076 - 008AG2.4.6 001/SRO/T1/G1/3.7/4.7/NEW//

Unit-1 was operating at 100% power when a SGTR occurred on "B" S/G.

- The crew manually tripped the Reactor and completed 1-E-0 "Reactor Trip or Safety Injection"
- Transition was made to 1-E-3 "Steam Generator Tube Rupture"
- RCPs are secured.
- Both PRZR PORV's are available.
- Crew is currently depressurizing the RCS to minimize break flow and refill PRZR using 1-RC-PCV-1456 "PRZR PORV".
- Ruptured S/G pressure is 1035 psig
- TSC has been activated and has directed using 1-ES-3.1 "Post-SGTR cooldown using backfill" for recovery

Current conditions:
- RWST level = 93%
- "B" S/G level = 86% NR
- PRZR level is 29% and increasing.
- RCS Pressure is 1034 psig.
- Crew is stopping the RCS Depressurization when 1-RC-PCV-1456 will not close.
- While attempting to close 1-RC-MOV-1535, the breaker trips.

Based on Current Conditions the proper procedure transition will be:

A✓ 1-ECA-3.1 "SGTR with Loss of Reactor Coolant Subcooled Recovery Desired"

B. 1-ECA-3.2 "SGTR with Loss of Reactor Coolant Saturated Recovery Desired"

C. 1-E-1 "Loss of Reactor or Secondary Coolant"

D. 1-ES-3.1 "Post-SGTR Cooldown using backfill"

Distractor Analysis:

This requires the examinee to have the knowledge of EOP mitigation strategies as the apply to Pressurizer Vapor Space Accident

CORRECT

A. 1-ECA-3.1 "SGTR with Loss of Reactor Coolant Subcooled Recovery Desired"

Transition to 1-ECA-3.1 from 1-E-3 is made due to the following:
Ruptured S/G not isolated from one intact S/G
Ruptured S/G $\leq$ 350 psig
No Intact S/G
< 250 psid between faulted S/G and highest intact S/G
RCS subcooling $\leq$ 45 DEFG
PRZR level $\leq$ 21% with uncontrolled decrease
RCS Press Decreasing
Nonisolable PZR PORV stuck open

INCORRECT

B. 1-ECA-3.2 "SGTR with Loss of Reactor Coolant Saturated Recovery Desired"

Incorrect but plausible if the SRO candidate is unaware that the only transition to 1-ECA-3.2 "SGTR with Loss of Reactor Coolant Saturated Recovery Desired is from 1-ECA-3.1. There are no direct transitions from 1-E-3. The transition entry conditions from ECA-3.1 is RWST level < 58% or Containment sump level is lower than expected. or Ruptured S/G level > 90% [85%]

INCORRECT

C. 1-E-1 "Loss of Reactor or Secondary Coolant"

Incorrect but plausible because the SRO candidate knows that E-1 provides guidance for operating personnel to recover from a loss of reactor or secondary coolant. The question states that the crew went directly from E-0 to E-3 and then the PORV and block valve fails open. The candidate may think the proper transition would be to E-1 to combat the loss of reactor coolant prior to transitioning to an ECA procedure.

INCORRECT

D. 1-ES-3.1 "Post-SGTR Cooldown using backfill"

Incorrect but plausible if the SRO candidate focus in on the TSC directive to recover using 1-ES-3.1. 1-ES-3.1, 1-ES-3.2, and 1-ES-3.3 are recovery procedures and only transitioned to after 1-E-3 is completed.

K/A:
008AG2.4.6
Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
Knowledge of EOP mitigation strategies.

Technical References:

1-E-3 "Steam Generator Tube Rupture"
Westinghouse background for E-3
1-ECA-3.1 "SGTR with Loss of Reactor Coolant Subcooled Recovery Desired"
Westinghouse background for ECA-3.1
1-ECA-3.2 "SGTR with Loss of Reactor Coolant Saturated Recovery Desired"
Westinghouse background for ECA-3.2
1-ECA-3.3 "SGTR without Pressurizer Pressure Control"
Westinghouse background for ECA-3.3
1-E-1 "Loss of Reactor or Secondary Coolant"
Westinghouse background for E-1
11715-FM-093B (Sh 1 of 3) Reactor Coolant system

References provided to applicants: None

Learning Objective:
U 13844
Entry conditions for 1-ECA-3.1 "SGTR with Loss of Reactor Coolant Subcooled Recovery Desired"
Conditions that result in leaving 1-ECA-3.1 "SGTR with Loss of Reactor Coolant Subcooled Recovery Desired"

U 9594
Explain how it is determined that a transition from ECA-3.1 "SGTR with Loss of Reactor Coolant Subcooled Recovery Desired" to ECA-3.2 "SGTR with Loss of Reactor Coolant Saturated Recovery Desired" is appropriate

U 13845
List the purpose of 1-ECA-3.2 "SGTR with Loss of Reactor Coolant Saturated Recovery Desired"

U 13846
List the purpose of 1-ECA-3.3 "SGTR Without Pressurizer Pressure Control"
List the entry Conditions of 1-ECA-3.3

U 13882
Explain the purpose of post SGTR cooldown procedures 1-ES-3.1, 3.2, 3.3.

U 13683
List various information associated with E-1 "Loss of Reactor or Secondary Coolant"

Question Source: NEW

Question History: None

Question Cognitive Level: Comprehension/Analysis

10 CFR Part 55 Content:
SRO only 10 CFR-55.43(b)(5)

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal. and emergency situations.
10 CFR-55.41(a)(b)(10)
Administrative, normal, abnormal, and emergency operating procedures at the facility.

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the knowledge of EOP mitigation strategies as they apply to Pressurizer Vapor Space Accident. To arrive at the correct answer the SRO applicant must recognize the proper procedure during a emergency situation that would be required to mitigate from a pressurizer vapor space accident when the PORV fails open and can not be isolated during the performance of 1-E-3 "Steam Generator Tube Rupture".

Given the following conditions:

Unit 1 was at 100% power when a Loss of all Main Feed Water occurred

The crew has just transitioned to 1-ES-0.1, Reactor Trip Response

The following indications are noted on PCS:

RCS Tave = 548° F and stable

Based on the indications of the attached PCS trend which ONE of the following will the SRO direct the crew to perform?

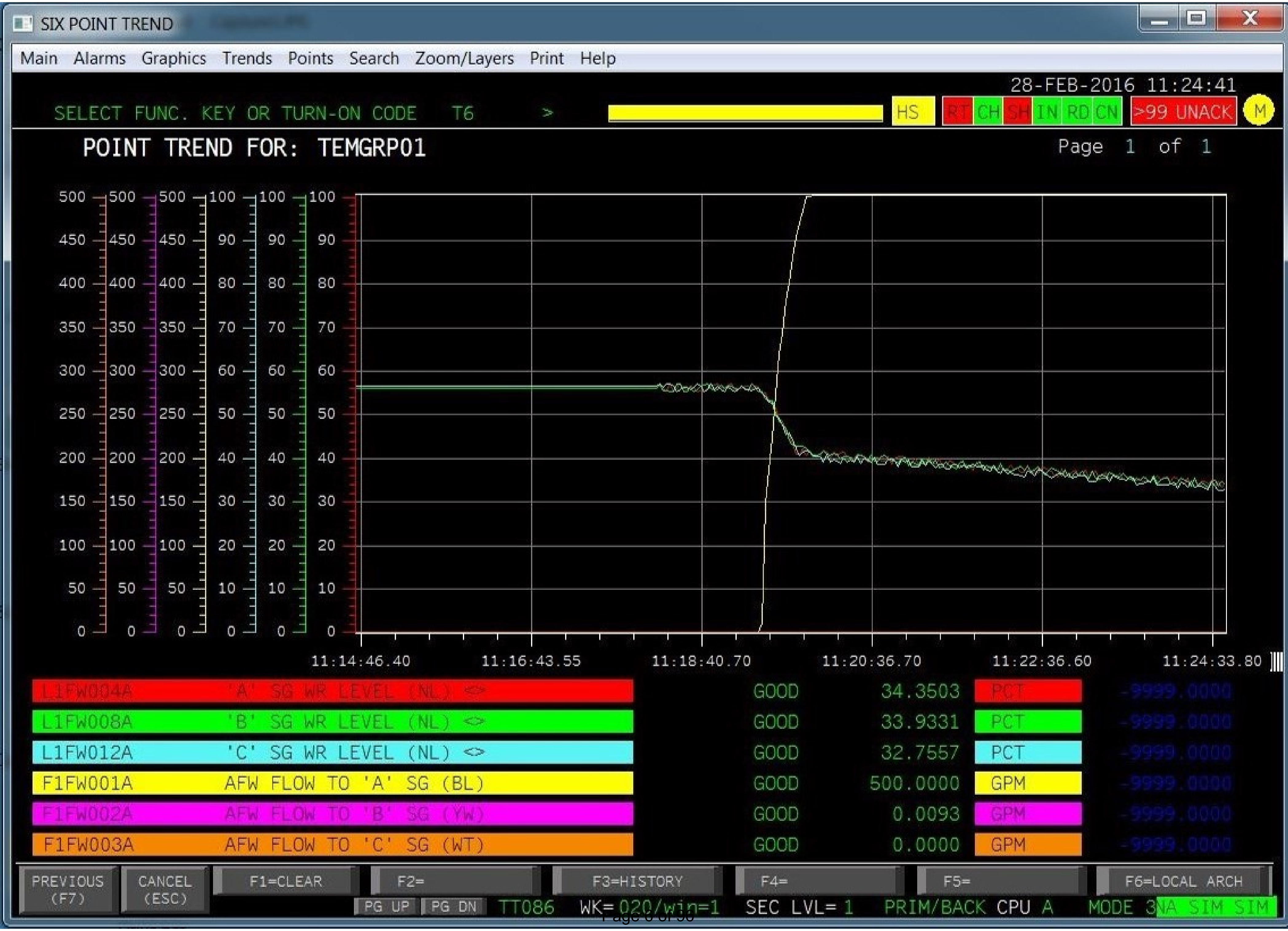
(Reference provided)

A✓ Transition to 1-FR-H.1, Response to Loss of Secondary Heat Sink

B. Remain in 1-ES-0.1 and perform 1-AP-22.4, Loss of Both Motor-Driven AFW Pumps

C. Transition to 1-FR-H.5, Response to Steam Generator Low Level

D. Return to 1-ES-0.0, Re-Diagnosis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Main Alarms Graphics Trends Points Search Zoom/Layers Print Help
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100 90 80 70 60 50 40 30 20 10 0
11:14:46.40
11:16:43.55
11:18:40.70
11:20:36.70
11:22:36.60
11:24:33.80
L1FW004A 'A' SG WR LEVEL (NL) <> GOOD 34.3503 PCT -9999.0000
L1FW008A 'B' SG WR LEVEL (NL) <> GOOD 33.9331 PCT -9999.0000
L1FW012A 'C' SG WR LEVEL (NL) <> GOOD 32.7557 PCT -9999.0000
F1FW001A AFW FLOW TO 'A' SG (BL) GOOD 500.0000 GPM -9999.0000
F1FW002A AFW FLOW TO 'B' SG (YW) GOOD 0.0093 GPM -9999.0000
F1FW003A AFW FLOW TO 'C' SG (WT) GOOD 0.0000 GPM -9999.0000
PREVIOUS (F7)
CANCEL (ESC)
F1=CLEAR
F2=
F3=HISTORY
F4=
F5=
F6=LOCAL ARCH
PG UP
PG DN
TT086
WK=020/win=1
SEC LVL= 1
PRIM/BACK CPU A
MODE 3NA SIM SIM

Distractor Analysis:

This requires the examinee to have the ability to use plant computer to evaluate system or component status associated with a Loss of Main Feedwater

A. CORRECT
Transition to 1-FR-H.1, Response to Loss of Secondary Heat Sink

Given the indications showing on the PCS graph, (AFW Flow to "A" SG At 500 GPM & "A" SG WR Level Decreasing) AFW is not reaching the A SG and therefore a loss of heat sink is indicated.

B. INCORRECT
Remain in 1-ES-0.1 and perform 1-AP-22.4, Loss of Both Motor-Driven AFW Pumps

Plausible because if the candidate thinks that the indicated AFW flow is reaching the SG (Based on AFW Flow to "A" SG At 500 GPM) then this would be the required action.

C. INCORRECT
Transition to 1-FR-H.5, Response to Steam Generator Low Level

Plausible because FR-H.5 conditions are met since all narrow range levels are below 11%.

D. INCORRECT
Return to 1-ES-0.0, Re-Diagnosis

Plausible because re-diagnosis can be used in many situations after E-0 has been exited but the candidate must know that SI should be in service or required to enter ES-0.0.

K/A:
054AG2.1.19
Loss of Main Feedwater
Ability to use plant computer to evaluate system or component status

Technical References:
PCS
1-F-0

References provided to applicants: PCS Printout

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only 10 CFR-55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations
10 CFR-55.41(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility

Comments:
This question matches the K/A statement by requiring the SRO applicant show the ability to use plant computer to evaluate system or component status associated with a Loss of Main Feedwater. To arrive at the correct answer the SRO applicant must recognize what kind of trend is associated with feed water flow going to A S/G, and then understand what affect this flow will have on indicated WR "A" S/G wide range level, then selecting the proper procedural flowpath.

1. 078 - 055EA2.03 001/SRO/T1/G1/3.9/4.7/MODIFIED//

Unit 1 was operating at 75% power when a loss of AC power occurred. The following plant conditions exist:

The crew is performing the actions of 1-ECA-0.0 "Loss of All AC Power"

- The operators completed Step 10 placing various control switches in the Pull-To-Lock position, including the charging pump switches.
- E Transfer bus is being supplied by the SBO
- 1J 4160 bus has been restored
- The Shift Technical Advisor reports a red path still exist on heat sink.

Based on the above information, the crew should ____________ .

A.✔ Proceed to step 31 of 1-ECA-0.0, to facilitate recovery actions.

B. Continue on sequentially with 1-ECA-0.0 from step 11 until transition, to appropriate recovery guildelines.

C. Transition to 1-E-0 "Reactor Trip or Safety Injection", then when directed, transition to 1-FR-H.1 "Response to Loss of Secondary Heat Sink".

D. Complete 1-ECA-0.0, then immediately transition to 1-FR-H.1 "Response to Loss of Secondary Heat Sink".

Distractor Analysis:
This requires the examinee to have the ability to determine or interpret actions necessary to restore power as they apply to a Station Blackout.

A. CORRECT
Proceed to step 31 of 1-ECA-0.0, to facilitate recovery actions.

Correct as per the Westinghouse background documents for this caution in ECA-0.0, which is prior to step 10, "To minimize the deterioration of plant conditions, recovery actions should be started as soon as AC power is restored". ECA-0.0 is written such that step 31 can be entered from any step that follows this caution.

B. INCORRECT
Continue on sequentially with 1-ECA-0.0 from step 11 until transition to appropriate recovery guildelines.

Incorrect but plausible if the candidate is not familiar with the Caution in 1-ECA-0.0 "When power is restored to any AC emergency bus, then, to facilitate recovery actions, recovery should continue with Step 31. Also, at the beginning of ECA-0.0, there is a note "CSF Status Trees should be monitored for information only. FRs should not be implemented".

C. INCORRECT
Transition to 1-E-0 "Reactor Trip or Safety Injection", then when directed, transition to 1-FR-H.1 "Response to Loss of Secondary Heat Sink".

Incorrect but plausible if the candidate feels the need to verify equipment and ESF actuation per 1-E-0 after the AC bus is returned to service.

D. INCORRECT
Complete 1-ECA-0.0, then immediatly transition to 1-FR-H.1 "Response to Loss of Secondary Heat Sink".

Incorrect but plausible Note in ECA-0.0 "CSF Status Trees should be monitored for information only. FRs should not be implemented". This priority is necessary since all FRGs are written on the premise that at least one ac emergency bus is energized. But in this case the recovery procedure from 1-ECA-0.0 will be either 1-ECA-0.1 "Loss of all AC power Recovery without SI required" which contains a note at the beginning of the procedure "CSF status trees should be monitored for information only. FRs should not be initiated before completion of Step 9" or ECA-0.2 "Loss of all AC power Recovery with SI required" which contains an note at the beginning of the procedure "CSF status trees should be monitored for information only. FRs should not be initiated before completion of Step 11". Actions in guideline ECA-0.0 defeat automatic loading of major plant equipment on the energized AC emergency bus. Steps 1 through 9 of ECA-0.1 starts normal operational equipment as necessary based on plant conditions, and Steps 1 through 11 of ECA-0.2 start safeguards equipment as necessary based on plant conditions. Both of these procedures have priority over the FRGs.

K/A:
055EA2.03
Loss of Offsite and Onsite Power (Station Blackout)
Ability to determine or interpret the following as they apply to a Station Blackout:
Actions necessary to restore power

Technical References:
1-ECA-0.0 "Loss of all AC Power"
Westinghouse background information ECA-0.0 "Loss of all AC Power"
0-OP-6.4 "Operation of the SBO Diesel (SBO event)"
DWG 11715-FE-1BB.
DWG 11715-FE-21M
NAPs UFSAR
1-F-0 "Critical Safety Function Status Trees"

References provided to applicants: None

Learning Objective:
U 13830
List the following information associated with 1-ECA-0.0 "Loss of All AC Power"
Purpose of the procedure
Modes of applicability
Entry Conditions
Major action categories
Conditions that result in leaving the procedure

U 13831
Explain why the FR procedures are not implemented during the performance of 1-ECA-0.0

Question Source: Modified NAPS Vision Exam Bank

Question History:
Unit 1 was operating at 60% power when a loss of all AC power occurred. The following plant conditions exist:
The crew is performing the actions of 1-ECA-0.0 "Loss of All AC Power"
The operators have just completed placing various control switches in the PULL-TO-LOCK position, including the charging pump switches
The "J" EDG has been started locally and 1J bus has been restored
The STA reports a red path still exists on heat sink

Based on the above information, the crew should _________.

A. Proceed to step 29 of 1-ECA-0.0 (CORRECT)
B. Continue on with 1-ECA-0.0 from step in effect until transition to appropriate recovery procedure
C. Transition to 1-E-0 "Rx Trip or SI" then when directed transition to 1-FR-H.1 "Response to Loss of Secondary Heat Sink"
D. Transition directly to 1-FR-H.1 "Response to Loss of Secondary Heat Sink"

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures, during normal, and emergency situations.
10 CFR-55.45(13)
Demonstrate the applicants ability to function within the control room team as appropriate to the assigned position, in such a way that the facility licensee's procedures are adhered to and that the limitations in its license and amendments are not violated.

Comments:
This question matches the K/A statement by requiring the SRO applicant to correctly interpret a set of given plant conditions during a Station Blackout (Loss of All AC Power) casualty, and then determine the appropriate procedure to take actions necessary to mitigate the accident. To arrive at the correct answer, the SRO applicant must recognize that in ECA-0.0 scenario FR procedures are not implemented due to inadequate electrical power situation, and also to recognize the need to jump ahead in the procedure to steps that facilitate recovery actions after power is restored to any AC emergency bus. Also, have knowledge that FRs will be monitored until equipment is started based on plant conditions before FRs become applicable in the recovery procedures that are transitioned to from ECA-0.0

2. 079 - 058AG2.1.20 001/SRO/T1/G1/4.6/4.6/MODIFIED//

Unit-2 in operating at 100% power when DC bus 2-III is lost.

- The crew is performing the steps of attachement 9 "Loss of DC Bus 2-III of 0-AP-10 "Loss of Elect Power"
- The operating crew can not re-energize 125-volt DC bus
- VCT level = 20%
- Pressurizer level > 28%

The SRO will initiate ______(1)_______, and direct the crew to _____(2)_______.

A. (1) 2-AP-49.1 "Loss of Normal and Excess Letdown"
(2) Swap Charging suction to RWST

B. (1) 2-AP-49 "Loss of Normal Charging"
(2) Isolate letdown due to 2-CH-FCV-2122 being closed

C. (1) 2-AP-49 "Loss of Normal Charging"
(2) Place 2-CH-FCV-2122 in local control at Aux Shutdown Panel and control flow

D✓ (1) 2-AP-49.1 "Loss of Normal and Excess Letdown"
(2) Reduce Makeup flow and Maximize RCS sample flow

Distractor Analysis:
This requires the examinee to have the ability to interpret and execute procedure steps for Loss of DC power

A. INCORRECT
(1) 2-AP-49.1 "Loss of Normal and Excess Letdown"
(2) Swap Charging suction to RWST

Incorrect but plausible, first part is correct, 0-AP-10 "Loss of Power", attachment 19 will direct the crew to initiate 2-AP-49.1 "Loss of Normal and Excess Letdown" If the operating crew can not re-energize 125-volt DC bus 2-III and pressurizer level is greater than 28%, 2-AP-49.1 will then direct the crew to minimize Reactor Coolant System (RCS) makeup flow and to maximize RCS sample flow, to reduce the rate of Pressurizer level increase, because of the Loss of Normal and Excess Letdown. Second part is incorrect but plausible if the candidate associates the loss of normal and excess letdown to a reduction is VCT level, then swapping to the RWST could be plausible, also the step prior to checking Pressurizer level in 0-AP-10, attachment 9, checks VCT level and if < 15% requires transferring charging pump suction to the RWST.

B. INCORRECT
(1) 2-AP-49 "Loss of Normal Charging"
(2) Isolate letdown due to 2-CH-FCV-2122 being closed

Incorrect but plausible, first part is incorrect; there is not any transitions to 2-AP-49

"Loss of Normal Charging" from 0-AP-10. But the candidate may think normal charging is lost due to the DC bus loss and assume that it is the correct procedure for this event. Second part is incorrect but could be plausible if the candidate does not understand that the signal generated to the E/P for 2-CH-FCV-2122 comes from the process plant cabinet 6, which has two separate power supplies.

C. INCORRECT
(1) 2-AP-49 "Loss of Normal Charging"
(2) Place 2-CH-FCV-2122 in local control at Aux Shutdown Panel and control flow

Incorrect but plausible, first part is incorrect; there are no transitions to 2-AP-49 "Loss of Normal Charging" from 0-AP-10. But the candidate may think normal charging is lost due to the DC bus loss and assumes that it is the correct procedure for this event. Second part is incorrect but could be plausible if the candidate knows that 2-CH-FCV-2122 can be controlled from the Aux Shutdown Panel and 2-AP-49 does have a step for the operators to use 2-AP-20, Operation from the Auxiliary Shutdown Panel to shift 2-CH-FCV-2122 to local control in the ASP and control charging flow if 2-CH-FCV-2122 demand in not indicated in the control room.

D. CORRECT
(1) 2-AP-49.1 "Loss of Normal and Excess Letdown"
(2) Reduce Makeup flow and Maximize RCS sample flow

Both the first part and second part are correct. 0-AP-10 "Loss of Power", attachment 19 will direct the crew to initiate 2-AP-49.1 "Loss of Normal and Excess Letdown" If the operating crew can not re-energize 125-volt DC bus 2-III and pressurizer level is greater than 28%, 2-AP-49.1 will then direct the crew to minimize Reactor Coolant System (RCS) makeup flow and to maximize RCS sample flow, to Reduce the rate of Pressurizer level increase because of the Loss of Normal and Excess Letdown.

K/A:
058AG2.1.20
Loss of DC Power
Ability to interpret and execute procedure steps.

Technical References:
0-AP-10 "Loss of Electrical Power"
1-AP-49.1 "Loss of Normal and Excess Letdown"
1-AP-49 "Loss of Normal Charging"
UFSAR for NAPS Unit 1 and 2 Section 8.3.2 Direct Current Power System
12050-CH-001 "CVCS Charginig flow control to the Regen Hx 2-CH-E-3

References provided to applicants: None

Learning Objective:
U 11902 Given the results of the electrical power system diagnostic (0-AP-10), sequence in order of priority the actions necessary to restore the Electrical Distribution System (Dealing with the effects of the loss of power)

U 11549
Explain concepts associated with responding to the loss of 125-volt DC bus 1-III (why RCS makeup flow is minimized and RCS sample flow is maximized, if the DC bus can not be re-energized. with PZR level >28%)

U 18050
Recognize plant conditions that result in a transition to or from 1-AP-49.1 "Loss of Normal and Excess Letdown"
Explain the high level actions, major action categories, key mitigating strategies, and their basis.

Question Source: Modified - NAPs Vision BANK (60591)

Question History:
Unit-2 in operating at 100% power when DC bus 2-III is lost.

• The crew is performing the steps of attachement 9 "Loss of DC Bus 2-III of 0-AP-10 "Loss of Elect Power"

If the operating crew can not re-energize 125-volt DC bus 2-III and pressurizer level is greater than 28%, the crew is directed to minimize Reactor Coolant System (RCS) makeup flow and to maximixe RCS sample flow.

This action is taken in order to _______(a)________ due to ________(b)_________.

A. (a) Increase the rate of RCS inventory turnover
(b) Loss of blender flow path
B. (a) Reduce the rate or Pressurizer level increase
(b) Loss of normal and excess letdown (CORRECT)
C. (a) Reduce the rate of Volume Control Tank level decrease
(b) Loss of normal and excess letdown
D. (a) Reduce the rate of RCS inventory turnover
(b) Loss of blender flow path

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations
10 CFR-55.41 (a)(b)10
Administrative, normal, abnormal, and emergency operating procedures for the facility,

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the ability to interpret and execute procedure steps for Loss of DC power. To arrive at the correct answer the SRO applicant must recognize the proper procedure and procedure steps that would be required to mitigate the given DC bus conditions. Also the applicant will be required to have the knowledge of various system lineups associated with the stated DC bus loss.

3. 080 - 077AA2.05 001/SRO/T1/G1/3.2/3.8/NEW//

Unit-1 and Unit-2 are both operating at 100% power.

Shortly after midnight, both units are experiencing mega-watt swings along with 500kv bus, voltage and frequency fluctuations.

As per 0-AP-8, Response to Grid Instability, describe the operation of the voltage regulator and when offsite power should be declared inoperable.

A. Place voltage regulators in MANUAL; grid voltage increases to 550 KV

B.✓ Maintain voltage regulators in AUTO; grid voltage decreases to 504 KV

C. Maintain voltage regulators in AUTO; grid voltage increases to 550 KV

D. Place voltage regulators in MANUAL; grid voltage decreases to 504 KV

Distractor Analysis:
This requires the examinee to have the ability to determine and interpret operational status of offsite circuit as they apply to Generator Voltage and Electric Grid Disturbances.

A) INCORRECT
Place voltage regulators in MANUAL; grid voltage increases to 550 KV

First part is incorrect but plausible if operator associates mega watt, voltage and frequency swing as a requirement for placing Voltage Regulator in Manual. Note in 0-AP-8 = Fluctuations in Mega-watts indicate grid disturbance, Voltage Regulator failures have little to no affect on mega-watt output. The Voltage Regulators should be kept in AUTO during grid disturbances and has the operator verify Voltage Regulator is in AUTO or place in AUTO if available. Second part is incorrect because high switchyard voltage is NOT an immediate operability concern. The concern is more of a long-term degradation issue for energized equipment, which is covered by a note in 0-AP-8. This is plausible because grid voltage of 550 KV is used as a trigger point in 0-AP-8 to contact System engineer as soon as possible to evaluate GDC-17 requirements and Offsite power source operability.

B) CORRECT
Maintain voltage regulators in AUTO; grid voltage decreases to 504 KV

First part is correct: Note in 0-AP-8, Fluctuations in Mega-watts indicate grid disturbance, Voltage Regulator failures have little to no affect on mega-watt output. The Voltage Regulators should be kept in AUTO during grid disturbances and has the operator verify Voltage Regulator is in AUTO or place in AUTO if available. Should the candidate apply 1-AP-26 Failure of Main Generator Voltage Regulator base on voltage swings, then Mega-watts are verified to be stable before the voltage regulator can be placed in Manual. Second part is correct: < 505 KV is the trigger point in 0-AP-8 and 1-LOG-4 to declare offsite power inoperable and enter the applicable actions of T.S. 3.8.1

C) INCORRECT
Maintain voltage regulators in AUTO; grid voltage increases to 550 KV

Plausible because first part is correct, second part is plausible (see above)

D) INCORRECT
Place voltage regulators in MANUAL; grid voltage decreases to 504 KV

First part incorrect but plausible (see above), Second part is correct

K/A:
077AA2.05
Generator Voltage and Electric Grid Disturbances
Ability to determine and interpret the following as they apply to | Generator Voltage and Electric Grid Disturbances:
Operational status of offsite circuit

Technical References:
0-AP-8, Response to Grid Instability
1-OP-26.8, 500 KV Switchyard Voltage
1-AP-26 Failure of Main Generator Voltage Regulator High
2-AP-26 Failure of Main Generator Voltage Regulator High
T.S. 3.8.1 "AC Sources-Operating"
T.S. 3.8.1 Bases
1-LOG-4 "Unit-1 Control Board (Modes 1-4)"
1-LOG-4A "Unit-1 Control Board (Modes 5 & 6)"
2-LOG-4 "Unit-2 Control Board (Modes 1-4)"
2-LOG-4A "Unit-2 Control Board (Modes 5 & 6)"

References provided to applicants: None

Learning Objective:
U 11991

Given a set of plant conditions, evaluate Main Generator Control and Protection System operations in light of the following issues:

- Effect of a failure, malfunction, or loss of a related system or component on a system.
- Effect of a failure, malfunction, or loss of components in a system or related systems
- Impact on Technical Specifications
- Response if limits or setpoints associated with a system or its components have been exceeded
- Proper operator response to the condition as stated

U 18006
Perform action of 0-AP-8, "Response to Grid Instabiltiy"

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental knowledge

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
10 CFR-55.41 (a)(b)(5)
Facility operating characteristics during steady state and transient conditions, including

coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating linitations and reasons for these operating characteristics.

Comments:
This question matches the K/A statement by requiring the SRO applicant to analysis and interpretation of operational status of offsite circuit as they apply to Generator Voltage and Electric Grid Disturbances. To arrive at the correct answer the SRO applicant must recognize if the voltage regulator is functioning properly and the condition is due to a perturbation on the grid (MW swing), then the best course of action is to leave the voltage regulator in automatic and allow it to control voltage. Also to possess knowledge of the setpoint that requires declaring the offsite power inoperable.

Given the following conditions:

Unit-2 was at 100% when a total loss of S/G Feedwater occurred.

- "A" S/G wide-range level is 32%
- "B" S/G wide-range level is 12%
- "C" S/G wide-range level is 12%
- Containment pressure is 11 psia
- Containment radiation peaked at 1.0E1 R/hr
- The Crew has just entered 2-FR-H.1 "Response to Loss of Secondary Heat Sink
- Secondary Heat Sink is required

For the given plant conditions, which one of the following, list the proper order of priority for cooling restoration per 2-FR-H.1?

A. 1) AFW, 2) MFW, 3) RCS bleed and feed, 4) Condensate, 5) Fire Protection or Service Water

B. 1) RCS bleed and feed, 2) MFW, 3) Condensate, 4) AFW, 5) Fire Protection or Service Water

C. 1) AFW, 2) MFW, 3) Condensate, 4) Fire Protection or Service Water, 5) RCS bleed and feed.

D.✓ 1) RCS bleed and feed, 2) AFW, 3) MFW, 4) Condensate, 5) Fire Protection or Service Water

Distractor Analysis:

This requires the examinee to have the knowledge for adherence to appropriate procedures and operation within the limitations in the facility license and amendments as they apply to the Loss of Secondary Heat Sink.

A) INCORRECT
1) AFW, 2) MFW, 3) RCS bleed and feed, 4) Condensate, 5) Fire Protection or Service Water

This could be plausible if the candidate is not familiar with the step that checks if RCS bleed and feed is required, the candidate may think it's appropriate to try and recover flow to S/G using less intrusive sources AFW then MFW before the bleed and feed is started, Condensate and Fire Protection/Service water requires lowering S/G pressure so they would be last.

B) INCORRECT
1) RCS bleed and feed, 2) MFW, 3) Condensate, 4) AFW, 5) Fire Protection or Service Water

This could be plausible if the candidate knows that the RCS bleed and feed is required: (Any 2 S/G wide range level < 14% [24%]). but is not familiar with the correct sequence that the procedure, has AFW, falling into the recovery plan.

C) INCORRECT
1) AFW, 2) MFW, 3) Condensate, 4) Fire Protection or Service Water, 5) RCS bleed and feed.

This could be plausible if the candidate is not familiar with the urgency to start a RCS bleed and feed based on loss of heat sink but is familiar with the order in which the operator attempts to establish flow. (AFW flow, MFW flow, Condensate flow then low pressure water source (FP or SW) in that order)

D) CORRECT
1) RCS bleed and feed, 2) AFW, 3) MFW, 4) Condensate, 5) Fire Protection or Service Water

Based on the units stated conditions the following can be established.
1) RCS bleed and feed is required: (Any 2 S/G wide range level < 14%). FR-H.1 is to restore and/or maintain adequate secondary heat removal capability and to establish RCS bleed and feed heat removal if secondary heat removal capability cannot be maintained. This is a continous action step within the procedure and the operators will transition when the proper criteria is met. Then the operator attempts to restore or establish AFW flow, MFW flow, Condensate flow then low-pressure water source (FP or SW) in that order,

K/A:
WE05EA2.2
Loss of Secondary Heat Sink

Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

Technical References:
2-FR-H.1, "Response to loss of secondary heat sink"
Background information for Westinghouse owners group Emergency Response Guidelines (HP-Rev. 2) FR-H.1 Response to Loss of Secondary Heat Sink

References provided to applicants: None

Learning Objective:
U 11254
Given a set of plant conditions, evaluate criteria that require bleed and feed to be initiated

U 11304
Why AFW is the preferred source of steam generator feedwater versus MFW

Question Source: Bank (NAPs Vision Bank)

Question History:
<u>Given the following conditions:</u>
Unit-1 was at 100% when a loss of all S/G feedwater occurred
A S/G wide range level is 30%
B and C S/G wide rage levels are 12%
Containment pressure is 13 psia
Containment radiation peaked at 1.0E1 R/hr

The crew has just entered 1-FR-H.1 "Response to Loss of Secondary Heat Sink, and has completed Step 1, CHECK IF SECONDARY HEAT SINK IS REQUIRED (answer Yes)

For the Stated Plant conditions, which ONE of the following lists the proper order of priority of cooling restoration per 1-FR-H.1?

A. 1) RCS bleed and feed; 2) AFW; 3) MFW; 4) Condensate 5) FP/SW
B. 1) RCS bleed and feed; 2) MFW; 3) Condensate; 4) AFW; 5) FP/SW
C. 1) AFW; 2) MFW; 3) Condensate; 4) RCS bleed and feed; 5) FP/SW
D. 1) AFW; 2) MFW; 3) Condensate; 4) FP/SW; 5) RCS bleed and feed

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
10 CFR-55.45(a)(13)
Demonstrate the applicant's ability to function within the control room team as appropriate to the assigned position, in such a way that the facility licensee's

procedures are adhered to and that the limitatioons in its license and amendments are not violated.

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the ability to determine and interpret plant conditions, and maintain adherence to appropriate procedures, and operation within the limitations in the facility license, and amendments, as they apply to the Loss of Secondary Heat Sink. To arrive at the correct answer, the SRO applicant must recognize the proper procedure sequence, by assessing plant conditions, and then, selecting a flow path used by 2-FR-H.1. This also requires knowledge of a decision point, that envolves transition, within 2-FR-H.1, to mitigate recovery.

4. 082 - 001AA2.04 001/SRO/T1/G2/4.2/4.3/NEW//

Unit 1 is being returned to 100% power following a scheduled refueling outage. Reactor power has been stabilized at 50% for shift turnover.

- RCS Pressure 2235 psia
- Tavg and Tref are matched
- Control Rods are in Auto

After assuming the duties as the Unit-1 Unit Supervisor, the selected First Stage Pressure channel fails High.

Which one of the following answers list the proper procedure that addresses **all** the failures of this event and power response to complete the statement?

Direct the operator to take immediate actions as per _____**(1)**______ and initial reactor power response will _____**(2)**______.

A. (1) 1-AP-1.1 "Uncontrolled Rod Motion"
(2) Decrease

B. (1) 1-AP-1.1 "Uncontrolled Rod Motion"
(2) Increase

C.✓ (1) 1-AP-3 "Loss of Vital Instrumentation"
(2) Increase

D. (1) 1-AP-3 "Loss of Vital Instrumentation"
(2) Decrease

Distractor Analysis:
This requires the examinee to have the ability to determine and interpret reactor power and its trend as they apply to continuous rod withdraw.

A) INCORRECT
(1) 1-AP-1.1 "Uncontrolled Rod Motion"
(2) Decrease

First part is incorrect but plausible because when impulse (1st stage pressure) fails high rate of change is sensed by the power mismatch unit and a large Delta is seen between Tavg and Tref. This large temperature error signal and power mismatch error signal are summed and rods will begin to withdraw at maximum speed. At this point candidate may select 1-AP-1.1 because it will address rod motion from a failure within the rod control circuitry. But will not address other parameters controlled by first stage pressure instrumentation. The operators should use 1-AP-3 as the controlling procedure for the loss of vital instrumentation, This will provide the correct response to place rods in manual, control Steam Generator level, and address steam dumps. It also stabilizes, defeats, and corrects the failed indication along with directing the proper T.S requirements. Second part is incorrect but plausible if the candidate thinks rods will insert, attempting to drive first stage pressure down to match Tavg.

B) INCORRECT
(1) 1-AP-1.1 "Uncontrolled Rod Motion"
(2) Increase

First part is incorrect but plausible because when impulse (1st stage pressure) fails high rate of change is sensed by the power mismatch unit and a large Delta is seen between Tavg and Tref. This large temperature error signal and power mismatch error signal are summed and rods will begin to withdraw at maximum speed. At this point candidate may select 1-AP-1.1 because it will address rod motion from a failure within the rod control circuitry. But will not address other parameters controlled by first stage pressure instrumentation. The operators should use 1-AP-3 as the controlling procedure for the loss of vital instrumentation, This will provide the correct response to place rods in manual, control Steam Generator level, and address steam dumps. It also stabilizes, defeats, and corrects the failed indication along with directing the proper T.S requirements. Second part is correct, when First stage pressure fails HIGH, a large rate of change in the power mismatch circuit is sensed, also a large difference is seen between Tavg and Tref indicating turbine power increasing greater than reactor power and rods will begin to withdraw at maximum speed. and reactor power will increase

C) CORRECT
(1) 1-AP-3 "Loss of Vital Instrumentation"
(2) Increase

1-AP-3 "Loss of Vital Instrumentation is the correct procedure to stabilize, defeat, and correct the failed indication. Rods are placed in manual, steam generator levels are maintained, steam dumps are transferred to steam pressure mode, the controlling channel is then swapped and the appropriate MOP is entered for repairs.
When First stage pressure fails HIGH, a large rate of change in the power mismatch circuit is sensed, also a large difference is seen between Tavg and Tref indicating turbine power increasing greater than reactor power and rods will begin to withdraw at maximum speed. and reactor power will increase

D) INCORRECT
(1) 1-AP-3 "Loss of Vital Instrumentation"
(2) Decrease

First part is correct, 1-AP-3 is the correct procedure to cover all issues of concern. Second part is incorrect but plausible, if the candidate thinks rods will insert, attempting to drive first stage pressure down to match Tavg.

K/A:
001AA2.04
Continuous Rod Withdrawal

Withdrawal :
Reactor power and its trend

Technical References:
1-AP-3 "Loss of vital instrumentation
1-AP-1.1 "Continuous uncontrolled rod motion"
1-AR-B-A7 "Median/HI TAVG <> TREF Deviation"

References provided to applicants: None

Learning Objective:
U 1739
Descripe the response of the Rod Control System to a failure of the controlling turbine impulse pressure channel

- Response if limits or setpoints associated with a system or its components have been exceeded
- Proper operator response to the condition as stated

U 17998
Explain the purpose of 1-AP-3 "Loss of Vital Instrumentation"
Explain the symptoms and entry conditions
Explain applicable TS/TRM/Reportability
Explain High Level actions
Recognize plant conditions that result in transition to 1-AP-3

U 17995
Explain the purpose of 1-AP-1.1 "Continuous Uncontrolled Rod Motion"
Explain the purpose
Recognize plant conditions that result in transition to 1-AP-1.1
Explain High Level actions

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
82 - 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

Comments:
This question matches the K/A statement by requiring the SRO applicant to have a ability to determine and interpret reactor power and its trend as they apply to continuous rod withdraw. To arrive at the correct answer the SRO applicant must have the systems knowledge to assess plant conditions by understanding the effect a particular failure of the selected channel of 1st stage pressure will have on Rod Control

System. Then the candidate must choose a procedure that has actions to stabilize and recover the plant that address all associated failures.

Both Units are at 100% power. Chemistry is in the process of sampling "A" WGDT (1-GW-TK-1A) when the following occurs:

22:04	Unit 2 receives annunciator 2B-B5 PROCESS VENT VNT STACK A&B HI HI RADIATION.
22:05	1-VG-RI-180-2, Vent Stack B, is reading 4E+5 uCi/sec
22:08	Chemistry reports diaphragm on 1-GW-TK-1A sample isolation valve is severely cracked and cannot be isolated.
22:09	HP is notified to sample and survey release at 1-GW-TK-1A
22:10	Mech notified to determine time required to repair/terminate leak.
22:15	1-VG-RI-180-2, Vent Stack B, peaks at 4.06E+6 uCi/sec
22:17	1-VG-RI-180-2, Vent Stack B, is reading 3E+6 uCi/sec and is decreasing
22:21	1-VG-RI-180-2, Vent Stack B, is reading a steady 2E+5 uCi/sec
22:23	HP takes two samples and confirms the release rate from 1-GW-TK-1A sample isolation valve is three times ODCM limit
22:45	Mechanics report it will take them 30 minutes to isolate the leak.

The highest required emergency classification is ___(1)___ and the NRC is required to be notified within ___(2)___ of event declaration.

REFERENCE PROVIDED

A. (1) NOUE
(2) 15 minutes

B✓ (1) NOUE
(2) 1 hour

C. (1) ALERT
(2) 15 minutes

D. (1) ALERT
(2) 1 hour

Distractor Analysis:
Abnormal plant evolution associated with an accidental gaseous radwaste release event, Knowledge of events related to system operations/status that must be reported to internal organizations or external agencies such as the State, the NRC, or the transmission system operator.

A) INCORRECT
NOUE; 15 minutes

First part is correct SM will declare NOUE, 19 min into the release HP notifies operations that sample results at 1-GW-TK-1A is 3 time ODCM limit, this meets the limits of RU1.4, the 60 min time frame is met because of Note 2 of the EAL's which states the SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the conditon will likely exceed the applicable time. This occurs at 41 minutes into the event when Mech report back they will need an additional 30 minutes to repair/terminate the leak. Second part is incorrect but plausible because 15 minutes is the required time the SM has to notified the state and local Emergency Operations Centers (EOC) of declaring the emergency class.

B) CORRECT
NOUE; 1 hour

First part is correct, SM will declare NOUE, 19 min into the release HP notifies operations that sample results at 1-GW-TK-1A is 3 time ODCM limit, this meets the limits of RU1.4, the 60 min time frame is met because of Note 2 of the EAL's which states the SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the conditon will likely exceed the applicable time. This occurs at 41 minutes into the event when Mech report back they will need an additional 30 minutes to repair/terminate the leak. Second part is correct and can be confirmed by the Note in EPIP-2.02 Notification of NRC "NRC notification shall be made immediately after notification of State and local governments and in all cases within 1 hour from the time of event declaration.

C) INCORRECT
ALERT; 15 minutes

First part in incorrect but plausible, candidate could pick this because EAL RA1.2 threshold set point for Vent stack B 1-VG-RI-180-2 is > 4.07E+6, and reading on monitor at 22:15 peaks at 4.06E+6 and never is greater than and only maintained this peak value for 2 min of the required $\geq$ 15 min., the value is well below the threshold when mechanics inform the SEM the repair of the leak will take another 30 min. If candidate does not notice that readings are below the set point then at 22:30 the Alert classification would be legitimate. Second part is incorrect but plausible because 15 minutes is the required time the SM has to notified the state and local Emergency Operations Centers (EOC) of declaring the emergency class.

D) INCORRECT
ALERT; 1 hour

First part in incorrect but plausible, candidate could pick this because EAL RA1.2 threshold set point for Vent stack B 1-VG-RI-180-2 is > 4.07E+6, and reading on monitor at 22:15 peaks at 4.06E+6 and never is greater than and only maintained this peak value for 2 min of the required $\geq$ 15 min, the value is well below the threshold when mechanics inform the SEM the repair of the leak will take another 30 min. If candidate does not notice that readings are below the set point then at 22:30 the Alert classification would be legitimate. Second part is correct and can be confirmed by the Note in EPIP-2.02 Notification of NRC "NRC notification shall be made immediately after notification of State and local governments and in all cases within 1 hour from the time of event declaration.

K/A:
060AG2.4.30
Accidental Gaseous Radwaste Release
Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Technical References:
EAL chart
EAL bases for RU1.3
EAL bases for RU1.4
EAL bases for RA1.2
2-AR-B-B5 PROCESS VENT VNT STACK A&B HI HI RADIATION
0-AP-5.2 "MGP Radiations Monitoring System"
EPIP-2.02 "Notification Of NRC"
EPIP-1.02 "Response To Notification Of Unusual Event"
EPIP-2.01 "Notification Of State And Local Governments"
0-AP-54 "Accidental, unplanned, or uncontrolled radioactive gaseous waste release.

References provided to applicants: EAL chart

Learning Objective:
U 14319
Evaluate a set of plant conditions associated with the emergency plan implementing procedures.

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR 55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.
10 CFR 55.41.10(a)(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the Knowledge of events related to system operations/status that must be reported to internal organizations or external agencies such as the State, the NRC, or the transmission system operator. Under an abnormal plant evolution associated with an accidental gaseous radwaste release event. This question requires the candidate to assess various parameters and make decisions and then selecting a notification based on the diagnostics. To arrive at the correct answer the SRO applicant must have the

use of EAL basis, that provides guidance for EAL decision points. Along with the procedure knowledge, required to make notification within the prescribed time frame.

5. 084 - 061AA2.03 001/SRO/T1/G2/3.0/3.3/NEW//

Unit-2 is defueled and fuel assembly insert shuffle is in progress with the Fuel Building Radiation Automatic Interlock key switch in enabled.

- Annunciator 1K-D2 RAD MONITOR SYSTEM HI RAD LEVEL, actuates.
- Annunciator 1K-D4 RAD MONITOR SYSTEM HI-HI RAD LEVEL actuates.

1-RMS-RM-153 "Fuel Pit Bridge Radiation Monitor" is noted to be pegged high and unresponsive to a source check or reset. 1-RMS-RM-152 "New Fuel Storage Radiation Monitor" indication has not changed and Health Physics reports radiation levels in the fuel building have not changed.

Which of the following actions will be required?

A✔ Place the fuel building radiation automatic interlock switch in Disable AND Enter (CR) Condition Report; No other action required.

B. Leave the fuel building radiation automatic interlock switch in Enable AND Enter (CR) Condition Report: No other action required.

C. Place the fuel building radiation automatic interlock switch in Disable AND Enter action of TR 3.3.7(c)

D. Leave the fuel building radiation automatic interlock switch in Enable AND Enter action of TR 3.3.7(c)

Distractor Analysis:
The candidate requires the ability to determine and interpret setpoints for alert and high alarms as they apply to the Area Radiation Monitoring (ARM) system alarms.

CORRECT
A. Place the fuel building radiation automatic interlock switch in Disable AND Enter (CR) Condition Report; No other action required.

On a HI-HI alarm on 1-RM-RMS-152 or 153 and the key switch in ENABLE position the following will occur: After a 2 minute time delay will automatically dump the MCR bottled air, close the MCR dampers and start the MCR emergency ventilation fans. Based on 1-RMS-RM-153 high indications along with 1-RMS-RM-152 indication being normal, and local reports of normal radiations levels in the fuel Building, the crew will be in 0-AP-5.1 "Common Radiation Monitors System" and would make the assessment, 1-RMS-RM-153 has malfunctioned and the SRO will direct the key switch to be placed in DISABLE this is a critical decision because going to Disable is required to be done within 2 minutes of receiving the alarm. Submitting the CR will be the only other action required because, both 1-RMS-RM-152 and 1-RMS-RM-153 are located in the fuel building but TRM 3.3.7 "Radiation Monitoring Instrumentation" Table 3.3.7-1 list the required area monitor as "Fuel Storage Pool Area Criticality Monitor" and only if you go to the bases for the TRM does it specifically call out 1-RMS-RM-152 as the required monitor.

INCORRECT

B. Leave the fuel building radiation automatic interlock switch in Enable AND
Enter (CR) Condition Report: No other action required.

Plausible selection for leaving the fuel building radiation automatic interlock switch in enable is because the candidate may assume that the only RM that feeds into the fuel building radiation automatic interlock switch comes from 1-RMS-RM-152. 0-AP-5.1 list the actions for 1-RMS-RM-152 first, second possibility is the fact that 1-RMS-RM-152 is associated with TRM 3.3.7 "Radiation Monitoring Instrumentation" and TRM 3.3.9 "Regulatory Guide (RG) 1.97 Instrumentation", 1-RM-RMS-153 is not listed in either TRM. Second part is correct, submitting the CR will be the only other action required because, both 1-RMS-RM-152 and 1-RMS-RM-153 are located in the fuel building but TRM 3.3.7 "Radiation Monitoring Instrumentation" Table 3.3.7-1 list the required area monitor as "Fuel Storage Pool Area Criticality Monitor" and only if you go to the bases for the TRM does it specifically call out 1-RMS-RM-152 as the required monitor.

INCORRECT

C. Place the fuel building radiation automatic interlock switch in Disable AND
Enter action of TR 3.3.7(c)

Plausible selection because first part is correct, based on 1-RMS-RM-153 high indications along with 1-RMS-RM-152 indication being normal, and local reports of normal radiations levels in the fuel Building, the crew will be in 0-AP-5.1 "Common Radiation Monitors System" and would make the assessment, 1-RMS-RM-153 has malfunctioned and the SRO will direct the key switch to be placed in DISABLE, this is a critical decision because going to Disable is required to be done within 2 minutes of receiving the alarm. On a HI-HI alarm 1-RM-RMS-152 or 153 and the key switch in ENABLE position the following will occur: After a 2 minute time delay will automatically dump the MCR bottled air, close the MCR dampers and start the MCR emergency ventilation fans. Second part is incorrect but plausible because, TRM 3.3.7 "Radiation Monitoring Instrumentation" Table 3.3.7-1 list the required area monitor as "Fuel Storage Pool Area Criticality Monitor" and only if you go to the bases for the TRM does it specifically call out 1-RMS-RM-152 as the required monitor the candidate may assume the TRM is referring to 1-RMS-RM-153 .

INCORRECT

D. Leave the fuel building radiation automatic interlock switch Interlock switch in Enable AND
Enter action of TR 3.3.7(c).

Plausible selection for leaving the fuel building radiation automatic interlock switch in enable is because the candidate may assume that the only RM that feeds into the fuel building radiation automatic interlock switch comes from 1-RMS-RM-152. 0-AP-5.1 list the actions for 1-RMS-RM-152 first, second possibility is the fact that 1-RMS-RM-152 is associated with TRM 3.3.7 "Radiation Monitoring Instrumentation" and TRM 3.3.9 "Regulatory Guide (RG) 1.97 Instrumentation", 1-RM-RMS-153 is not listed in either TRM. Second part is incorrect but plausible because, TRM 3.3.7 "Radiation Monitoring Instrumentation" Table 3.3.7-1 list the required area monitor as "Fuel Storage Pool

Area Criticality Monitor" and only if you go to the bases for the TRM does it specifically call out 1-RMS-RM-152 as the required monitor the candidate may assume the TRM is referring to 1-RMS-RM-153 .

K/A:
061AA2.03
Area Radiation Monitoring (ARM) System Alarms
Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms:
Setpoints for alert and high alarms

Technical References:
0-OP-21.11 "Fuel Bldg ventilation lineup for evolutions over spent fuel pool
0-AP-5.1 "Common unit radiation monitoring system"
1-AR-K-D4 (1K-D4) "RAD MONITOR SYST HI-HI RAD LEVEL"
1-AR-K-D2 (1K-D2) "RAD MONITOR SYSTEM HI RAD LEVEL"
TR 3.3.7 "Radiation Monitoring Instrumentation" and bases
TR 3.3.9 Regulatory guide (RG) 1.97 instrumentation
11715-RM-035 (1-RMS-RM-152)
11715-RM-036 (1-RM-RMS-153)
0-LOG-6A, Backboards Logs

References provided to applicants: TR 3.3.7

Learning Objective:
U 5241
List means provided to locally determine high radiation in a monitored area as it applies to the area radiation monitors.

U 5242
Explain the response of all equipment which is directly affected by HIGH-HIGH alarms associated with the New fuel storage area and Fuel pit bridge area Westinghouse area radiation monitors

U 17486
Explain the concepts associated with the Radiation Monitoring Instrumentation technical requirement and bases (TR-3.3.7)

U 18003
Perform the following actions of 0-AP-5.1 "Common Unit Radiation Monitoring Sys"
•Recognize the symptoms and entry conditions

•Explain high level actions, key mitigating strategies, and their basis.

Question Source: NEW

Question History:

Question Cognitive Level:

10 CFR Part 55 Content:
SRO only 10 CFR-55.43 (b)(4)
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
10 CFR-55.45 (a)(13)
Demonstrate the applicants ability to function within the control room team as appropriate to the assigned position, in such a way that the facility licensee's procedures are adhered to and that the limitations in its license and amendments are not violated.

Comments:
This question matches the K/A statement by requiring the SRO applicant to have a ability to determine and interpret setpoints for alert and high alarms as they apply to the Area Radiation Monitoring (ARM) system alarms, in particular 1-RMS-152 "New Fuel Storage Area RM and 1-RM-RMS-153 "Fuel Pit Bridge RM" have an understanding of the bases behind the limits set forth in technical requirement manual. To arrive at the correct answer the SRO applicant must recognize normal and abnormal situation based on indication and field personnel feedback. Have a systems knowledge of applicable actions performed by the RMs. Have a knowledge of steps within abnormal procedure that correspond to this particular failure and then to apply required TRM actions.

6. 085 - WE14EG2.4.21 001/SRO/T1/G2/4.0/4.6/MODIFIED//

Unit-1 experienced a reactor trip and safety injection from 100% power. The following conditions exist:

- The operating crew just entered 1-ECA-1.1 "Loss of Emergency Coolant Recirculation"
- No Containment sump blockage exists
- Containment pressure is 61 psia
- Two recirculation spray pumps are available (Not Running)
- Recirculation spray sump level is 3 feet 0 inches
- Two quench spray pumps are running

Which one of the following describes the proper sequence?

A. Remain in 1-ECA-1.1 "Loss of Emergency Coolant Recirculation"
FRs not applicable during 1-ECA-1.1.

B.✔ Transition to 1-FR-Z.1 "Response to High Containment Pressure"
Do not start any recirculation spray pumps.

C. Remain in 1-ECA-1.1 "Loss of Emergency Coolant Recirculation"
FRs should not be initiated before completion of step 9.

D. Transition to 1-FR-Z.1 "Response to High Containment Pressure"
Start available recirculation spray pumps.

Distractor Analysis:
Based on Loss of Containment Integrity it requires the candidate to have knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc..

A. INCORRECT
Remain in 1-ECA-1.1 "Loss of Emergency Coolant Recirculation"
FRs not applicable during 1-ECA-1.1.

Incorrect but plausible because of the beginning note in 1-ECA-1.1 IF Containment Sump Blockage has occurred, THEN FRs should not be implemented until directed in this procedure. If the candidate is not sure of the details, mitigative strategy or has knowledge of this step then this answer may be chosen.

B. CORRECT
Transition to 1-FR-Z.1 "Response to High Containment Pressure"
Do not start any recirculation spray pumps.

Correct, under the questions given conditions Containment pressure is in the required red path transition path for 1-FR-Z.1 (Containment Pressure > 60 psia and

Containment sump blockage does not exists). However there is a Caution in 1-FR-Z.1 "If 1-ECA-1.1 Loss of Emergency coolant recirculation, is in effect then to preserve RWST inventory, Step 2 through Step 5 of this procedure should not be performed. This caution warns the operator that the operation of the containment spray pumps indicated in guideline 1-ECA-1.1 takes precedence. Step 2 through Step 5 of 1-FR-Z.1 Checks if CDA is required, verifies proper operation of Containment quench spray systems, verifies proper operation of the SW system and verifies proper operation of the Containment Recirc spray system. These steps will not be performed based on the Caution.

C. INCORRECT

Remain in 1-ECA-1.1 "Loss of Emergency Coolant Recirculation"
FRs should not be initiated before completion of step 9

Incorrect but plausible because of the beginning note in 1-ECA-1.1 IF Containment Sump Blockage has occurred, THEN FRs should not be implemented until directed in this procedure. If the candidate is not sure of the details or has knowledge of this step then this answer may be chosen. The statement FRs should not be initiated before completion of step 9 is a requirement listed by a note in 1-ECA-0.1 and is also repeated closely in 1-ECA-0.2

D. INCORRECT

Transition to 1-FR-Z.1 "Response to High Containment Pressure"
Start available recirculation spray pumps.

Incorrect but plausible because, under the questions given conditions, Containment pressure is in the required red path transition path for 1-FR-Z.1 (Containment Pressure $\geq$ 60 psia and Containment Containment sump blockage does not exists). If the candidate does not possess the knowledge of the procedure content in the Caution at the beginning of the procedure "If 1-ECA-1.1 Loss of Emergency coolant recirculation, is in effect then to preserve RWST inventory, Step 2 through Step 5 of this procedure should not be performed", and the candidate is only familiar with the overall mitigative strategy, then this answer could be chosen.

K/A:
WE14EG2.4.21
High Containment Pressure
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Technical References:
1-ECA-1.1 "Loss of Emergency Coolant Recirculation"
Westinghouse Owners Group background information ECA-1.1
1-FR-Z.1 "Response to High Containment Pressure"

1-F-0 "Critical Safety Function Status Tree"

References provided to applicants: None

Learning Objective:
U 13008
Explain why if 1-ECA-1.1 is in effect, the steps which place quench spray in service should not be performed in 1-FR-Z.1

U 5829
Explain how ECA-1.1 is used when containment sump blockage occurs during recirculation mode, including use of FRs

Question Source: Callaway 2007 SRO question 85

Question History: Modified
Given the following conditions:
•A LOCA has occurred.
•Due to several component failures, the crew was required to perform ECA-1.1 "Loss of emergency coolant recirculation.
•The crew is now entering FR-Z.1 "Response to high containment pressure
•Containment pressure is 61 psig and stable
•Both Containment spray pumps are off
•RWST level is 8%

Which (1) ONE of the following describes the strategy for reducing containment pressure?

A. Start both containment spray pumps in accordance with FR-Z.1 Red path take precedence over ECA actions
B. Operate Containment spray pumps in accordance with the guidance in ECA-1.1 as directed by FR-Z.1. Continue in FR-Z.1 until exit criteria is met. (CORRECT)
C. Perform ONLY the FR-Z.1 actions that do NOT conflict with or undo the action taken in ECA-1.1, Two containment Coolers will provide adequate depressurization to meet the containment safety function requirements
D. Do NOT perform actions of FR-Z.1 until the RWST LO LO Level alarm is clear and Containment Spray Pumps may be restarted. Ensure all other automatic actions related to containment isolation have occurred as required.

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
10 CFR-55.41 (a)(b)(7)

Design, components, and functions of control and safety systems including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments:
Question requires the SRO candidate to possess knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. Needs to know content of the procedure verses knowledge of the procedures overall stategy or purpose.

Given the following:

- Unit-1 startup has commenced following welding repairs to S/G blowdown line in containment.
- Reactor Power is at 4% of rated thermal power
- Tavg is 550 degrees being maintained by steam dumps
- Pressurizer Pressure Control is in Automatic and controlling pressure at 2235 psig
- 1-RC-PT-1456 Pressure Protection Channel II fails Hi
- The crew is in 1-AP-3 and is directed to completed 1-MOP-55.73 Pressurizer Pressure Protection Instrumentation.
- 1-MOP-55.73 directs the Unit Supervisor to refer to TS 3.3.1 "Rx trip instrumentation and TS 3.3.2 "Engineered Safety Feature Actuation System instrumentation"

Which instrument **FUNCTION** requires entry into a LCO Condition **AND** what is the **PROTECTION** provided by this function?

	Function	Protection
A.	Overpower Delta-T Reactor Trip	Ensures the design limit DNBR is met
B.	Overtemperature Delta-T Reactor Trip	Ensures the integrity of the fuel
C.✓	Overtemperature Delta -T Reactor Trip	Ensures the design limit DNBR is met
D.	Overpower Delta-T Reactor Trip	Ensures the integrity of the fuel

Distractor Analysis:

This requires the examinee to have the ability to (a) predict the impacts of faulty or erratic operation of detectors and function generators on the RPS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

A) INCORRECT

Function	Protection
Overpower Delta-T Reactor Trip	Ensures the design limit DNBR is met

Function is incorrect but plausible because, TS 3.3.1 Overpower Delta-T is applicable in Modes 1,2 but it does not use Pressurizer Pressure in its calculation like Overtemperature Delta-T does, so it does not require LCO entry Protection is also incorrect but plausible because it would be correct for Overtemperature Delta-T reactor trip

B) INCORRECT

Function	Protection
Overtemperature Delta-T Reactor Trip	Ensures the integrity of the fuel

Function is correct TS 3.3.1 Overtemperature Delta-T setpoints are impacted by RCS pressure and is applicable in Mode 1,2 to place the channel in trip within 72 hours. Protection is incorrect but plausible because it would be correct for Overpower Delta-T reactor trip

C) CORRECT

Function	Protection
Overtemperature Delta -T Reactor Trip	Ensures the design limit DNBR is met

TS 3.3.1 Overtemperature Delta-T setpoints are impacted by RCS pressure and is applicable in Mode 1,2 to place the channel in trip within 72 hours and ensures protection from violating the DNBR limit is met.

D) INCORRECT

Function	Protection
Overpower Delta-T Reactor Trip	Ensures the integrity of the fuel

Function is incorrect but plausible because, TS 3.3.1 Overpower Delta-T is applicable in Modes 1,2 but it does not use Pressurizer Pressure in its calculation like Overtemperature Delta-T does, so it does not require LCO entry. Also Overpower Delta-T limits the required range of Overtemperature Delta-T. Protection is correct, Overpower Delta-T ensures the integrity of the fuel, no fuel pellet melting and < 1% cladding strain.

K/A:
012A2.05
Reactor Protection System (RPS)
Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Faulty or erratic operation of detectors and function generators

Technical References:
1-AP-3 "Loss of Vital Instrumentation"
1-MOP-55.73 "Pressurizer Pressure Protection Instrumentation"
TS 3.3.1 & Bases "Rx Trip System Instrumentation"
TS 3.3.2 & Bases "Engineered Safety Feature Actuation System Instrumentation"

References provided to applicants: None

Learning Objective:
U 9689
Explain the concepts associated with the low pressurizer pressure function of the reactor trip system instrumentation TS and bases

U 9690
Explain the concepts associated with the high pressure function of the reactor trip system instrumentation TS and bases

U 16192
Explain the concepts associated with the low-low pressurizer safety injection function of the reactor trip system instrumentation TS and bases

U 16221
Explain the concepts associated with the low power reactor trips block P-11
Question Source:

U 9687
Explain the concepts associated with the overtemperature delta-T function of the reactor trip system instrumentation TS and bases

U 9688
Explain the concepts associated with the overpower delta-T function of the reactor trip system instrumentation TS and bases

Question Source:
Modification of 2011 Kewanee SRO ILT Exam question. (K/A 012A2.05)

Given the following:
Unit restart has commenced 16 hours after a spurious reactor trip
Rx Pwr is at 4% of rated thermal power
Tave is 547 degrees maintained by the steam dumps
Pressurizer Pressure Control is in auto controlling at 2235 psig
PT-429, Red Channel Pressurizer Pressure, has FAILED HIGH
The crew has implemented AOP-MISC-001, "Response to Instrument Failure"
AOP-MISC-001 directs the unit supervisor to refer to TS 3.3.1 and TS 3.3.2
LCO 3.3.1 Ther RPS instrumentation for each function in table 3.3.1-1 shall be operable
LCO 3.3.2 The ESFAS instrumentation for each function in table 3.3.2-1 shall be operable

Which instrumentation function requires ENTRY into a LCO CONDITION AND what is the protection provided by this function?

	FUNCTION	PROTECTION
A.	Low pressure safety injection function	Prevents exceeding RCS pressure safety limit
B.	Overtemperature Delta-T reactor trip function	Prevents exceeding DNBR limits
C.	Pressurizer Low Pressure reactor trip function	Prevents exceeding subcooling limits during Mode 1 operation
D.	Pressurizer High Pressure reactor trip function	Prevents exceeding the peak centerline fue temperature limit specification in TS

Question History: Kewaunee 2011

Question Cognitive Level: Memory and Fundamental Knowledge

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(5)
Assessment fo facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations
10 CFR-55.41 (a)(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and ractivity changes, effects of load changes, and operating linitations and reasons for these operating characteristics.

Comments:
This question matches the K/A statement by requiring the SRO applicant to analysis and predict the impacts of faulty or erratic operation of Pressure Protection channel failure on the RPS. To arrive at the correct answer the SRO applicant must recognize which affected protection requires entry into a LCO based on the questions given conditions and have knowledge of the Technical Specification basis for each functions protection provided.

7. 087 - 013A2.04 001/SRO/T2/G1/3.6/4.2/BANK//

The actions of 0-AP-41 "Severe Weather Conditions" are being performed due to a thunderstorm warning, when lightning strikes the SWYD and a loss of offsite power occurs.

- 1-ECA-0.0 "Loss of all AC power" was entered.
- Operations have completed all actions up to determining the correct recovery method and procedure.
- One AC emergency bus has been energized.
- RCS subcooling is 35 DEGF.
- Pressurizer level is 5%.
- Adverse Containment Conditions do not exist.
- Safeguards equipment is not operating.

Which answer will complete the following statement?

SRO will direct procedure transition to __________ .

A. 1-ES-0.0 "Re-Diagnosis", to determine or confirm the most appropriate post-accident recovery procedure.

B. 1-ECA-0.1 "Loss of all AC power without SI required", recover the plant based on stable RCS conditions following a lose of all AC power.

C. 1-E-0 "Reactor trip or safety injection", verify proper response of Reactor Protection and safety injection following restoration of at least one emergency bus.

D.✓ 1-ECA-0.2 "Loss of all AC power recovery with SI required" to manually load safeguards equipment as appropriate.

Distractor Analysis:
This requires the examinee to have the ability to (a) predict the impacts of the loss of Instrument bus on the Engineered Safety Features Actuation System (ESFAS); and (b) based on those predictions. use procedures to correct, control, or mitigate the consequences of those malfunction or operations.

A. INCORRECT
1-ES-0.0 "Re-Diagnosis", to determine or confirm the most appropriate post-accident recovery procedure.

Incorrect but plausible because the candidate may decide, since one emergency bus has been recovered, the crew no longer needs to be in the 1-ECA-0 series procedures, and with a low pressurizer level, 1-E-1 should be your recovery procedure and the quickest way is through 1-ES-0.0 "Re-Diagnosis" or 1-ES-0.0 could be used to determine/confirm the most appropriate post-accident recovery procedure the crew should be in. 1-ES-0.0 should only be used when SI is in service or is required and 1-E-0 Reactor Trip or Safety injection has been exited. Based on the given condition entry into the EOP was directly through 1-ECA-0.0 so 1-ES-0.0 can not be used.

B. INCORRECT

1-ECA-0.1 "Loss of all AC power without SI required", recover the plant based on stable RCS conditions following a lose of all AC power.

Incorrect but plausible if the candidate reflects only on Subcooling and the fact that one emergency bus has been recovered and safeguards equipment is not operating, but following the restoration of AC power, selection depends on the existence of RCS subcooling 25DEGF [75], existence of pressurizer level > 21% [26], and verification that SI equipment has not automatically actuated upon power restoration. If RCS conditions have not deteriorated significantly (i.e. all criteria satisfied), the operator is directed to guidline ECA-0.1 to recover the plant using normal operational systems.

C. INCORRECT

1-E-0 "Reactor trip or safety injection", verify proper response of Reactor Protection and emergency core cooling systems following restoration of at least one emergency bus.

Incorrect but plausible because the candidate may think that since one emergency bus has been recovered, the crew no longer needs to be in the 1-ECA-0 series procedures and there is a need to verify that the Rx is tripped and initiate SI if required.

D. CORRECT

1-ECA-0.2 "Loss of all AC power recovery with SI required" to manually load safeguards equipment as appropriate.

Correct, 1-ECA-0.0 provides recovery guidelines based on existing RCS conditions. The criteria for recovery selection includes (1) existence of RCS subcooling (>25 DEGF), (2) existence of Pressurizer level (>21%) and (3) verification that SI equipment has not automatically actuated upon ac power restoration, which would occur if power was restored after SI signal actuation but before SI signal was reset. If **any** of the criteria is not satisfied the operator is directed to 1-ECA-0.2. This procedure will start safeguards equipment as appropriate and then direct the operator to guideline 1-E-1 "Loss of Reactor or Secondary Coolant, for subsequent recovery actions.

K/A:
013A2.04
Engineered Safety Features Actuation System (ESFAS)
Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations;
Loss of instrument bus

Technical References:
1-ECA-0.0 "Loss of all AC power"
1-ECA-0.0 Westinghouse owners group background document.
1-ECA-0.1 "Loss of all AC power recovery without SI required".
1-ECA-0.1 Westinghouse owners group background document.
1-ECA-0.2 "Loss of all AC power recovery with SI required".
1-ECA-0.2 Westinghouse owners group background document.

References provided to applicants: None

Learning Objective:
U 13830
List conditions that result in leaving 1-ECA-0.0 "Loss of all ac power"

U 9630
Explain why SI is reset after depressurization of S/G's

U 13837
List concepts of 1-ECA-0.1 "Loss of all ac power without SI required"

U 13838
List concepts of 1-ECA-0.2 "Loss of all ac power with SI required"

Question Source: NAPS Vision Bank (7999)

Question History:
Assume the following plant conditions:
- ECA-0.0 is in effect due to a loss of all AC power
- Operators have completed all actions up to determining the correct recovery method and procedure
- One AC emergency bus is energized
- The RCS is 44DEGF subcooled
- PZR level is 50%
- SFGD's equipment is not operating

The operator should go to procedure _______________.

A. ECA-0.1, Loss of all ac power recovery W/O SI required, because all conditions required are met
B. ES-0.0, Re-Diagnosis, to determine or confirm the most appropriate post-accident recovery procedure
C. ECA-0.2, Loss of all ac power recovery with SI required, to use engineered safeguards equipment to recover plant conditions following restoration of AC emergency power to at least one bus
D. E-0, Rx trip or SI, to verify proper response of the reactor protection and emergency core cooling systems following restoration of AC emergency power to at least on bus.

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
10 CFR-55.41 (a)(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes,

effects of load changes, and operating limitations and reasons for these characteristics

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the ability to (a) predict the impacts of the loss of Instrument bus on the Engineered Safety Features Actuation System (ESFAS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunction or operations. To arrive at the correct answer the SRO applicant must demonstrate the ability to make the proper diagnostics bases on given conditions, and then use the information to transition to an event specific procedure that address the plant configuration based on ESFAS systems and electrical availability.

Unit-1 was operating at 100% power when the following sequence of events occurred:

- Annuciator 1K-G6, N-16 RAD DET, alarms.
- N-16 indication on "B" S/G AND Main Steam Header are in alarm and trending up.
- 1-RM-SV-121, Condenser Air Ejector Radiation Monitor indication has not changed.
- PRZR level is rapidly decreasing.
- Controller demand on 1-FW-FCV-1488 "B MFRV" shows a slight decrease.
- The crew trips the Reactor and initiates SI.

Current Status
- The immediate actions of 1-E-0, Reactor Trip or Safety Injection have just been completed,
- All S/G are below Narrow Range level indication
- The BOP recommends isolating AFW flow to "B" S/G

Which ONE of the choices below states the actions that the SRO will direct the crew to perform?

A. Isolate AFW to "B" S/G;
Contact chemistry and HP for confirmation of the affected S/G prior to initiating 1-E-0, Attachment 8 "Ruptured S/G isolation.

B. Isolate AFW to "B" S/G;
Initiate 1-E-0, attachment 8, Ruptured S/G isolation

C. Do not isolate AFW to "B" S/G;
Contact chemistry and HP for confirmation of the affected S/G prior to initiating 1-E-0, Attachment 8 "Ruptured S/G isolation

D✓ Do not isolate AFW to "B" S/G;
Initiate 1-E-0, Attachment 8, "Ruptured S/G isolation.

Distractor Analysis:
Ability to (a) predict the impacts of indications and alarms for main steam and area radiation monitors (during SGTR) on the MRSS and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

A INCORRECT
Isolate AFW to "B" S/G;
Contact chemistry and HP for confirmation of the affected S/G prior to initiating 1-E-0, Attachment 8 "Ruptured S/G isolation.

First part is plausible because reducing AFW flow could confirm to the crew that indeed "B" S/G has a tube rupture. The Westinghouse background document mentions "Since primary-to-secondary leakage adds additional inventory which accumulates in the ruptured steam generator, level should return to the narrow range in that steam generator(s) significantly earlier and will continue to increase more rapidly. This response provides confirmation of a steam generator tube rupture event and also identifies the affected steam generator(s)". Second part is also plausible because 1-RM-SV-121 should be showing something and it is not (malfunction); the candidate who does not feel N-16 indication is enough to go on may delay action that are important to limiting offsite dose (attachment 8 isolates steam supply to AFW terry turbine, which is a local action). Also the continuous action page for 1-E-0 (step 7) states Attachement 8 <u>MAY</u> be used, it does not directly instruct the operator to use it.

B INCORRECT

Isolate AFW to "B" S/G;
Initiate 1-E-0, attachment 8, Ruptured S/G isolation.

First part is plausible because reducing AFW flow could provide the crew with more time before over filling the S/G, but knowledge of Westinghouse background guidelines provides the importance of maintaining the water level above the top of the U-tubes. 1) Provides Insulation between the ruptured steam generators steam space and the U-tubes. 2) Ensures a heat sink is available if no other intact steam generator is available. 3) Prevents misdiagnosis of the ruptured steam generator due to imbalances in feedwater flow. Second part is correct, attachment 8 should be implemented for identification (which is confirmed by both "B" S/G and Main Steam header N-16 Radiation monitor indication along with a slight decrease of demand on 1-FW-FCV-1488) and isolation which is important to limiting offsite dose.

C INCORRECT

Do not isolate AFW to "B" S/G;
Contact chemistry and HP for confirmation of the affected S/G prior to initiating 1-E-0, Attachment 8 "Ruptured S/G isolation.

First part is correct need U-tubes covered before isolation. Second part is incorrect but plausible because 1-RM-SV-121should be showing something and it is not (malfunction); the candidate who does not feel N-16 indication is enough to go on may delay action that are important to limiting offsite dose (attachment 8 isolates steam supply to AFW terry turbine, which is a local action.

D CORRECT

Do not isolate AFW to "B" S/G;
Initiate 1-E-0, Attachment 8, "Ruptured S/G isolation".

First part is correct the Westinghouse background documents Basis describes the requirement to maintain the S/G U-tubes covered before isolation 1) Provides Insulation between the ruptured steam generators steam space and the U-tubes. 2) Ensures a heat sink is available if no other intact steam generator is available. 3) Prevents misdiagnosis of the ruptured steam generator due to imbalances in feedwater flow. and second part is correct because attachment 8 should be implemented for identification (which is confirmed by both "B" S/G and Main Steam header N-16 Radiation monitor indication along with a slight decrease of demand on 1-FW-FCV-1488) and isolation which is important to limiting offsite dose.

039A2.03
Main and Reheat Steam System (MRSS)
Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Indications and alarms for main steam and area radiation monitors (during SGTR)

Technical References:
1-AP-24 "S/G tube leak"
1-AP-5 "Unit-1 Radiation Monitoring System
1-E-0 and Westinghouse Owners Group Background document
1-E-3 and Westinghouse Owners Group Background document
1-EI-CB-21K Annunciator G6
TR 3.4.4 Primary to Secondary Leakage
TR 3.4.4 Primary to Secondary Leakage Bases

References provided to applicants: None

Learning Objective:
U 18027
Perform actions of 1-AP-24 "Steam Generator Tube leak"

U 18002
Perform actions of 1-AP-5 "Unit-1 Radiation Monitoring System"

U 12174
Given a set of plant conditions, determine the appropriate recovery procedure

U 12677
Explain the concepts associated with identifying and isolating a ruptured steam generator in accordance with 1-E-3

Question Source: Modified Bank (2012 North Anna SRO written Exam, Question #82)

Question History:
<u>Unit 1 was operating at 100% power when the following sequence of events occurred</u>:
- Annunciator K-G6, N-16 RAD DET, alarms
- RO reports N-16 modules for "C" SG AND Main Steam Header are in alarm and trending up
- BOP reports 1-RM-SV-121, Condenser Air Ejector Radiation Monitor indication has not changed
- RO reports PRZR level rapidly decreasing
- The crew trips the reactor and initiates SI

<u>Current Status</u>:
- The immediate actions of 1-E-0 Reactor Trip of Safety Injection, have just been completed.
- All SG's are below the Narrow Range.
- The BOP recommends isolating AFW flow to the "C" SG.

Based on the above, which ONE of the following identifies how the US should proceed?

A. allow the BOP to isolate AFW to "C" SG; contact HP & Chemistry for confirmation of the affected SG prior to initiating 1-E-0, Attachment 8, Ruptured SG Isolation.

B. allow the BOP to isolate AFW to "C" SG; have the RO initiate 1-E-0, attachment 8, Ruptured SG Isolation.

C. DO NOT allow the BOP to isolate AFW to "C" SG; contact HP & Chemistry for confirmation of the affected SG prior to initiating 1-E-0, Attachment 8, Ruptured SG Isolation.

D. DO NOT allow the BOP to isolate AFW to "C" SG; have the RO initiate 1-E-0, attachment 8, Ruptured SG Isolation.

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
10 CFR-55.41 (a)(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:
This question matches the K/A statement by requiring the SRO applicant to possess the ability to predict the impacts of indications and alarms for main steam and area radiation monitors (during SGTR). To arrive at the correct answer the SRO must be aware of the radiation monitor flowpath within the Main Steam system. The SRO must also show knowledge of diagnostic steps, and decision points, based on indications the candidate is given by plant conditions, and then use this information in deciding when and how to implement attachments of the Emergency Operating Procedures.

Unit-2 has been at 100% power for 90 days following a 26 day refueling outage
Unit-1 is currently in Day 2 of a scheduled refueling outage, with the following conditions:

- RHR discharge Temp = 115 DEGF
- PZR level is at 28% and on VCT float
- RCS loop stop valves are all open

At 01:30 due to an Electrical fault a fire starts in the SW pump house. The following conditions exist;

- The crew is performing actions of 0-FCA-9 "Service Water Pump House Fire"
- All SW pumps have been secured.
- Component Cooling temperatures are rapidly increasing.
- Annunciator SFP Hi/Hi-Hi Temp has illuminated.
- Spent fuel pool level is +1 inches.
- 01:55 Fire Brigade Scene Leader reports that the fire has been extinguished at the SWPH.

Per Attachment 7 of 0-AP-27 "Malfunction of the Spent Fuel Pit System" the expected heat up rate for the spent fuel pit under these conditions is ____(1)_____ DEGF per hour

And

Based on conditions and not on Shift Manager judgment what is the highest EAL classification that will be entered ___________ ?

REFERENCE PROVIDED

A✓ (1) 3.20
(2) Alert

B. (1) 3.99
(2) NOUE

C. (1) 3.20
(2) NOUE

D. (1) 3.99
(2) Alert

Distractor Analysis:

The candidate is required to have the ability to interpret reference materials, such as graphs, curves, tables, etc.. that would be associated with the Service Water system.

A. CORRECT
(1) 3.20
(2) Alert

First part is correct 3.20 heat up rate is correct using the table Conditions for Back to Back Refuelings, which is correct for less than 120 days between refuelings. Second part is also correct the highest EAL classification from the table will be HA2.1 Fire or explosion in table H-1 area AND Plant personnel report visible damage to any safety related structure, system, or component within the area OR Affected system parameter indications show degraded performance As per the Bases for the EAL the wording of this EAL does not imply that an assessment of safety related structure system and component performance should be performed; rather that safety related structure system and component are degraded as a result of the event. In this case all four SW pumps have been affected. Or CA3.1 An unplanned event results in RCS temperature > 200 DEGF for > Table C-3 duration (Note 3 = The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.)

B. INCORRECT
(1) 3.99
(2) NOUE

The first part is incorrect but plausible if the candidate incorrectly uses the table based on conditions for Non-Back to Back Refueling outages, then the expected heat up rate would be 3.99 DEGF/hour. Second part is also incorrect but plausible if the candidate selects EAL HU2.1 from the table due to the fire at the SWPH not extinguished within 15 min. or CU3.1 An unplanned event results in RCS temperature > 200 DEGF

C. INCORRECT
(1) 3.20
(2) NOUE

First part is correct See above B answer. Second part is incorrect See above answer A

D. INCORRECT
(1) 3.99
(2) Alert

First part is incorrect See above answer A. Second part is correct See above answer B

K/A:
076G2.1.25

Service Water System (SWS)
Ability to interpret reference materials, such as graphs, curves, tables, etc.

Technical References:
EAL Bases
EAL Table
0-AP-27 "Malfunction of spent fuel pit system"
0-AP-12 "Loss of Service Water"
0-FCA-9 "Service Water Pump House Fire"
1-EI-CB-21E Annuciator C5 SFP HI/HI-HI TEMP (1-AR-E-C5)
11715-FM-79C (sh 3 of 5)
UFSAR 9.1.3.3.1

References provided to applicants:
• 0-AP-27 Attachment 7 "Spent Fuel Pool Heat-Up rate following loss of cooling"
• EAL Tables

Learning Objective:
U 18030
Perform various actions of 0-AP-27 "Malfunction of spent fuel pit system"

U 18011
Perform the actions of 0-AP-12 "Loss of Service Water"

U 13910
Explain the concepts associated with responding to a fire in the service water pump house in accordance with 0-FCA-9

U 13699
Explain importance of correctly applying classification criteria to actual plant conditions when classifying an emergency.

Question Source: Modified NAPS Vision Bank

The following conditions exist
Unit-1 is currently in a scheduled refueling outage
The refueling team is currently latching control rods in the unit one containment
Unit 2 has been at 100% power for 3 month following a 26 day refueling outage
Both SFP cooling pumps are currently isolated due to a leak on a common line
Annunciator E-C5, SFP Hi/Hi-Hi TEMP has illuminated

Per 0-AP-27 "Malfunction of Spent Fuel Pit System" the expected heat-up rate for the spent fuel pit under these conditions is ________ degrees Fahrenheit per hour (F/hr)

A. 3.99
B. 4.29 (Correct)
C 9.91
D 11.40

Question History:None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43.(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
10 CFR-55.41.(a)(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the ability to interpret reference materials, such as graphs, curves, tables, etc. associated with the SW system. On a loss of all service water, component cooling temperatures are expected to increase, with one unit having just been shutdown, RHR in service and the large Decay Heat factor, this will have a large effect on the Spent Fuel Pit temperatures. To arrive at the correct answer the SRO applicant must posses the ability to determine the proper heat up rate of the Spent Fuel Pool based on the ability to use the correct table for the questions given conditions, and then determine the correct heat up rate based on the status of the core. Second the SRO candidate must be able to demonstrate the correct use of the EAL tables to determine the highest EAL classification for the stated conditions.

8. 090 - 103G2.2.38 001/SRO/T2/G1/3.6/4.5/NEW//

Unit-1 is at 100%, when Annuciator 1J-G1 "Containment Partial Pressure +0.25 PSI CH I-II", locks in with the following conditions:

- RWST = 48 DEGF
- Service Water Temp = 83 DEGF
- Partial Pressure **Setpoint** is set at 10.85 psia on Ch I and II
- Containment Temperature = 96.0 DEGF

Which of the following completes the statement?

Based on these conditions, containment air partial pressure is ____________ for operations AND the limits on pressure and temp ensures that the ___________.

A. Acceptable,
Peak containment pressure will be limited to the upper containment pressure of 45 psig

B. Unacceptable,
Containment pressure will not reach the lower design pressure of 8.6 psia for an inadvertent containment spray actuation.

C.✓ Unacceptable,
Peak containment pressure will be limited to the upper containment pressure of 45 psig

D. Acceptable,
Containment pressure will not reach the lower design pressure of 8.6 psia for an inadvertent containment spray actuation.

Distractor Analysis:
The candidate is required to have the knowledge of conditions and limitations in the facility license as they relate to containment.

A. INCORRECT

Acceptable, Peak containment pressure will be limited to the upper containment pressure of 45 psig.

First part is incorrect but is plausible because the partial pressure is given to the candidate as the alarm which is +0.25 PSI greater than the setpoint which is given as 10.85 psia, so the sum of the two (11.1 psia) is the point used on the graph. if the candidate does not add the two then the point will fall in the acceptable region. Second part is the correct limit as per the bases of TS 3.6.4 "Containment Pressure". If the candidate is not familiar with the bases value, it can also be found in section 5.5.15(b) Administrative Controls for Containment Leakage Rate Testing Program.

B. INCORRECT

Unacceptable, Containment pressure will not reach the lower design pressure of 8.6 psia for an inadvertent containment spray actuation.

First part is correct the pressure falls in the unacceptable region of the gragh Figure 3.6.4-1 (alarm which is +0.25 PSI greater than the setpoint which is given as 10.85 psia, so the sum of the two (11.1 psia) is the point used on the graph). and the second part in incorrect, but plausible because, the design pressure is 5.5 psia, but the analysis calculation used in the TS bases shows the minimum pressure that should be reached inside of containment would be 8.6 psia.

C. CORRECT
Unacceptable, Peak containment pressure will be limited to the upper containment pressure of 45 psig

First part is correct the pressure falls in the unacceptable region of the gragh Figure 3.6.4-1 (alarm which is +0.25 PSI greater than the setpoint which is given as 10.85 psia, so the sum of the two (11.1 psia) is the point used on the graph). and the Second part is the correct limit as per the bases of TS 3.6.4 "Containment Pressure". If the candidate is not familiar with the bases value, it can also be found in section 5.5.15(b) Administrative Controls for Containment Leakage Rate Testing Program.

D. INCORRECT
Acceptable, Containment pressure will not reach the lower design pressure of 8.6 psia for an inadvertent containment spray actuation.

First part is incorrect but is plausible because the partial pressure is given to the candidate as the alarm which is +0.25 PSI greater than the setpoint which is given as 10.85 psia, so the sum of the two (11.1 psia) is the point used on the graph. if the candidate does not add the two, the point will fall in the acceptable region. The second part is incorrect, but plausible because, the design pressure is 5.5 psia, but the analysis calculation used in the TS bases shows the minimum pressure that should be reached inside of containment would be 8.6 psia.

K/A:
103G2.2.38
Containment System
Knowledge of conditions and limitations in the facility license.

Technical References:
1-AR-J-G1 "Containment Partial Press +0.25 PSI CH I-II"
11715-LM-013
11715-LM-014
1-AP-18 "Increasing containment pressure"
TS 3.6.4 Containment Pressure and Bases
TS 3.6.4 Figure 3.6.4-1

References provided to applicants: Figure 3.6.4-1 of TS 3.6.4

Learning Objective:
U 5756 List the following information concerning the containment stucture.

• Design Pressure

U 16273
Explain the concepts associated with the containment pressure technical specification and bases

U 16274
Explain the concepts associated with the containment air temperature technical specification and bases

U 18017
Perform actions of 1-AP-18 "Increasing Containment Pressure"

U 4375
List all indications available to the control room operator for abnormal containment pressure, including all annunciator setpoints.

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only 10 CFR 55.43 (b)(1)
Conditions and limitations in the facility license
10 CFR 55.45 (a)(13)
Demostrate the applicant's ability to function within the control room team as appropriate to the assigned position, in such a way that the facility licensee's procedures are adhered to and that the limitations in its license and amendments are not violated
10 CFR 55.41 (a)(b)(10)
Administrative, normal, abnormal, and emergency procedures for the facility
10 CFR 55.41 (a)(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failures modes , and automatic and manual features.

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the knowledge of conditions and limitations in the facility license as they relate to containment. To arrive at the correct answer the SRO applicant must have the knowledge to correctly assess plant conditions, and interpret those conditions, to determine the proper technical specification requirement that needs to be applied. Also the applicant is asked to possess the knowledge of the importance behind the requirement as outlined in the technical specifications bases document.

Unit-1 status is as follows:

- A reactor startup is in progress.
- Both Intermediate Range channels indicate approximately 5 E -11 amps.
- Source Range Channel N-31 fails low.
- The Crew initiates 1-AP-4.1 "Malfunction of Source Range Nuclear Instrumentation".

Which One of the following describes (1) the 1-AP-4.1 required response and (2) the Technical Specification Bases for the response?

A. (1) Continue the reactor startup;
(2) With only one source range channel operable 48 hours is allowed to restore two channels to service, Protection is provided by the logic of 1/2 SR channels > 10 E 5 cps.

B. (1) Suspend the reactor startup;
(2) One decade of overlap between both SR and IR channels cannot be verified.

C.✓ (1) Suspend the reactor startup;
(2) With only one source range channel operable, a single random failure will disable the SR Trip function.

D. (1) Continue the reactor startup;
(2) Intermediate Range and Power Range low setpoint trip functions provide the necessary redundant protection.

Distractor Analysis:

This requires the examinee to have the knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

A. INCORRECT
(1) Continue the reactor startup;
(2) With only one source range channel operable 48 hours is allowed to restore two channels to service. Protection is provided by the logic of 1/2 SR channels > 10 E 5 cps.

The first part is incorrect but plausible if the operator is unfamiliar with the abnormal procdedure guidance associated with the SR failing Low. The Trip Logic is 1/2 SR channels > 10 E 5 cps, so a Rx trip will not occur. The fact that this occurs at < P-6 value (1 X 10 E -10 amps) then the AP checks Reactor power constant, and then restores affected channel to service before exceeding P-6 permissive. Because the procedure does not required inserting rods or opening the Rx trip breakers, it could make this a plausible choice for the candidate. The second part is also incorrect but a plausible choice matched with the first answer because T.S. 3.3.1, Table 3.3.1-1 also list action J as a condition associated with one source range channel inoperable. If the candidate associates this completion time of 48 hours with a Mode 2 requirement then this answer could be matched together with the first part and be chosen. Also combining in the idea that protection is still being provided with the Rx trip logic also makes this a plausble Tech Spec bases choice.

B. INCORRECT

(1) Suspend the reactor startup;

(2) One decade of overlap between both SR and IR channels cannot be verified.

The first part is correct. The source range trip logic is 1/2 SR channels > 10 E 5 cps, so a Rx trip will not occur. The fact that this occurs at < P-6 value (1 X 10 E -10 amps) then the AP checks Reactor power constant and then restores affected channel to service before exceeding P-6. The second part is incorret but plausible because the start up procedures (OP-1.5 / 1.7) has the operators verify that proper overlap is occuring between the SR and IR instrumentation before maually blocking the Source Range Neutron Flux trip at the P-6 permissive. The candidate may assume this is listed in the bases for the SR intrumentation.

C. CORRECT

(1) Suspend the reactor startup;

(2) With only one source range channel operable, a single random failure will disable the SR Trip function.

The first part is correct. The source range trip logic is 1/2 SR channels > 10 E 5 cps, so a Rx trip will not occur. The fact that this occurs at < P-6 value (1 X 10 E -10 amps) then the AP checks Reactor power constant and then restores affected channel to service before exceeding P-6. The second part is also correct, The Tech Spec 3.3.1 bases for SR specifies " Two Operable channels are sufficient to ensure no single random failure will disable this Trip function.

D. INCORRECT

(1) Continue the reactor startup;

(2) Intermediate Range and Power Range low setpoint trip functions provide the necessary redundant protection.

The first part is incorrect but plausible if the operator is unfamiliar with the abnormal procdedure guidance associated with the SR failing Low. The Trip Logic is 1/2 SR channels > 10 E 5 cps, so a Rx trip will not occur. The fact that this occurs at < P-6 value (1 X 10 E -10 amps) then the AP checks Reactor power constant, and then restores affected channel to service before exceeding P-6 permissive. Because the procedure does not required inserting rods or opening the Rx trip breakers, it could make this a plausible choice for the candidate. Second part is incorrect but plausible and could be matched with the first part because the basis associated with SR neutron flux mentions "Above P-6 setpoint, the IR and PR low setpoint trip will provide core protection for reactivity accidents.

K/A:
015G2.2.25
Nuclear Instrumentation System
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Technical References:
Technical specification basis 3.3.1 "Reactor Trip System (RTS) Instrumentation"

1-AP-4.1 "Malfunction of Source Range Nuclear Instrumentation"
1-OP-1.7 "Unit Startup from Mode 3 to Mode 2 following refueling"
1-OP-1.5 "Unit Startup from Mode 3 to Mode 2"

References provided to applicants: None

Learning Objective:

Question Source: Modified Bank Wolf Creek 2006 SRO ILT exam #96 (015G2.2.25)

Given the following conditions:
A Rx startup is in progress
Both IR channels indicate approximately 5 E -11
SR channel N-31 fails DOWNSCALE

Which ONE (1) on the following describes the required operator response and the Technical Specification basis for the response?

A. Continue the Rx startup; with only one source range channel operable; 48 hours is allowed to restore two channels to service
B. Suspend the reactor startup; SR channels are not required to trip the reactor however the SR monitoring functions must be available.
C. Continue the reactor startup; the IR flux trip and the PR flux low trp provide the necessary core protection
D. Suspend the Rx startup; with only one SR channel operable, the minimum required SR high Flux Trip protection is not met. (CORRECT)

Question History: None

Question Cognitive Level: Memory

10 CFR Part 55 Content:
SRO only 10 CFR-55.43(b)(2)
Facility operating limitations in the technical specifications and their bases
10 CFR-55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments:
This question matches the K/A statement by requiring the SRO applicant to demonstrate a knowledge of the bases in technical specifications for limiting conditions for operations and safety limits. To arrive at the correct answer the SRO applicant must recognize the operational requirements, associated with a loss of the source range nuclear instrumentation, as they would apply to a Reactor Startup. Along with the technical specification bases behind the limit imposed.

A LOCA has occurred on Unit-1 with the following conditions:

- Crew is in 1-E-1 "Loss of Reactor or Secondary Coolant".
- The crew is at Step 19 to check containment Hydrogen Concentration.
- Containment Pressure is 22 psia
- Containment Hydrogen Concentration is 3.5%
- The Technical Support Center is fully manned and activated.

Which of the following answers completes the statement?

The Analyzer will provide an accurate reading _____(a)_____ minutes after being placed in service and based on containment Hydrogen concentration the Hydrogen recombiner will _____(b)______ .

A. (a) 15
(b) be placed in service, using 1-OP-63.1 "Post Accident thermal H2 Recombiner"

B. (a) 15
(b) not be placed in service, Consult TSC for additional recovery actions

C.✓ (a) 5
(b) be placed in service, using 1-OP-63.1 "Post Accident thermal H2 Recombiner"

D. (a) 5
(b) not be placed in service, Consult TSC for additional recovery actions

Distractor Analysis:
Malfunctions or operations on the HRPS; and (b) based on those predictions, use

procedures to correct, control or mitigate the consequences of those malfunctions or operations: LOCA condition and related concern over hydrogen.

A INCORRECT
(a) 15
(b) be placed in service, using 1-OP-63.1 "Post Accident thermal H2 Recombiner".

First part is incorrect but plausible if the candidate in unfamiliar with the Note in 1-E-1 just prior to step 19 to place the analyzer in service "There is a delay time of 5 minutes before the containment Hydrogen Analyzer will provide an accurate reading. Second part is correct concentration set point for placing the Hydrogen Recombiner in service is between 0.5% and 4%, for any concentration above or below this band, the procedures require the operators to contact the TSC/Plant staff for recovery actions. Even though the question provides a containment pressure of 22 psia, there are no alternate max and min setpoints associated with placing the recombiner in service under adverse containment conditions.

B INCORRECT
(a) 15
(b) not be placed in service, Consult TSC for additional recovery actions.

First part is incorrect but plausible if the candidate in unfamiliar with the Note in 1-E-1 just prior to step 19 to place the analyzer in service "There is a delay time of 5 minutes before the containment Hydrogen Analyzer will provide an accurate reading. Second part is incorrect but plausible because the recombiner can be placed in service at a concentration between 0.5% and 4%, but the given containment pressure of 22 psia is equal to an adverse containment condition, this could draw in the candidate, and provides a test on their knowledge of the adverse containment requirements. Older revisions of the emergency procedures had an adverse containment pressure maximum concentration value of 1.6% for placing the Recombiner in service.

C INCORRECT
(a) 5
(b) be placed in service, using 1-OP-63.1 "Post Accident thermal H2 Recombiner".

First part is correct Note in 1-E-1 just prior to step 19 to place the analyzer in service "There is a delay time of 5 minutes before the containment Hydrogen Analyzer will provide an accurate reading. Second part is correct concentration set point for placing the Hydrogen Recombiner in service is between 0.5% and 4%. Even though the question provides a containment pressure of 22 psia, there are no alternate max and min setpoints associated with placing the recombiner in service under adverse containment conditions.

D CORRECT
(a) 5
(b) not be placed in service, Consult TSC for additional recovery actions.

First part is correct Note in 1-E-1 just prior to step 19 to place the analyzer in service "There is a delay time of 5 minutes before the containment Hydrogen Analyzer will provide an accurate reading. Second part is incorrect but plausible because the recombiner can be placed in service at a concentration between 0.5% and 4%, but the given containment pressure of 22 psia is equal to an adverse containment condition, this could draw in the candidate, and provides a test on their knowledge of the adverse containment requirements. Older revisions of the emergency procedures had an adverse containment pressure maximum concentration value of 1.6% for placing the recombiner in service.

K/A:
092 - 028A2.02

Technical References:
1-E-1 "Loss of Reactor and Secondary Coolant" and Westinghouse background information
1-ECA-1.1"Loss of Emergency Coolant Recirculation" and Westinghouse background information
1-ECA-3.1"SGTR with Loss of Reactor Coolant Subcooled Recovery Desired" and Westinghouse background information
1-ECA-3.2 "SGTR with Loss of Reactor Coolant Saturated Recovery Desired" and Westinghouse background information
1-ES-1.2 "Post LOCA cooldown and Depressurization" and Westinghouse background information
1-FR-C.1 "Response Inadequate Core Cooling" and Westinghouse background information
1-FR-I.3 "Responds to voids in the Reactor Vessel" and Westinghouse background information
1-OP-63.1 "Post Accident Thermal Hydrogen Recombiner"
1-OP-63.2 "Containment Hydrogen Analyzer"
11715-FM-106A Sht 4

References provided to applicants: None

Learning Objective:
U 5451
List the design basis of the Containment Atmosphere Cleanup System

U 5340
Assuming that the purge blowers and hydrogen recombiner are operable, explain the conditions under which the purge blowers and not the recombiner would be used for removing hydrogen from containment during a LOCA

U 5450
List information associated with hydrogen in containment

U 17487
Explain the concepts associated with the Containment Hydrogen Analyzers technical

Question Source: Modified St. Lucie 2012 SRO ILT exam bank (K/A 028A2.02)

Question History:
Unit 1 has entered 1-EOP-03, LOCA with the following conditions:
• Containment pressure is 12 psig
• All containment coolers and Containment spray pumps are operating
• Containment Hydrogen concentration is 4.5%

The TSC is fully manned and plant conditions have been communicated to the TSC

Which ONE of the following states the status of the Containment Temperature & Pressure Safety function AND when should the H2 purge system or Hydrogen Recombiners be placed in service?

1) Containment Temperature & Pressure Safety function is:
2) The H2 purge system or Hydrogen Recombiner should be placed in service:

A. 1) Met
   2) only as directed by the Technical Support Center
B. 1) Met
   2) when the Unit Supervisor deems it necessary based on Containment H2 concentration
C. 1) Not Met (CORRECT)
   2) only as directed by the Technical Support Center
D. 1) Not Met
   2) when the Unit Supervisor deems it necessary based on Containment H2 concentration

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal abnormal, and emergency situations.
10 CFR-55.41 (a)(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes and operating limitations and reasons for these operating characteristics.

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the ability to correctly direct operation of the Hydrogen Recombiner under LOCA conditions. To arrive at the correct answer the SRO applicant must have the knowledge to determine when accurate and reliable information is being obtained by available indications, which will be used to coordinate the crew to properly make the decision concerning operation of the Hydrogen Recombiner and coordination with supporting staff members.

9. 093 - 071G2.4.21 001/SRO/T2/G2/4.0/4.6/BANK/NAPS 2010/

The Waste Gas Decay Tanks (WGDT) was sampled to determine the quantity of radioactive material contained in each gas storage tank.

<u>Confirmed sample results are as follows:</u>
- "A" WGDT – 7,000 curies of noble gas
- "B" WGDT – 16,000 curies of noble gas

Based on these sample results, which ONE of the following identifies the implications per TR 3.10.3 Gas Storage Tanks?

The quantity of radioactive material in the WGDTs ____(1)______. The TRM bases for restricting the quantity of radioactive material is to limit dose at the exclusion boundary to _____(2)______.

A. (1) is NOT within limits, enter actionTR 3.10.3 action
(2) 2.0 rem

B.✓ (1) is within limits, entry into TR 3.10.3 in NOT required
(2) 0.5 rem

C. (1) is NOT within limits, enter actionTR 3.10.3 action
(2) 0.5 rem

D. (1) is within limits, entry into TR 3.10.3 in NOT required
(2) 2.0 rem

Distractor Analysis:

This requires the examinee to have knowledge of the **Waste Gas Disposal** parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc..

A INCORRECT
(1) is NOT within limits, enter actionTR 3.10.3 action
(2) 2.0 rem

The first part is incorrect by plausible since the value of "B" tank is well in excess of those that would normally ever be encountered and the candidate may be aware that there is a limit, but if they are not sure of the specifiec value they may default to this distractor based on the large values given. Second part is also incorrect but plausible because the TRM bases states for an uncontrolled release the total body exposure to an individual at the nearest exclusion boundary will not exceed 0.5 rem in an event of 2 hours.

B CORRECT
(1) is within limits, entry into TR 3.10.3 in NOT required
(2) 0.5 rem

First part is correct 25,000 is the TR Limit so we are within it. The second part is also correct per the TR bases.

C INCORRECT
(1) is NOT within limits, enter actionTR 3.10.3 action
(2) 0.5 rem

The first part is incorrect by plausible since the value of "B" tank is well in excess of those that would normally ever be encountered and the candidate may be aware that there is a limit, but if they are not sure of the specifiec value they may default to this distractor based on the large values given. The second part is correct.

D INCORRECT
(1) is within limits, entry into TR 3.10.3 in NOT required
(2) 2 rem

The first part is correct. Second part is incorrect but plausible because the TRM bases states for an uncontrolled release the total body exposure to an individual at the nearest exclusion boundary will not exceed 0.5 rem in an event of 2 hours.

K/A:
071G2.4.21
Waste Gas Disposal System (WGDS)
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system

integrity, containment conditions, radioactivity release control, etc.

Technical References:
TR 3.10.3 "Gas Storage Tanks"
TR 3.10.3 Bases

References provided to applicants: None

Learning Objective:
U 17527
Explain the concepts associated with the Gas Storage Tanks technical requirements and bases (TR-3.10.3)

Question Source: 2010 NAPs ILT SRO test Question 97 (G2.3.6)

Question History:
The Waste Gas Decay Tanks (WGDT) was sampled to determine the quantity of radioactive material contained in each gas storage tank.

<u>Confirmed sample results are as follows:</u>
- "A" WGDT – 15,000 curies of noble gas
- "B" WGDT – 6,000 curies of noble gas

Based on these sample results, which ONE of the following identifies the implications per TR 3.10.3 Gas Storage Tanks AND includes the associated TRM BASES?

The quantity of radioactive material in the WGDTs ____(1)______. The TRM base for restricting the quantity of radioactive material is to limit dose at the exclusion boundary to _____(2)______.

A. is within limits, entry into TR 3.10.3 action is NOT required; 0.5 rem (CORRECT)
B. is within limits, entry into TR 3.10.3 action is NOT required; 0.1 rem
C. is NOT within limits, enter TR 3.10.3 action; 0.5 rem
D. is NOT within limits, enter TR 3.10.3 action; 0.1 rem

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content:
SRO only - 10 CFR 55.43 (b)(5)
Assessment of facility conditons and selection of appropriate procedures during normal, abnormal, and emergency situations
10 CFR-55.41 (a)(b)(7)
Design, components, and functions of control and sagety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
This question matches the K/A statement by requiring the SRO applicant to have a

knowledge of limits placed on the Gas storage tanks and to have an understanding of the bases behind the limits set forth in technical specifications. To arrive at the correct answer the SRO applicant must recognize what value the quantity of radioactive material contained in each storage tank is limited to AND why that limit provides protection.

Unit-1 has been shutdown from 100% power for a scheduled refueling outage.

In accordance with OP-AA-100 "Conduct of Operations", which one of the following choices answers the statement?

The _____________ is responsible for overall supervision and coordination of refueling operations, including fuel movement.

A. Operations Manager

B. Fuel Handling Supervisor

C. Shift Manager

D.✔ Refueling SRO

Distractor Analysis:

This requires the examinee to have the knowledge of refueling administrative requirements.

A. INCORRECT
   Operations Manager

Incorrect but plausible, during refueling operation at NAPS there is a Operations Manager assigned to be at the station during dayshift and nightshift activities. The candidate may assume that overall responsibility is assigned to this individual

B. INCORRECT
   Fuel Handling Supervisor

Incorrect but plausible, 1-OP-4.1 "Controlling Procedure for Refueling" and other OP-4 series procedures associated with refueling includes various steps where the Fuel Handing Supervisor authorization is required, and if the candidate associates overall supervision of refueling operations as his responsibility this distractor may be chosen.

C. INCORRECT
   Shift Manager

Incorrect but plausible because the Shift Manger has control room oversight and at all times the Shift Manger is in concurrence with the outage unit acitivities. The candidate may assume that overall responsibility is assigned to this individual.

D. CORRECT
   Refueling SRO

In accordance with OP-AA-100 "Conduct of Operations" Attachment 6, The Refueling SRO is responsible for overall supervision and coordination of refueling operations, including fuel movement. When assigned, this refueling SRO has <u>NO</u> other concurrent responsibilities.

K/A:
G2.1.40
Knowledge of refueling administrative requirements.

Technical References:
1-OP-4.1 "Controlling Procedure for refueling"
OP-AA-100 "Conduct of Operations"

References provided to applicants: None

Learning Objective:
U 13561
Explain Personnel requirements associated with core alterations (OP-AA-100)

Question Source: NEW

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content:
SRO only 10 CFR-55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, and emergency situations.
10 CFR-55.41(a)(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the knowledge of refueling administrative requirements. To arrive at the correct answer the SRO applicant must have the knowledge of OP-AA-100 "Conduct of Operations", which lists the duties and responsibilities of the Refueling SRO.

Mechanics have a work order to replace the cable spreading to control room stairwell door due to a broken hinge.

- The crew has entered action of T.S. 3.7.10 "Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS).

- 1-LOG-17 "Unit 1 & 2 Control Room Boundary Breaching Log" has been initiated. The control room boundary can NOT be maintained operable by use of administrative control.

Which one of the following will complete the statement?

In accordance with T.S. bases, which Emergency Switchgear Emergency Filtered Air Supply Fan, can **NOT** be used to supply the main control room and relay rooms in the Pressurization Mode.

A. 2-HV-F-41

B✓ 1-HV-F-41

C. 1-HV-F-42

D. 2-HV-F-42

Distractor Analysis:
This requires the examinee to have the knowledge of the process for controlling equipment configuration or status. In accordance with ER-NA-CRH-100 "Control Room Habitability Program" Control room boundary breeches shall be controlled by procedurally administrative controls, NAPS has adopted the use of 1-LOG-17 "Unit 1 & 2 Control Room Boundary Breaching Log" as the procedure to establish an equipment configuration for appropriate compensatory actions depending on the nature of the breach.

A. INCORRECT
2-HV-F-41
As per the Bases associated with T.S. 3.7.10 "Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS) 3 of the 4 Emergency Switchgear Emergency Filtered Air Supply Fans can be used to support control room habitability. 1-HV-F-41 **CAN NOT** be used due to the location of its air intake with respect to Vent Stack B

B. CORRECT
1-HV-F-41

See above explanation of the bases associated with T.S. 3.7.10

C. INCORRECT
1-HV-F-42
See Above explanation of the bases associated with T.S. 3.7.10

D. INCORRECT
2-HV-F-42
See Above explanation of the bases associated with T.S. 3.7.10

K/A:
095 - G2.2.14

Technical References:
1-LOG-17 "Unit 1 & 2 Control Room Boundary Breaching Log"
T.S. 3.7.10 "Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)
Bases of T.S. 3.7.10
ER-NA-CRH-100 "Control Room Habitability Program"

References provided to applicants: None

Learning Objective:
U 16288
Explain the various concepts associated with the main control room/emergency switchgear room (MCR/ESGR) emergency ventilation system (EVS)

U 5782
Information associated with the control room ventilation fans and filters

U 5784
Explain the purpose of the control room emergency fans

Question Source: NEW

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(3)
Facility licensee procedures required to obtain authority for design and operating changes in the facility.
10 CFR-55.41 (a)(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility

Comments:
This question matches the K/A statement by requiring the SRO applicant to possess the knowledge for controlling equipment configuration to maintain the Unit 1 & Unit 2 Control Room Pressure Boundary. In accordance with ER-NA-CRH-100 "Control Room Habitability Program" Control room boundary breeches shall be controlled by proceduralized administrative controls, NAPS has adopted the use of 1-LOG-17 "Unit 1 & 2 Control Room Boundary Breaching Log" as the procedure to establish an equipment configuration for appropriate compensatory actions depending on the nature of the breach. To arrive at the correct answer the SRO must know the requirements stated in T.S. 3.7.10 bases pertaining to the ventilation fans and their use, and also be aware of the administrative procedure requirement of 1-LOG-17.

Unit-2 Mode 1, 100% power.

The safeguard operator discovers a small packing leak on 2-MS-TV-201B ("B" Main Stream Trip Valve). The leak comes in surges but is not steaming. The leak has been quantified at 30 dpm. CR was submitted and WO is issued to adjust packing to the last known torque value IAW component engineering recommendation.

The WOs Post Maintenance Testing sheet requires the following:

- Valve timed stroke according to 2-PT-212.9 "Valve inservice inspection"
- External leak check

Based on plant conditions, the PMT requirement to perform 2-PT-212.9 will need to be ____(1)______, and at the minimum, justification need to be provided by ____(2)____.

A. (1) Deferred
(2) Engineering

B. (1) Waived
(2) FSRC

C.✓ (1) Waived
(2) Engineering

D. (1) Deferred
(2) FSRC

Distractor Analysis:

The candidate is required to have the knowledge of pre- and post-maintenance operability requirements

A. INCORRECT
(1) Deferred
(2) Engineering

First part is incorrect but plausible if the candidate is not familiar with the definition section of VPAP- 2003 "Post Maintenance Testing Program" which defines Deferred Test as a test requirement that will not be performed by the time the test data sheet is closed. Test deferrals still means that the test requirements need to be completed, the VPAP also states "All post maintenance testing requirements normally should be completed prior to declaring the equipment operable. 2-MS-TV-201B is required to be operable in Modes 1, 2 and 3 as per TS 3.7.2 "Main Steam Trip Valves (MSTVs)". Second part is correct, at the minimum, justification shall be provided by engineering. VPAP-2003 includes "Providing written justification when a test listed in the PMT requirements section of the Test Data Sheet is to be waived, as a responsibility of Engineering. Engineering would also determine if a higher level of justification is required by FSRC

B. INCORRECT
(1) Waived
(2) FSRC

First part is correct, VPAP-2003 defines a waived test as " a test which appears on the Test Data Sheet but will not be completed as specified or performed at all based on a special set of plant or maintenance conditions. Tests may only be waived prior to the test being performed; it is not permissible to disposition a failed test by waiving it. Second part is incorrect but plausible because, at the minimum, justification shall be provided by engineering. VPAP-2003 includes "Providing written justification when a test listed in the PMT requirements section of the Test Data Sheet is to be waived, as a responsibility of Engineering. Engineering could then determine if a higher level of justification is required by FSRC

C. CORRECT
(1) Waived
(2) Engineering

VPAP-2003 defines a waived test as " a test which appears on the Test Data Sheet but will not be completed as specified or performed at all based on a special set of plant or maintenance conditions. Tests may only be waived prior to the test being performed; it is not permissible to disposition a failed test by waiving it. The minimum, justification shall be provided by engineering. VPAP-2003 includes "Providing written justification when a test listed in the PMT requirements section of the Test Data Sheet is to be waived, as a responsibility of Engineering.

D. INCORRECT
(1) Deferred
(2) FSRC

Both sections are incorrect. First part is incorrect but plausible if the candidate is not familiar with the definition section of VPAP- 2003 "Post Maintenance Testing Program" which defines Deferred Test as a test requirement that will not be performed by the time the test data sheet is closed. Test deferrals still means that the test requirements need to be completed, the VPAP also states "All post maintenance testing requirements normally should be completed prior to declaring the equipment operable. 2-MS-TV-201B is required to be operable in Modes 1, 2 and 3 as per TS 3.7.2 "Main Steam Trip Valves (MSTVs)". Second part is incorrect but plausible because, at the minimum, justification shall be provided by engineering. VPAP-2003 includes "Providing written justification when a test listed in the PMT requirements section of the Test Data Sheet is to be waived, as a responsibility of Engineering. Engineering could then determine if a higher level of justification is required by FSRC

K/A:

Knowledge of pre- and post-maintenance operability requirements.

Technical References:
Operations Site Specific Instructions OP-AA-100-1001 "Operations Department Instructions"
(Title: Operations PMT Review)

VPAP-2003 "Post Maintenance Testing Program"

OP-AA-200 "Equipment Clearance"

WM-AA-100 "Work Management"

North Anna PMT Test Data Sheet

Work Order #59102710332

References provided to applicants: None

Learning Objective:
U 13130
List the conditions that are acknowledged and approved by the shift manager upon approval of a work order.

U 13634
Explain the Operations Department's responsibilities associated with post-maintenance testing)

U 13120
Explain the how to determine whether testing is required prior to restoring safety-related equipment to service.

Question Source: NEW

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43(b)(2)
Facility operating limitations in the technical specifications and their bases.
10 CFR-44.41(a)(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the knowledge of pre- and post-maintenance operability requirements. To arrive at the correct answer the SRO applicant must have the knowledge to correctly assess plant conditions, and interpret those conditions, during the review and authorization of a WO. Specifically, the candidate needs to know that 2-MS-TV-201B is required under the

given plant conditions. Also, the candidate needs the knowledge of VPAP-2003 to process the proper course of actions that will allow the required maintenance to be performed, <u>and</u> maintain operability of the system.

10. 097 - G2.3.14 001/SRO/T3//3.4/3.8/MODIFIED/SURRY 2010/

Unit-1 is at 100% full power.

- 1-GM-F-1 "Isophase bus duct cooling fan No 1 is tagged out due to bad contactor, and is currently being worked by Electricians.
- 1-GM-F-2 "Isophase bus duct cooling fan No 2" suddenly trips.
- Crew has entered 1-AP-2.2 "Fast Load Reduction".
- Chemistry has been notified to perform and Isotopic Analysis for power reductions geater than 15 percent in 1 hour.

Which one of the following completes the following statement?

The RCS activity must be limited to _____(1)_______ DOSE EQUIVALENT IODINE-131. Which ensures the 2 hour doses at the site boundary will not exceed limits following _______(2)_____ accidents.

A. (1) Less than or equal to 0.1 uCi/gm
(2) LOCA

B.✔ (1) Less than or equal to 1.0 uCi/gm
(2) SGTR

C. (1) Less than or equal to 1.0 uCi/gm
(2) LOCA

D. (1) Less than or equal to 0.1 uCi/gm
(2) SGTR

Distractor Analysis:
This requires the examinee to have the knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

A INCORRECT
(1) Less than or equal to 0.1 uCi/gm
(2) LOCA

First part is incorrect but plausible because 0.1 uCi/gm is the activity limit for the Secondary Specific Activity per TS 3.7.7, Second part is incorrect but plausible if the candidate is unfamiliar with the TS bases background or applicable safety analysis, which includes the SGTR but not a LOCA.

B CORRECT
(1) Less than or equal to 1.0 uCi/gm
(2) SGTR

First part is correct Reactor Coolant DOSE EQUIVALENT I-131 specific activity verified to be less than or equal to 1.0 uCi/gm, and the second part is correct the LCO limits are established to minimize the dose consequences in the event of a steam generator tube rupture (SGTR)

C INCORRECT
(1) Less than or equal to 1.0 uCi/gm
(2) LOCA

Incorrect but plausible because the first part is correct Reactor Coolant DOSE EQUIVALENT I-131 specific activity verified to be less than or equal to 1.0 uCi/gm Second part is incorrect but plausible if the candidate is unfamiliar with the TS bases background or applicable safety analysis, which includes the SGTR but not a LOCA.

D INCORRECT
(1) Less than or equal to 0.1 uCi/gm
(2) SGTR or SLB

First part is incorrect but plausible because 0.1 uCi/gm is the activity limit for the Secondary Specific Activity per TS 3.7.7, and the second part is correct the LCO limits are established to minimize the dose consequences in the event of a steam generator tube rupture (SGTR)

K/A:
G2.3.14
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Technical References:
TS 3.4.16
TS Bases 3.4.16
1-PT-53.5 "RCS Specific Activity - Iodine Isotopic Analysis

References provided to applicants: None

Learning Objective:
U 16260
Explain the concepts associated with the RCS specific activity technical specification and bases (TS-3.4.16)

U 16285
Explain the concepts associated with the secondary specific activity technical specification and bases (TS-3.7.7)

Question Source: Modified Surry 2010 ILT SRO exam (G2.3.14)

Question History: Modified

Current Conditions
• Unit-1 is at full power

Which one of the following completes the statement conserning:
1) the DOSE EQUIVALENT IODINE-131 limit for RCS activity in accordance with TS 3.1.D Maximum Reactor Coolant Activity AND 2) the assumed release duration through the main steam safety valves and atmospheric relief valves in accordance with TS bases?

The RCS activity must be limited to _____________ DOSE EQUIVALENT IODINE-131. Primary water assumed to enter the secondary system and be release for a period of _______________

A. ≤ to 0.1uCi/cc: 60 minutes
B. ≤ to 0.1uCi/cc: 30 minutes
C. ≤ to 1.0uCi/cc: 60 minutes
D. ≤ to 1.0uCi/cc: 30 minutes (CORRECT)

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content:
97 - 10 CFR 55.43 (b)(4)
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
10 CFR 55.41 (a)(b)(12)
Radiological safety principles and procedures

Comments:
This question matches the K/A statement by requiring the SRO applicant to analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits, by understanding the bases behind the limits set forth in technical specifications. To arrive at the correct answer the SRO applicant must recognize what value RCS activity is limited to AND under which accident the limit provides protection.

Due to elevated turbine vibrations, the night shift crew is ramping Unit-1 to Mode 2 for a Turbine Balance Shot.

At 20:30 and 79% power, annuciator 1K-G6 "N-16 Rad Det" alarms.

1-MS-RI-191 "B S/G N-16 and 1-MS-RI-193 "MS Header" are in Alert at 5 GPD, the crew enters 1-AP-5, " Unit 1 Radiation Monitoring Systems."

During the ramp the following occurs:

1-MS-RI-191 and 1-MS-RI-193 indications:
- 21:30 and 61% power = 9 GPD
- 22:30 and 43% power = 14 GPD
- 23:30 and 26% power = 20 GPD

1-SV-RM-121 "Condenser Air Ejector RM" indicated a consistent but slowly increasing count rate.

The shift STA has confirmed leakage rate trend data is correct.

During the ramp, TRM 3.4.5 Primary to Secondary Leakage Detection System Condition A will be entered when power goes below ____(1)______ % power, and Mandatory frequency of grab samples is _____(2)____ hours.

REFERENCE PROVIDED

A. (1) 30
(2) 12

B. (1) 30
(2) 4

C. (1) 25
(2) 12

D✓ (1) 25
(2) 4

Distractor Analysis:

This requires the examinee to have the knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

A. INCORRECT
(1) 30
(2) 12

First part is incorrect but plausible, If the candidate would consider both N-16 and condenser air ejector radiation monitors non functional at 30% power. As the unit is ramping down, at 30% power 1-SV-RM-121 "Condenser Air Ejector RM" is declared Non-functional, this is described in TRM 3.4.5 bases, "RCS Ar-41 activity is expected to be sufficient to support functionality of the condenser air ejector exhaust system shortly after 30% power is reached. At this point N-16 is still functional and meeting the requirement for the continuous readout radiation monitoring systems. Second part is also incorrect but plausible if the candidate incorrectly reads table 3.4.5-1 of the TRM. When leakage is ≥ 5 GPD and < 30 GPD grab samples shall be once per 12 hours, but if leakage is ≥ 5 GPD and < 30 GPD AND rate of increase is > 5 GPD then grab samples will be taken once per 4 hours.

B. INCORRECT
(1) 30
(2) 4

First part is incorrect but plausible, if the candidate would consider both N-16 and condenser air ejector radiation monitors not functional at 30% power (SEE plausibility explanation for answer A). Second part is correct. Because based on N-16 trends the rate of increase is > 5 GPD and grab samples should be taken once per 4 hours

C. INCORRECT
(1) 25
(2) 12

First part is correct, During a down power ramp at 30% power the condenser air ejector RM is declared non-functional, and as per the 3.4.5 TRM bases "RCS N-16 activity will be sufficient to support FUNCTIONALITY of the N-16 system with reactor power at or above 25% power. At NAPs the shutdown procedures then declare N-16 non-functional at 25% power. At that point we do not have a continuous readout radiation monitoring system functional and Condition A will be entered. The second part is incorrect as stated above (SEE plausibility explanation for answer B)

D. CORRECT
(1) 25
(2) 4

First part is correct, As the unit is ramping down, at 30% power 1-SV-RM-121 "Condenser Air Ejector RM" is declared Non-functional, this is described in TRM 3.4.5 bases, "RCS Ar-41 activity is expected to be sufficient to support functionality of the condenser air ejector exhaust system shortly after 30% power is reached. At 25% power N-16 is then also declared Non-functional, as per the 3.4.5 TRM bases "RCS N-16 activity will be sufficient to support FUNCTIONALITY of the N-16 system with reactor power at or above 25% power. Second part is also correct, because based on N-16 trends, the rate of increase is > 5 GPD. Between 61% and 43% power the rate of increase was at 5 gpd and then between 43% and 26% the rate of increase was 6 gpd, so based on Note (b) of table 3.4.5-1 grab samples should be taken once per 4 hours, and the STA has confirmed the leakage rate trends are correct.

K/A:
G2.3.15
Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Technical References:
TRM 3.4.5 "Primary to Secondary Leakage detection systems"
TRM 3.4.5 Bases
1-OP-2.2 "Unit Power Operation From Mode 1 to Mode 2"

References provided to applicants:
TRM 3.4.5 "Primary to Secondary Leakage Detection Systems
Page 3.4.5-1
Page 3.4.5-2
Page 3.4.5-4 (Table 3.4.5-1)
NOTE: Page 3.4.5-3 TRM Surveillance Requirements WILL NOT BE INCLUDED

Learning Objective:
U 18002
Explain High Level Actions of 1-AP-5 "Unit-1 Radiation Monitoring System"
Explain applicable TRMs

U 5263
List the means provided to the Control Room to determine abnormal conditions as they apply to the N-16 main steam radiation monitor.

U 5264
Explain Relationship between indicated Rx power and N-16 RM readings
Why the N-16 MS RM indication is invalid below 25% Rx power

Question Source: NEW

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR 55.43 (b)(4)
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
10 CFR 55.41 (a)(b)(12)
Radiological safety principles and procedures

Comments:
This question matches the K/A statement by requiring the SRO applicant to analysis and interpret Main Steam RM readings, including comparison to operational procedures and then regulatory limits, by understanding the bases behind the limits set

applicant must recognize what kind of trend is associated with the N-16 radiation monitors. Having a backup confirmation of that trend. Then applying TRM actions, based on unit conditions outlined in the bases, that support functionality of the monitoring equipment.

Unit-1 is at 100% power when a fire occurs in the Unit-1 Emergency Switchgear Room. The Control Room crew enters 1-FCA-2 "Emergency Switchgear Room Fire".

- The operating crew is ready to establish RCS cooldown.
- H emergency bus is operable.
- J emergency bus is de-energized.
- Two CRDM fans are available and running.
- Charging is aligned through the BIT.
- An operator has been dispatched to perform attachment 15 "Fuel Building and Mitigating Spurious Valve Operations"

Which one of the following completes the statement below?

The operator assigned to establish communications with the control room from the remote monitoring panel in the fuel building shall have a minimum qualification as a _____(1)______ and the cooldown rate limit will be _____(2)______.

A.✓ (1) Licensed Operator
(2) < 15 DEGF/HR

B. (1) Non-Licensed operator who has completed step 7
(2) < 15 DEGF/HR

C. (1) Licensed Operator
(2) < 25 DEGF/HR

D. (1) Non-Licensed operator who has completed step 7
(2) < 25 DEGF/HR

Distractor Analysis:
This requires the examinee to have knowledge of "fire in the plant" procedures.

A. CORRECT
(a) Licensed Operator
(b) < 15 DEGF/HR

(a) Is correct because 1-FCA-2 procedure require the assignment of a Licensed Operator for performing Remote Monitoring operations. (b) is correct, the RCS cooldown limits are posted frequently through out the cooldown sequence and also as a CAUTION of attachment 15 in 1-FCA-2 "Emergency Switchgear Room". 1) With three CRDM fans running cooldown rate is limited to < 25 DEGF/HR 2) With less than three CRDM fans running cooldown rate is limited to < 15 DEGF/HR.

B. INCORRECT
(a) Non-Licensed operator who has completed step 7
(b) < 15 DEGF/HR

(a) Is incorrect but plausible if the candidate is not sure of the procedure qualification requirements, it would be very likely the candidate would consider a Non-Licensed operator who has completed step 7 as having the qualifications since the operator is not operating equipment and only providing indicated readouts to the control room and also the Fuel Building is a watchstation of a Non-Licensed operators who has completed step 7. Also, a watchstation qualified operator is assigned to perform actions within other FCA series procedures/attachments, example is controlling S/G levels locally in 0-FCA-1 and if remote control of PORV's is not available they can be locally operator by a operator qualified for the watchstation but with constant communication and direction from the MCR such as in attachment 6 in 1-FCA-2. Attachment 15 is the only attachment in 1-FCA-2 out of 17 attachments that requires actions to be performed by a Licensed operator (b) is correct for the questions stated condition that only two CRDM fans are available and running so the cooldown graph that would be chosen for use would be limited to <15 DEGF/HR which is explained by a CAUTION in attachment 15 and posted frequently through out the cooldown sequence of 1-FCA-2.

C. INCORRECT
(a) Licensed Operator
(b) < 25 DEGF/HR

(a) Is correct, 1-FCA-2 procedure requires the assignment of a Licensed Operator for performing remote monitoring operations. (b) is incorrect but plausible if candidate is not sure of FCA requirements vs EOP procedure requirements to determine which cooldown graph to use based on CRDM fans in service, of either < 25 or < 15 DEGF/HR.

D. INCORRECT

(a) Non-Licensed operator who has completed step 7
(b) < 25 DEGF/HR

(a) Is incorrect but plausible if the candidate is not sure of the procedure qualification requirements, it would be very likely the candidate would consider a Non-Licensed operator who has completed step 7 as having the qualifications since the operator is not operating equipment and only providing indicated readouts to the control room and also the Fuel Building is a watchstation of a Non-Licensed operators who has completed step 7. Also, a watchstation qualified operator is assigned to perform actions within other FCA series procedures/attachments, example is controlling S/G levels locally in 0-FCA-1 and if remote control of PORV's is not available they can be locally operator by a operator qualified for the watchstation but with constant communication and direction from the MCR such as in 1-FCA-3 attachment 10 (b) is incorrect but plausible if candidate is not sure of FCA requirements vs EOP procedure requirements, to determine which cooldown graph to use based on CRDM fans in service, of either < 25 or < 15 DEGF/HR.

K/A:
099 - G2.4.27

Technical References:
1-FCA-2 "Emergency Switchgear Room Fire"

References provided to applicants: None

Learning Objective:
U 13906
Determine the maximum Reactor Coolant System cooldown rate allowed while responding to a fire in the emergency switchgear room in accordance with 1-FCA-2

U 9113
Explain the qualifications of the operator assigned to conduct remote monitoring operations in the fuel building.

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:
SRO only - 10 CFR-55.43 (b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
10 CFR-55.41 (a)(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
This question matches the K/A statement by requiring the SRO applicant to possess a working knowledge of NAPs Fire contingency actions (FCA). To arrive at the correct answer the SRO must be aware of the procedure administrative requirement to send a License operator to the fuel building, who in turn will provide remote monitoring to the operators in the MCR or Auxiliary shutdown panel. The SRO must also show knowledge of diagnostic steps and decision point to obtain the proper sub-procedure in terms of selecting the proper cooldown graph.

Unit-2 is defueled with maintenance activities in progress in the Containment Building. A call comes into the control room that a worker has fallen from a scaffold platform in containment and is unconscious.

- The crew enters 0-AP-51 "Personnel Injury - Operations Response"
- The injured worker has various lacerations and abrasions.
- The First Aid Team determines the worker must be transported to a hospital for treatment.
- HP survey of the fall area and the workers DAD, indicates no radiological overexposure exist.
- Some of the workers wounds are contaminated and initial decon efforts have been unsuccessful.

Due to the nature of the injuries a decision is made to immediately transport the worker to the appropriate hospital.

Assuming no news releases or notification to other government agencies are planned, a ___________ notification to the NRC is required in accordance with VPAP-2802, Notifications and Reports, and 10 CFR-50.72.

REFERENCE PROVIDED

A. Immediate

B. One hour

C. Four hour

D.✓ Eight hour

Distractor Analysis:

The candidate is required to have the knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

INCORRECT
A. Immediate

Incorrect but plausible if the candidate is unfamiliar with using VPAP-2802, and is directed to section 6.3.2 which is plausible if the candidate uses the topic of Personnel overexposures under Radiation or Exposure Events or goes to section 6.28.2 (see above) also the candidate could be directed to section 6.3.2 using Miscellaneous Events or conditions under injuries.

INCORRECT
B. One hour

Incorrect but plausible if the candidate is unfamiliar with using VPAP-2802, and is directed to section 6.3.3, or goes to section 6.28.2 for internal (Dominion) notification. This has a step requiring the Manager of Nuclear Operations or Operations Manager on Call to notify the Site Vice President or a director of a potentially media significant event. This step is referenced in section 6.3.5.a.6 and in section 6.2.1(i) bullet for Injuries or accidental deaths or Ambulance transport of personnel to an off-site medical facility.

INCORRECT
C. Four hour

Incorrect but plausible if the candidate is unfamiliar with using VPAP-2802, and is directed to section 6.3.4 which could be required if a news release is planned

CORRECT
D. Eight hour

The candidate can use section 6.2 of VPAP-2802 "Non-Scheduled Notification and Reports" and search for the matching criteria. In this case section 6.2.1(i) "Miscellaneous Events or Conditions" (•) Transport of contaminated injured person, and will be directed to step 6.3.5.a.6. Section 6.3.5 is for eight hour notifications; a - As soon as practical, but within eight hours, the Shift Manager shall notify the NRC Operations Center via the ENS of; 6 - Any event requiring the transport of a radioactively contaminated person to an off-site medical facility for treatment.

K/A:
100 - G2.4.30
Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Technical References:
VPAP-2802 "Notifications and Reports"

References provided to applicants:
VPAP-2802 "Notifications and Reports"

Learning Objective:
U 9391
Define immediate notification and Reportable as they apply to reportability requirements (VPAP-2802)

U 9390
List Documents to be used and purpose of worksheet as it applies to reportability requirements (VPAP-2802)

U 9389
Explain the process for notifying NRC during Non-emergency and emergency events (VPAP-2802)

U 14323
Given a copy of VPAP-2802 "Plant Reporting Requirement" evaluate a set of plant conditions associated with reportability requirements in light of the following issues (SRO)

Question Source: NAPS Vision Bank

Question History:
Unit-2 is defueled with maintenance activities in progress in the Containment Building. During one of these activities, a worker falls form a scaffold ladder. The worker is rendered unconscious and sustains various lacerations and abrasions. The first aid team is dispatched and determines that the worker must be transported to a hospital for treatment. Based on HP survey of the fall area and the workers DAD, no radiological overexposure is suspected, however, it is found that some of the workers wounds are contaminated. Initial efforts to decontaminate the worker on site are not successful further attempts are suspended. Due to the nature of the injuries, a decision is made to immediately transport the worker to the appropriate hospital. The worker arrives at the hospital by ambulance within the next hour to receive further medical treatment.

Assuming no news releases or notifications to other government agencies are planned _____________ notification to the NRC is required in accordance with 10CFR50.72

A. an eight hour (correct)
B. no
C. a four hour
D. a one hour

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content:
SRO only 10 CFR-55.43(b)(5)

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
10 CFR-55.41(a)(b)(10)
Administrative, normal abnormal, and emergency operating procedures for the facility.

Comments:
This question matches the K/A statement by requiring the SRO applicant to have the knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. To arrive at the correct answer the SRO applicant must have the knowledge to correctly assess plant conditions, and interpret those conditions, to determine the proper reporting requirement. Also the candidate must demonstrate proficiency negotiating through VPAP-2803 "Notifications and Reports" to determine the reporting requirements and limits that needs to be applied.