136 psid
FIS-B21-1N009C	134 psid
FIS-B21-1N009D	134 psid

What Technical Specification required action and completion time, if any, is applicable at the time of discovery?

A. Restore Isolation capability within 1 hour.

B. Enter the Condition referenced in Table immediately.

C. None, LCO is met.

D. Be in MODE 2 in 7 hours.

Question Data

A Restore Isolation capability within 1 hour.

Explanation/Justification:

A. is correct. MSL isolation function is not operable if 4 channels from D MSL are inoperable. The issue of calibration error adds complexity since have to make decision if equipment is broke or not. Have to determine from data if sufficient number of instruments and then determine if function is available.

B. Entering the Condition from Table 3.3.6.1-1 is not done until the Condition A or B completion time is exceeded.

C. LCO is not met, Table 3.3.6.1-1 requires each trip system to have 2 channels/steam line operable.

D. time from LCO 3.0.3 which is not applicable.

Sys # System Category **Conduct of Operations** KA Statement **Ability to apply technical specifications for a system.**

K/A# **2.1.12** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **Tech Spec** Technical References: **TS 3.3.6.1**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 3

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: 55.43

Training Objective: **1642** **Given a set of U-1 (U-2) Technical Specifications, determine the ability to determine Main Steam System operability by locating the applicable LCO Action Statement. (SRO only)**

Training Task: **00TS001** **Ensure Plant Operates In Accordance With The Operating License, Technical Specifications (TS), and Technical Requirements Manual (TRM)**

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20 Technical Specification Surveillance Requirement SR 3.3.1.1.2 was last performed at 2145 on 10/5/03 prior to a reactor scram.

Given the following times and data:

- Plant Start-up on 10/14/03.
- 1115 on 10/15/03 MODE 1 entered.
- 1740 on 10/15/03 power initially exceeded 25%.
- 1830 on 10/15/03 power was subsequently reduced to 22% before SR 3.3.1.1.2 was completed.
- 2020 on 10/15/03 power exceeded 25%.
- No LCO required actions were entered.

What is the maximum time for completion of SR 3.3.1.1.2 to comply with Technical Specification requirements without using frequency interval extensions?

A. 0540 on 10/16/03

B. 0940 on 10/16/03

C. 0820 on 10/16/03

D. 2315 on 10/15/03

Question Data

C 0820 on 10/16/03

Explanation/Justification:

A. This time and date is based on initially starting the clock for performance of SR 3.3.1.1.2. When power was reduced below 25% the clock is reset since the conditions are no longer met to perform the surveillance.

B. This time and date is based on the initial power increase above 25% plus the 25% frequency interval extension.

C. correct. This time and date is based on meeting the conditions for performance of the surveillance the second time. There is no violation, even with the 7 day frequency not met, provided operation does not exceed 12 hours with power greater than 25%.

D. This time and date is based upon entering MODE 1. Entering MODE 1 is not a trigger to complete the surveillance. SR 3.3.1.1.2 is modified by a note for the 7 day frequency such that it is not required to be performed until 12 hours after thermal power is greater than 25%.

Sys #	System	Category	KA Statement
		Equipment Control	**Knowledge of surveillance procedures.**

K/A# 2.2.12 K/A Importance 3.4 Exam Level **SRO**

References provided to Candidate **Tech Spec** Technical References: **TS 3.3.1**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 3

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: 55.43

Training Objective: **1398** **Determine if a component or system is required to be operable per Technical Specifications.**

Training Task: **00TS001** **Ensure Plant Operates In Accordance With The Operating License, Technical Specifications (TS), and Technical Requirements Manual (TRM)**

21 Unit 1 is in REFUELING mode with the following plant conditions:

- Refueling cavity water level is 22.5 feet above the top of the RPV flange and stable
- B and D RHR pumps are out of service for maintenance
- C RHR pump is running
- A RHR pump is in standby
- Irradiated fuel assemblies are in the RPV
- An irradiated fuel assembly is being loaded into the RPV

Engineering reports the 'A' SBGT fan is not seismically qualified.

Immediately AFTER receiving the SBGT status report, the C RHR pump trips on overcurrent and cannot be restarted. The PCOM attempts to start A RHR pump, however, it will NOT start.

What Technical Specification actions, if any, are REQUIRED within 1 hour, for these conditions?

A. Verify two alternate methods of decay heat removal are available, AND verify reactor coolant circulation by an alternate method AND monitor reactor coolant temperature.

B. Immediately suspend loading irradiated fuel assemblies into the RPV.

C. No Technical Specification actions required, RHR may be removed from service for 2 hours per 8 hour period.

D. Verify an alternate method of decay heat removal is available, AND verify reactor coolant circulation by an alternate method AND monitor reactor coolant temperature.

Question Data

D Verify an alternate method of decay heat removal is available, AND verify reactor coolant circulation by an alternate method AND monitor reactor coolant temperature.

Explanation/Justification:

A. Incorrect. These are the required actions if Refueling cavity water level is < 22 feet

B. Incorrect. These actions need to be taken only if Condition A actions are NOT met. These actions would only be required if BOTH SBGT fans were inoperable.

C. Incorrect. TS 3.9.7 Note that allows RHR shutdown for 2 hours in any 8 hour period is an allowance for specific planned evolutions and does NOT apply to unplanned losses of RHR. If a candidate does not understand these restrictions, the candidate will incorrectly choose this distracter

D. Correct answer. Information given in the stem of the question makes TS 3.9.7 applicable. Actions A and C are appropriate since this is the last operating RHR shutdown cooling subsystem.

Sys #	System	Category	KA Statement
		Equipment Control	**Knowledge of refueling administrative** requirements.

K/A# 2.2.26 K/A Importance 3.7 Exam Level SRO

References provided to Candidate **TS 3.9.7** Technical References: **TS 3.9.7**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 3

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: 55.43

Training Objective: **496** **Perform refueling prerequisites and requirements.**

Training Task: **49ON003** **Implement Loss Of RHR Shutdown Cooling Mode**

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22 The Unit Supervisor is preparing a prejob brief per OP-AD-004, "Operations Standards For Error And Event Prevention" with Unit 1 at 100% power. The prejob brief is to support valve lineup checks for Maintenance in the Reactor Water Cleanup (RWCU) Backwash Receiving Tank Room while RWCU system remains in service.

Due to a broken reach rod, entry is required to check position of 166004, RWCU BKWSH TK DRAIN TO LRW and other valves as shown on the attached Area Survey Map .

The operator being sent in the area has a total dose for the year of 400 mrem TEDE.

A 600 mrem allowance for checking the other valves and to exit the area must be factored into the maximum stay time calculations.

SSES Administrative dose limits shall not be exceeded and no dose extensions have been authorized.

Based on these conditions, how should system blocking requirements and maximum stay time be addressed during the ALARA briefing?

A. System blocking is not required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 60 minutes.

B. System blocking is not required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 24 minutes.

C. System blocking is required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 36 minutes.

D. System blocking is required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 12 minutes.

Question Data

D System blocking is required to prevent introducing resin into the Backwash Receiving Tank.
Maximum stay time is 12 minutes.

Explanation/Justification:

A. 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.

B. 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.

C. 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.

D. Correct Answer, 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. 1000 divided by 5rem/hr is 20 minutes.

Sys # System Category KA Statement

SSES SRO NRC Re-Exam

Radiological Controls

Ability to perform procedures **to reduce excessive** levels of **radiation and** guard against **personnel exposure**.

K/A# **2.3.10** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **None** Technical References: **NDAP-QA-0696**, 1191

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 2

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: 55.43

Training Objective: **4347** **DESCRIBE the access and control requirements for:**

a. **Radiation Areas**

b. **High and Very High Radiation Areas**

Training Task: **00AD018** **Implement Appropriate Portions Of Radiologically Controlled Area Access** And RWP **System**

PP&L Form 3104 (1-97)

SUSQ SES - AREA SURVEY MAP H2 scfm Rx PWR 100 %

UNIT: 1 BUILDING: REACTOR Elev. 761' ROOM: RWCU BWRT Room I-509

RWP# ##### DATE: Today TIME: Now SURVEY BY: H. P. Tech

RAD. INST. ### HP # #1 CAL DUE 12/30/03 SOURCE CHECK SAT Yes

AIR SAMPLER N/A HP # N/A CAL DUE N/A ACTIVITY N/A µCi/cc

CONTAM. INST. N/A HP # N/A CAL DUE N/A EFF. N/A % BKGD. N/A cpm

SMEAR RESULTS (dpm/100cm^2)

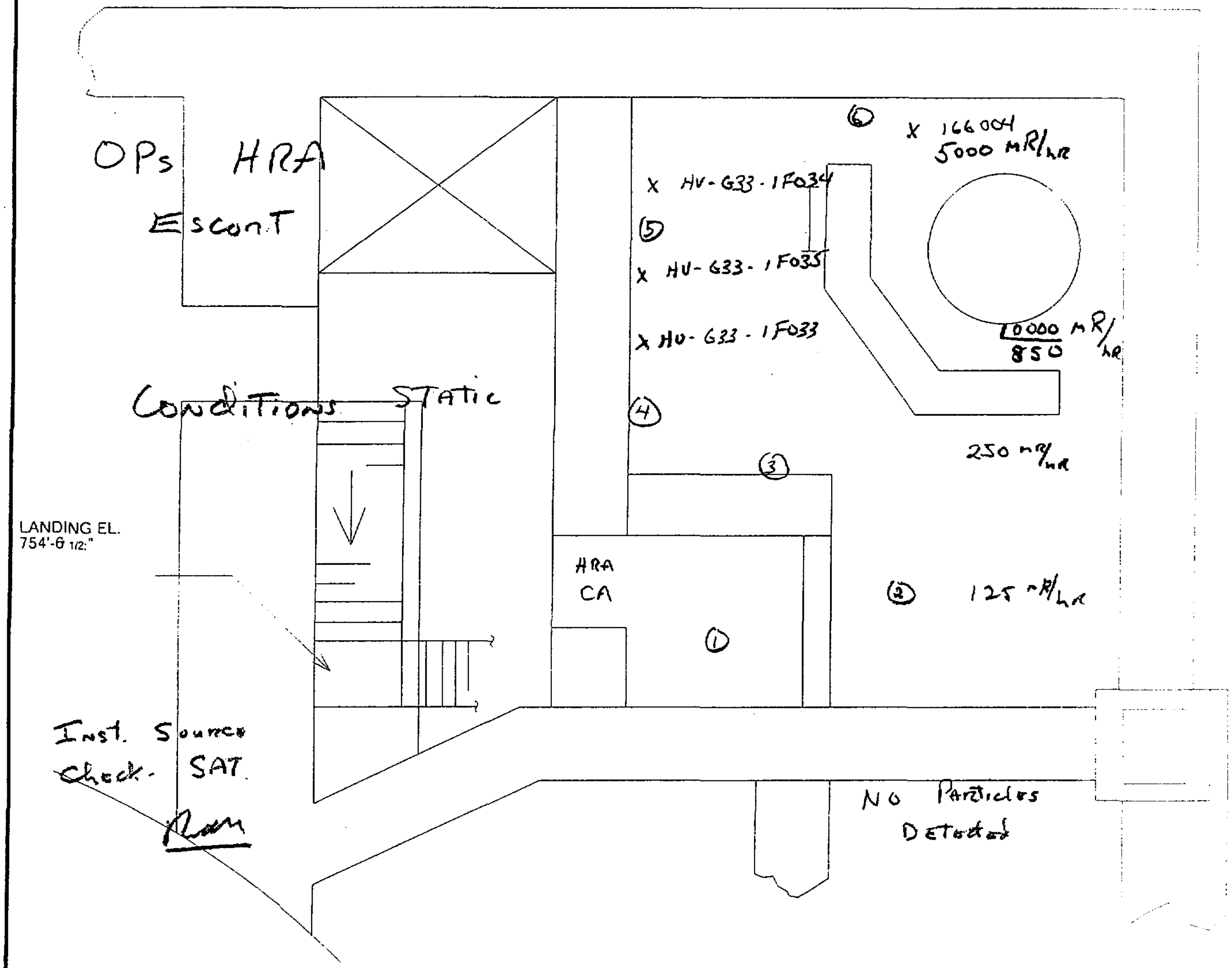
1. 4K Floor
2. 6K Floor
3. 4K Wall
4. 4K Wall
5. 1K Pipe
6. 1K Insulation
7.
8.
9.
10.
11.
12.
13.
14.
15.
16.
17.
18.

REASON FOR SURVEY

RAD READINGS IN mR/hr SMEAR LOCATIONS CIRCLED. CONTACT RAD READINGS ■ = S.O.P.

β- = BETA DOSE RATE (mRad/hr) Ⓜ = LARGE AREA SMEAR (ccpm) GENERAL AREA DOSE RATES → N

- - - = RAD TAPE -X-X- = RAD TAPE & ROPE X X X X = RAD ROPE @ = A/S LOCATION



REVIEWED: Supv HP Tech / Today

Health Physics Date

SSES SRO NRC Re-Exam

23 OP-069-050, "Release of Liquid Radioactive Waste" is being performed for the Laundry Drain Sample Tank (OT312). All required channel checks have been completed satisfactorily with the EXCEPTION of the Unit 1 Cooling Tower Blowdown Flow Instrumentation Channel Check, which failed.

What actions need to be completed for disposition of the release permit initiated for the Laundry Drain Sample Tank (OT312)?

Release of the Laundry Drain Sample Tank (OT312) may:

A. be completed with Shift Supervision approval, and greater than 5500 gpm flow from Unit 2 Cooling Tower Blowdown Flow.

B. be completed with Shift Supervision approval, and analyze at least two independent samples in accordance with TRO 3.11.1.1 AND Independently determine release rates for samples analyzed per Action B.1 actions.

C. NOT be completed. Discharging Laundry Drain Sample Tank requires all Blowdown Flow instrumentation to be operable.

D. NOT be completed, until TR 3.11.1.4 Condition E actions complete and post release samples are analyzed in accordance with Table 3.11.1.1-1.

Question Data

A be completed with Shift Supervision approval, and greater than 5500 gpm flow from Unit 2 Cooling Tower Blowdown Flow.

Explanation/Justification:

A. correct answer, there are three possible flow instruments that may be selected to satisfy the blowdown flow interlock and to satisfy the procedure and Technical Requirements manual. The three position switch is labeled; Unit 1 or 2 or BOTH.

B. TR 3.11.1.1 is not required to be entered and the sampling requirement is for an inop rad monitor.

C. only one channel is required by the procedure and the technical requirements.

D. TR 3.11.1.4 condition is applicable for an inoperable rad monitor and the post sampling is required always per the ODCM to verify the composite of all samples has not exceeded any limits.

Sys #	System	Category	KA Statement
		Radiological Controls	**Knowledge of the requirements for reviewing and approving release permits.**

K/A# 2.3.6 K/A Importance 3.1 Exam Level **SRO**

References provided to Candidate **TRO 3.11** Technical References: **TR 3.11.1.4**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 2

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: 55.43

Training Objective: **789** **Complete Form OP-069-050, Attachment F for a Liquid Radwaste Release.**

Training Task: **69OP001** **Complete Form OP-069-050 ATT F For A Liquid Radwaste Release**

24 A loss of coolant accident has occurred on Unit 1 with the following conditions:

Drywell Pressure	- 42 psig by alarm and indication.
Drywell Temperature	- 270 deg F.
Reactor Pressure	- 20 psig
Reactor Level	- below Top of Active Fuel as indicated on the Fuel Zone.
Containment Radiation	- 2100 R/hr

Dose Projections indicate a 545 mrem Thyroid CDE at two (2) miles from the plant.

As the Control Room Emergency Director which of the following actions need to be taken?

A. Declare Site Area Emergency. Evacuate 0-2 miles downwind sectors and shelter 2-10 miles downwind sectors.

B. Declare Site Area Emergency. No protective actions required at this time, continue assessment.

C. Declare General Emergency. Evacuate 0-10 miles.

D. Declare General Emergency. Evacuate 0-2 miles and shelter 2-10 miles.

Question Data

D Declare General Emergency. Evacuate 0-2 miles and shelter 2-10 miles.

Explanation/Justification:

A. SSES does not issue protective actions by sector.

B. A General Emergency has been declared and continued assessment is not valid.

C. A valid dose projection has been performed thus a PAR of evacuation of 0-10 miles is not valid.

D. Correct answer, using PAR Airborne Releases Tab 5 provided. Dose projection indicates less than 5 Rem CDE requiring partial evacuation.

Sys #	System	Category	KA Statement
		Emergency Procedures and Plan	**Knowledge of emergency plan protective action recommendations.**

K/A# **2.4.44** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **Control Room Emergency Director Procedure** Technical References: **Tab 5 EP-PS-100**

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 3

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: 55.43

Training Objective: **EP-010-6** **Apply guidance for PAR. {SRO and STA}**

Training Task: **00.EP.003** **Recommend Protective Actions To Safeguard The Public And To Protect Personnel Working Near The Plant.**

25 Unit 1 operating at 100% power when the following alarm and indication is received:

- MAIN STEAM DIV 2 SRV OPEN (AR-110-E03)
- Division 2 Acoustic Monitor Red light LIT on 1C601 vertical panel

Which initial set of indications validate the SRV alarm and indication for an SRV opening?

What procedure(s) must be implemented upon receipt of the alarm and indication?

What follow-up actions must be performed assuming the SRV closes in 3 minutes?

A. - INDICATED total steam flow constant, indicated Reactor Pressure constant, and Generator MWE lowering,

- Enter ON-156-001 "Unexplained Reactivity Change".

- PLOT position on Power/Flow Map, Form NDAP-QA-0338-11.

B. - INDICATED total feedflow constant, ACTUAL steam flow constant and the Suppression Pool temperature rising.

- Enter ON-183-001 "Stuck Open Relief Valve".

- Verify vacuum breakers CLOSED within 2 hours and perform Functional Test within 12 hours.

C. - Red indicating light above SRV control Switch is LIT, ACTUAL total steam flow constant, and Generator MWE lowering.

- Enter ON-156-001 "Unexplained Reactivity Change".

- PLOT position on Power/Flow Map, Form NDAP-QA-0338-11.

D. - Red indicating light above SRV control Switch is NOT LIT, ACTUAL total feedwater flow constant, and indicated Reactor Pressure constant.

- Scram the reactor within 2 minutes of receiving the alarm in accordance with ON-183-001 "Stuck Open Relief Valve".

- Initiate Cooldown at < 100 deg F/ hr in accordance with EO-100-102.

Question Data

B - INDICATED total feedflow constant, ACTUAL steam flow constant and the Suppression Pool temperature rising.

- Enter ON-183-001 "Stuck Open Relief Valve".

- Verify vacuum breakers CLOSED within 2 hours and perform Functional Test within 12 hours.

Explanation/Justification:

A. Total steam flow indications will show a decrease with an SRV open since the SRVs are located on the steam line upstream from the main steam flow instruments. Generator MWE would indicate lower with the steam going to the Suppression chamber instead of the main turbine. Entry into Unexplained Reactivity ON is not totally correct since there is indication of the cause for the transient I.e. the SRV Open alarm. Plot power to flow is correct action if entered the unexplained reactivity procedure for a lowering of power.

B. Correct, acoustic monitor indicates SRV is open, total feedflow is not an indicator of SRV open since same feedflow required whether SRV open or not, Suppression Pool temperature increasing is indication that SRV is open requiring entry into ON-183-001 and when the valve is closed Tech Specs TS 3.6.1.6 require that containment vacuum breakers be checked closed and cycled.

C. Red indicating light would be lit if the control switch were used for opening the valve. Actual steam flow would remain constant where indicated steam flow would lower as explained. Generator MWE lowering is indicative of a stuck open SRV. Entry into ON-156-001 would not be proper actions.

D. The procedure no longer requires that the reactor be scrammed after an SRV is open for 2 minutes.

Sys # System

Category: **Emergency Procedures and Plan**

KA Statement: **Ability to verify that the alarms are consistent with the plant conditions.**

K/A# 2.4.46 K/A Importance 3.6 Exam Level **SRO**

References provided to Candidate **Tech Specs** Technical References: **ON-183-001**, TS 3.6.1.6

Question Source: **New** **Susquehanna, 12/15/2003** Level Of Difficulty: (1-5) 3

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: 55.43

Training Objective: **2104** **Predict how each supported system will be affected by any of the following SRV/ADS system failures:**

a. **Inadvertent initiation**
b. **Failure to initiate**
c. **SRV failure to open**
d. **Stuck open SRV**

Training Task: **83.ON.003** **Implement Stuck Open Safety-Relief Valve**