Question Documentation Anatomy

Prefix List:

A.0 – Question
- A.1 – Distractor Analysis
- A.2 – Misc Comments/Feedback/History
- A.3 – NUREG ES-401-5
- A.4 – Original Question (if question was MODIFIED from a BANK question)
- A.5 – Supplied Reference

B.0 – Regulator Documents
- B.0 – Tech Specs
- B.1 – Tech Spec Bases
- B.2 – TRM
- B.3 – COLR
- B.4 – PTLR
- B.5 – ODCM
- B.6 – PLS
- B.7 – PTDB

C.0 – Procedures
- C.1 – EOP
- C.2 – AOP
- C.3 – SOP
- C.4 – UOP
- C.5 – ARP
- C.6 – Admin
- C.9 – Other

D.0 – Drawings
- D.1 – P&IDs
- D.2 – Oneline
- D.3 – Elementary
- D.4 – Logic
- D.5 - Other

E.0 – Misc Other
- E.1 – Photographs
- E.2 – Maps
- E.3 – Rad Surveys
- E.4 – Lesson Plan

The bookmark sidebar will be very helpful in viewing this document as it will show the breakdown of each question more easily.

1. 001AA2.04 001/LOIT/SRO/C/A 4.2/4.3/001AA2.04/LO-TA-63013///

Initial conditions:

- Unit 1 is at 70% reactor power.
- Main Turbine load is stable.
- Rods are in automatic with CBD at 190 steps.
- Control rods start stepping out without demand.

Current conditions:

- Rod motion stops when the Rod Bank Selector Switch is placed in Manual.
- Reactor power indication has risen to 75%.
- CBD rods are at 211 steps, EXCEPT control rod H8, which is at 190 steps by DRPI indication.

With NO other actions taken, which one of the following completes the following statement?

When the inadvertent rod motion stops, reactor power indication will __(1)__,

and

per the Bases of Tech Spec 3.1.4, "Rod Group Alignment Limits," CBD rod H8 is __(2)__ at this time.

A. (1) stabilize and remain near 75%

(2) OPERABLE

B. (1) stabilize and remain near 75%

(2) inoperable

C. (1) trend down from the peak value observed to slightly above 70%

(2) OPERABLE

D✓ (1) trend down from the peak value observed to slightly above 70%

(2) inoperable

K/A

001 **Continuous Rod Withdrawal**

AA2.04 **Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal:**

- Reactor power and its trend

K/A MATCH ANALYSIS

The question sets up a plausible scenario where rods withdrawal due to some internal failure inserts some amount of positive reactivity. Once operator action is taken rod motion stops and the candidate must address how the core will respond to the inserted reactivity and the impact of the stuck rod, therefore the two elements are in place that the KA requires. The OPERABILITY determination for the affect rod is SRO required knowledge.

EXPLANATION OF REQUIRED KNOWLEDGE

As control rods are withdrawn, an increase in fission rate will occur. The resulting fuel centerline temperature increase from the positive reactivity addition will be offset by FTC and MTC feedback. As a result, reactor power will trend down toward the orginial power level with and increase RCS average temperature. Since RCS temperature is higher than the orginial temperature, SG pressure will also be slightly elevated. The higher SG pressure will result in slight increase in steam flow as compared to the original conditions. Therefore, reactor power will be slightly higher than the pre-transient value. This is a fundamental reactor theory concept.

Per TS 3.1.4 Bases, the OPERABILITY requirements (i.e. trippability) are seperate from the alignment requirements. Even if a rod is >12 steps misaligned, it is still OPERABLE. Where rod(s) are not moving, the rod(s) must be considered untrippable unless there is verification that a rod control system failure is preventing rod motion. Since rod motion was demanded and did not occur and there is nothing given in the stem to explain the loss of motion, the candidate must declare rod H8 inoperable.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Part 1 is 'plausible' however incorrect in that the candidate may determine that reactor power will rise to a new higher equilibrium value and stabilize. Thus, the candidate failed to take into count the reactivity feedback mechanisms of MTC and FTC on core behavior. In addition, the answer would be correct under some plant condition like during startup steam dumps in steam pressure mode.

Part 2 is 'plausible' but is also incorrect in that the candidate must consider the affected rod inoperable until proven trippable as stated in Tech Spec 3.1.4 ‘Rod Group Alignment Limits’ bases; However, where rod(s) are not moving, the rod(s) must be considered untrippable unless there is verification that a rod control system failure is preventing rod motion. If the rod control system is demanding motion properly and no motion occurs, the rod is considered untrippable (i.e., inoperable). This operability call is contrary to the standard philosophy that a component is considered OPERABLE until determined otherwise.

B. Incorrect. Plausible. Part 1 is 'plausible' however incorrect. See Part 1 of choice A above.

Part 2 is correct requiring the candidate to recall Tech Spec 3.1.4 'Rod Group Alignment Limits' bases which ties OPERABILITY to trippability of the rods. The rod OPERABILITY (i.e., trippability) requirement is satisfied provided that the rod will fully insert in the required rod drop time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability. However, where rod(s) are not moving, the rod(s) must be considered untrippable unless there is verification that a rod control system failure is preventing rod motion. If the rod control system is demanding motion properly and no motion occurs, the rod is considered untrippable (i.e., inoperable).

C. Incorrect. Plausible. Part 1 is correct which is addressing core response and correctly predicts that the positive reactivity will be offset by FTC and MTC feedback. Over time the reactor power will lower to near original values, only slightly higher due to T_{AVG} being raised which results in an increase in Steam Generator pressure. This would result in a small increase in steam flow.

Part 2 is 'plausible' but is also incorrect. See Part 2 of choice A above.

D. Correct. Part 1 is correct. See Part 1 of choice C above.

Part 2 is correct. See Part 2 of choice B above.

<u>SRO JUSTIFICATION (10CFR43(b))</u>

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No the question is not addressing any TS action times.
-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" No, the question is not addressing above-the-line TS information. The required knowledge is TS Bases.
-Can question be answered *solely* by knowing the TS Safety Limits? No, the question is not related to TS Safety Limits.
-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- **Knowledge of TS bases that is required to analyze TS required actions and terminology. Yes. Specific knowledge of TS Bases is required to make the OPERABILITY call required and this call is contrary to the standard philosophy for OPERABILITY calls.**

Level:	SRO
Tier # / Group #	T1 / G2
K/A#	001AA2.04
Importance Rating:	4.2 / 4.3

Technical Reference: Tech Spec 3.1.4 Bases Rev 1-8/03

References provided: None

Learning Objective: LO-LP-36990-07 State the Technical Specification bases for the restrictions on control rod insertion limit, and alignment (SRO only).

LO-LP-39205-02 Given a set of Tech Specs and the Bases, determine for a specific set of plant conditions, equipment availability, and operational mode: Whether any Tech Spec LCO's of section 3.1 are exceeded.The required actions for all section 3.1. LCO's.

LO-TA-63013 Implement Technical Specification LCO using 10008-C (SRO Only)

Question origin: MODIFIED - Vogtle HL18 Question #005AG2.1.07 001

Cognitive Level: C/A

10 CFR Part 55 Content: 41.1 / 43.2

Comments:

You have completed the test!

1. 005AG2.1.07 001/1/2/STUCK CRDM/C/A - 4.4/4.7/BANK-HL-17/HL-18 NRC/SRO/AML

Initial conditions:

Original Question

- Time = 0900.
- Unit 1 is at 60% power following a refueling outage.
- The OATC is withdrawing rods when one DRPI is seen not moving with its group.
- The OATC immediately stops withdrawing rods, and all rod motion stops.
- CBD, Group 2, Rod H-8 DRPI indicates 198 steps.
- CBD, Group 2, step counters indicate 209 steps.

Current conditions:

- Time = 0945.
- No rod motion has occurred since 0900.
- I&C has verified no faults on the DRPI system.
- I&C has verified that the rod lift coil for Control Rod H-8 is failed.

Which one of the following completes the below statements?

Based on the initial conditions, at 0900 Control Rod H-8 was __________ in accordance with the Bases of Tech Spec 3.1.4, Rod Group Alignment Limits.

Based on the current conditions, at 0945 Control Rod H-8 was __________ in accordance with the Bases of Tech Spec 3.1.4, Rod Group Alignment Limits.

	Rod H-8 status <u>at 0900</u>	Rod H-8 Status <u>at 0945</u>
A.	OPERABLE	inoperable
B.	inoperable	inoperable
C.	OPERABLE	OPERABLE
D.✓	inoperable	OPERABLE

BASES (continued)

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The rod OPERABILITY (i.e., trippability) requirement is satisfied provided that the rod will fully insert in the required rod drop time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability. However, where rod(s) are not moving, the rod(s) must be considered untrippable unless there is verification that a rod control system failure is preventing rod motion. If the rod control system is demanding motion properly and no motion occurs, the rod is considered untrippable (i.e., inoperable).

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. When required, movable incore detectors may be used to determine rod position and verify the rod alignment requirement of this LCO is met.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which a self-sustaining chain reaction ($K_{eff} \geq 1$) occurs, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are fully inserted and the reactor is shut down, with no self-sustaining chain reaction. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

(continued)

1. 001G2.1.19 001/LOIT/SRO/C/A 3.9/3.8/001G2.1.19/LO-TA-05002///G2.1.37

Initial conditions:

- Unit 1 reactor power lowered due to an inadvertent turbine runback.
- ALB10-D04 ROD BANK LO-LO LIMIT alarm was received.

Current conditions:

- Main Turbine load has been stabilized.
- RCS Tavg is 2°F below Tref.
- The OATC has requested a 2-step rod withdrawal for temperature control.

Which one of the following completes the following statement?

Based on the current conditions and using the Plant Computer data provided, ALB10-D04 __(1)__ valid,

and

based on the current conditions and NMP-OS-001, "Reactivity Management Program," guidance, the Shift Supervisor is __(2)__ to authorize a 2-step control rod withdrawal for Tavg control.

REFERENCE PROVIDED

	__(1)__	__(2)__
A.✓	is	allowed
B.	is	NOT allowed
C.	is NOT	allowed
D.	is NOT	NOT allowed

K/A

001 **Control Rod Drive**

G2.1.19 **Ability to use plant computers to evaluate system or component status.**

K/A MATCH ANALYSIS

The question test the candidate ability to evaluate and validate a plant computer (IPC) generated alarm for the current plant conditions associated with the Rod Control system. Then the SRO is required to determine if control rod withdrawal is both prudent and allowed under the stated conditions in the stem per the NMP-OS-001 ‘Reactivity Management Program’.

EXPLANATION OF REQUIRED KNOWLEDGE

The IPC calculates the rod insertion limit using real time data. When the alarm is generated the candidate would be expected to verify the validity of the alarm. This evaluation would be accomplished using IPC and Tech Spec data with backup from QMCB indications. In addition, the Shift Supervisor is required the make a decision to withdrawal control rods based on NMP-OS-001 'Reactivity Management Program' guidance.

The control room team shall not immediately dilute or withdraw control rods in an attempt to restore RCS Tavg/Tref deviations caused by a secondary plant transient. Attempts to immediately restore RCS Tavg/Tref deviations caused by a secondary plant transient can be aggravated by withdrawing control rods or reducing boron concentration with reactor power rising. Per NMP-OS-001, once turbine load has been stabilized and RCS Tavg has been restored to within 3°F of Tref, positive reactivity can be added by withdrawing control rods.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. Per the IPC printout, RCS DeltaT Power is approximately 83% and CBD position is 116 steps. The IPC uses Auctioneered High DeltaT Power for the RIL calculation. Per the COLR, the RIL for 83% power is 122 steps on CBD. Therefore, ALB10-D04 is a valid alarm.

The second part is correct. Per NMP-OS-001 step 6.1.2.4, once turbine load has been stabilized and RCS Tavg has been restored to within 3° of Tref, positive reactivity can be added by withdrawing control rods. Both of these conditions have been met. Therefore, control rod withdrawal is not restricted.

B. Incorrect. Plausible. The first part is correct. See the first part of Choice A above.

The second part is incorrect. Per NMP-OS-001 step 6.1.2.4, rod withdrawals are allowed if Tavg/Tref devation is within 3°F. However, NMP-OS-001 step 6.1.2.4 states it is non-conservative to withdraw control rods in response to a transient and anomalies are to be mitigated utilizing the secondary plant. It is reasonable for a candidate not familiar with the specific guidance of NMP-OS-001 to believe that the Tavg/Tref of 2°F is still considered a "transient" and not allow the use of control rods. Therefore, this distractor is plausible.

C. Incorrect. Plausible. The first part is incorrect. Per the IPC printout, RCS DeltaT Power is approximately 83% and CBD position is 116 steps. Per the COLR, the RIL for 83% power is 122 steps on CBD. Therefore, ALB10-D04 is a valid alarm. However, if the candidate does not understand how RIL is calculated and uses NIS Power instead of DeltaT Power, an RIL of 105 steps would be determined and the candidate would conclude the alarm was

not valid. Therefore, this distractor is plausible.

The second part is correct. See the second part of choice B above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

SRO JUSTIFICATION (10CFR43(b))

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, specific knowledge of Reactivity management during a transient per NMP-OS-001 is required.
-Can the question be answered *solely* by knowing immediate operator actions? No, there are not associated IOA's.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, the only associated AOP (18013-C) does not directly address the RIL alarm or the associated temperature requirement.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, the conditions are specific to NMP-OS-001.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- **Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. Yes, the question requires specific knowledge of NMP-OS-001, which is an administrative procedure on Reactivity Management and more specifically the guidance associated with management of a transient condition and the duties of the SRO (Reactivity Management SRO) when reactivity manipulations are being performed.**

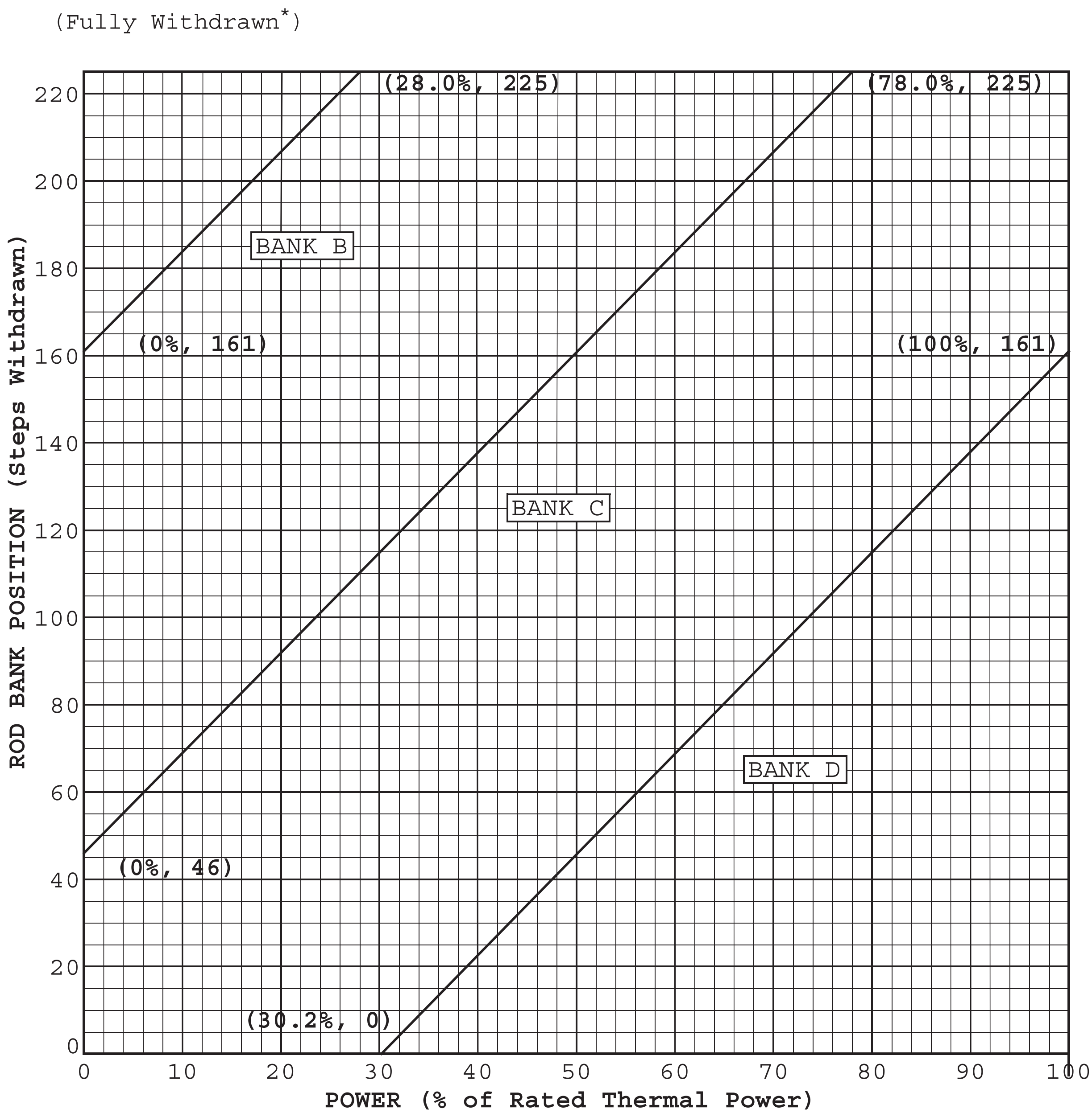
(6) Procedures and limitations involved in initial core loading, alternations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

The question is about "Procedures and limitations involved in control rod movement as it relates to internal/external effects on core reactivity". Although not listed as an example for this category in the "Clarification Guidance for SRO-only Questions", this question does include procedural administrative requirements and controls associated with external effects on core reactivity.

Level:	SRO
Tier # / Group #	T2 / G2
K/A#	001G2.1.19
Importance Rating:	3.9 / 3.8
Technical Reference:	NMP-OS-001 Rev 17.0, page 10 Unit 1 Cycle 18 COR, Figure 3 ARP 17010-1 Rev 50, page 3 and 41
References provided:	Plant Computer RIL screenshot and COLR Figure 3
Learning Objective:	LO-LP-39205-07 State the reasons for maintaining rods above the RIL. LO-LP-60301-08 Describe how placing the delta T defeat switch to a failed channel will affect the response of the rod insertion limit computer. LO-PP-27101-21 State the alarms associated with the rod insertion limits; include set points and source of the set points. LO-TA-05002 Obtain Data From the Integrated Plant Computer using 13505-1/2
Question origin:	BANK - Hatch 2011 NRC Question # G2.1.37
Cognitive Level:	C/A
10 CFR Part 55 Content:	43.5
Comments:	Early submittal 401-9 response: Question appears to match the KA. Question appears to be at the SRO level. Question appears to be okay as-is. - JAT 12/19/13 (SAT)

You have completed the test!

FIGURE 3
ROD BANK INSERTION LIMITS VERSUS % OF RATED THERMAL POWER



*Fully withdrawn shall be the condition where control rods are at a position within the interval ≥ 225 and ≤ 231 steps withdrawn.

NOTE: The Rod Bank Insertion Limits are based on the control bank withdrawal sequence A, B, C, D and a control bank tip-to-tip distance of 115 steps.

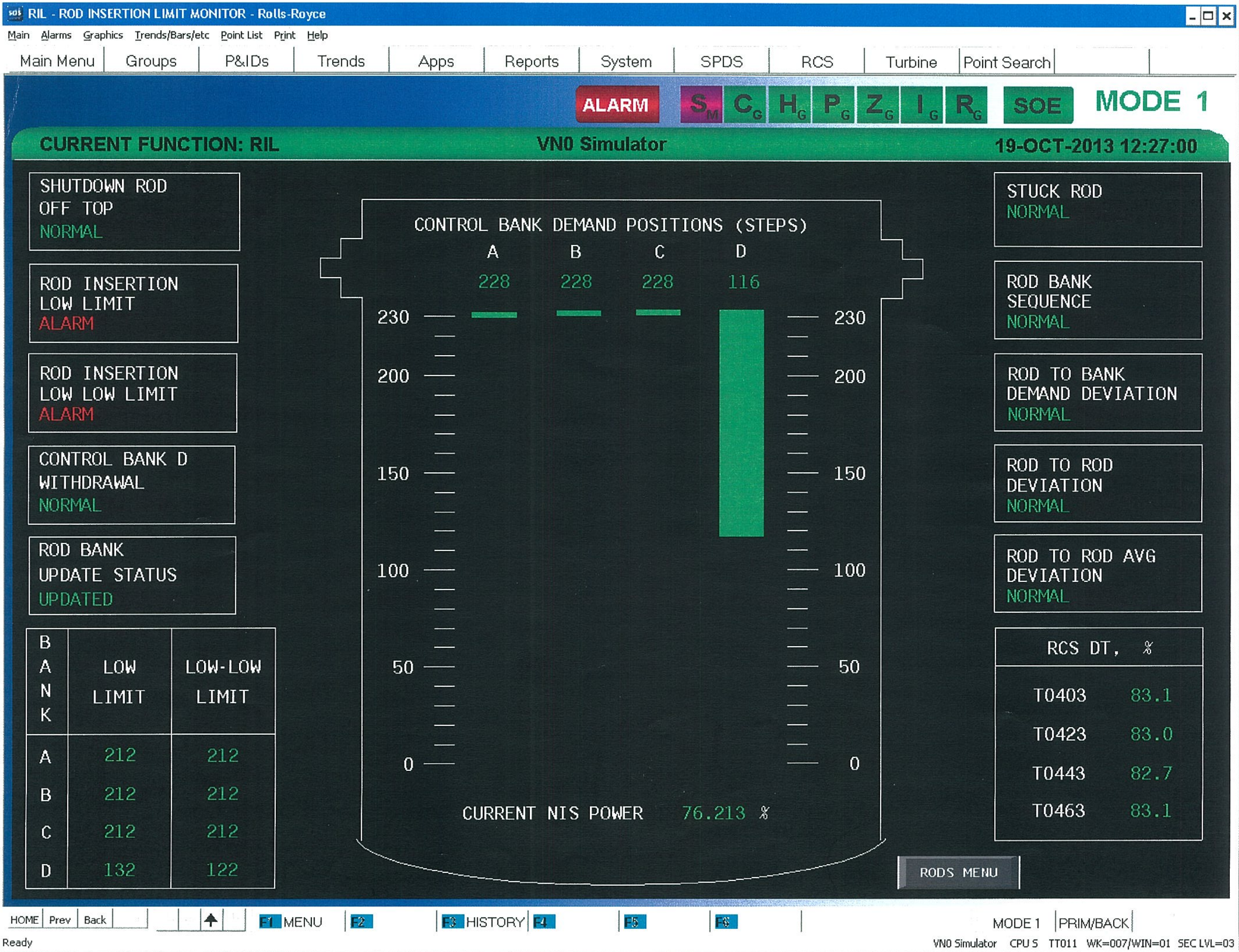
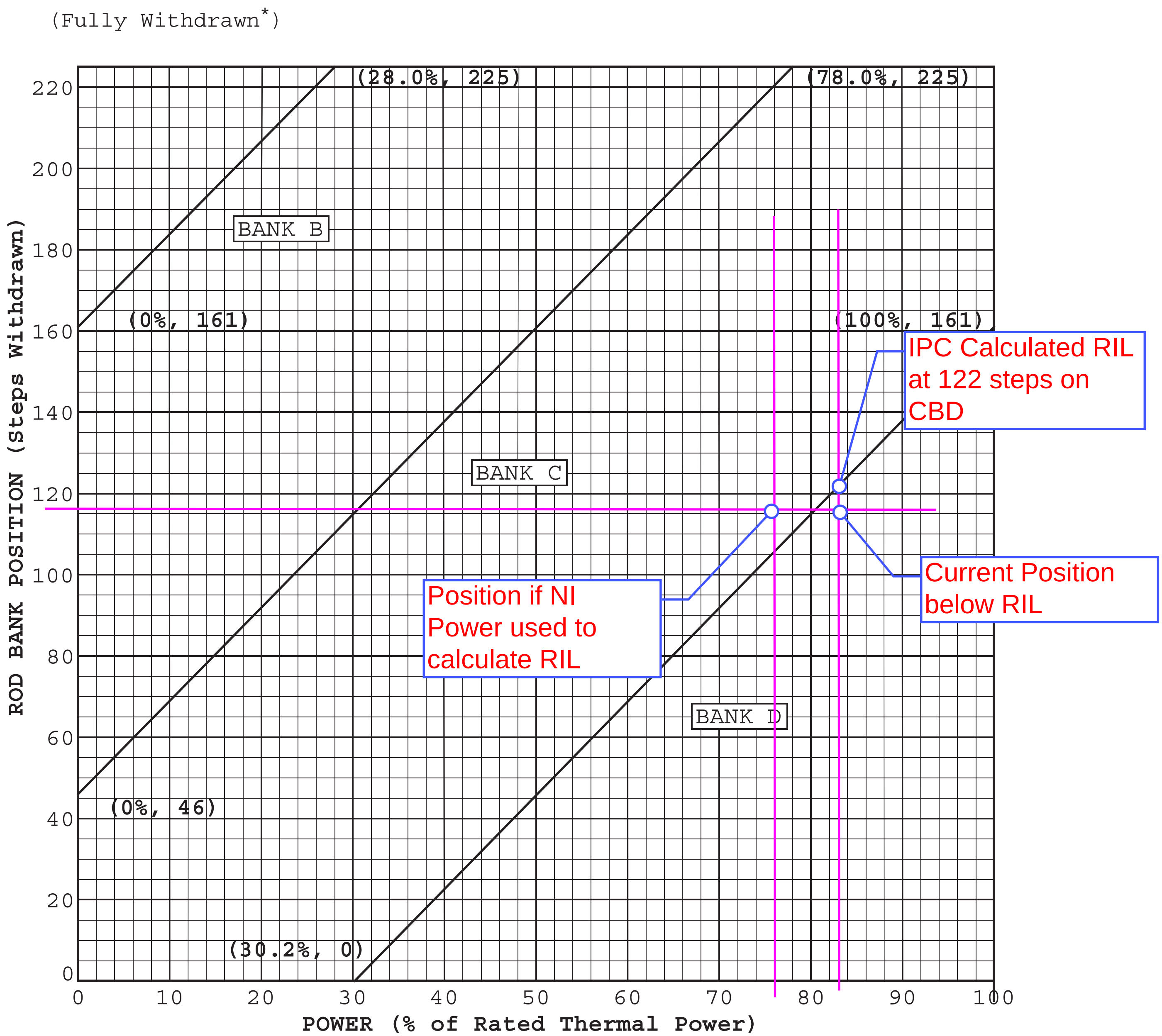

RIL - ROD INSERTION LIMIT MONITOR - Rolls-Royce
Main Alarms Graphics Trends/Bars/etc Point List Print Help
Main Menu Groups P&IDs Trends Apps Reports System SPDS RCS Turbine Point Search
ALARM
SOE
MODE 1
CURRENT FUNCTION: RIL
VN0 Simulator
19-OCT-2013 12:27:00
SHUTDOWN ROD OFF TOP NORMAL
ROD INSERTION LOW LIMIT ALARM
ROD INSERTION LOW LOW LIMIT ALARM
CONTROL BANK D WITHDRAWAL NORMAL
ROD BANK UPDATE STATUS UPDATED
BANK | LOW LIMIT | LOW-LOW LIMIT
A | 212 | 212
B | 212 | 212
C | 212 | 212
D | 132 | 122
CONTROL BANK DEMAND POSITIONS (STEPS)
A B C D
228 228 228 116
230 200 150 100 50 0
CURRENT NIS POWER 76.213 %
STUCK ROD NORMAL
ROD BANK SEQUENCE NORMAL
ROD TO BANK DEMAND DEVIATION NORMAL
ROD TO ROD DEVIATION NORMAL
ROD TO ROD AVG DEVIATION NORMAL
RCS DT, %
T0403 83.1
T0423 83.0
T0443 82.7
T0463 83.1
RODS MENU
HOME Prev Back
F1 MENU F2 F3 HISTORY F4 F5 F6
MODE 1 PRIM/BACK
Ready
VN0 Simulator CPU 5 TT011 WK=007/WIN=01 SEC LVL=03

FIGURE 3
ROD BANK INSERTION LIMITS VERSUS % OF RATED THERMAL POWER



*Fully withdrawn shall be the condition where control rods are at a position within the interval ≥ 225 and ≤ 231 steps withdrawn.

NOTE: The Rod Bank Insertion Limits are based on the control bank withdrawal sequence A, B, C, D and a control bank tip-to-tip distance of 115 steps.

ALB 10

	(1)	(2)	(3)	(4)	(5)	(6)
A	SR/IR SIG PROCESSOR TROUBLE	NIS SOURCE AND INTMD RANGE TRIP BYPASS	POWER RANGE HI NEUTRON FLX HI SETPOINT ALERT	REACTOR BYPASS BRKR BYA IN-OPERATE	REACTOR BYPASS BRKR BYA CLOSE	ROD CONTROL NON URGENT FAILURE
B	SOURCE RNG HI SHUTDOWN FLUX ALARM BLOCKED		POWER RANGE HI NEUTRON FLX LOW SETPOINT	REACTOR BYPASS BRKR BYB IN-OPERATE	REACTOR BYPASS BRKR BYB CLOSE	ROD CONTROL URGENT FAILURE
C	SOURCE RANGE HI FLUX LEVEL AT SHUTDOWN	POWER RANGE CHANNEL DEVIATION	OVERPOWER ΔT ROD BLOCK AND RUNBACK ALERT	ROD BANK LO LIMIT	RPI NON URGENT ALARM	NIS CHANNEL ON TEST
D	INTMD RANGE HI FLUX LEVEL ROD STOP	PWR RANGE UP DET HI FLX DEV	OVERPOWER ROD STOP	ROD BANK LO-LO LIMIT	RPI URGENT ALARM	ROD DEV
E	SR/IR REMOTE SIG PROCESSOR DPU-B TROUBLE	PWR RANGE LWR DET HI FLX DEV	OVERTEMP ΔT ROD BLOCK AND RUNBACK ALERT		ROD AT BOTTOM	RADIAL TILT
F	SR/IR AMPLIFIER TROUBLE	POWER RANGE HI NEUTRON FLX RATE ALERT		ROD DRIVE M-G SET TROUBLE	TWO OR MORE RODS AT BOTTOM	DELTA FLUX DEVIATION

IPC and Rod Control function

WINDOW D04

ORIGIN	SETPOINT	
IPC Calculated UD0366	Rod Insertion Limit	ROD BANK LO-LO LIMIT

1.0 **PROBABLE CAUSE**

RCS Boron concentration too low for present reactor power level due to:

1. Plant transient.
2. Xenon transient.
3. IPC failure

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** indications and determine if actual control bank rod position is below the Lo-Lo insertion limit by referring to the COLR and Technical Specification LCO 3.1.6.

2. IF actual control bank position is below the Lo-Lo Insertion Limit, **perform** the following:

    a. Within 1 hour:

    Verify shutdown margin is within the limits specified in the COLR per 14005-1 “Shutdown Margin Calculation”; **Refer To** TR 13.1.1 for applicability.

    OR

    Initiate and **maintain** Emergency Boration per 13009-1, “CVCS Reactor Makeup Control System”, until the Control Banks Lo-Lo Limit Annunciator clears.

    b. **Restore** the affected control bank(s) above the limit within 2 hours.

6.1.2.3 Administrative Controls

The following administrative requirements ensure that nuclear safety is maintained during activities that affect reactivity:

- Planned reactivity manipulations are peer checked.
- Calculations that involve reactivity control are independently verified prior to use.
- Reactor operators use redundant instrumentation when monitoring the effects of reactivity manipulations.
- Reactor engineering is actively engaged in activities that change reactivity significantly, including any special tests with the potential to affect reactivity.
- The reactor operator performing rod movement activities is free from distractions and will have no other duties while performing reactivity manipulations.

See Attachment 2 for summarized expectations for reactivity manipulations.

6.1.2.4 Conduct of Reactivity Changes

Operators anticipate the effects of reactivity manipulations and monitor core parameters carefully until parameters stabilize. Any unanticipated reactivity change is immediately brought to the attention of management and the resolution of the change is pursued to its conclusion.

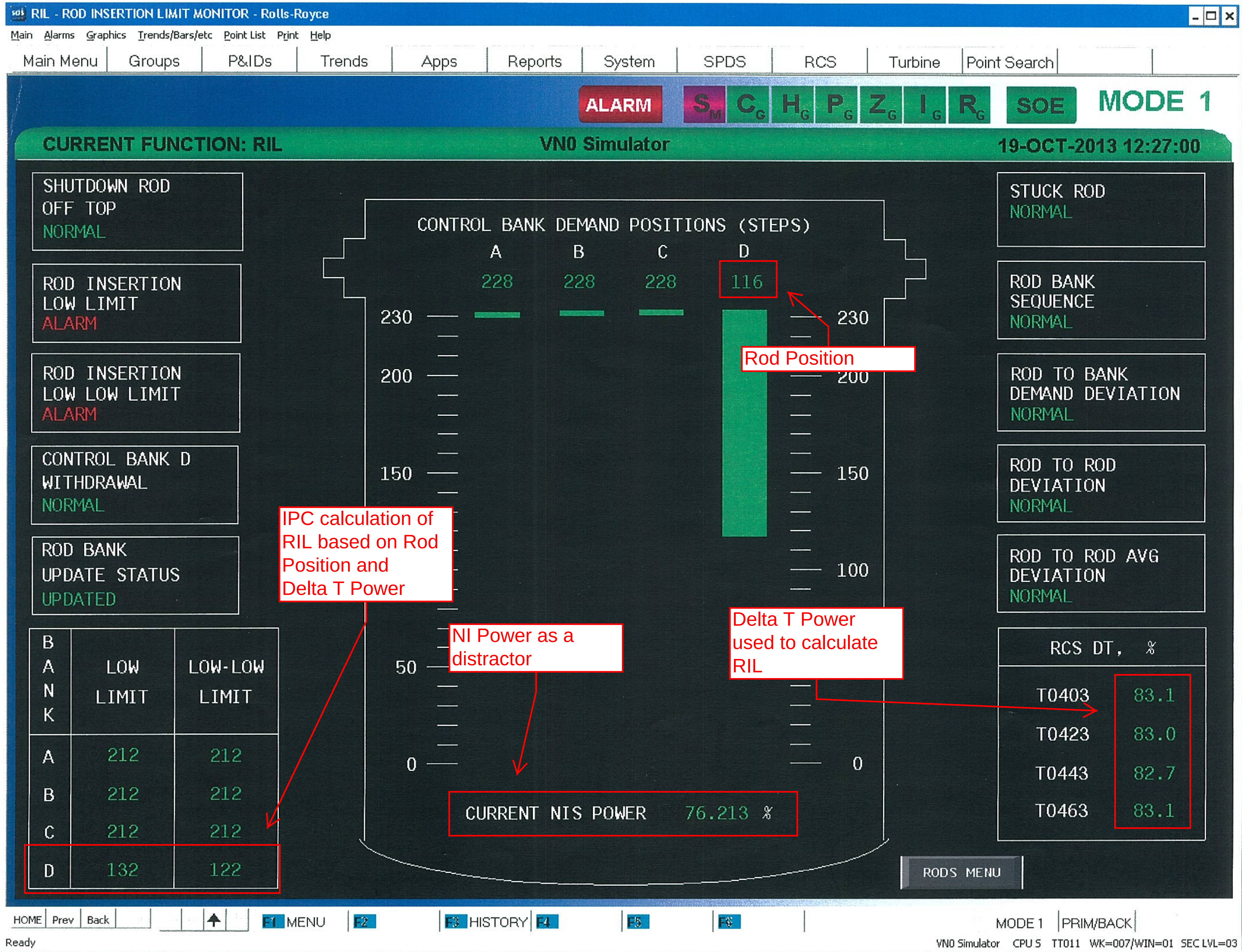
Adding positive reactivity is never an appropriate way to address unstable plant conditions. It is non-conservative to withdraw control rods in response to primary plant anomalies caused by unplanned secondary plant transients. For the PWRs, once turbine load has been stabilized and RCS Tavg has been restored to within 3 degrees of Tref, positive reactivity can be added by withdrawing control rods.

Whenever the status of reactor criticality becomes unknown, the reactor is shutdown.

Under normal conditions, positive reactivity changes will not be performed by more than one means at a time.

During approach to criticality, two positive reactivity additions will not be performed simultaneously.

The operator at the controls shall suspend turnover if a reactivity manipulation is required during turnover.



RIL - ROD INSERTION LIMIT MONITOR - Rolls-Royce
Main Alarms Graphics Trends/Bars/etc Point List Print Help
Main Menu Groups P&IDs Trends Apps Reports System SPDS RCS Turbine Point Search
ALARM
SOE
MODE 1
CURRENT FUNCTION: RIL
VN0 Simulator
19-OCT-2013 12:27:00
SHUTDOWN ROD OFF TOP NORMAL
ROD INSERTION LOW LIMIT ALARM
ROD INSERTION LOW LOW LIMIT ALARM
CONTROL BANK D WITHDRAWAL NORMAL
ROD BANK UPDATE STATUS UPDATED
CONTROL BANK DEMAND POSITIONS (STEPS)
A B C D
228 228 228 116
230 200 150 100 50 0
STUCK ROD NORMAL
ROD BANK SEQUENCE NORMAL
ROD TO BANK DEMAND DEVIATION NORMAL
ROD TO ROD DEVIATION NORMAL
ROD TO ROD AVG DEVIATION NORMAL
BANK LOW LIMIT LOW-LOW LIMIT
A 212 212
B 212 212
C 212 212
D 132 122
RCS DT, %
T0403 83.1
T0423 83.0
T0443 82.7
T0463 83.1
CURRENT NIS POWER 76.213 %
RODS MENU
Rod Position
IPC calculation of RIL based on Rod Position and Delta T Power
NI Power as a distractor
Delta T Power used to calculate RIL
HOME Prev Back
F1 MENU F2 F3 HISTORY F4 F5 F6
MODE 1 PRIM/BACK
Ready
VN0 Simulator CPU 5 TT011 WK=007/WIN=01 SEC LVL=03

1. 003G2.4.3 001/LOIT AND LOCT/SRO/M/F 3.7/3.9/003G2.4.3/LO-TA-37015///

Initial conditions:

- Unit 1 experienced a small break LOCA.
- 19000-C, "Reactor Trip or Safety Injection," is in progress.
- Loop 3 RCS Tcold instrumentation is not available.

Current condition:

- RCS pressure is 1350 psig and slowly lowering.

Which one of the following completes the following statement?

Stopping the RCPs is required to minimize the risk of __(1)__,

and

per the Bases of Tech Spec 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," for diverse indication of RCS Tcold temperature, the operator is directed to use __(2)__ instrumentation.

	__(1)__	__(2)__
A.	core uncovery	RCS Thot
B✓	core uncovery	steam generator pressure
C.	RCP damage	RCS Thot
D.	RCP damage	steam generator pressure

K/A

003 Reactor Coolant Pump

G2.3.4 Ability to identify post-accident instrumentation.

K/A MATCH ANALYSIS

The question tests the candidate's ability to identify the post-accident instrument that would be utilized as a diverse indication of RCS Tcold temperature with RCPs secured.

EXPLANATION OF REQUIRED KNOWLEDGE

Per WOG Background for RCP Trip, the reason for purposely tripping the RCPs during a small break LOCA is to prevent excessive depletion of RCS water inventory through a saml break in the RCS which might lead to severe core uncovery if the RCPs were tripped for some reason later in the accident. The RCPs should be tripped before the RCS inventroy is depleted to the point where tripping of the pumps would cause the

break to immediately uncover. The WOG gives options in parameter that may be utilzied as RCP trip criteria. Vogtle has chosen the option based solely on RCS pressure. Therefore, RCPs are manually tripped if RCS pressure is <1375 psig provided either CCPs or SIPs are injecting into the core.

Per TS 3.3.3 FU 2,3 Bases, steam line pressure provides diverse indication for the RCS cold leg temperature. With either forced or natural circulation flow through the steam generators, SGs will be at saturation pressure for the RCS Cold Legs.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Part 1 of the answer is correct and states the reason provided in the Westinghouse background documents for tripping the Reactor Coolant Pumps for small break LOCAs as the potential for core damage due to core uncover and exceeding peek centerline temperatures criteria.

Part 2 is not correct but is 'plausible' because T-hot instrumentation is addressed in the same bases as diverse indication for the CETCs as opposed to Tcold instrumentation. The candidates would see there is a relationship since both are measuring temperature as opposed to the correct instrument which is using pressure.

B. Correct. Part 1 is correct. See Part 1 of choice A above.

Part 2 is correct. Per Tech Spec 3.3.3 'Post Accident Monitoring Instrumentation,' steam line pressure provides diverse indication for the RCS cold leg temperature.

C. Incorrect. Plausible. Part 1 is not correct for the small break LOCA but would be true for large break LOCAs since the Reactor Coolant Pumps would be stopped under this condition due to loss of support conditions for continued operation and subsequent pump damage.

Part 2 is not correct. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is not correct. See Part 1 of choice C above.

Part 2 of the answer is correct. See Part 2 of choice B above.

SRO JUSTIFICATION (10CFR43(b))

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the question is not addressing Tech Spec action times.
-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" No, the question is not addressing Tech Spec above-the-line information.

-Can question be answered *solely* by knowing the TS Safety Limits? No, the question is not related to Tech Spec Safety Limits.

-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- **Knowledge of TS bases that is required to analyze TS required actions and terminology. Yes, the answer to the question is only found in Tech Spec bases.**

Level:	SRO	
Tier # / Group #	T2 / G1	
K/A#	003G2.4.3	
Importance Rating:	3.7 / 3.9	
Technical Reference:	Westinghouse Background - RCP Trip Rev 2, 4/30/2005 Tech Spec 3.3.3 Bases page B 3.3.3-6, Rev 0	
Reference Provided:	None	
Learning Objective:	LO-TA-37006	Conduct a Natural Circulation Cooldown per 19002-C
	LO-TA-37015	Perform the Initial Recovery Actions for a small Loss of Reactor or Secondary Coolant per 19010-C
	LO-TA-63013	Implement Technical Specification LCO using 10008-C (SRO Only)
	LO-LP-39207-04	Describe the bases for any given Tech Spec in section 3.3.
	LO-LP-39208-01	For any given item in section 3.4 of Tech Specs, be able to: State the LCO. State any one hour or less required actions.
Question origin:	NEW	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.10 / 43.2	
Comments:		

You have completed the test!

BASES

LCO

2,3. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures (Wide Range) (continued)

RCS hot and cold leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for unit stabilization and cooldown control.

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS.

Reactor outlet temperature inputs to the Reactor Protection System are provided by two fast response resistance elements and associated transmitters in each loop. The channels provide indication over a range of 50°F to 700°F.

The core exit thermocouples provide diverse indication for the RCS hot leg temperature.

Steam line pressure provides diverse indication for the RCS cold leg temperature.

4. Steam Generator Water Level (Wide Range)

Wide range SG water level (Loops 501, 502, 503, & 504) is a Type A variable used to determine if an adequate heat sink is being maintained through the SGs for decay heat removal, primarily for the response to a loss of secondary heat sink event when the level is below the narrow range. The wide range SG level indication may also be used in conjunction with auxiliary feedwater flow for SI termination. In addition, the wide range level is cold calibrated and provides a complete range for monitoring SG level during a cooldown. Auxiliary feedwater flow provides the diverse indication for wide range SG water level.

(continued)

of the RCPs during a LOCA cannot be guaranteed since tripping of the RCPs would occur upon a loss of offsite power or other essential support conditions which can be postulated to occur at any time. The reason for purposely tripping the RCPs during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCPs were tripped for some reason later in the accident. The RCPs should be tripped before RCS liquid inventory is depleted to the point where tripping of the pumps would cause the break to immediately uncover.

2.2.2 Non-LOCA Accidents

In virtually all non-LOCA accidents, it is advantageous to have the RCPs in operation. Either this provides additional margin to safety criteria limits or makes operator actions during recovery easier. However, whether or not the RCPs remain in operation or are tripped, safety criteria must be met and plant operators are provided with guidance to mitigate and to recover from the accident. For accidents involving loss of secondary coolant, control of RCS pressure, RCS temperature, and pressurizer level is the major concern, rather than core cooling. For the various types of SGTR events, (either single or multiple ruptures) control of the leak rate, RCS pressure, RCS temperature, and pressurizer level is important. In all cases, RCP operation provides enhanced core heat removal and makes RCS pressure control by the operator a more straight forward matter. In general, for non-LOCA accidents, it is desirable to have the RCPs in operation throughout the event.

The NRC (References 1 and 2) makes this preference clear in the requests for development of RCP trip setpoints based on parameters which will allow the operation of some (or all) of the RCPs during those accidents which will benefit from them, yet result in a trip of RCPs for SBLOCAs and others which require it. Development of these parameters is discussed in subsection 2.3.

2.3 RCP Trip Criteria

RCP trip criteria have been developed and incorporated into the ERGs to provide for RCP trip when required for SBLOCAs and to minimize the probability

1. 004G2.2.44 001/LOIT AND LOCT/SRO/C/A 4.2/4.4/004G2.2.44/LO-TA-16009///

Given the following procedure titles:

- 19000-C, "Reactor Trip or Safety Injection"
- 13003-1, "Reactor Coolant Pump Operation"
- 18005-C, "Partial Loss of Flow"

Initial condition:

- Unit 1 is at 16% reactor power with a startup in progress.

Current conditions:

- ALB08-A04 RCP 1 NO. 2 SEAL LKOF HI FLOW is received.
- ALB08-A05 RCP 1 CONTROLLED LKG HI/LO FLOW is received.
- 1FI-160A, #1 SEAL LEAK-OFF for RCP #1 is indicating 6.0 gpm.

Which one of the following completes the following statement?

RCP #1, seal number __(1)__ has failed,

and

per 13003-1, the Shift Supervisor will direct the crew to __(2)__.

	__(1)__	__(2)__
A.	one	initiate 18005-C
B.✓	one	trip the reactor and initiate 19000-C
C.	two	initiate 18005-C
D.	two	trip the reactor and initiate 19000-C

K/A

004 **Chemical and Volume Control**

G2.2.44 **Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.**

K/A MATCH ANALYSIS

The question tests the candidate’s ability to interpret the control room annunciators associated with RCP seal #1 and #2 leak-offs. Seal injection and leakoff are part of the CVCS system per the K/A catalog. The candidate is then required to utilize these annunciators to diagnose the issue and select the appropriate procedure path to

address the degraded conditions.

EXPLANATION OF REQUIRED KNOWLEDGE

Per ARP 17008-1 window A05, either low or high seal flow can actuate the alarm. High seal flow is an indication of Seal #1 failure, and low seal flow is an indication of Seal #2 failure. The candidate is given a seal flow of 6.0 gpm to allow differentiation between these two. Window A04 lists a probable cause of Seal #2 failure only. Knowledge of RCP seal package construction is required to understand why this annunciator would be consistent with a Seal #1 failure. As Seal #1 fails, the seal surfaces open and backpressure lowers within the package. Flow down the shaft lowers and increases both to the leak-off path and to Seal #2 .

Per 13003-1, Seal #1 leak-off >5.5 gpm requires stopping the RCP immediately per step 4.2.1.4. With reactor power greater than 15% RTP, the operator is directed to trip the reactor and initiate 19000-C. When the IOA's are complete, the operator is direct to perform steps 4.2.1.4.d thru h to stop the RCP, close the associated spray valve, and isolate seal #1 leak-off.

The question stem specifies 1FI-160A, #1 SEAL LEAK-OFF for RCP #1 is indicating 6.0 gpm. This is top of scale for the indicator. Simulations show that a seal failure equivalent to approximately 9 gpm leak leak-off is required for annunciators ALB08-A04 RCP 1 NO. 2 SEAL LKOF HI FLOW and ALB08-A05 RCP 1 CONTROLLED LKG HI/LO FLOW to be in alarm. At a seal leak-off of >4.8 gpm and <9 gpm, only annunciator ALB08-A05 RCP 1 CONTROLLED LKG HI/LO FLOW is in alarm.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. The combination of annunciator alarms and elevated seal #1 leak-off are symptoms of a seal #1 failure.

The second part is incorrect. With seal #1 leak-off > 5.5 gpm, an immediate shutdown of the RCP is required. Per 13003-1 step 4.2.1.4, the reactor must be tripped and 19000-C initiated. However, if a candidate does not realize RTP is >15%, then step 4.2.1.4 would direct initiation of AOP 18005-C.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. With seal #1 leak-off > 5.5 gpm, an immediate shutdown of the RCP is required. Per 13003-1 step 4.2.1.4, the reactor must be tripped and 19000-C initiated.

C. Incorrect. Plausible. The first part is incorrect. The combination of annunciator alarms and elevated seal #1 leak-off are symptoms of a seal #1 failure. However, if the candidate does not recognize 1FI-160A indicating 6.0 gpm as being abnormally high, then annunciator ALB08-A04 RCP 1 NO. 2 SEAL LKOF HI FLOW would be a symptom of a seal #2 failure.

The second part is incorrect. With seal #1 leak-off > 5.5 gpm,

an immediate shutdown of the RCP is required. Per 13003-1 step 4.2.1.4, the reactor must be tripped and 19000-C initiated. However, if a candidate has diagnosed a seal #2 failure, 13003-C still supports an immediate stop of the RCP given other conditions. Since the question does not give an option to leave the RCP running, the candidate must assume a valid reason to stop the RCP has been met. As such, if the candidate then does not realize RTP is >15%, then step 4.2.1.4 would direct initiation of AOP 18005-C.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

<u>SRO JUSTIFICATION (10CFR43(b))</u>

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the first part is system knowledge however, the second part is discriminating to the SRO level as it requires an operational decision.
-Can the question be answered *solely* by knowing immediate operator actions? No, the requrired actions are specific direction associated with ARPs and SOPs and a shutdown decision must be made.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, entry conditions to 18005-C and 19000-C both address plant conditions in a generic nature. Specific procedure knowledge is required to differentiate the required procedure flow path.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, the question requires specific knowledge of a specific SOP step.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. **Yes, the candidate must use specific knowledge of SOP 13003-C decision flow charts from memory as well as specific knowledge of step 4.2.1.4 in order to direct the specific actions to be taken to shutdown the reactor and address the RCP seal failure.**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Level:	SRO
Tier # / Group #	T2 / G1
K/A#	004G2.2.44
Importance Rating:	4.2 / 4.4

Technical Reference: ARP 17008-1, Rev 18.0, pages 10-12
SOP 13003-1, Rev 47.1, pages 13-15 & 38-40
AOP 18005-C, Rev 11.1, pages 1 & 3
V-LO-PP-16401, Rev 5.4, pages 12-14

References provided: None

Learning Objective:

LO-PP-16401-02	Describe the function of RCP seals 1, 2, and 3 including DP across each seal and expected flow rate.
LO-PP-16401-03	Describe the control room indications for a failure of a RCP seal.
LO-TA-60015	Respond to a Partial Loss of Flow per 18005-C
LO-TA-16009	Respond to abnormal RCP seal per 13003-1/2

Question origin: MODIFIED - HL14 NRC Question #015/017G2.4.4

Cognitive Level: C/A

10 CFR Part 55 Content: 43.5

Comments:

- JAT 12/19/13 (U/E)
  Early submittal 401-9 response:
  Amanda's comments incorporated to include removing the VCT pressure and trend in the stem and moved the question to procedure knowledge as opposed to Tech Spec bases.

- JAT 2/4/14 (U/E)
  Response following revision from early submittal:
  The first part of the question is improved. However, I am having difficulty seeing the TS completion time connection to the second part of the question, and the way it's written, it appears as though the question can be answered solely by knowing what specific direction is contained within the ARP (which is likely not SRO-only, because it does not involve selection of procedures or sections of a procedure, nor does it require knowledge of >1h TS.).

- JCC 2/5/14
  Question replaced with modified question from HL14 NRC associated with RCP seal abnormality and procedure

You have completed the test!

1. 015/017G2.4.4 002/1/1/RCP MALF - EOP ENTRY/C/A - 4.3/MODIFIED/SRO/HL-14 NRC/TNT / RLM

Unit 2 is at 12% power when the following annunciators are received.

- ALB08B05 "RCP # 3 CONTROLLED LKG HI / LO FLOW"

- ALB08B04 "RCP # 3 NO. 2 SEAL LKOF HI FLOW"

Original Question

The OATC reports the following indications:

- RCP # 3 seal leakoff flow Hi Range meter is 6.0 gpm.

- RCP # 3 seal injection flow is 9.9 gpm.

- RCP # 3 Seal Water Inlet temperature is 223°F and stable.

Which one of the the following is the correct procedurally directed action(s) for the SS to take?

A. Per 12004-C, "Power Operations (Mode 1)", commence a unit shutdown to be in Mode 3 in 8 hours.

B. Trip the reactor and enter 19000-C, "E-0 Reactor Trip or Safety Injection", per 13003-2, "RCP Operation", stop RCP # 3 and close seal leakoff valve HV-8141C.

C.✓ Per 13003-2, stop RCP # 3, close seal leakoff valve HV-8141C, enter 18005-C, "Partial Loss of RCS Flow", commence unit shutdown per 12004-C.

D. Per 12004-C, maintain reactor power at 25%, monitor the RCP per 13003-2 section 4.2.1 "Pump Operation With A Seal Abnormality", contact Duty Engineering.

ABNORMAL OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure addresses the loss of forced RCS flow during power operation below the P-8 (48%) setpoint.

SYMPTOMS

- ALB11-E04 RCP TRIP
- ALB12-A01(B01, C01, D01) RCP LOOP 1 (2, 3, 4) LOW FLOW ALERT
- ALB08-A01(B01, C01, D01) RCP 1 (2, 3, 4) MTR OVERLOAD
- ALB11-E06 UNDERVOLTAGE RCP BUS ALERT
- ALB11-F06 UNDERFREQUENCY RCP BUS ALERT

UNIT 1

- ALB33-A01(A02) 13.8KV SWGR 1NAA(1NAB) TROUBLE

UNIT 2

- ALB33-A01(A02) 13.8KV SWGR 2NAA(2NAB) TROUBLE

MAJOR ACTIONS

- ♦ Stabilize plant conditions.
- ♦ Shutdown to Mode 3.
- ♦ Restart RCP.
- ♦ Select appropriate UOP.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Check Reactor power - LESS THAN OR EQUAL TO 15%.	1. Perform the following: a. Trip the Reactor. b. Go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
2. Stop any power changes in progress.	
3. Initiate the Continuous Actions Page.	
*4. **Check affected loop SG NR Level - TRENDING TO 65%.**	*4. Control feed flow to maintain affected loop SG NR level between 60% and 70%.
5. Check Tavg - TRENDING TO PROGRAM.	5. Adjust control rods to restore Tavg.
6. Verify PRZR level - TRENDING TO PROGRAM.	
7. Verify PRZR pressure - TRENDING TO 2235 PSIG.	
8. Check RCP 1 and RCP 4 – RUNNING.	8. Close the affected loop spray valve: Loop 1: PIC-0455C Loop 4: PIC-0455B
9. Initiate shutdown to Mode 3 by initiating 12004-C, POWER OPERATION (MODE 1). (TS 3.4.4)	
10. Determine and correct the cause of the pump trip.	
11. Check shutdown to Mode 3 – COMPLETE.	11. Return to Step 9.

INITIALS

4.2 SYSTEM OPERATION

4.2.1 Pump Operation With A Seal Abnormality

4.2.1.1 IF the Plant Computer is available, **trend** the computer data points listed in Table 2. ________

4.2.1.2 IF the Plant Computer is NOT available, perform the following**:**

a. **Monitor** the QMCB indication listed in Table 2 at least hourly for the next 8 hours. ________

b. IF NO further seal degradation exists after 8 hours, **consult** with the Shift Supervisor (SS) for less frequent monitoring. ________

4.2.1.3 **Monitor** the No. 1 seal for further degradation using Figure 1 and RCP Trip Criteria as follows:

a. **Evaluate** the monitored indications using Figure 1, "RCP Seal Abnormalities Tree." ________

b. IF evaluation of the monitored indications using Figure 1 requires immediate pump shutdown, **Go to** Step 4.2.1.4. ________

c. IF any of the following RCP Trip Criteria is exceeded, **Go To** Step 4.2.1.4 for immediate RCP shutdown. ________

RCP TRIP CRITERIA	
Motor bearing temperature	>195°F
Motor stator-winding temperature	>311°F
Seal water inlet temperature	>230°F
RCP shaft vibration	≥20 mils
RCP Frame vibration	≥5 mils
#1 seal Differential Pressure	<200 psid
#1 seal leakoff flow (sum of #1 seal leakoff as indicated on the MCB and #2 seal leakoff read locally in containment)	< minimum on Figure 2 with pump bearing / seal inlet temperature increasing
Total loss of ACCW for a duration of 10 minutes	

INITIALS

d. WHEN directed by Figure 1, **stop** the affected RCP within 8 hours as follows:

(1) **Establish** 9 gpm or greater seal injection flow to the affected pump. ________

(2) **Stop** the affected RCP by continuing with Step 4.2.1.4. ________

4.2.1.4 WHEN directed by the SS, perform an RCP shutdown as follows:

a. **Start** the RCP Oil Lift Pump for affected RCP, if available. ________

b. IF Reactor Power is greater than 15% Rated Thermal Power:

(1) **Trip** the Reactor and **initiate** 19000-C, "E-0 Reactor Trip Or Safety Injection." ________

(2) WHEN the immediate operator actions of 19000-C are complete, **Go to** Step 4.2.1.4.d. ________

c. IF Reactor Power is less than 15% Rated Thermal Power, **initiate** 18005-C, “Partial Loss Of Flow.” ________

d. **Stop** the RCP by placing the RCP Non-1E Control Switch in STOP and then placing the RCP 1E Control Switch in STOP:

RCP	Non-1E Control Switch	1E Control Switch	
• Loop 1	1-HS-0495B	1-HS-0495A	________
• Loop 2	1-HS-0496B	1-HS-0496A	________
• Loop 3	1-HS-0497B	1-HS-0497A	________
• Loop 4	1-HS-0498B	1-HS-0498A	________

INITIALS

CAUTION

IF RCP #1 or #4 is stopped, the associated Spray Valve is placed in manual and closed to prevent spray short cycling.

e. IF RCP #1 OR #4 is stopped, **verify** its associated spray valve is placed in MANUAL and closed.

- RCP 1: 1-PIC-0455C ________
- RCP 4: 1-PIC-0455B ________

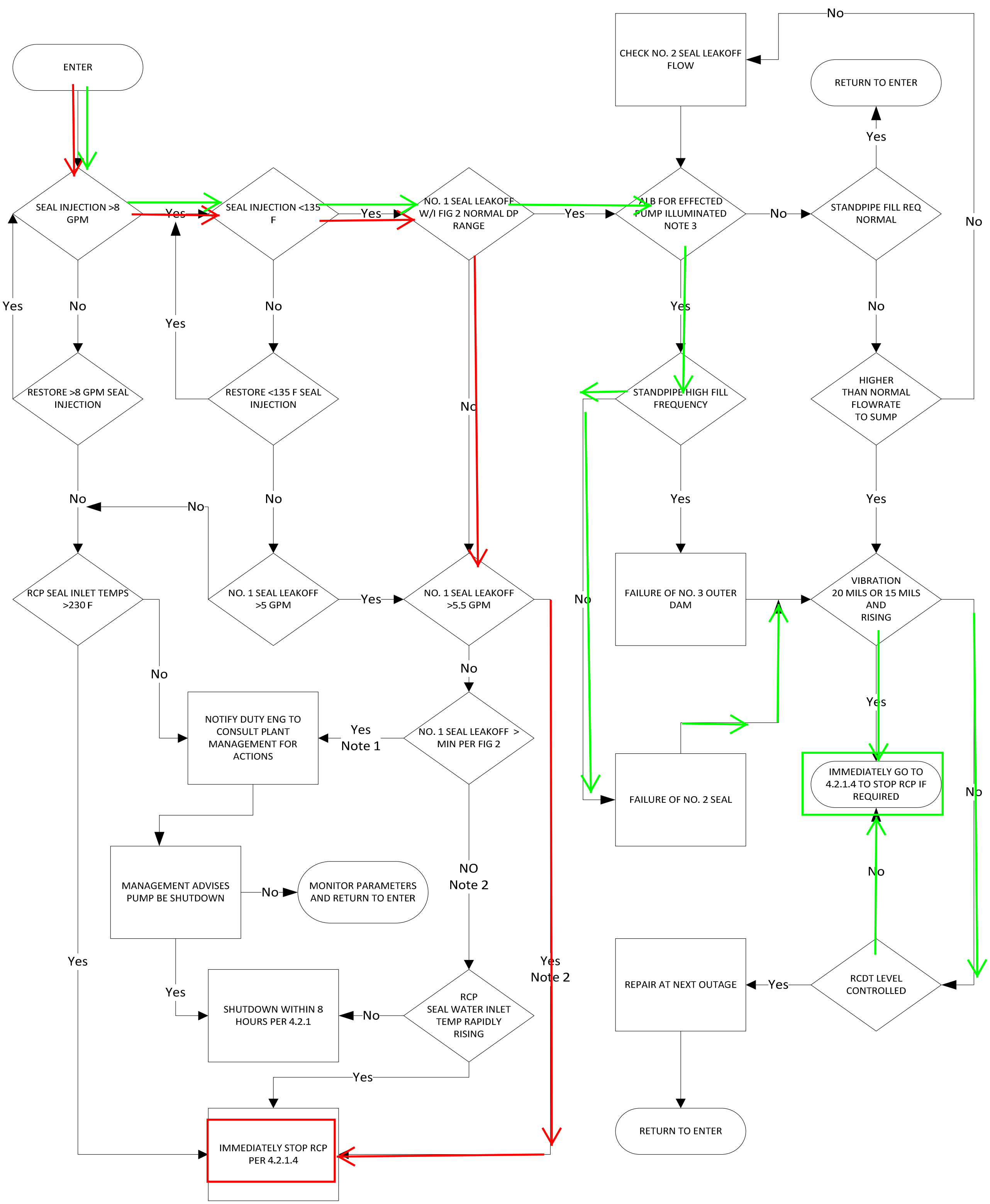
f. WHEN the RCP comes to a complete stop (as indicated by reverse flow), **close** the RCP Seal Leakoff Isolation valve for the affected pump.

- RCP 1: 1-HV-8141A ________
- RCP 2: 1-HV-8141B ________
- RCP 3: 1-HV-8141C ________
- RCP 4: 1-HV-8141D ________

g. **Secure** the associated RCP Oil Lift Pump. ________

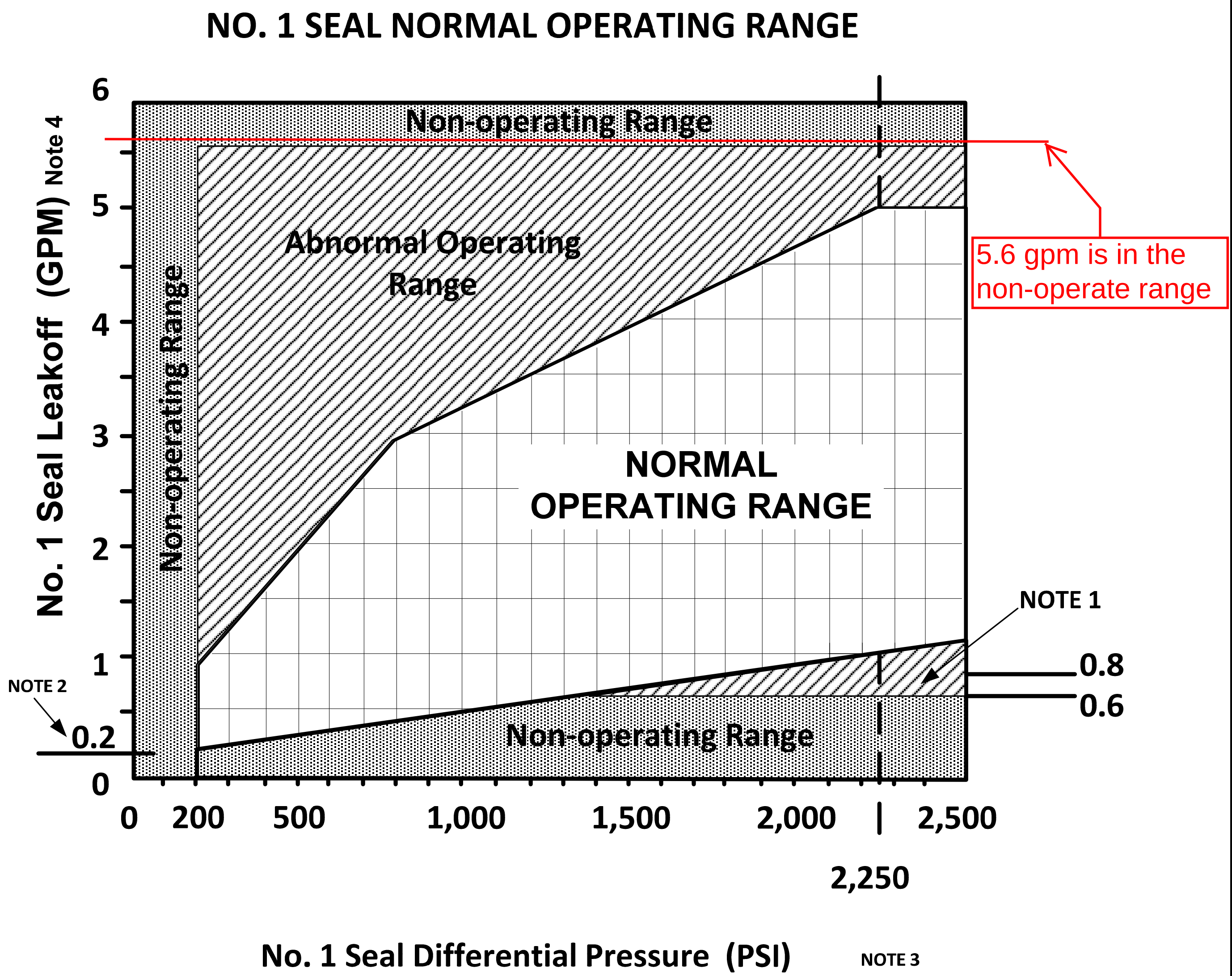
h. IF RCP shutdown was due to loss of RCP seal cooling, **review** Limitation 2.2.11 for recovery action. ________

FIGURE 1 - RCP SEAL ABNORMALITIES DECISION TREE



Note 1: Abnormal Operating Range of Figure 2
Note 2: Non-operating Range of Figure 2
Note 3: ALB08 A-04, B-04, C-04 or D-04

FIGURE 2



1. If the No. 1 seal leak rates are outside the normal (1.0-5.0 gpm) but within the operating limits ((0.8-5.5 gpm), continue pump operation. VERIFY that seal injection flow exceeds No. 1 seal leak rate for the affected RCP. Closely monitor pump and seal parameters and contact engineering for further instructions. IF the No.1 seal leak off is between 0.6 and 0.8 gpm within 8 hours determine the leak off with the No.1 plus No. 2 seals. IF the total leakoff is less than 0.8 gpm perform an orderly shutdown of the pump (see Note 4) within 8 hours. At 0.6 gpm on the No. 1 seal immediately shutdown the pump. (The 0.8 gpm and 0.6 gpm value includes the #2 seal leak off value from containment as well.)

2. Minimum startup requirements are 0.2 gpm at 200 PSID differential across the No. 1 seal. For startups at differential pressures greater than 200 PSID, the minimum No. 1 seal leak rate requirements are defined in the NO. 1 SEAL NORMAL OPERATING RANGE (e.g., at 1000 psi differential pressure, do not start the RCP with less than 0.5 gpm).

3. No.1 Seal Differential Press = RCS WR Press – VCT Press.

4. Per Westinghouse Technical Bulletin ESBU-TB-93-01-R1, total #1 seal leakoff is the sum of #1 seal leakoff and #2 seal leakoff. #1 seal leakoff is read directly at the MCB and #2 seal leakoff can be obtained from instrumentation in Containment.

	(1)	(2)	(3)	(4)	(5)	(6)
A	RCP 1 MTR OVERLOAD	RCP 1 STANDPIPE LO LEVEL	RCP 1 STANDPIPE HI LEVEL	RCP 1 NO. 2 SEAL LKOF HI FLOW	RCP 1 CONTROLLED LKG HI/LO FLOW	
B	RCP 2 MTR OVERLOAD	RCP 2 STANDPIPE LO LEVEL	RCP 2 STANDPIPE HI LEVEL	RCP 2 NO. 2 SEAL LKOF HI FLOW	RCP 2 CONTROLLED LKG HI/LO FLOW	
C	RCP 3 MTR OVERLOAD	RCP 3 STANDPIPE LO LEVEL	RCP 3 STANDPIPE HI LEVEL	RCP 3 NO. 2 SEAL LKOF HI FLOW	RCP 3 CONTROLLED LKG HI/LO FLOW	
D	RCP 4 MTR OVERLOAD	RCP 4 STANDPIPE LO LEVEL	RCP 4 STANDPIPE HI LEVEL	RCP 4 NO. 2 SEAL LKOF HI FLOW	RCP 4 CONTROLLED LKG HI/LO FLOW	RCP NO. 1 SEAL LO ΔP
E			RCP 1 VIBRATION ALERT	RCP 2 VIBRATION ALERT	RCP VIBRATION HIGH	RCP SEAL WATER INJ FILTER HI ΔP
F			RCP 3 VIBRATION ALERT	RCP 4 VIBRATION ALERT	RCP VIB MON PNL TROUBLE	RCP SEAL WATER INJ LO FLOW

WINDOW A04

ORIGIN	SETPOINT	RCP 1 NO. 2 SEAL LKOF HI FLOW
1-FIS-0194	0.9 gpm	

1.0 **PROBABLE CAUSE**

1. Number 2 Seal failure.
2. Sudden reduction in RCDT level or pressure.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** RCDT pressure on 1-PISL-9699 (QPCP) 3 psig or greater.
2. **Dispatch** Operator to check RCDT pressure and level at PLPP:
   a. Pressure 2-3 psig,
   b. Level 20-75%.
3. IF RCDT pressure and level are normal, **Go To** 13003-1, "Reactor Coolant Pump Operation" for instructions on RCP operation with seal malfunctions.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB114, 1X6AB09-119, PLS

WINDOW A05

ORIGIN	SETPOINT
1-FT-0161	4.8 gpm
1-FT-0157	0.8 gpm

RCP 1
CONTROLLED LKG
HI/LO FLOW

1.0 **PROBABLE CAUSE**

1. High Flow:
    a. Flashing in the Seal Leakoff Line due to loss of seal injection flow or high seal injection temperature,
    b. Failure of Number 1 Seal.
2. Low Flow:
    a. Low differential pressure across Number 1 Seal,
    b. High Volume Control Tank (VCT) pressure,
    c. Excess letdown in service,
    d. Failure of Number 2 Seal.

2.0 **AUTOMATIC ACTIONS**

NONE

WINDOW A05
(Continued)

3.0 INITIAL OPERATOR ACTIONS

NOTE

RCP 1 No. 1 seal water leakoff high range flow may be monitored using computer point F0161.

1. **Observe** seal injection flow and seal leakoff flow, as well as excess letdown temperature and pressure for indication of an actual seal anomaly.

2. IF a seal problem is indicated, **Go To** 13003-1, "Reactor Coolant Pump Operation".

3. IF an instrument problem is indicated, **initiate** maintenance as required.

4.0 SUBSEQUENT OPERATOR ACTIONS

NONE

5.0 COMPENSATORY OPERATOR ACTIONS

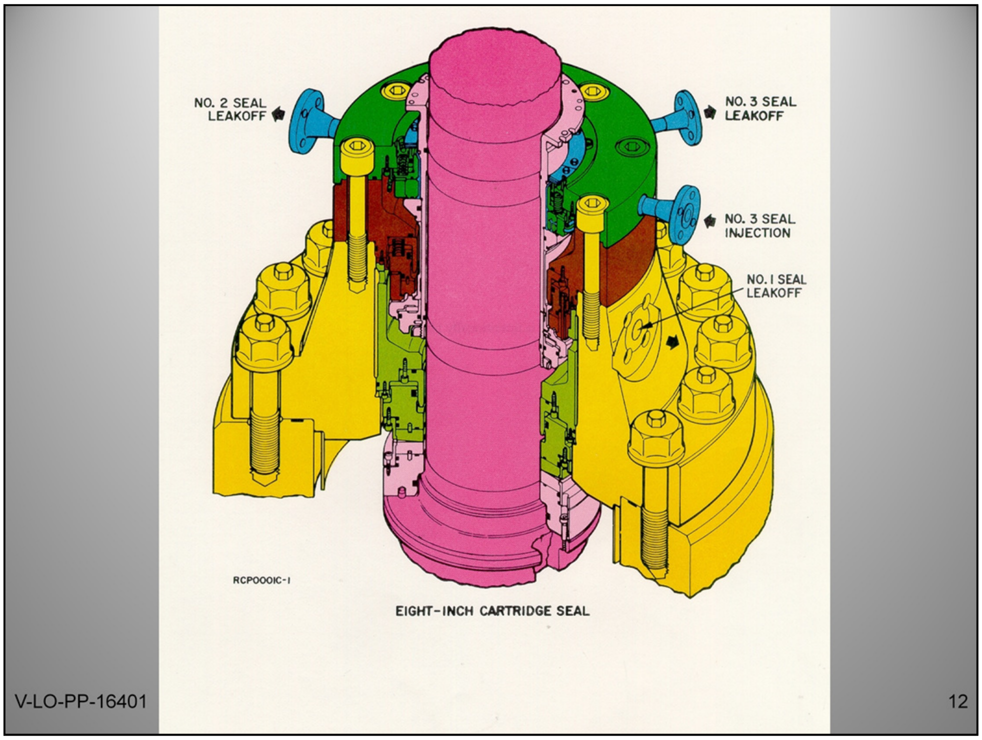
1. **Verify** proper seal leakoff using 1-FI-0156A and 1-FI-0160A once per shift, and refer to 13003-1, "Reactor Coolant Pump Operation" if leakoff is outside the limits.

2. **Log** corrective actions to repair the disabled annunciator or reasons for no action on 10018-C, "Annunciator Control", Figure 2.

3. **Log** compensatory actions on 10018-C, "Annunciator Control", Figure 5.

END OF SUB-PROCEDURE

REFERENCES: 1X4DB114, 1X6AB09-119, PLS

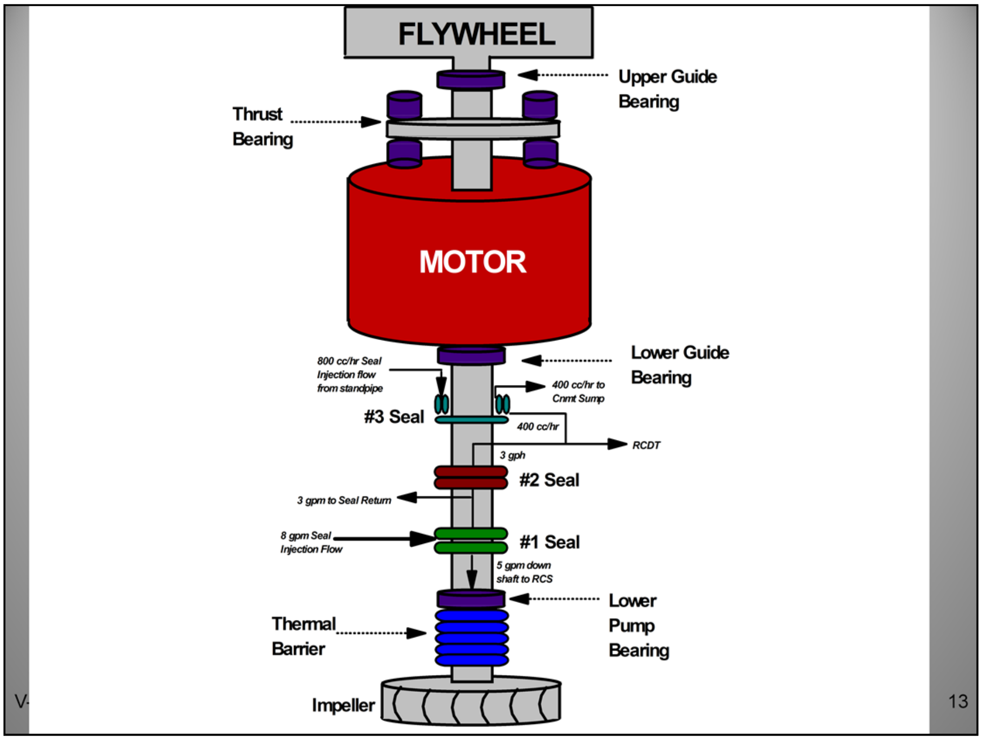


Objective 4

Demonstrate using the seal package model.

The seal package consist of three seals

1) **Number 1 seal (film riding)**
   - a) The primary seal
   - b) Seal is accomplished with a hydrostatic film between the shaft runner and seal ring.
   - c) No mechanical contact between seal ring and shaft runner (must keep ΔP >200 psid)
2) **Number 2 seal (face rubbing)**
   - a) Provides back up for #1 seal
   - b) Consist of carbon graphite (face rubbing seal)
   - c) Graphite makes contact with runner which rotates with shaft
   - d) If #1 seal fails , #2 seal converts to a film riding seal if #1 seal leak off valve is closed and seal is exposed to full RCS pressure. #2 seal designed to allow plant shutdown and should last approximately 24 hours.
   - e) Placing #2 seal in service with the RCP shaft still rotating will tend to score the shaft at the #2 seal area. This can require extensive repairs before placing the RCP back in service. Vogtle chooses to remove RCP from service and allow its shaft to come to a standstill before closing the #1 seal leak off valve to avoid this problem.
3) **Number 3 seal (face rubbing)**
   - a) Prevents the leakage of liquid and gases from the RCS into containment.
   - b) Consist of carbon graphite seal which makes contact with runner (face rubbing)
   - c) The runner is around the shaft and rotates with it.
   - d) The seal is actually two graphite sealing surfaces called dams.



Objective 1d

2) RCP Motor Auxiliaries

A) Motor Cooler

1) Containment Air is drawn into the motor by fan blades on motor's rotor

2) It is then exhausted through the motor cooler

3) ACCW is the cooling medium used in the cooler (cools the outgoing air)

4) This arrangement limits containment air temperature rise and in turn limits motor temperature.

3) **Flywheel**

A) Addressed in tech spec administrative section

B) Stores rotational energy of the pump and motor while running then releases energy by maintaining pump motion to slowly reduce core flow following loss of power for core protection.

4) RCP motor space heater

A) Each RCP motor has a electric resistance heater.

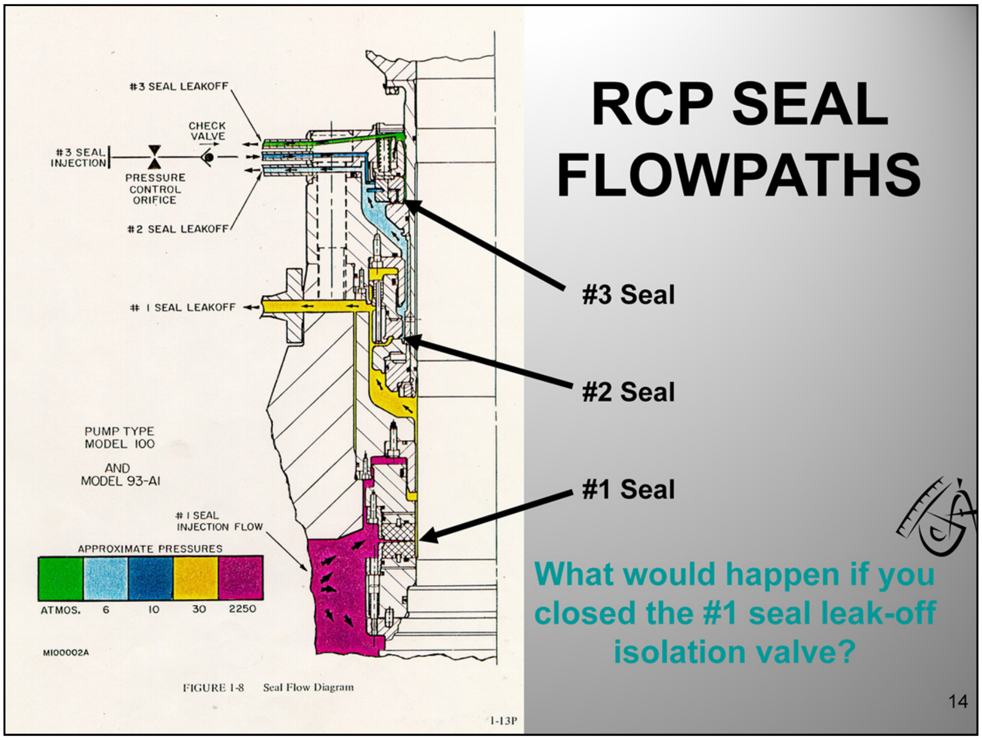
B) Used to prevent moisture accumulation in windings when motor is shutdown.

C) Not needed when motor is in operation because of heat generation from motor windings.

D) Heaters are automatically energized when either of the RCP motor breakers are opened.

E) Heaters are supplied from 480 V MCCs

RCP motor burned up at Vogtle that was attributed to moisture from space heater breaker being open when pump was shutdown. The space heater did not energize therefore moisture accumulated while the pump was shutdown during the outage. Upon restart the motor windings shorted out. Motor rebuild was required.



Objective 2

RCP Seal Injection

a) Provided from CVCS
b) 8 gpm per pump
c) 5 gpm is directed through to lower pump radial bearing and into the RCS loop.
d) The remaining 3 gpm supplies #1 and #2 seals
e) #3 seal injection is from small tanks called Standpipes. (Gravity Fed)
f) Flow path
   1) 8 gpm from CVCS enters RCP at 2250 psig
   2) 5 gpm passes through the lower pump bearing lubricating and cooling it.
   3) Seal injection at 2250 psig prevents RCS water from escaping the loop.
   4) 3 gpm is directed through #1 seal
   5) A pressure drop at 2220 psid across the #1 seal occurs.
   6) Approximately 3 gph (0.05 gpm) leak off from #1 seal is used as seal injection to #2 seal.
   7) The remainder of #1 seal leak off is directed back to the VCT via seal water return.
   8) 3 gph passes through #2 seal and the leak off is directed to the RCDT (~5-6 psig)
   9) 800 cc/hr seal injection for #3 seal is provided by standpipe (~10 psig)
   10) The standpipes are located at a higher elevation than the RCP and gravity feeds #3 seal; standpipes Auto fill from RMWST.
   11) #3 seal injection is injected between the two dams and sealing surfaces.
   12) #3 seal injection pressure is slightly higher than #2 seal injection leak off.
   13) This prevents RCS liquids or gases from escaping to the containment environment.
   14) #3 seal has two leak off paths
      a) The outer dam leak off (400 cc/hr) combines with #2 seal leak off and is routed to RCDT
      b) The inner dam leak off (400 cc/hr) is directed to the containment sump.

SMART – Solid Knowledge. If the #1 seal leakoff was isolated, the #2 seal would become a film riding seal due to increased pressure across the #2 seal facing.

1. 008A2.02 001/LOCT AND LOIT/SRO/C/A 3.2/3.5/008A2.02/LO-TA-60026//HL-18 NRC/

Initial conditions:

- Unit 1 is at 100% reactor power.
- CCW pump #5 is tagged out for maintenance.
- 1-LSLL-1854, CCW Surge Tank level switch for CCW pump #3, has failed.
- ALB02-A05 CCW TRAIN A SURGE TK LO-LO LVL is received.

Current conditions:

- ALB02-A06 CCW TRAIN A LO HDR PRESS is received.
- ALB02-B06 CCW TRAIN A LO FLOW is received.

Which one of the following completes the following statement?

Demin Water Makeup Valve to the CCW Train 'A' Surge Tank __(1)__ automatically open,

and

per Tech Spec 3.7.7, "Component Cooling Water (CCW) System," Train 'A' CCW is declared __(2)__.

	__(1)__	__(2)__
A.	will	OPERABLE
B.	will	inoperable
C.	will NOT	OPERABLE
D.✓	will NOT	inoperable

008 Component Cooling Water System (CCWS)

A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

- High / Low surge tank level

<u>K/A MATCH ANALYSIS</u>

The candidate is presented with a plausible scenario where a CCW Surge Tank level

transmitter fails low and is required to determine the impact of the failure on the system. With one CCW pump tagged out and the inability to start the pump with the failed level transmitter, a decision of operability must be determined which is SRO required knowledge.

EXPLANATION OF REQUIRED KNOWLEDGE

Each train of CCW is comprised of 3 pumps. Two pumps are required to met LCO 3.7.7. The pumps automatically start on SI, LOSP, Lo header pressure, and trip of a running pump. Only the SI and LOSP starts are required per SR 3.7.7.2.

When level switch 1-LSLL-1854 fails low, CCW pump #3 trips. With CCW pump #5 already tagged out, the system header pressure lowers and annunciators ALB02-A06 & B06 alarm. Since automatic Demin Water Makeup Valve LV-1850 is controlled by level switches 1-LSL-1850 & 1-LSH-1850, its operation is not affected. (Reference P&ID 1X4DB136) No actual low level condition exists, therefore LV-1850 will not open.

Since 1-LSLL-1854 failing low resulted in a pump trip, SR 3.7.7.2 to verify each CCW pump starts automatically on an actual or simulated actuation signal can not be satsified. Per TS SR 3.0.1, failure to meet a Survelliance, whether such failure is experienced during the performance of the Survelliance or between performances of the Survelliance, shall be failure to meet the LCO. Therefore, CCW Pump #3 must be declared inoperable. Since CCW Pump #5 was already inoperable, TS 3.7.7 LCO is not met and RAS 'A' must be entered.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Since automatic makeup valve LV-1850 is controlled by level switches 1-LSL-1850 & 1-LSH-1850, its operation is not affected. No actual low level condition exists, therefore LV-1850 will not open. However, if the candidate believes makeup is controlled off the same level transmitter or misses that the level transmitter failed and believes there is an actual lo level, then makeup valve LV-1850 would be expected to open.

The second part is incorrect. SR 3.7.7.2 can not be met and CCW Pmp #3 and #5 are both inoperable. LCO 3.7.7 cannot be met. However, a candidate not familiar with the requirements of SR 3.7.72 & SR 3.0.1 or the pump start logic could conclude that the non-safety related surge tank low level trip would be bypassed on an SI or LOSP or that auto makeup would restore a low level condition and CCW Pmp #3 would remain OPERABLE.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. SR 3.7.7.2 can not be met and CCW Pmp #3 and #5 are both inoperable. LCO 3.7.7 cannot be met.

C. Incorrect. Plausible. The first part is correct. Since automatic makeup valve LV-1850 is controlled by level switches 1-LSL-1850 & 1-LSH-1850, its operation is not affected. No actual low level condition exists, therefore LV-1850 will not open.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

SRO JUSTIFICATION (10CFR43(b))

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, generic LCO SR3.0.1 knowledge is required. No 1 hr or less actions exist.
-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" No, the required knowledge is found in generic LCO SR3.0.1 and SR 3.7.7.2, which is below the line.
-Can question be answered *solely* by knowing the TS Safety Limits? No, this question does not involve a TS Safety Limit.
-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- **Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4) Yes, the question requires an OPERABILITY call be made by utilizing generic LCO SR3.0.1 as applied to SR 3.7.7.2.**
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Level:	SRO
Tier # / Group #	T2 / G1
K/A#	008A2.02
Importance Rating:	3.2 / 3.5

Technical Reference:	ARP 17002-1, Rev 24.1, pages 3, 14, & 26 P&ID 1X4DB136, Rev 33.0 TS 3.7.7, Amendment No. 96, pages 3.7.7-1 & 2 TS 3.0.1, Amendment No. 125, page 3.0-4
References provided:	None
Learning Objective:	LO-PP-10101-04 From memory, describe the expected system response and operator corrective actions for each of the following: d. Surge Tank Low Level LO-LP-39211-04 Describe the bases for any given Tech Spec in section 3.7. LO-TA-60026 Respond to a Loss of CCW per 18020-C LO-TA-10006 Draw the CCW System
Question origin:	BANK - Reuse - HL18 Question # 008A2.02
Cognitive Level:	C/A
10 CFR Part 55 Content:	43.2
Comments:	

You have completed the test!

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	-------------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. ----------------------------------	
		Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	------------------------------NOTE---------------------------- Isolation of CCW flow to individual components does not render the CCW System inoperable. -- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

	(1)	(2)	(3)	(4)	(5)	(6)
A	**NSCW TRAIN A F-1 HI VIB**	**NSCW TRAIN A F-2 HI VIB**	**NSCW TRAIN A F-3 HI VIB**	**NSCW TRAIN A F-4 HI VIB**	**CCW TRAIN A SURGE TK LO-LO LVL**	**CCW TRAIN A LO HDR PRESS**
B	**NSCW TRAIN A LO HDR PRESS**	**NSCW TRAIN A TRANSF PMP LO DISCH PRESS**			**CCW TRAIN A SURGE TK HI/LO LVL**	**CCW TRAIN A LO FLOW**
C	**NSCW TRAIN A CLG TWR BASIN HI/LO LVL**		**NSCW TRAIN A DG CLR LO FLOW**	**NSCW TRAIN A RHR PMP & MTR CLR LO FLOW**	**CCW TRAIN A SURGE TK MAKE UP LVL**	**CCW TRAIN A RHR HX HI FLOW**
D		**NSCW TRAIN A CNMT CLR 1 & 2 LO FLOW**	**NSCW INTERTIE TRN A TO TRN B HI FLOW**			**CCW TRAIN A RHR HX LO FLOW**
E		**NSCW TRAIN A CNMT CLR 5 & 6 LO FLOW**	**NSCW TRAIN A NORM/BYP VLV MISPOSITIONED**	**RMWST VAC DEGASIFIER PNL ALARM**	**CCW TRAIN A RHR PMP SEAL LO FLOW**	**PRIMARY EQUIPMENT HI TEMP**
F		**NSCW TRN A RX CVTY CLG COIL LOW FLOW**	**RX MAKE UP STOR TK LO-LO LVL**	**RX MAKE UP STOR TK HI/LO LVL**		

WINDOW A05

ORIGIN	SETPOINT	
1-LSLL-1852 1-LSLL-1854 1-LSLL-1856	4.75 in. below CL (equal to 42%)	CCW TRAIN A SURGE TK LO-LO LVL

1.0 **PROBABLE CAUSE**

1. Failure of automatic make-up from Demineralized Water System.
2. Failure of manual make-up from Reactor Makeup Water System.
3. Leak in Component Cooling Water System.

2.0 **AUTOMATIC ACTIONS**

LO-LO level will trip Component Cooling Water Pumps.

3.0 **INITIAL OPERATOR ACTIONS**

Go To 18020-1, "Loss Of Component Cooling Water."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB136, 1X3D-BD-L01A, 1X3D-BD-L01C, 1X3D-BD-L01E, 1X5DN091-1, -2, -3, 1X5DT0022, CX5DT101-96

WINDOW C05

ORIGIN	SETPOINT	
1-LSL-1850	1.25 in. above CL (equal to 52%)	CCW TRAIN A SURGE TK MAKE UP LVL

1.0 **PROBABLE CAUSE**

Component Cooling Water (CCW) System leakage.

2.0 **AUTOMATIC ACTIONS**

Makeup Valve 1-LV-1850 opens.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Monitor** level using 1-LIT-1846 or computer point L2671.

2. IF 1-LV-1850 fails to open:

   a. **Open** the valve using 1-HS-1850 on QMCB,

   b. **Continue to monitor** level,

   c. **Open** 1-LV-1848 using 1-HS-1848 if level continues to fall.

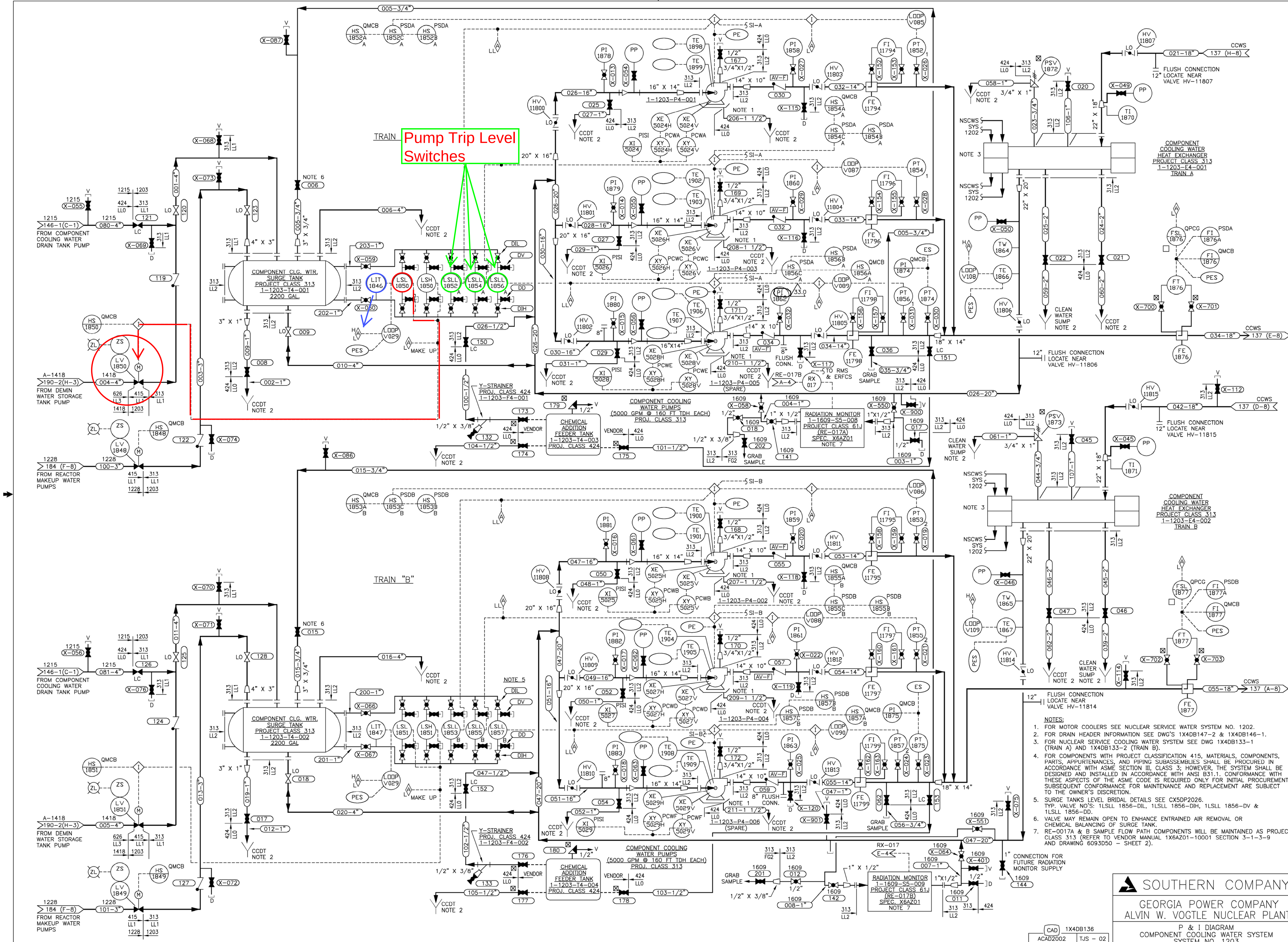
3. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB136, 1X3D-BD-L01G, 1X5DT0022, CX5DT101-95



NOTES:

1. FOR MOTOR COOLERS SEE NUCLEAR SERVICE WATER SYSTEM NO. 1202.
2. FOR DRAIN HEADER INFORMATION SEE DWG'S 1X4DB147-2 & 1X4DB146-1.
3. FOR NUCLEAR SERVICE COOLING WATER SYSTEM SEE DWG 1X4DB133-1 (TRAIN A) AND 1X4DB133-2 (TRAIN B).
4. FOR COMPONENTS WITH PROJECT CLASSIFICATION 415, MATERIALS, COMPONENTS, PARTS, APPURTENANCES, AND PIPING SUBASSEMBLIES SHALL BE PROCURED IN ACCORDANCE WITH ASME SECTION III, CLASS 3; HOWEVER, THE SYSTEM SHALL BE DESIGNED AND INSTALLED IN ACCORDANCE WITH ANSI B31.1. CONFORMANCE WITH THESE ASPECTS OF THE ASME CODE IS REQUIRED ONLY FOR INITIAL PROCUREMENT. SUBSEQUENT CONFORMANCE FOR MAINTENANCE AND REPLACEMENT ARE SUBJECT TO THE OWNER'S DISCRETION.
5. SURGE TANKS LEVEL BRIDAL DETAILS SEE CX5DP2026. TYP. VALVE NO'S: 1LSLL 1856-DIL, 1LSLL 1856-DV & 1LSLL 1856-DD.
6. VALVE MAY REMAIN OPEN TO ENHANCE ENTRAINED AIR REMOVAL OR CHEMICAL BALANCING OF SURGE TANK.
7. RE-0017A & B SAMPLE FLOW PATH COMPONENTS WILL BE MAINTAINED AS PROJECT CLASS 313 (REFER TO VENDOR MANUAL 1X6AZ01-10001 SECTION 3-1-3-9 AND DRAWING 6093D50 – SHEET 2).

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
COMPONENT COOLING WATER SYSTEM
SYSTEM NO. 1203

CAD 1X4DB136
ACAD2002 | TJS – 02

NO.		DATE	DR	CHK	APPV	DTL
33.0	REVISED PER ABN- V00871, VER. 1.0	7-23-07	TJS	SRG	WLN	X

SCALE: NONE
JOB NO. 10604
DRAWING NO. 1X4DB136
VER. 33.0
SIZE E 34X44
DRAWING CATEGORY: CRITICAL

1. 008AG2.4.41 001/LOCT AND LOIT/SRO/C/A 2.9/4.6/008AG2.4.41/LO-TA-40002///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- ALB12-F01 PRZR SAFETY RELIEF DISCH HI TEMP is received.
- PRT level, temperature, and pressure are all increasing.
- CVCS charging flow is 110 gpm.
- All RCP seal injection and #1 seal return flows are within normal operating range.
- CVCS letdown is isolated.
- Pressurizer level is 55% and stable.

Which one of the following completes the following statement?

Per Tech Spec 3.4.13, "RCS Operational Leakage," the RCS leakage is classified as __(1)__,

and

per NMP-EP-110, "Emergency Classification Determination and Initial Action," the Shift Manager is required to declare as a minimum a(n) __(2)__.

REFERENCE PROVIDED

	__(1)__	__(2)__
A.✓	identified	NOUE
B.	identified	Alert
C.	unidentified	NOUE
D.	unidentified	Alert

K/A

008 **Pressurizer Vapor Space Accident**

G2.4.41 **Knowledge of the emergency action level thresholds and classifications.**

K/A MATCH ANALYSIS

The question sets up plausible scenario where Main Control annunciators are received and various plant parameters and indications are provided. Based on the information provided the SRO candidate is required to assess the type of leakage indicated and determine if any emergency action level thresholds were exceeded. The type of

leakage reflects the first part of the KA requirement for a vapor space accident, the second action to quantify the leakage amount and determine any Emergency Plan requirements meets the second part of the KA and brings the question to the SRO knowledge level.

EXPLANATION OF REQUIRED KNOWLEDGE

The given plant conditions are indicitive of a Pressuirzer Code Safety valve relieving into the PRT. Per TS 3.4.13 Bases, identified leak is defined as being from a specifically known and located source, ie it must be collected and quantifiable, and not pressure boundary. Based on the given conditions, the leakrate is approximately 98 gpm, [charging-(letdown + seal leakoff)]. Total seal leakoff is normally about 12 gpm. Since the leakage is across a vavle seat, it is not pressure boundary. Therefore, the leakage described fully meets the definition of IDENTIFIED.

Per NMP-EP-110-GL03 Figure 1, a potential loss of the RCS barrier exists if RCS leak rate is non-isolatabe and >120 gpm. Since leakrate is calculated to be 98 gpm, this threshold has not been exceeded. Per NMP-EP-110-GL02 Figure 2, NOUE threshold SU5 is exceeded if identified leakage is > than 25 gpm. Therefore, this threshold has been exceeded.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Part 1 is correct it required the candidate to determine the type of RCS leakage based on information provided and knowledge of the Tech Spec definition of the various leakage categories. From the information the candidate should determine this is identified leakage. *LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank.*

Part 2 required the candidate to take the information gathered in Part 1 and apply it to the various Emergency Action Level classification thresholds. From that a leak rate of 98 gpm is determined and limits for NOUE recognized as exceeded.

B. Incorrect. Plausible. Part 1 is correct. See Part 1 of choice A above.

Part 2 is incorrect but ‘plausible’ because the candidate may determine that 110 gpm charging added to 32 gpm seal injection (which is alreay included in the charging flow indicator reading) going into RCS is 142 GPM leak. This minus the RCP seal return flow of 3.2 each gpm would be above the ALERT threshold of 120 gpm for the potential loss of the RCS barrier. (Reference P&IDs 1X4DB114 and 1X4DB116-1)

C. Incorrect. Plausible. Part 1 is incorrect but 'plausible' in that the candidate must first identify the type of RCS leakage. As the PRT continues to recieve effluent from the Przr Code Safety, it will eventually

rupture. If the candidate assumes that once this occur the leakage is no longer being "collected" in the PRT and not think about the leakage now being colleted in the containment sump, then it would be reasonable to interpret the indications as UNIDENTIFIED RCS leakage.

Part 2 is correct. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice C above.

Part 2 is incorrect. See Part 2 of choice B above.

SRO JUSTIFICATION

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the answer requires specific knowledge of emergency classification thresholds.
-Can the question be answered *solely* by knowing immediate operator actions? No, IOAs are not addressed by this question.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, the question does not address AOP or EOP entry conditions.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, the answer requires specific knowledge of emergency classification thresholds.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- **Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. Yes, the answer requires specific knowledge of emergency classification thresholds and determination of the specific classification based on current plant conditions. This is an SRO ONLY job link associated with an SRO ONLY objective. [LO-LP-40101-13 Given an emergency scenario, and the procedure, classify the emergency (SRO only).]**

Level:	SRO
Tier # / Group #	T1 / G1
K/A#	008G2.4.41
Importance Rating:	2.9 / 4.6

Technical Reference: NMP-EP-110-GL03, Rev 3.0, page 58
NMP-EP-110-GL03, Figure 1, Rev 3.0, page 121
NMP-EP-110-GL03, Figure 2, Rev 3.0, page 122
P&ID 1X4DB114, Rev 50.0
P&ID 1X4DB116-1, Rev 41.0
TS 3.4.13 Bases, Rev 2-9/06, page B.3.4.13-2
TS 3.4.13 Bases, Rev 1-9/03, page B.3.4.13-3

References provided: NMP-EP-110-GL03, Figure 1
NMP-EP-110-GL03, Figure 2
NMP-EP-110-GL03, Figure 3

Learning Objective:

LO-TA-40002	Emergency Classification and Implementing Instructions using NMP-EP-110 (SRO Only)
LO-TA-60014	Respond to Reactor Coolant System Leakage per 18004-C
LO-LP-40101-13	Given an emergency scenario, and the procedure, classify the emergency (SRO only).
LO-LP-39202-02	Demonstrate a working knowledge of the application of all Technical Specification definitions.
LO-LP-39202-01	Define the following terms, as per Plant Vogtle Tech Specs: i. identified leakage q. unidentified leakage
LO-LP-60304-04	Given the symptoms of RCS leakage into an area or system, correctly identify the leakage area or system.
LO-LP-60304-10	Given conditions and/or indications of leaks identified in Attachment "A" of AOP 18004-C, determine the probable location of the leakage per 18004-C.
LO-LP-60304-12	Discuss how approximate RCS leak rate is determined.

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 43.5

Comments:

You have completed the test!

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of an off-normal condition. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary

(continued)

BASES

LCO

c. Identified LEAKAGE (continued)

LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, “The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day.” The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

(continued)

	VOGTLE ELECTRIC GENERATING PLANT **Figure 1 – Fission Product Barrier Evaluation**		NMP-EP-110- GL03 Rev 3.0
General Emergency	**Site Area Emergency**	**Alert**	**Unusual Event**
FG1 Loss of ANY Two Barriers **AND** Loss or Potential Loss of Third Barrier	**FS1** Loss or Potential Loss of ANY Two Barriers	**FA1** ANY Loss or ANY Potential Loss of **EITHER** Fuel Clad **OR** RCS	**FU1** ANY Loss or ANY Potential Loss of Containment

Fuel Clad Barrier	
Loss	**Potential Loss**
1. Critical Safety Function Status Core-Cooling RED	**1. Critical Safety Function Status** Core Cooling-ORANGE **OR** Heat Sink-RED
2. Primary Coolant Activity Level Indications of RCS Coolant Activity greater than 300 µCi/gm Dose Equivalent I-131	2. Primary Coolant Activity Level Not Applicable
3. Core Exit Thermocouple Readings Core Exit TCs greater than 1200°F	**3. Core Exit Thermocouple Readings** Core Exit TCs greater than 711°F
4. Reactor Vessel Water Level Not Applicable	**4. Reactor Vessel Water Level** RVLS LEVEL less than 63%
5. Containment Radiation Monitoring Containment Radiation Monitor RE-005 **OR** 006 greater than 6E+6 mR/hr	5. Containment Radiation Monitoring Not Applicable
6. Other Indications Not applicable	6. Other Indications Not applicable
7. Emergency Director Judgment Judgment by the ED that the Fuel Clad Barrier is lost. Consider conditions not addressed and inability to determine the status of the Fuel Clad Barrier	**7. Emergency Director Judgment** Judgment by the ED that the Fuel Clad Barrier is potentially lost. Consider conditions not addressed and inability to determine the status of the Fuel Clad Barrier.
RCS Barrier	
Loss	**Potential Loss**
1. Critical Safety Function Status Not Applicable	**1. Critical Safety Function Status** RCS Integrity-RED **OR** ~~Heat Sink-RED~~
2. RCS Leak Rate RCS subcooling less than 24°F {less than 38° F Adverse} due to an RCS leak greater than Charging / RHR capacity	**2. RCS Leak Rate** Non-isolable RCS leak (including SG tube Leakage) greater than 120 gpm
3. SG Tube Rupture SGTR resulting in an SI actuation	3. SG Tube Rupture Not Applicable
4. Containment Radiation Monitoring CTMT Rad Monitor RE-005 **OR** 006 greater than 2.0E+4 mR/hr	4. Containment Radiation Monitoring Not Applicable
5. Other Indications Not applicable Distractor	**5. Other Indications** Unexplained level rise in ANY of the following: Containment sump Reactor Coolant Drain Tank (RCDT) Waste Holdup Tank (WHT)
6. Emergency Director Judgment Judgment by the ED that the RCS Barrier is lost. Consider conditions not addressed and inability to determine the status of the RCS Barrier	**6. Emergency Director Judgment** Judgment by the ED that the RCS Barrier is potentially lost. Consider conditions not addressed and inability to determine the status of the RCS Barrier.
Containment Barrier	
Loss	**Potential Loss**
1. Critical Safety Function Status Not Applicable	**1. Critical Safety Function Status** Containment-RED
2. Containment Pressure Rapid unexplained CTMT pressure lowering following initial pressure rise **OR** Intersystem LOCA indicated by CTMT pressure or sump level response not consistent with a loss of primary or secondary coolant	**2. Containment Pressure** CTMT pressure greater than 52 psig **OR** CTMT hydrogen concentration greater than 6% **OR** CTMT pressure greater than 21.5 psig **AND** Less than the following minimum operable equipment: Four CTMT fan coolers **AND** One train of CTMT spray
3. Core Exit Thermocouple Reading Not applicable	**3. Core Exit Thermocouple Reading** CORE COOLING CSF - RED **OR** - ORANGE for greater than 15min **AND** RVLS LEVEL less than 63%
4. SG Secondary Side Release with Primary to Secondary Leakage RUPTURED S/G is also FAULTED outside of containment **OR** Primary-to-Secondary leakrate greater than 10 gpm with nonisolable steam release from affected S/G to the environment	4. SG Secondary Side Release with P-to-S Leakage Not applicable
5. CNMT Isolation Valves Status After CNMT Isolation CTMT isolation valve(s) **OR** damper(s) are **NOT** closed resulting in a direct pathway to the environment after containment isolation is required	5. CNMT Isolation Valves Status After CNMT Isolation Not Applicable
6. Significant Radioactive Inventory in Containment Not Applicable	**6. Significant Radioactive Inventory in Containment** CTMT Rad monitor RE-005 **OR** 006 greater than 2.4E+8 mR/hr
7. Other Indications Pathway to the environment exists based on VALID RE-2562C Alarm **AND** RE-12444C **OR** RE-12442C Alarms	7. Other Indications Not applicable
8. Emergency Director Judgment Judgment by the ED that the CTMT Barrier is lost. Consider conditions not addressed and inability to determine the status of the CTMT Barrier	**8. Emergency Director Judgment** Judgment by the ED that the CTMT Barrier is potentially lost. Consider conditions not addressed and inability to determine the status of the CTMT Barrier

FIGURE 2

Vogtle Electric Generating Plant

Procedure Number **NMP-EP-110-GL03** Rev 3.0

EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS

Figure 2 - HOT INITIATING CONDITION EMERGENCY ACTION LEVEL MATRIX - MODES 1, 2, 3, and 4 ONLY

RADIOLOGICAL: EFFLUENTS | RAD LEVELS — FISSION PRODUCT BARRIERS — SYSTEM MALFUNCTIONS: AC/DC POWER | RX and CORE | ALARMS / COMMUNICATIONS — HAZARDS: NATURAL / DESTRUCTIVE | FIRE / EXPLOSION | TOXIC / FLAMMABLE | SECURITY | CR EVACUATION | ED DISCRETION

GENERAL EMERGENCY

RG1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mR TEDE **OR** 5000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology (pg. 36)

See Notes 1 and 2 →

1. VALID reading on any of the following radiation monitors that exceeds **OR** expected to exceed the reading shown for 15 minutes **OR** longer:
   - RE-12839 1.5×10^{5} µCi/cc
   - RE-12444 7.0×10^{2} µCi/cc
   - RE-13119 thru 13122 3.1×10^{5} µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 1000 mR TEDE **OR** 5000 mR thyroid CDE at **OR** beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates exceeding 1000 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 5000 mR for one hour of inhalation, at **OR** beyond the site boundary.

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note 2: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

FG1 - Loss of ANY Two Barriers **AND** Loss or Potential Loss of Third Barrier (pg. 39)

See Fission Product Barrier Matrix

SG1 - Prolonged Loss of All Offsite Power **AND** Prolonged Loss of All Onsite AC Power to essential buses (pg. 71)

1. Loss of ALL AC power indicated by:
   a. Loss of power to **OR** from transformers 1(2)NXRA **AND** 1(2)NXRB resulting in loss of all off-site electrical power to **BOTH** 1(2)AA02 **AND** 1(2)BA03 for greater than 15 minutes
   AND
   b. Failure of emergency diesel generators to supply power to emergency busses

AND EITHER

2. Restoration of at least one 4160V ESF bus, AA02 **OR** BA03, within 4 hr. of time of loss is **NOT** likely

OR

3. Fuel Clad Barrier Evaluation indicates continuing degradation (Loss or Potential Loss) of core cooling

SG2 - Failure of the RPS to complete an Automatic Trip and Manual Trip was NOT successful **AND** there is Indication of an Extreme Challenge to the ability to Cool the Core **Modes 1, 2 Only** (pg. 73)

NOTE Heat Sink CSF should not be considered – RED if total AFW flow is less than 570 gpm due to operator action.

1. Subcriticality CSF - RED
   AND
   Core Cooling CSF - RED

OR

2. Subcriticality CSF - RED
   AND
   Heat Sink CSF - RED

HG1 - HOSTILE ACTION resulting in loss of physical control of the facility. (pg. 117)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

HG2 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency (pg. 120)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity
   OR
   HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

RS1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE **OR** 500 mR Thyroid CDE for the Actual or Projected Duration of the Release. (pg 34)

See Notes 3 and 4 →

1. VALID reading on any of the following radiation monitors that exceeds **OR** is expected to exceed the reading shown for 15 minutes OR longer:
   - RE-12839 1.5×10^{4} µCi/cc
   - RE-12444 7.0×10^{1} µCi/cc
   - RE-13119 thru 13122 3.1×10^{4} µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE **OR** 500 mR thyroid CDE at **OR** beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at **OR** beyond the site boundary.

Note 3: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note 4: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

FS1 - Loss or Potential Loss of ANY Two Barriers (pg. 39)

See Fission Product Barrier Matrix

SS1 - Loss of All Offsite Power **AND** Loss of All Onsite AC Power to Essential Busses (pg. 65)

1. Loss of all AC power indicated by:
   a. Loss of power to **OR** from transformers 1(2)NXRA **AND** 1(2)NXRB resulting in loss of all off-site electrical power to **BOTH** 1(2)AA02 **AND** 1(2)BA03 for greater than 15 minutes
   AND
   b. Failure of emergency diesel generators to supply power to emergency busses.
   AND
   c. Restoration of at least one 4160V ESF bus, AA02 **OR** BA03, has **NOT** occurred within 15 minutes of time of loss of all AC power.

SS3 - Loss of All Vital DC Power (pg. 67)

1. Loss of Vital DC power to 125 VDC Buses AD1, BD1, CD1, **AND** DD1 indicated by bus voltage indications less than 105 VDC for greater than 15 minutes.

SS2 - Failure of RPS Instrumentation to complete or Initiate an Automatic Reactor Trip Once a RPS Setpoint Has Been Exceeded AND Manual Trip Was NOT Successful. **Modes 1, 2 Only** (pg. 66)

NOTE A successful manual trip for purposes of declaration is any action taken from the Main Control Board (MCB) that rapidly inserts the control rods, this can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB.

1. Automatic **AND** manual trip was **NOT** successful as indicated by:
   a. An automatic reactor trip setpoint was exceeded
   AND
   b. An automatic reactor trip did **NOT** occur
   AND
   c. A successful manual trip did **NOT** occur from the Control Room (NOTE)

SS4 - Complete Loss of Heat Removal Capability (pg. 68)

NOTE: Heat Sink CSF should NOT be considered RED if total AFW flow is less than 570 gpm due to operator action.

1. Complete Loss of Heat Removal Capability as indicated by:
   a. Core Cooling CSF - ORANGE
   AND
   b. Heat Sink CSF - RED

SS6 - Inability to Monitor a SIGNIFICANT TRANSIENT in Progress (pg. 69)

NOTE: SIGNIFICANT TRANSIENT: is an UNPLANNED event involving one or more of the following: (1) automatic turbine runback >25% thermal reactor power, (2) electrical load rejection >25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations >10%.

1. Inability to monitor a SIGNIFICANT TRANSIENT in progress as indicated by:
   a. Loss of most **OR** all of the MCB annunciators **OR** indicators associated with safety systems
   AND
   b. Compensatory non-alarming indications are **NOT** available
   AND
   c. Indications needed to monitor the Critical Safety Function Status Tree parameters are **NOT** available.

HS4 - HOSTILE ACTION within the PROTECTED AREA (pg. 118)

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by security shift supervision.

HS2 - Control Room Evacuation Has Been Initiated **AND** Plant Control Cannot Be Established (pg. 116)

1. a. Control Room evacuation has been initiated
   AND
   b. Control of the plant **CANNOT** be established per 18038 – Operation From The Remote Shutdown Panels, within 15 minutes.

HS3 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency (pg. 117)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public
   OR
   HOSTILE ACTION that results in intentional damage **OR** malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels that exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

ALERT EMERGENCY

RA1 - Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer. (pg. 27)

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

1. VALID reading on any of the following effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes OR longer.

Monitor	200 X Setpoint Value
RE-0018	2.8×10^{-3} µCi/cc
RE-0021	4.0×10^{-3} µCi/cc
RE-0848	3.5×10^{-3} µCi/cc
ARE-0014	$2.4 \times 10^{+2}$ µCi/cc
RE-12839	7.9×10^{-1} µCi/cc
RE-12442C/12444C	1.3×10^{-1} µCi/cc

OR

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes **OR** longer:
   Main Steam RE-13119 thru 13122 2.5 µCi/cc

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations **OR** release rates in excess of 200 times Technical Specification 5.5.4 as confirmed by the ODCM, with a release duration of 15 minutes **OR** longer.

RA2 - Damage to Irradiated Fuel **OR** Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel (pg. 30)

1. UNPLANNED VALID alarm on any of the following radiation monitors:
   - Fuel Handling Building RE-008
   - Fuel Handling BLDG EFFL. ARE-2532 A/B
   - Fuel Handling BLDG EFFL. ARE-2533 A/B

OR

2. Loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel as indicated by ANY of the following:
   Personnel report during fuel assembly movements;
   Spent Fuel Pool Storage Less than El 193'-5";
   Transfer Canal Transit El Less than El 186'10";
   Reactor Core El Less than El 181'-10" (63% on Full Range RVLIS)

RA3 Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown (pg. 32)

1. VALID radiation monitor readings greater than 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:
   Control Room radiation monitor RE-001 **OR** Central Alarm Station (by survey)

OR

2. UNPLANNED VALID radiation readings greater than 1 R/hr values in areas requiring infrequent access to maintain plant safety functions:
   Turbine Building, South & North Main Steam Valve Rooms, Control Building Fire Pump House, Aux Feed Water Pump House, Low Voltage Switchyard, Diesel Generator Buildings, NSCW Towers

FA1 - ANY Loss or ANY Potential Loss of **EITHER** Fuel Clad Barrier **OR** RCS Barrier (pg. 39)

See Fission Product Barrier Matrix

Distractor

SA5 - AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout. (pg. 64)

1. a. AC power capability to 1(2)AA02 **AND** 1(2)BA03 reduced to a single power source for greater than 15 minutes
   AND
   b. ANY additional single failure will result in station blackout.

SA2 - Failure of RPS Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a RPS System Setpoint Has Been Exceeded **AND** Manual Trip was Successful. Modes 1, 2, 3 Only (pg. 61)

NOTE A successful manual trip for purposes of declaration is any action taken from the Main Control Board (MCB) that rapidly inserts the control rods, this can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB.

1. Automatic trip was **NOT** successful but manual trip was successful as indicated by:
   a. An automatic Reactor Protection System trip setpoint exceeded
   AND
   b. An automatic reactor trip did **NOT** occur
   AND
   c. A successful manual trip occurred from the Control Room (NOTE)

SA4 - UNPLANNED Loss of Most or All Annunciation or Indication With **EITHER** a SIGNIFICANT TRANSIENT in Progress, **OR** Compensatory Non-Alarming Indicators are Unavailable (pg. 62)

NOTE: SIGNIFICANT TRANSIENT: is an UNPLANNED event involving one or more of the following: (1) automatic turbine runback >25% thermal reactor power, (2) electrical load rejection >25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations >10%.

"Compensatory non-alarming indications" in this context includes computer based information such as SPDS. The indications needed to monitor safety functions necessary for protection of the public must include control room indications, computer generated indications and dedicated annunciation capability. The specific indications are those used to determine such functions as the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

1. UNPLANNED loss of most **OR** all MCB annunciators **OR** indicators associated with safety systems for greater than 15 minutes

AND EITHER

2. a. A SIGNIFICANT TRANSIENT is in progress
   OR
   b. Compensatory non-alarming indications are **NOT** available

HA1 - Natural and Destructive Phenomena Affecting the Plant VITAL AREA (pg. 105)

1. Seismic monitoring system confirms seismic event greater than OBE Earthquake acceleration of 0.12 g.

OR

2. Tornado **OR** high winds greater than 100 mph within the PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures/equipment (Table 1) **OR** the Control Room has indication of degraded performance of any listed systems.

OR

3. Vehicle crash within PROTECTED AREA boundary AND resulting in VISIBLE DAMAGE to any of the following plant structures (Table 1) OR equipment therein OR control indication of degraded performance of those systems.

OR

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to **OR** penetration of any area containing safety-related equipment, their controls **OR** their power supplies.

OR

6. Sustained hurricane winds onsite resulting in VISIBLE DAMAGE to plant structures within the PROTECTED AREA boundary containing equipment necessary for safe shutdown.

TABLE 1

- Containment Building
- Fuel Handling Building
- Diesel Generator Building
- Diesel Fuel Oil Storage Tank Pumphouse
- Auxiliary Building
- Auxiliary Feedwater Pumphouse
- Control Building
- NSCW CoolingTowers

HA2 - FIRE **OR** EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown (pg. 108)

1. FIRE **OR** EXPLOSION in a plant VITAL AREA **AND** affected system parameter indications show degraded performance **OR** plant personnel report VISIBLE DAMAGE to permanent structures **OR** safety related equipment in any of the following:
   - Containment Building
   - Fuel Handling Building
   - NSCW Cooling Towers
   - Control Building
   - Diesel Generator Building
   - Auxiliary Building
   - Diesel Fuel Oil Storage Tank Pumphouse
   - Auxiliary Feedwater Pumphouse

HA3 - Release of Toxic, Asphyxiant, or Flammable Gases Within or contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown (pg. 110)

1. Report **OR** detection of toxic **OR** asphyxiant gas within **OR** contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report **OR** detection of flammable gases in concentration greater than the LOWER FLAMMABILITY LIMIT within **OR** contiguous to a VITAL AREA.

HA4 – HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat. (pg. 112)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by security shift supervision.

OR

2. A validated notification from the NRC of an airliner attack threat within 30 minutes of the site.

HA5 - Control Room Evacuation Has Been Initiated. (pg. 114)

1. Entry into 18038 – Operation From the Remote Shutdown Panels, procedure for Control Room evacuation.

HA6 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. (pg. 115)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant
   OR
   a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

NOTIFICATION OF UNUSUAL EVENT

RU1 – Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. (pg. 23)

Note: The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes **OR** longer.

Monitor	2 X Setpoint Value
RE-0018	2.8×10^{-5} µCi/cc
RE-0021	4.0×10^{-5} µCi/cc
RE-0848	3.5×10^{-5} µCi/cc
ARE-0014	2.4 µCi/cc
RE-12839	7.9×10^{-3} µCi/cc
RE-12442C/12444C	1.3×10^{-3} µCi/cc

OR

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes **OR** longer:
   Main Steam (RE-13119 thru 13122) 2.5×10^{-2} µCi/cc

OR

3. Confirmed sample analyses for gaseous **OR** liquid releases indicates concentrations **OR** release rates, with a release duration of 60 minutes **OR** longer, in excess of two times Technical Specification 5.5.4, as confirmed by the ODCM.

RU2 – Unexpected Rise in Plant Radiation (pg. 25)

1. a. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, **OR** fuel transfer canal with all irradiated fuel assemblies remaining covered by water by any of the following:
   - Personnel report of low water level; **OR**
   - LSHL-0625 offscale low; **OR**
   - Personnel report of cavitation **OR** low discharge pressure for SFP (1/2-1213-P6-002 **OR** -005) **AND/OR** RHR (1/2-1205-P6-001 **OR** -002) Pumps
   AND
   b. UNPLANNED VALID Direct Area Radiation Monitor readings rise on any of the following:
   - RE-0008 in the fuel building
   - RE-0002, -0003, -0004 in containment
   - RE-0011 at the seal table
   - RE-0005, -0006 in containment.

OR

2. UNPLANNED VALID Direct Area Radiation Monitor readings rise by a factor of 1000 over normal* levels.
   - RE-0001 Control Room
   - RE-0002/0003 Cnmt Bldg Low Range
   - RE-0004 Cnmt Bldg Per. Hatch
   - RE-0005/0006 Cnmt Bldg High Range
   - RE-0007B Sampling Room
   - RE-0008 FHB
   - RE-0011 Cnmt Bldg Incore Instr
   - ARE-16971 RPF North Wall General Area
   - ARE-16972 RPF East Wall General Area
   - ARE-16973 RPF Dress Out Area
   - *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

FU1 - ANY Loss **OR** ANY Potential Loss of Containment Barrier (pg. 39)

See Fission Product Barrier Matrix

SU1 - Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes (pg. 54)

1. a. Loss of power to OR from transformers 1(2)NXRA AND 1(2)NXRB resulting in loss of all off-site electrical power to BOTH 1(2)AA02 AND 1(2)BA03 for greater than 15 minutes
   AND
   b. Emergency diesel generators supplying power to BOTH 1(2)AA02 AND 1(2)BA03.

SU2 - Inability to Reach Required Shutdown Within Technical Specification Limits (pg. 55)

1. Plant is **NOT** brought to required operating mode within Technical Specifications LCO Action Statement Time limit.

SU5 - RCS Leakage (pg. 58)

1. Unidentified **OR** pressure boundary leakage greater than 10 gpm.

OR

2. Identified leakage greater than 25 gpm.

SU4 - Fuel Clad Degradation (pg. 57)

1. RE48000 radiation monitor reading greater than 5.55µCi/gm indicating fuel clad degradation greater than Technical Specification allowable limits.

OR

2. RCS coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits as indicated by ANY of the following:
   - Dose Equivalent I-131 greater than 1 µCi/gm for greater than 48 hours
   - Dose Equivalent I-131 greater than Technical Specification figure 3.4.16-1 limits
   - RCS specific activity greater than 100/E µCi/gm gross radioactivity.

SU8 - Inadvertent Criticality Modes 3, 4 Only (pg. 60)

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

SU3 - UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes (pg. 56)

1. UNPLANNED loss of most **OR** all of the MCB annunciators **OR** indicators associated with safety systems for greater than 15 minutes.

SU6 - UNPLANNED Loss of All Onsite **OR** Offsite Communications Capabilities (pg. 59)

1. UNPLANNED loss of ALL of the following on-site communications capability affecting the ability to perform routine operations:
   - In plant telephones
   - Public address system
   - Plant radio systems

OR

2. UNPLANNED loss of ALL of the following off-site communications capability:
   - ENN (Emergency Notification Network)
   - ENS (Emergency Notification System)
   - Commercial phones (Radio, PBX, Satellite, Wireless)
   - VOIP (Voice Over Internet Protocol)
   - OPX (Off Premise Extension)

HU1 - Natural and Destructive Phenomena Affecting the PROTECTED AREA (pg. 97)

1. Seismic monitoring system indicates seismic event **OR** plant operators report an earthquake was "felt".

OR

2. Report by plant personnel of tornado **OR** high winds greater than 100 mph striking within PROTECTED AREA.

OR

3. Crash of vehicle, large enough to cause significant damage, into plant structures containing functions or systems required for safe shutdown within the PROTECTED AREA boundary.

OR

4. Report by plant personnel of an unanticipated EXPLOSION within the PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure **OR** equipment.

OR

5. Report of turbine failure resulting in casing penetration **OR** damage to turbine **OR** generator seals.

OR

7. Sustained hurricane force winds greater than 73 mph forecast to be at the plant site in the next four hours in accordance with 11889-C, Severe Weather Checklist.

HU2 - FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection (pg. 99)

1. FIRE in buildings **OR** areas contiguous to any of the following areas **NOT** extinguished within 15 minutes of control room notification **OR** verification of a control room alarm unless disproved by personnel observation:

Containment Building
Fuel Handling Building
NSCW Cooling Towers
Control Building
Diesel Generator Building
Auxiliary Building
Diesel Fuel Oil Storage Tank Pumphouse
Auxiliary Feedwater Pumphouse

HU3 - Release of Toxic, Asphyxiant, **OR** Flammable Gases Deemed Detrimental to Normal Operation of the Plant (pg. 101)

1. Report **OR** detection of toxic, asphyxiant, **OR** flammable gas that has **OR** could enter the Owner Controlled Area in amounts greater than life threatening **OR** flammable concentrations that can affect NORMAL PLANT OPERATIONS.

OR

2. Report by Local, County, **OR** State Officials for evacuation **OR** sheltering of site personnel based on an offsite toxic, asphyxiant, **OR** flammable gas event.

HU4 - Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant (pg. 102)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by security shift supervision.

OR

2. A CREDIBLE VEGP security THREAT notification.

OR

3. A validated notification from NRC providing information of an aircraft threat.

HU5 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE. (pg. 104)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Classification

SU5

Initiating Condition

RCS Leakage.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

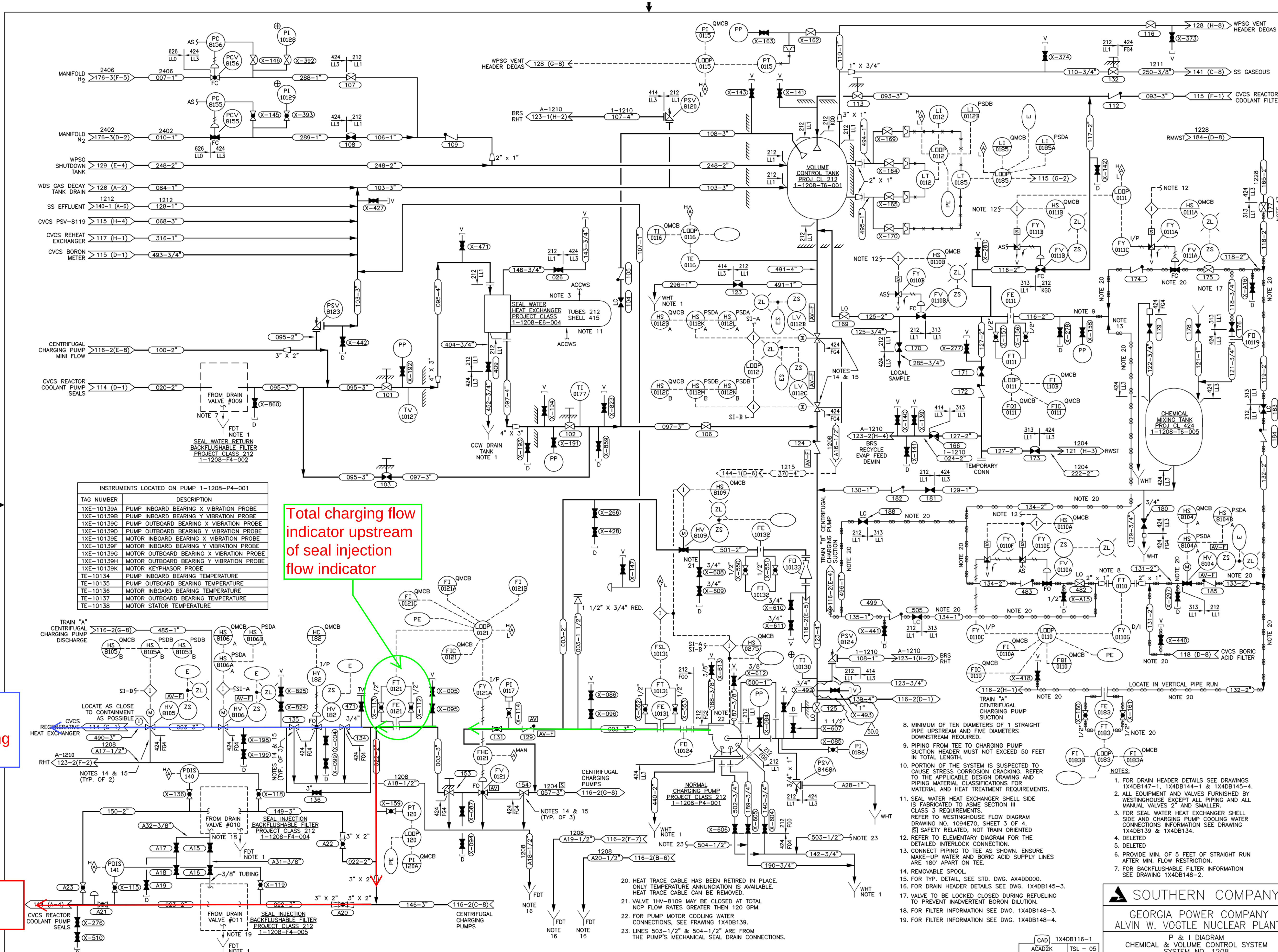
Threshold Values: (1 **OR** 2)

1. Unidentified **OR** pressure boundary leakage greater than 10 gpm.

2. Identified leakage greater than 25 gpm.

Basis:

This IC is included as a NOUE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The Threshold Value for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.



INSTRUMENTS LOCATED ON PUMP 1-1208-P4-001	
TAG NUMBER	DESCRIPTION
1XE-10139A	PUMP INBOARD BEARING X VIBRATION PROBE
1XE-10139B	PUMP INBOARD BEARING Y VIBRATION PROBE
1XE-10139C	PUMP OUTBOARD BEARING X VIBRATION PROBE
1XE-10139D	PUMP OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139E	MOTOR INBOARD BEARING X VIBRATION PROBE
1XE-10139F	MOTOR INBOARD BEARING Y VIBRATION PROBE
1XE-10139G	MOTOR OUTBOARD BEARING X VIBRATION PROBE
1XE-10139H	MOTOR OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139K	MOTOR KEYPHASOR PROBE
TE-10134	PUMP INBOARD BEARING TEMPERATURE
TE-10135	PUMP OUTBOARD BEARING TEMPERATURE
TE-10136	MOTOR INBOARD BEARING TEMPERATURE
TE-10137	MOTOR OUTBOARD BEARING TEMPERATURE
TE-10138	MOTOR STATOR TEMPERATURE

NOTES:

1. FOR DRAIN HEADER DETAILS SEE DRAWINGS 1X4DB147-1, 1X4DB144-1 & 1X4DB145-4.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER.
3. FOR SEAL WATER HEAT EXCHANGER SHELL SIDE AND CHARGING PUMP COOLING WATER CONNECTIONS INFORMATION SEE DRAWING 1X4DB139 & 1X4DB134.
4. DELETED
5. DELETED
6. PROVIDE MIN. OF 5 FEET OF STRAIGHT RUN AFTER MIN. FLOW RESTRICTION.
7. FOR BACKFLUSHABLE FILTER INFORMATION SEE DRAWING 1X4DB148-2.
8. MINIMUM OF TEN DIAMETERS OF 1 STRAIGHT PIPE UPSTREAM AND FIVE DIAMETERS DOWNSTREAM REQUIRED.
9. PIPING FROM TEE TO CHARGING PUMP SUCTION HEADER MUST NOT EXCEED 50 FEET IN TOTAL LENGTH.
10. PORTION OF THE SYSTEM IS SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
11. SEAL WATER HEAT EXCHANGER SHELL SIDE IS FABRICATED TO ASME SECTION III CLASS 3 REQUIREMENTS.
    REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E70, SHEET 3 OF 4.
    S SAFETY RELATED, NOT TRAIN ORIENTED
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. CONNECT PIPING TO TEE AS SHOWN. ENSURE MAKE-UP WATER AND BORIC ACID SUPPLY LINES ARE 180° APART ON TEE.
14. REMOVABLE SPOOL.
15. FOR TYP. DETAIL, SEE STD. DWG. AX4DD000.
16. FOR DRAIN HEADER DETAILS SEE DWG. 1X4DB145-3.
17. VALVE TO BE LOCKED CLOSED DURING REFUELING TO PREVENT INADVERTENT BORON DILUTION.
18. FOR FILTER INFORMATION SEE DWG. 1X4DB148-3.
19. FOR FILTER INFORMATION SEE DWG. 1X4DB148-4.
20. HEAT TRACE CABLE HAS BEEN RETIRED IN PLACE. ONLY TEMPERATURE ANNUNCIATION IS AVAILABLE. HEAT TRACE CABLE CAN BE REMOVED.
21. VALVE 1HV-8109 MAY BE CLOSED AT TOTAL NCP FLOW RATES GREATER THEN 120 GPM.
22. FOR PUMP MOTOR COOLING WATER CONNECTIONS, SEE FRAWING 1X4DB139.
23. LINES 503-1/2" & 504-1/2" ARE FROM THE PUMP'S MECHANICAL SEAL DRAIN CONNECTIONS.

CAD 1X4DB116-1
ACAD2K | TSL - 05

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
50.0	REVISED PER ABN-1081944801M002, VER. 1.0	09/28/12	TSL	CFC	JMR	

SOUTHERN COMPANY

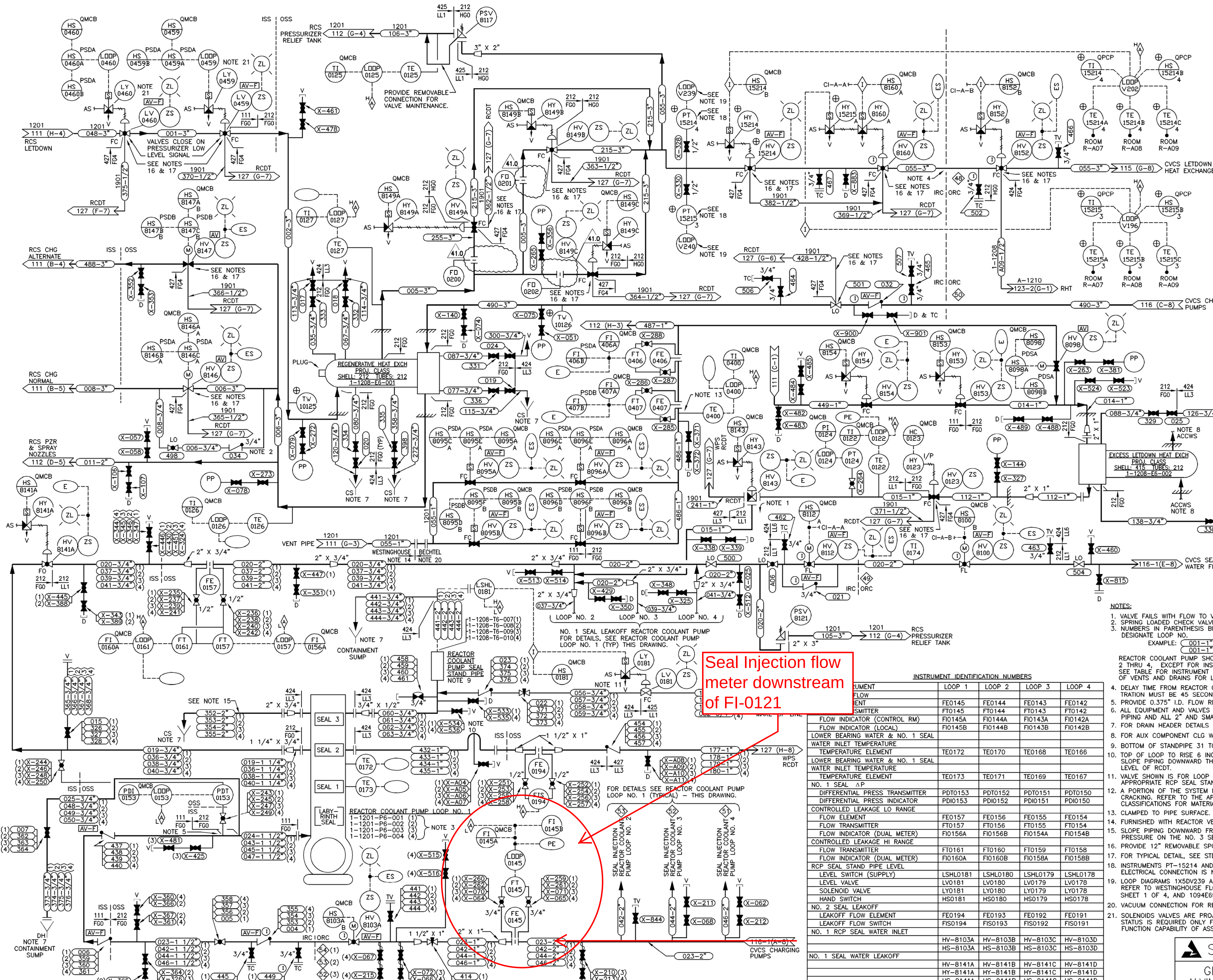
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SYSTEM NO. 1208

SCALE: 0.0 | JOB NO.10604 | DRAWING NO. 1X4DB116-1 | VER. 50.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



INSTRUMENT IDENTIFICATION NUMBERS

RUMENT	LOOP 1	LOOP 2	LOOP 3	LOOP 4
FLOW				
ENT	FE0145	FE0144	FE0143	FE0142
SMITTER	FT0145	FT0144	FT0143	FT0142
FLOW INDICATOR (CONTROL RM)	FI0145A	FI0144A	FI0143A	FI0142A
FLOW INDICATOR (LOCAL)	FI0145B	FI0144B	FI0143B	FI0142B
LOWER BEARING WATER & NO. 1 SEAL				
WATER INLET TEMPERATURE				
TEMPERATURE ELEMENT	TE0172	TE0170	TE0168	TE0166
LOWER BEARING WATER & NO. 1 SEAL				
WATER INLET TEMPERATURE				
TEMPERATURE ELEMENT	TE0173	TE0171	TE0169	TE0167
NO. 1 SEAL ΔP				
DIFFERENTIAL PRESS TRANSMITTER	PDT0153	PDT0152	PDT0151	PDT0150
DIFFERENTIAL PRESS INDICATOR	PDI0153	PDI0152	PDI0151	PDI0150
CONTROLLED LEAKAGE LO RANGE				
FLOW ELEMENT	FE0157	FE0156	FE0155	FE0154
FLOW TRANSMITTER	FT0157	FT0156	FT0155	FT0154
FLOW INDICATOR (DUAL METER)	FI0156A	FI0156B	FI0154A	FI0154B
CONTROLLED LEAKAGE HI RANGE				
FLOW TRANSMITTER	FT0161	FT0160	FT0159	FT0158
FLOW INDICATOR (DUAL METER)	FI0160A	FI0160B	FI0158A	FI0158B
RCP SEAL STAND PIPE LEVEL				
LEVEL SWITCH (SUPPLY)	LSHL0181	LSHL0180	LSHL0179	LSHL0178
LEVEL VALVE	LV0181	LV0180	LV0179	LV0178
SOLENOID VALVE	LY0181	LY0180	LY0179	LY0178
HAND SWITCH	HS0181	HS0180	HS0179	HS0178
NO. 2 SEAL LEAKOFF				
LEAKOFF FLOW ELEMENT	FE0194	FE0193	FE0192	FE0191
LEAKOFF FLOW SWITCH	FIS0194	FIS0193	FIS0192	FIS0191
NO. 1 RCP SEAL WATER INLET				
	HV–8103A	HV–8103B	HV–8103C	HV–8103D
	HS–8103A	HS–8103B	HS–8103C	HS–8103D
NO. 1 SEAL WATER LEAKOFF				
	HV–8141A	HV–8141B	HV–8141C	HV–8141D
	HY–8141A	HY–8141B	HY–8141C	HY–8141D
	HS–8141A	HS–8141B	HS–8141C	HS–8141D

NOTES:

1. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK.
2. SPRING LOADED CHECK VALVE.
3. NUMBERS IN PARENTHESIS BEHIND VALVE AND LINE BUBBLES, DESIGNATE LOOP NO.
   EXAMPLE: 001–1" (1) – LOOP NO. 1
   001–1" (2) – LOOP NO. 2
   REACTOR COOLANT PUMP SHOWN FOR LOOP NO. 1, TYPICAL FOR LOOPS 2 THRU 4, EXCEPT FOR INSTRUMENT AND VALVE IDENTIFICATION NUMBERS. SEE TABLE FOR INSTRUMENT IDENTIFICATION NUMBERS. RELATIVE POSITION OF VENTS AND DRAINS FOR LOOPS 2 THRU 4 MAY ALSO VARY.
4. DELAY TIME FROM REACTOR COOLANT SYSTEM TO THE CONTAINMENT PENETRATION MUST BE 45 SECONDS MINIMUM AT MAXIMUM LETDOWN FLOW.
5. PROVIDE 0.375" I.D. FLOW RESTRICTION AS SHOWN ON BPC DWG CX4DG001.
6. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND ALL 2" AND SMALLER MANUAL VALVES.
7. FOR DRAIN HEADER DETAILS SEE DWGS 1X4DB143, 144–1 & 144–2.
8. FOR AUX COMPONENT CLG WATER SYSTEM INFORMATION SEE DWG 1X4DB138–2.
9. BOTTOM OF STANDPIPE 31 TO 34 FEET ABOVE CONNECTION TO NO. 3 SEAL.
10. TOP OF LOOP TO RISE 6 INCHES ABOVE NO. 2 SEAL LEAKOFF CONNECTION; SLOPE PIPING DOWNWARD THROUGH RUN TO RCDT. LOCATE CHECK VALVE AT LEVEL OF RCDT.
11. VALVE SHOWN IS FOR LOOP ONE; VALVES IN OTHER LOOPS WILL USE THE APPROPRIATE RCP SEAL STANDPIPE INSTRUMENTATION FOR THAT LOOP.
12. A PORTION OF THE SYSTEM IS SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
13. CLAMPED TO PIPE SURFACE.
14. FURNISHED WITH REACTOR VESSEL HEAD.
15. SLOPE PIPING DOWNWARD FROM THIS POINT TO ASSURE 0 PSIG BACK PRESSURE ON THE NO. 3 SEAL.
16. PROVIDE 12" REMOVABLE SPOOL.
17. FOR TYPICAL DETAIL, SEE STD. DWG. AX4DD000.
18. INSTRUMENTS PT–15214 AND PT–15215 ARE TO BE ABANDONED IN PLACE. ELECTRICAL CONNECTION IS NOT REQUIRED.
19. LOOP DIAGRAMS 1X5DV239 AND 1X5DV240 ARE VOIDED. REFER TO WESTINGHOUSE FLOW DIAGRAMS, DRAWINGS NO. 1094E70 SHEET 1 OF 4, AND 1094E69 SHEET 2 OF 2.
20. VACUUM CONNECTION FOR REACTOR COOLANT VACUUM REFILL SYSTEM.
21. SOLENOIDS VALVES ARE PROJECT CLASSIFICATION 11J. SAFETY RELATED STATUS IS REQUIRED ONLY FOR THE SOLENOIDS TO ENSURE ACTIVE VALVE FUNCTION CAPABILITY OF ASSOCIATED AOV.

Q

CAD 1X4DB114
ACAD2002 | TSL – 02

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SYSTEM NO. 1208

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
41.0	REVISED PER ABN 1090006501M002, VER. 1.0	10/07/09	TSL	MWD	JMR	X

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB114	41.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. 022G2.1.19 001/LOCT AND LOIT/SRO/M/F 3.9/3.8/022G2.1.19/LO-TA-63013///

Initial conditions:

- Unit 1 is at 100% reactor power.
- Containment Cooler #1 high speed fan breaker trips.

Currect conditions:

- ALB01-E06 CNMT HI TEMP is received.
- IPC data for containment temperature is collected.

Which one of the following completes the following statement?

Based on the IPC data provided, the Tech Spec Surveillance for containment temperature (Tech Spec SR 3.6.5.1) __(1)__ within Tech Spec limits,

and

per the applicable Annunciator Response Procedure and System Operating Procedure, the crew is required, as a minimum, to __(2)__.

REFERENCE PROVIDED

A. (1) is

(2) start one additional Containment Cooler

B.✓ (1) is

(2) stop Containment Cooler #2, then start an additional pair of Containment Coolers

C. (1) is NOT

(2) start one additional Containment Cooler

D. (1) is NOT

(2) stop Containment Cooler #2, then start an additional pair of Containment Coolers

K/A

022 **Containment Cooling**

AA2.01 **Ability to use plant computers to evaluate system or component status.**

K/A MATCH ANALYSIS

The question sets up plausible scenario where the candidate must first determine which instruments are required to be referenced when complying with the Containment temperature monitoring verification surveillance of Technical Specifications. A decision is required based on the operating limit in comparison with an IPC screen shot. The second part matches the KA for the Shift Supervisor to evaluate Containment Cooling requirements used to control within environmental limits and the first part makes this SRO required knowledge when the surveillance requirement is tested.

EXPLANATION OF REQUIRED KNOWLEDGE

Per TS SR 3.6.5.1, containment average temperature is required to be verified <120F on a periodic bases. The IPC screen for Control Room TS Rounds lists values for CMNT Levels 2, C, & B and an average temperature. Individual readings above 120F are acceptable as long as the average temperature is <120F.

Per OSP 14000-1, page 15, if the IPC points for containment temperature are not available, annunciator ALB01-E06 can be verified extinquished as an alternate. This annunciator utilzes temperature elements from all three levels and has a setpoint of <120F. If the annunciator is in alarm and the IPC is unavailable, then local temperature readings are required.

Per ARP 17001-1 for ALB01-E06, if any of the three containment temperature indicators rise above 120F, the operator is instructed to start an additional pair of Containment Coolers. Containment Coolers must be started in specific pairs (1&2, 3&4, 5&6, and 7&8) due to the backdrafter damper for the individual fans being de-energized open on a common plentum (reference P&ID 1X4DB212 and SOP 13120-1 section 4.1). Starting the fans one at a time will result in the non-running fan of the pair to spinning backwards. When the second fan is subsequently started, the supply breaker will trip open from high in-rush current produced by the increased electrical slip angle.

Since fan #1 tripped, fan #2 should be stopped prior to starting an additional pair. Fan #2 is mostly recirculating short-cycled flow in the current configuration and is producing little cooling to containment. Simply starting a second fan on another pair would only create the same condition on a second pair and would also do little to lower containment temperature.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Part 1 is correct and sets up a circumstance were the candidate must determine which instrumentation in particular is required to be monitored the ensure compliance with SR 3.6.5.1 for Containment temperature. The SR requires the average temperature of three levels in Containment be noted to ensure Containment temperature remains within the accident analyses.

Part 2 is incorrect but 'plausible' because the candidate may determine that the only required action is to start an additional Containment Cooler to replace the tripped Containment cooler not recalling that the system operating procedure would require

the fans be operated in pairs.

B. Correct. Part 1 is correct. See Part 1 of choice A above.

Part 2 is correct per the ARP 17001-1, and the SOP 13120-1, an additional pair of Containment Coolers would be started.

C. Incorrect. Plausible. Part 1 is incorrect but 'plausible' in that the candidate may determine that Technical Specification if any of the three levels listed on the IPC printout exceeded 120F. This assumption would be consistent with the use of ALB01-E06 as an alternate as described in OSP 14000-1.

Part 2 is incorrect. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice C above.

Part 2 is correct. See part 2 of choice B above.

SRO JUSTIFICATION

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the question does not address 1 hour Tech Spec actions.
-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" No, the question is not related to above-the-line information in Tech Spec.
-Can question be answered *solely* by knowing the TS Safety Limits? No, the question is not related the Tech Spec Safety Limits.
-Does the question involve one or more of the following for TS,TRM, or ODCM?

- **Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1). Yes, the question requires specific knowledge related to Tech Spec surveillance requirements and instruments used to satisfy the requirement.**
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4). **No, the question is not related to these aspects of Tech Spec.**
- Knowledge of TS bases that is required to analyze TS required actions and terminology. **No, the question is not related to Tech Spec bases.**

Level:	SRO
Tier # / Group #	T2 / G1
K/A#	022G2.1.19
Importance Rating:	3.9 / 3.8

Technical Reference: TS 3.6.5, Amendment No. 158, page 3.6.5-1
TS 3.6.5 Bases Rev 14.0, page 3.6.5-4
SOP 13120-1, Rev 24.0, pages 3 & 5-8
ARP 17001-1, Rev 31.1, page 3 & 47
OSP 14000-1, Rev 88.2, page 15
P&ID 1X4DB212, Rev 12.0

References provided: IPC Screen shot of CNMT Temps

Learning Objective:
LO-TA-63013 Implement Technical Specification LCO using 10008-C (SRO Only)
LO-LP-39209-01 For any given item in section 3.5 of Tech Specs, be able to: State the LCO. State any one hour or less required actions
LO-LP-39209-03 Describe the bases for any given Tech Spec in section 3.5.
LO-PP-29101-15 State the starting interlocks associated with the Containment Cooling fans. Include set points and coincidence where applicable.

Question origin: MODIFIED - LORQ Question # V-LO-PP-29101-09 001

Cognitive Level: M/F

10 CFR Part 55 Content: 41.9 / 43.2

Comments:

You have completed the test!

1. V-LO-PP-29101-09 001/SRO/0022A2.04/LO-TA-29003/C/A/3/DIABLO CANYON 05/SOP 13120-1/2/TECH SPEC 3.6.5

Due to the high summer air temperature and fouling of the Containment Air Coolers are causing the following conditions to occur on Unit 1:

Original Question

- ALB01-E06 “CNMT HI TEMP” is in alarm
- Containment Level 2 temperature - 122°F
- Containment Level C temperature - 114°F
- Containment Level B temperature - 120°F
- Containment Coolers 1,2,5, and 6 are running in high speed
- Containment pressure is currently 1.0 psig and increasing slowly

Based on the current plant conditions, which of the following actions should the Shift Supervisor the operator to perform?

A.✓ Direct a start of the Train “B” Containment Coolers in Hi speed, and consider venting Containment.

B. Direct a start of Train “B” Containment Coolers in Hi speed, and restore pressure within limits within the next 4 hours or be in MODE 3 in the next 6 hours.

C. Direct a start of the Train “B” Containment Coolers in Hi speed, reduce Containment temperature to less than 120°F within the next 8 hours or be in MODE 3 in the next 6 hours and consider venting Containment.

D. Direct a start of the Train “B” Containment Coolers in Hi speed, restore pressure within limits within the next 4 hours or be in MODE 3 in the next 6 hrs and consider venting Containment.

P002
ROOM TECH SPEC ROUNDS

DESCRIPTION	VALUE	UNITS
CNMT NARROW RANGE PRESS	0.10	PSIG
NSCW SUPPLY HEADER TEMP TRAIN A TEMP	71.6	DEG F
NSCW SUPPLY HEADER TRAIN B TEMP	67.3	DEG F
CNMT LEVEL 2 TEMP	122	DEG F
CNMT LEVEL C TEMP	114	DEG F
CNMT LEVEL B TEMP	115	DEG F
AVERAGE CNMT TEMP	117	DEG F
CNMT SOUTH SUMP LEVEL	21.0	INCHES
REACTOR CAVITY SUMP LEVEL	21.4	INCHES
CNMT NORTH SUMP LEVEL	16.3	INCHES
PRI MET TWR 10M 15 MIN AVE WIND SPEED	2.6	MPH
PRI MET TWR 60M 15 MIN AVE WIND SPEED	6.2	MPH
PRI MET TWR 10M 15 MIN AVE WIND DIR	156	DEGREES
PRI MET TWR 60M 15 MIN AVE WIND DIR	136	DEGREES
PRI MET TOWER 60-10M 15 MIN AVE DELTA TEMP	0.615	DEG F

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5.1 Verify containment average air temperature is within limit.	In accordance with the Surveillance Frequency Control Program

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Location	Tag Number
a. Level 2	TE-2563
b. Level B	TE-2613
c. Level C	TE-2612

NOTE: A local sample may be taken at a corresponding location in lieu of using one of the instruments designated above.

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.2.

2. 10 CFR 50.49.

1.0 PURPOSE

This procedure provides instructions for operation of the Containment Building Cooling Systems, which consist of these subsystems: Containment Heat Removal System (CHRS), Control Rod Drive Mechanism (CRDM) Cooling Units, and the Containment Building (CTB) Cavity Cooling System. Instructions are provided in the following subsections:

4.1 CTB Cooling System Startup To Standby

4.2 Containment Heat Removal System Startup

4.3 CRDM Cooling Units Startup

4.4 Shifting CRDM Cooling Units

4.5 CTB Cavity Cooling System Startup

4.6 Shifting CTB Cavity Cooling Units

4.7 Shifting CTB Reactor Support Cooling Fans

4.8 Shifting CTB Cooling Unit Fans

4.9 Shifting Auxiliary Coolers

4.10 Post LOCA Purge Cavity Fan Operation

4.11 Containment Heat Removal System Shutdown

4.12 CRDM Cooling Units Shutdown

4.13 CTB Cavity Cooling System Shutdown

INITIALS

4.0 INSTRUCTIONS

4.1 CTB COOLING SYSTEM STARTUP TO STANDBY

4.1.1 **Perform** Table 1 to align the system handswitches for startup. ________

4.1.2 If required, **perform** the system startup alignment per 11120-1, "Containment Building Cooling System Alignment". ________

4.1.3 **Close** the links for the K2 Relay and **close** the breakers for the CNMT COOLING UNITS (IV REQUIRED), **document** on Checklist 1:

- 1ABE-26 ________
- 1ABE-27 ________
- 1ABC-07 ________
- 1ABC-08 ________
- 1BBE-26 ________
- 1BBE-27 ________
- 1BBC-07 ________
- 1BBC-08 ________

INITIALS

NOTE

Panel LR01 (1-1816-U3-019) is located in Control Building, Corridor R-149, outside the Radiochemistry Lab.

4.1.4 **Open** the CTB Cooling Units Outlet Dampers at Panel LR01. Damper Position may also be verified by 1-ZLB indications on 1-QHVC. (IV REQUIRED), IF available, the IPC may also be used to **verify** damper position, **document** on Checklist 1.

	DAMPER	DESCRIPTION	HANDSWITCH	1-QHVC	INITIALS
•	1-HV-2582A	CTB CLG UNIT 1	1-HS-2582G	1ZLB-43	________
•	1-HV-2582B	CTB CLG UNIT 2	1-HS-2582H	1ZLB-43	________
•	1-HV-2583A	CTB CLG UNIT 3	1-HS-2583G	1ZLB-44	________
•	1-HV-2583B	CTB CLG UNIT 4	1-HS-2583H	1ZLB-44	________
•	1-HV-2584A	CTB CLG UNIT 5	1-HS-2584G	1ZLB-43	________
•	1-HV-2584B	CTB CLG UNIT 6	1-HS-2584H	1ZLB-43	________
•	1-HV-2585A	CTB CLG UNIT 7	1-HS-2585G	1ZLB-44	________
•	1-HV-2585B	CTB CLG UNIT 8	1-HS-2585H	1ZLB-44	________

INITIALS

4.1.5 **Open** the links for the K2 Relay for the following breakers. (IV REQUIRED), **document** on Checklist 1:

- 1ABE-26 ________
- 1ABE-27 ________
- 1ABC-07 ________
- 1ABC-08 ________
- 1BBE-26 ________
- 1BBE-27 ________
- 1BBC-07 ________
- 1BBC-08 ________

INITIALS

NOTES

- CTB Cooling Units Outlet Dampers are OPEN and DE-ENERGIZED with their breakers LOCKED OPEN to preclude inadvertent closure of dampers.
- Damper indication on 1ZLB-43 and 1ZLB-44 will have no indication after performing the following step.
- Indication will still be available for damper indication on the IPC. Computer points for damper indication are; ZD9430, ZD9432, ZD9434, ZD9436, ZD9438, ZD9440, ZD9442 and ZD9444.

4.1.6 **Open** and **lock** the following breakers for the CTB Cooling Units Outlet Dampers. (IV REQUIRED), **document** on Checklist 1:

	DESCRIPTION	BREAKER	INITIALS
•	CTB CLG UNIT A7-001 1-HV-2582A,	1ABE-26	________
•	CTB CLG UNIT A7-002 1-HV-2582B,	1ABE-27	________
•	CTB CLG UNIT A7-003 1-HV-2583A,	1BBE-26	________
•	CTB CLG UNIT A7-004 1-HV-2583B,	1BBE-27	________
•	CTB CLG UNIT A7-005 1-HV-2584A,	1ABC-07	________
•	CTB CLG UNIT A7-006 1-HV-2584B,	1ABC-08	________
•	CTB CLG UNIT A7-007 1-HV-2585A,	1BBC-07	________
•	CTB CLG UNIT A7-008 1-HV-2585B,	1BBC-08	________

	(1)	(2)	(3)	(4)	(5)	(6)
A	CIRC WTR P-1 MOTOR OVERLOAD	CIRC WTR P-2 MOTOR OVERLOAD	TPCW PUMP 1 TRIPPED	TPCW PUMP 2 TRIPPED	SERVICE AIR SWING CMPSR MISALIGNED	AIR CMPSR MSTR SEP DISCH HI TEMP
B	CIRC WTR P-1 DISCH VLV TROUBLE	CIRC WTR P-2 DISCH VLV TROUBLE	TPCW PMP DISCH HDR LO PRESS		SERVICE AIR CMPSR TROUBLE	INSTR AIR EQUIP LO PRESS
C	CIRC WTR P-1 LO PIT LVL	CIRC WTR P-2 LO PIT LVL	CLG TOWER BASIN HI LVL		UNIT 1 SERV AIR HDR TIED TO UNIT 2	SERVICE AIR HDR LO PRESS
D	CIRC WTR P-1 SCREEN WTR HI DIFF LVL	CIRC WTR P-2 SCREEN WTR HI DIFF LVL	TPCCW PUMP 1 TRIPPED	TPCCW PUMP 2 TRIPPED		INSTR AIR CNMT SPLY LINE BREAK
E	CONDR CIRC WTR ISO VLV CLOSED		TPCCW SURGE TK HI/LO LVL	TPCCW DISCH HDR LO PRESS		CNMT HI TEMP
F						CNMT HI MSTR

WINDOW E06

ORIGIN	SETPOINT	
1-TSH-2563 1-TSH-2612 1-TSH-2613	120°F	CNMT HI TEMP

1.0 **PROBABLE CAUSE**

Insufficient number of Containment Building Cooling Units operating.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Start** an additional pair of Containment Cooling Units or a Containment Auxiliary Cooling Unit per 13120-1, "Containment Building Cooling Systems".

2. **Verify** Nuclear Service Cooling Water flow to coolers, and IF necessary, **dispatch** an operator to inspect the Containment Heat Removal System.

3. **Refer to** Technical Specification LCO 3.6.5 and 3.6.6.

4. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 **COMPENSATORYOPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES:1X4DB212, CX5DT101-66, CX5DT101-71

Sheet 9 of 10

DATA SHEET 1
MODE 1 & 2

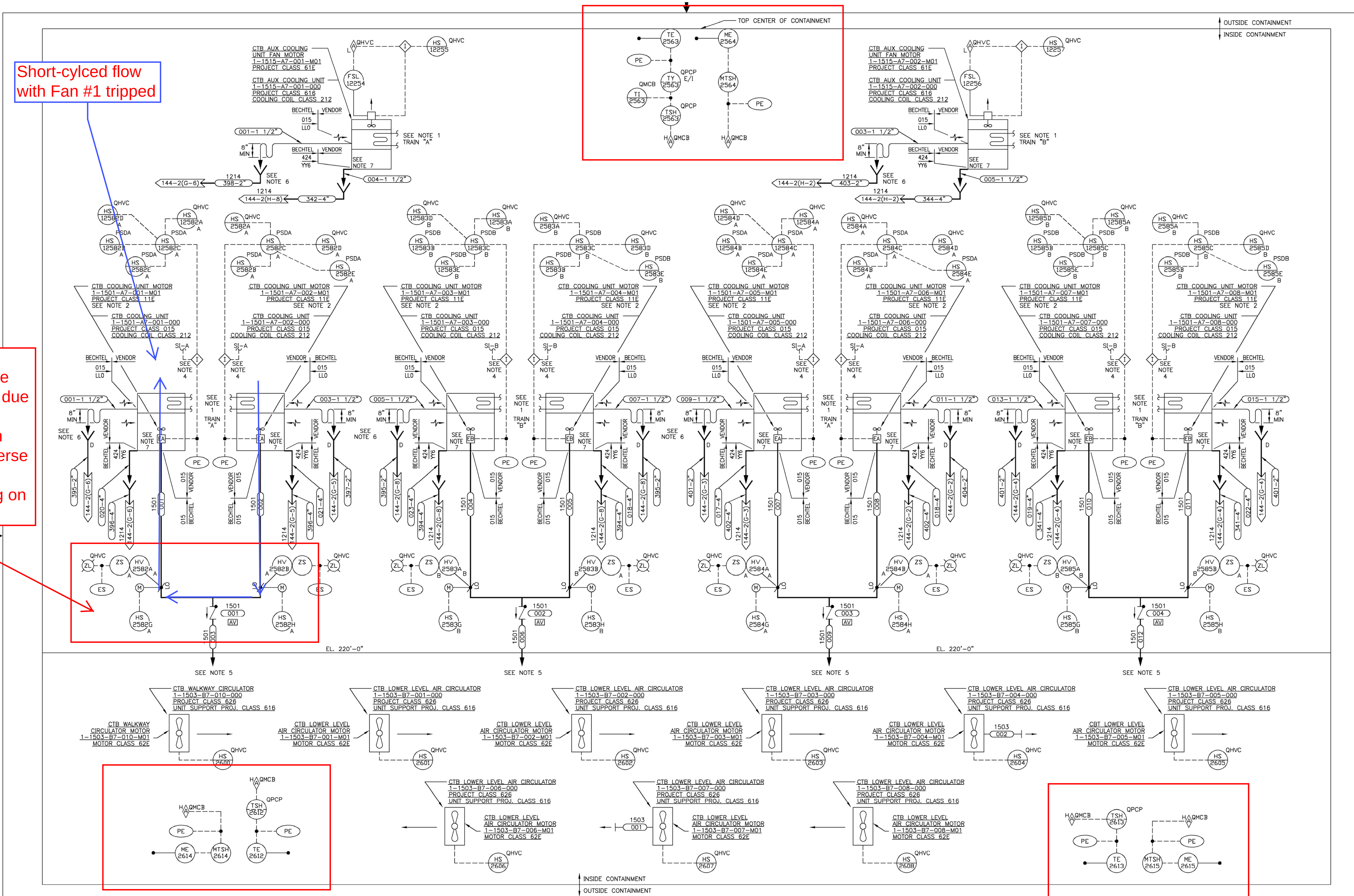
MODE _______________

DATE _______________

LCO METHOD OF VERIFICATION	TECH SPEC SURV REQ	PARAMETER	INSTRUMENT	INDICATION DAY	INDICATION NIGHT	LIMIT(S) TOLERANCE	LCO/PROC
CREFS ACTUATION OPERABLE CHANNEL CHECK	SR 3.3.7.1 FCN 3	CR INTAKE RADIATION MONITORS (INIT)	1RE-12116			CHANNEL CHECK REQUIRED 2	3.3.7
			1RE-12117				
FHB ACTUATION OPERABLE CHANNEL CHECK	TRS 13.3.6.1	FHB EFFL RADIOGAS FHB ISO (INIT)	ARE-2532A			* REQUIRED 1	13.3.6
			ARE-2532B				
FHB ACTUATION OPERABLE CHANNEL CHECK	TRS 13.3.6.1	FHB EFFL RADIOGAS FHB ISO (INIT)	ARE-2533A			* REQUIRED 1	13.3.6
			ARE-2533B				
		*INDICATING NORMALLY. ALL STATUS AND ALARM LIGHTS EXTINGUISHED.					
DG1A FUEL OIL INVENTORY VERIFY FUEL OIL STORAGE TANK LEVEL	SR 3.8.3.1	DG 1A LEVEL (%)	1-LI-9024			≥82%	3.8.3
DG1B FUEL OIL INVENTORY VERIFY FUEL OIL STORAGE TANK LEVEL	SR 3.8.3.1	DG 1B LEVEL (%)	1-LI-9025			≥ 82%	3.8.3
TWO INDEPENDENT CONTROL ROOM EMERGENCY FILTRATION SYSTEMS SHALL BE OPERABLE VERIFY CONTROL ROOM TEMP	SR 3.7.10.1 SR 3.7.11.1	NOTE: TEMPERATURE INDICATION IS OBTAINED FROM HAND-HELD TEST EQUIPMENT. RECORD INSTRUMENT INFORMATION BELOW.					
		INSTRUMENT ID NO.				N/A	
		CAL DUE DATE					
		CONTROL ROOM TEMPERATURE (°F)	M&TE			≤85°F	3.7.10 3.7.11
THE RWST SHALL BE OPERABLE VERIFY TEMPERATURE	SR 3.5.4.1 TRS 13.1.7.1	RWST TEMPERATURE (°F)	1TIS-10980			≥51°F * ≤109°F *	3.5.4 13.1.7
		*WITH INDICATED RWST TEMPERATURE OUTSIDE THE LIMITS, THEN VERIFY RWST TEMPERATURE IS WITHIN TECHNICAL SPECIFICATION LIMITS BY PLACING THE RWST ON RECIRC USING SLUDGE MIXING PUMP WITH HEATER OFF AND OBSERVING 1-TI-10982 TO BE WITHIN ≥44°F AND ≤116°F.					
THE ULTIMATE HEAT SINK SHALL BE OPERABLE VERIFY WATER TEMPERATURE AND LEVEL	SR 3.7.9.2	TEMPERATURE (°F)	COMPUTER POINT T2601*			≤90°F	3.7.9
			-OR- 1TJI-1692 POINT 2*				
			COMPUTER POINT T2602*				
			-OR- 1TJI-1692 POINT 17*				
		*IF COMPUTER POINT AND RECORDER POINT ARE NOT AVAILABLE, TEMPERATURE READING MUST BE OBTAINED LOCALLY USING HAND-HELD TEST EQUIPMENT. RECORD INSTRUMENT INFORMATION BELOW.					
		INSTRUMENT ID NO.				N/A	
		CAL DUE DATE					
	SR 3.7.9.1	LEVEL (%)	1LI-1606			≥73%	
			1LI-1607				
CONTAINMENT AIR TEMPERATURE SHALL NOT EXCEED 120°F VERIFY AVERAGE AIR TEMPERATURE	SR 3.6.5.1	TEMPERATURE (°F)	COMPUTER POINT T2501			NA	3.6.5
			COMPUTER POINT T2502				
			COMPUTER POINT T2503				
			COMPUTER POINT UT2501 (AVG)			≤120°F	
		*IF COMPUTER POINT IS NOT AVAILABLE VERIFY CNMT HI TEMP ALARM ALB-01 (E06) IS NOT IN ALARM.				ALB-01 (E06) NOT IN ALARM	
		*IF COMPUTER POINT AND ALB-01 (E06) ARE NOT AVAILABLE, TEMPERATURE READING MUST BE OBTAINED LOCALLY USING HAND-HELD TEST EQUIPMENT FOR 1TE-2612 FOR POINT T2502 AND 1TE-2613. FOR POINT T2503 RECORD INSTRUMENT INFORMATION BELOW. USE MCB INDICATOR 1TI-2563 FOR POINT T2501 AND AVERAGE THE THREE.					
		INSTRUMENT ID NO.				≤120°F	
		CAL DUE DATE					

COMPLETED BY: DAY: _______________ TIME: ________ NIGHT: _______________ TIME: ________

SS REVIEW: DAY: _______________ TIME: ________ NIGHT: _______________ TIME: ________



NOTES:

1. FOR NUCLEAR SERVICE COOLING WATER SEE P&ID 1X4DB135.
2. CTB COOLING UNIT FAN MOTORS ARE 2-SPEED.
3. FOR DRAIN LINES SEE P&ID 1X4DB144-2.
4. CTB COOLING UNIT FANS SHALL OPERATE AT LOW SPEED UPON RECEIPT OF THE SAFETY INJECTION (SI) SIGNAL.
5. REFER TO DWG. 1X4DB251-1 FOR AIR DISTRIBUTION.
6. SCREEN(S) INSTALLED TO PREVENT TRASH AND MISC. DEBRIS GETTING INTO AND BLOCKING DRAINS.
7. FLOOR DRAIN SCREEN(S) MAY BE INSTALLED IN THE AUX & MAIN COOLERS AS SHOWN IN MDD 92-V1M119. A RER SHOULD BE INITIATED TO ENSURE DWG IS REVISED TO REFLECT THE SCREEN(S) INSTALLATION.

NO.		DATE	DR	CHK	APPV	DTL
12.0	REDRAWN PER ABN-52221, VER. 1.0	11-2-05	NJ	LCF	EPD	X

CAD 1X4DB212 ACAD2002 NJ - 01

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CONTAINMENT HEAT REMOVAL SYSTEM
SYSTEM NO. 1501 & 1515

SCALE: NONE | JOB NO. 10604 | DRAWING NO. 1X4DB212 | VER. 12.0

SIZE E 34X44 | 1 | DRAWING CATEGORY: CRITICAL

1. 025AG2.4.30 001/LOCT AND LOIT/SRO/C/A 2.7/4.1/025AG2.4.30/LO-TA-40002///

At time 1055:

- Unit 1 is in Mode 5.
- Containment integrity is NOT established.
- Pressurizer level is 27%.
- All Pressurizer Safety Valves are removed.

At time 1100:

- An LOSP occurs.
- DG1A trips on overspeed.
- RHR pump 'B' will NOT start following the load sequence.
- RCS temperature is 200°F and increasing.

Which one of the following completes the following statement?

At time 1115, the Shift Manager must declare a(n) __(1)__ per NMP-EP-110, "Emergency Classification Determination and Initial Action,"

and

no later than **time 1130**, as a minimum, the __(2)__ must be notified of the declaration per NMP-EP-111, "Emergency Notifications."

REFERENCE PROVIDED

	__(1)__	__(2)__
A.	NOUE	NRC, state, and local authorities
B.	NOUE	state and local authorities
C.	Alert	NRC, state, and local authorities
D.✓	Alert	state and local authorities

K/A

025 **Loss of RHR System**

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.**

K/A MATCH ANALYSIS

The question sets up a plausible scenario which includes the first required element of

the KA loss of RHR cooling, then has the candidate evaluate plant conditions and make determination of reporting requirements which meets the second part of the KA and brings the knowledge to the SRO required level.

EXPLANATION OF REQUIRED KNOWLEDGE

Based on the conditions in the stem, 2 EPIPs classification thresholds have been exceed. First, an LOSP has existed with only the 'B' DG energizing 1BA03. As such NOUE CU3 has been exceeded. Additonally, due to both RHR pumps not running, RCS temperature has risen above 200F while in Mode 4. Containment integrity is not established. Since all pressurizer safeties are removed, RCS integrity is also not established. As shuch, ALERT CA4 has also been exceeded. Therefore, the Emergency Director is required to declare an ALERT emergency on or before 11:15 per NMP-EP-110.

Per NMP-EP-111 steps 5.1.1 ad 5.1.3, state and local agencies must be notified within 15 minutes of the declaration, which would be 11:30. Additionally, the NRC shall be notified immediately follwoing the state and local agencies and within an hour of the declaration.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Part 1 is incorrect but plausible because the candidate may determine the loss of power event is driving the initial classification not recognizing the loss of RHR Cooling and CU3 would be the correct.

Part 2 is incorrect but ‘plausible’ because the candidate may determine that the NRC must be notified in this condition within 15 minutes. Per NMP-EP-111, Notification of the NRC shall be completed immediately following notifications to the state and local agencies and within an hour of the declaration of an emergency. Therefore, the NRC is not REQUIRED to be notified by 11:30, but instead is allowed additonal time if needed up to an hour, but is to be done as soon as possible following the state and locals.

B. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice A above.

Part 2 is correct in that within 15 minutes of the classification of the NOUE the state and local authorities must be notified per NMP-EP-111, "Emergency Notifications". Notifications of applicable State and Local Agencies shall be accomplished as soon as practicable and within 15 minutes of the declaration of an emergency, an upgrade to a higher emergency classification level, or the approval of protective actions recommendations.

C. Incorrect. Plausible. Part 1 is correct and has the candidate evaluate plant conditions and determine the event meets the ALERT threshold for CA4 due to being in Mode 5 with RCS temperature >200F without containment and RCS integrity established.

Part 2 is incorrect. See Part 2 of choice A above.

D. Correct. Part 1 is correct. See Part 1 of choice C above.

Part 2 is correct. See Part 2 of choice B above.

SRO JUSTIFICATION

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the answer requires specific knowledge of emergency classification thresholds.
-Can the question be answered *solely* by knowing immediate operator actions? No, IOAs are not addressed by this question.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, the question does not address AOP or EOP entry conditions.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, the answer requires specific knowledge of emergency classification thresholds.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- **Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. Yes, the answer requires specific knowledge of emergency classification thresholds and determination of the specific classification based on current plant conditions. This is an SRO ONLY job link associated with an SRO ONLY objective. [LO-LP-40101-13 Given an emergency scenario, and the procedure, classify the emergency (SRO only).]**

Level:	SRO
Tier # / Group #	T1 / G1
K/A#	025G2.4.30
Importance Rating:	2.7 / 4.1

Technical Reference: NMP-EP-110-GL03, Rev 3.0, page 58
NMP-EP-110-GL03, Figure 3, Rev 3.0, page 123
NMP-EP-111, Rev 8.0, page 7

References provided: NMP-EP-110-GL03, Figure 1, Rev 3.0, page 121
NMP-EP-110-GL03, Figure 2, Rev 3.0, page 122
NMP-EP-110-GL03, Figure 3, Rev 3.0, page 123

Learning Objective:

LO-TA-40002	Emergency Classification and Implementing Instructions using NMP-EP-110 (SRO Only)
LO-TA-40003	Emergency Notifications using NMP-EP-111
LO-LP-40101-13	Given an emergency scenario, and the procedure, classify the emergency (SRO only).
LO-LP-40101-16	List the state and federal authorities that are notified in an emergency.

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 43.5

Comments:

You have completed the test!

FIGURE 3

Vogtle Electric Generating Plant

Procedure Number: NMP-EP-110-GL03 Rev 3.0

EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS

Figure 3 - COLD INITIATING CONDITION EMERGENCY ACTION LEVEL MATRIX - MODES 5, 6 AND DEFUELED ONLY

RADIOLOGICAL: EFFLUENTS | RAD LEVELS
System Malfunctions: AC / DC POWER | RX and CORE | HEAT REMOVAL | COMMUNICATIONS
HAZARDS: NATURAL / DESTRUCTIVE | FIRE / EXPLOSION | TOXIC / FLAMMABLE | SECURITY | CR EVACUATION | ED DISCRETION

GENERAL EMERGENCY

EFFLUENTS

RG1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mR TEDE **OR** 5000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology (pg. 36)

See Notes 1 and 2 →

1. VALID reading on any of the following radiation monitors that exceeds **OR** expected to exceed the reading shown for 15 minutes **OR** longer:

Monitor	Value
RE-12839	1.5×10^5 µCi/cc
RE-12444	7.0×10^2 µCi/cc
RE-13119 thru 13122	3.1×10^5 µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 1000 mR TEDE **OR** 5000 mR thyroid CDE at **OR** beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates exceeding 1000 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 5000 mR for one hour of inhalation, at **OR** beyond the site boundary.

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note 2: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

RX and CORE

CG1 - Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV. **Modes 5 and 6 Only.** (pg. 93)

1. Loss of RPV inventory as indicated by unexplained Containment sump level rise, Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

AND

2. RPV Level:
   a. Less than 63% on RVLIS full range for greater than 30 minutes
   OR
   b. RPV level **CANNOT** be monitored **WITH** indication of core uncovery for greater than 30 minutes as evidenced by ANY of the following:
   - RE-005 **OR** 006 greater than 162 R/hr
   - RE-011 greater than 15 mR/hr
   - Erratic Source Range Monitor Indication.

AND

3. Containment challenged as indicated by ANY of the following:
   - Explosive mixture inside containment greater than **OR** equal to 6% H_2.
   - Containment Pressure greater than **OR** equal to 3 psig **WITH** CONTAINMENT CLOSURE established
   - Containment Pressure greater than **OR** equal to 52 psig **WITH** Tech Spec containment integrity intact
   - CONTAINMENT CLOSURE **NOT** established.

SECURITY

HG1 - HOSTILE ACTION resulting in loss of physical control of the facility (pg. 119)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

ED DISCRETION

HG7 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency. (pg. 120)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity
OR
HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

EFFLUENTS

RS1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE **OR** 500 mR Thyroid CDE for the Actual or Projected Duration of the Release. (pg. 34)

See Notes 3 and 4 →

1. VALID reading on any of the following radiation monitors that exceeds **OR** is expected to exceed the reading shown for 15 minutes OR longer:

Monitor	Value
RE-12839	1.5×10^4 µCi/cc
RE-12444	7.0×10^1 µCi/cc
RE-13119 thru 13122	3.1×10^4 µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE **OR** 500 mR thyroid CDE at **OR** beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at **OR** beyond the site boundary.

Note 3: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note 4: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

RX and CORE

CS1 - Loss of RPV Inventory Affecting Core Decay Heat Removal Capability **Mode 5 Only** (pg. 89)

1. Loss of Reactor Pressure Vessel (RPV) inventory affecting core decay heat removal capability with CONTAINMENT CLOSURE NOT established as indicated by:
   a. RPV level less than 72% on RVLIS full range [6" below Bottom ID of loop]
   OR
   b. RPV level CANNOT be monitored for greater than 30 minutes with a possible loss of RPV inventory as indicated by unexplained level rise in Containment sump, Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

OR

2. Loss of RPV inventory affecting core decay heat removal capability with CONTAINMENT CLOSURE established as indicated by:
   a. RPV level less than 63% on RVLIS full range [TOAF]
   OR
   b. RPV level CANNOT be monitored for greater than 30 minutes with a possible loss of RPV inventory as indicated by EITHER of the following: Unexplained Containment sump level rise or Erratic Source Range Monitor Indication.

HEAT REMOVAL

CS2 - Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV. **Mode 6 Only** (pg. 91)

1. **WITH** CONTAINMENT CLOSURE **NOT** established:
   a. RPV level less than 72% on RVLIS full range [6" below Bottom ID of loop]
   OR
   b. RPV level CANNOT be monitored WITH indication of core uncovery as evidenced by ANY of the following: RE-002, 003, OR 004 greater than 3.1 mR/hr or Erratic Source Range Monitor Indication.

OR

2. **WITH** CONTAINMENT CLOSURE established:
   a. RPV level less than 63% on RVLIS full range [TOAF]
   OR
   b. RPV level CANNOT be monitored WITH indication of core uncovery as evidenced by ANY of the following: RE-005 OR 006 greater than 162 R/hr, RE-011 greater than 15 mR/hr, Erratic Source Range Monitor Indication.

SECURITY

HS4 - HOSTILE ACTION within the PROTECTED AREA (pg. 118)

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the security shift supervision.

CR EVACUATION

HS2 - Control Room Evacuation Has Been Initiated AND Plant Control Cannot Be Established (pg. 116)

1. a. Control Room evacuation has been initiated
   AND
   b. Control of the plant **CANNOT** be established per 18038 – Operation From the Remote Shutdown Panels, within 15 minutes.

ED DISCRETION

HS3 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency (pg. 117)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public
OR
HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels that exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Correct Answer

ALERT EMERGENCY

EFFLUENTS

RA1 - Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer. (pg. 27)

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

1. VALID reading on any of the following effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes OR longer.

Monitor	200 X Setpoint Value
RE-0018	2.8×10^{-3} µCi/cc
RE-0021	4.0×10^{-3} µCi/cc
RE-0848	3.5×10^{-3} µCi/cc
ARE-0014	$2.4 \times 10^{+2}$ µCi/cc
RE-12839	7.9×10^{-1} µCi/cc
RE-12444C	1.3×10^{-1} µCi/cc

OR

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes **OR** longer:

RE-13119 thru 13122 2.5 µCi/cc

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations **OR** release rates in excess of 200 times Technical Specification 5.5.4 as confirmed by the ODCM, with a release duration of 15 minutes **OR** longer.

RAD LEVELS

RA2 - Damage to Irradiated Fuel **OR** Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. (pg. 30)

1. UNPLANNED VALID alarm on any of the following radiation monitors:
   - Fuel Handling Building RE-008
   - CNMT BLDG Low Range** RE-002/003
     ** Mode 6 during fuel transfer
   - Fuel Handling BLDG EFFL. ARE-2532 A/B
   - Fuel Handling BLDG EFFL. ARE-2533 A/B

OR

2. Loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel as indicated by ANY of the following:
   - Personnel report during fuel assembly movements.
   - Spent Fuel Pool Storage Less than El 193'-5"
   - Transfer Canal Transit El Less than El 186'10"
   - Reactor Core El Less than El 181'-10" (63% on RVLIS full range)

RA3 Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown (pg. 32)

1. VALID radiation monitor readings greater than 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:
   Control Room radiation monitor RE-001 OR Central Alarm Station (by survey)

OR

2. UNPLANNED VALID radiation readings greater than 1 R/hr values in areas requiring infrequent access to maintain plant safety functions.

South & North Main Steam Valve Rooms, Turbine Building, Control Building, Fire Pump House, Aux Feed Water Pump House, Low Voltage Switchyard, Diesel Generator Buildings, NSCW Towers

AC / DC POWER

CA3 - Loss of All Offsite Power **AND** Loss of All Onsite AC Power to Essential Busses. (pg. 86)

1. a. Loss of power to **OR** from transformers 1(2)NXRA **AND** 1(2)NXRB resulting in loss of all off-site electrical power to **BOTH** 1(2)AA02 **AND** 1(2)BA03 for greater than 15 minutes.
   AND
   b. Failure of emergency generators to supply power to emergency busses.
   AND
   c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Distractor

RX and CORE

CA1 - Loss of RCS Inventory. **Mode 5 Only** (pg. 84)

1. Loss of RCS inventory as indicated by RPV level less than elevation 185'-9.5" (185.79') (73% on RVLIS full range)

OR

2. a. RCS level **CANNOT** be monitored for greater than 15 minutes
   AND
   b. A possible loss of RCS inventory may be occurring as indicated by unexplained level rise in Containment sump, Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

CA2 - Loss of RPV Inventory with Irradiated Fuel in the RPV. **Mode 6 Only** (pg. 85)

1. Loss of inventory as indicated by RPV level less than 185'-9.5" (185.79') (73% on RVLIS full range)

OR

2. a. RPV level **CANNOT** be monitored for greater than 15 minutes
   AND
   b. A possible loss of RCS inventory may be occurring as indicated by unexplained level rise in Containment sump, , Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

HEAT REMOVAL

CA4 - Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV. **Modes 5 and 6 Only.** (pg. 87)

Note The Emergency Director should not wait until 20 minutes has elapsed, but should declare the event as soon as it is determined that the duration has or will likely exceed 20 minutes.

1. An UNPLANNED event results in RCS temperature exceeding 200°F with:
   a. CONTAINMENT CLOSURE **NOT** established
   AND
   b. RCS integrity **NOT** established.

OR

NOTE If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this Threshold Value is not applicable.

2. An UNPLANNED event results in RCS temperature exceeding 200°F for greater than 20 minutes (Note) with:
   a. CONTAINMENT CLOSURE established
   AND
   b. RCS integrity NOT established
   OR
   c. RCS inventory reduced.

OR

3. An UNPLANNED event results in:
   a. RCS temperature exceeding 200°F for greater than 60 minutes (Note)
   OR
   b. RCS pressure increasing greater than 10 psig.

NATURAL / DESTRUCTIVE

HA1 - Natural and Destructive Phenomena Affecting the Plant VITAL AREA (pg. 105)

1. Seismic monitoring system confirms seismic event greater than OBE Earthquake acceleration of 0.12 g.

OR

2. Tornado **OR** high winds greater than 100 mph within the PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures/equipment (Table 1) **OR** the Control Room has indication of degraded performance of any listed systems: (Table 1)

OR

3. Vehicle crash within PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures **OR** equipment therein (Table 1) **OR** control indication of degraded performance of those systems: (Table 1)

OR

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to **OR** penetration of any area containing safety-related equipment, their controls **OR** their power supplies. (Table 1)

OR

6. Sustained hurricane winds onsite resulting in VISIBLE DAMAGE to plant structures within the PROTECTED AREA boundary containing equipment necessary for safe shutdown. (Table 1)

Table 1
- Containment Building
- Fuel Hand Building
- NSCW Cooling Towers
- Control Building
- Diesel Generator Building
- Auxiliary Building
- Diesel Fuel Oil Storage Tank Pumphouse
- Auxiliary Feedwater Pumphouse

FIRE / EXPLOSION

HA2 - FIRE **OR** EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown (pg. 108)

1. FIRE **OR** EXPLOSION in a plant VITAL AREA **AND** affected system parameter indications show degraded performance **OR** plant personnel report VISIBLE DAMAGE to permanent structures **OR** safety related equipment in any of the following:
   - Containment Building
   - Fuel Hand Building
   - NSCW Cooling Towers
   - Control Building
   - Diesel Generator Building
   - Auxiliary Building
   - Diesel Fuel Oil Storage Tank Pumphouse
   - Auxiliary Feedwater Pumphouse

TOXIC / FLAMMABLE

HA3 - Release of Toxic, Asphyxiant, or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown (pg. 110)

1. Report **OR** detection of toxic **OR** asphyxiant gas within **OR** contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report **OR** detection of flammable gases in concentration greater than the LOWER FLAMMABILITY LIMIT within **OR** contiguous to a VITAL AREA.

SECURITY

HA4 – HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat. (pg. 112)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by security shift supervision.

OR

2. A validated notification from the NRC of an airliner attack threat within 30 minutes of the site.

CR EVACUATION

HA5 - Control Room Evacuation Has Been Initiated. (pg. 114)

1. Entry into 18038 – Operation From the Remote Shutdown Panels, procedure for Control Room evacuation.

ED DISCRETION

HA6 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. (pg. 115)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant
OR
a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

NOTIFICATION OF UNUSUAL EVENT

EFFLUENTS

RU1 – Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. (pg. 23)

Note: The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes **OR** longer.

Monitor	2 X Setpoint Value
RE-0018	2.8×10^{-5} µCi/cc
RE-0021	4.0×10^{-5} µCi/cc
RE-0848)	3.5×10^{-5} µCi/cc
ARE-0014	2.4 µCi/cc
RE-12839	7.9×10^{-3} µCi/cc
RE-12442C or 1RE-12444C	1.3×10^{-3} µCi/cc

OR

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes **OR** longer:
   Main Steam (RE-13119-RE-13122) 2.5×10^{-2} µCi/cc

OR

3. Confirmed sample analyses for gaseous **OR** liquid releases indicates concentrations **OR** release rates, with a release duration of 60 minutes **OR** longer, in excess of two times Technical Specification 5.5.4, as confirmed by the ODCM.

RAD LEVELS

RU2 – Unexpected Rise in Plant Radiation (pg. 25)

1. a. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, **OR** fuel transfer canal with all irradiated fuel assemblies remaining covered by water indicated by any of the following:
   - Personnel report of low water level
   - LSHL-0625 offscale low
   - Personnel report of cavitation OR low discharge pressure for SFP (1/2-1213-P6-002 OR -005) AND/ OR RHR (1/2-1205-P6-001 OR -002) Pumps

   AND

   b. UNPLANNED VALID Direct Area Radiation Monitor readings rise on any of the following:
   - RE-0008 in the fuel building
   - RE-0002, -0003, -0004 in containment
   - RE-0011 at the seal table
   - RE-0005, -0006 in containment

OR

2. UNPLANNED VALID Direct Area Radiation Monitor readings rise by a factor of 1000 over normal* levels.
   - RE-0001 Control Room
   - RE-0002 Cnmt Bldg Low Range
   - RE-0003 Cnmt Bldg Low Range
   - RE-0004 Cnmt Bldg Per. Hatch
   - RE-0005 Cnmt Bldg High Range
   - RE-0006 Cnmt Bldg High Range
   - RE-0007B Sampling Room
   - RE-0008 FHB
   - RE-0011 Cnmt Bldg Incore Instr
   - ARE-16971 RPF North Wall General Area
   - ARE-16972 RPF East Wall General Area
   - ARE-16973 RPF Dress Out Area

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

AC / DC POWER

CU3 - Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes. (pg. 78)

1. a. Loss of power to **OR** from transformers 1(2)NXRA **AND** 1(2)NXRB resulting in loss of all off-site electrical power to **BOTH** 1(2)AA02 **AND** 1(2)BA03 for greater than 15 minutes
   AND
   b. One emergency diesel generators supplying power to **EITHER** 1(2)AA02 **OR** 1(2)BA03.

CU7 - UNPLANNED Loss of Required DC Power for Greater Than 15 Minutes. **Modes 5 and 6 Only.** (pg. 82)

1. a. UNPLANNED loss of Vital DC power to 125 VDC Buses AD1, BD1, CD1, AND DD1 indicated by bus voltage indications less than 105 VDC
   AND
   b. Failure to restore power to at least one DC bus within 15 minutes from the time of loss.

RX and CORE

CU8 - Inadvertent Criticality **Modes 5 and 6 Only.** (pg. 83)

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

CU1 - RCS Leakage **Mode 5 Only.** (pg. 76)

1. Unable to establish or maintain pressurizer level greater than 17%.

CU2 - Unplanned Loss of RCS Inventory with Irradiated Fuel in the RPV **Mode 6 Only** (pg. 77)

1. UNPLANNED RCS level lowering below 194' (RPV flange) for greater than 15 minutes

OR

2. a. RPV level **CANNOT** be monitored
   AND
   b. A possible loss of RPV inventory may be occurring as indicated by unexplained level rise in Containment sump, Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

HEAT REMOVAL

CU4 - Unplanned Loss of Decay Heat Removal Capability with Irradiated Fuel in the RPV. **Modes 5 and 6 Only.** (pg. 79)

NOTE The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the duration has or will likely exceed 15 minutes.

1. An UNPLANNED event results in RCS temperature exceeding 200°F.

OR

2. Loss of all RCS temperature **AND** RPV level indication for greater than 15 minutes.

COMMUNICATIONS

CU6 - UNPLANNED Loss of All Onsite **OR** Offsite Communications Capabilities (pg. 81)

1. UNPLANNED loss of ALL of the following on-site communications capability affecting the ability to perform routine operations:
   - In plant telephones
   - Public address system
   - Plant radio systems

OR

2. UNPLANNED loss of ALL of the following off-site communications capability:
   - ENN (Emergency Notification Network)
   - ENS (Emergency Notification System)
   - Commercial phones (Radio, PBX, Satellite, Wireless)
   - VOIP (Voice Over Internet Protocol)
   - OPX (Off Premise Extension)

NATURAL / DESTRUCTIVE

HU1 - Natural and Destructive Phenomena Affecting the PROTECTED AREA (pg. 97)

1. Seismic monitoring system indicates seismic event OR plant operators report an earthquake was "felt".

OR

2. Report by plant personnel of tornado **OR** high winds greater than 100 mph striking within PROTECTED AREA.

OR

3. Crash of vehicle, large enough to cause significant damage, into plant structures containing functions or systems required for safe shutdown within the PROTECTED AREA boundary.

OR

4. Report by plant personnel of an unanticipated EXPLOSION within the PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure **OR** equipment.

OR

5. Report of turbine failure resulting in casing penetration **OR** damage to turbine **OR** generator seals.

OR

7. Sustained hurricane force winds greater than 73 mph forecast to be at the plant site in the next four hours in accordance with 11889-C, Severe Weather Checklist.

FIRE / EXPLOSION

HU2 - FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection (pg. 99)

1. FIRE in buildings **OR** areas contiguous to any of the following areas **NOT** extinguished within 15 minutes of control room notification **OR** verification of a control room alarm unless disproved by personnel observation:
   - Containment Building
   - Fuel Hand Building
   - NSCW Cooling Towers
   - Control Building
   - Diesel Generator Building
   - Auxiliary Building
   - Diesel Fuel Oil Storage Tank Pumphouse
   - Auxiliary Feedwater Pumphouse

TOXIC / FLAMMABLE

HU3 - Release of Toxic, Asphyxiant, or Flammable Gases Deemed Detrimental to Normal Operation of the Plant (pg. 101)

1. Report **OR** detection of toxic, asphyxiant, **OR** flammable gas that has **OR** could enter the Owner Controlled Area in amounts greater than life threatening **OR** flammable concentrations that can affect NORMAL PLANT OPERATIONS.

OR

2. Report by Local, County, **OR** State Officials for evacuation **OR** sheltering of site personnel based on an offsite toxic, asphyxiant, **OR** flammable gas event.

SECURITY

HU4 - Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant (pg. 102)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by security shift supervision.

OR

2. A CREDIBLE VEGP security THREAT notification.

OR

3. A validated notification from NRC providing information of an aircraft threat.

ED DISCRETION

HU5 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE (pg. 104)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

4.3.3 The ENS functions DO NOT normally transfer to the EOF. The EOF has an ENS communicator position that coordinates with the TSC ENS communicator to maintain a continuous open communication line with the NRC. The role of the EOF ENS Communicator is to assist in communications with the NRCOC relevant to activities performed from the EOF (i.e., offsite interface, public information, PAR development, Dose assessment activities, etc.).

5.0 PROCEDURE

5.1 Precautions and Limitations

5.1.1 Notifications of applicable State and Local Agencies shall be accomplished as soon as practicable and within 15 minutes of the declaration of an emergency, an upgrade to a higher emergency classification level, or the approval of protective actions recommendations.

5.1.2 Electronic notification using the electronic Emergency Notification Form (ENF) in WebEOC is the preferred method of notification of State and Local Agencies. Should WebEOC be unavailable the back-up notification method is completion of a hard copy ENF and reading the form to the applicable State and Local Agencies via the ENN. Whether using the electronic or back-up ENF the emergency notification to an agency is considered complete when the agency verbally confirms receipt of the message via the ENN. To expedite availability of WebEOC in an emergency, the crew members responsible for completing the ENF and making electronic notifications should login to WebEOC as soon as possible and remain logged-in.

5.1.3 Notification of the NRC shall be completed immediately following notifications to the state and county agencies and within an hour of the declaration of an emergency. Follow-up notifications of the NRC shall be made promptly after any further degradation in the plant conditions, any change from one emergency class to another, or for the termination of an emergency. NRC notifications are typically performed utilizing the Federal Telephone System (FTS). The Emergency Notification System (ENS) line is normally utilized. An open line is maintained for the duration of the event at the request of the NRC communicator receiving the initial notification.

5.1.4 For security based emergencies, notifications to the NRC should be performed within 15 minutes of discovery of an imminent threat or attack against the plant to ensure proper mobilization of federal resources.

CAUTION

An initial notification of an upgrade in emergency Classification should take precedence over a follow-up message of a lower ranking emergency. (i.e., an initial site area emergency notification takes precedence over an alert follow-up notification.)

5.1.5 If the plant condition degrades and a higher emergency classification is declared before the notifications are confirmed for the lesser emergency declaration, then a notification reflecting the higher emergency classification should be made. This notification should be made within 15 minutes of the lesser emergency declaration. This should be performed IF the notification can be made within 15 minutes of the lesser (first) classification.

1. 026AA2.06 001/LOIT AND LOCT/SRO/C/A 2.8/3.1/026AA2.06/LO-TA-60005///

At time 1000:

- Unit 1 is at 100% reactor power.

At time 1005:

- ALB04-A03 ACCW RCP 1 CLR LOW FLOW is received.
- ALB04-A04 ACCW RCP 1 CLR OUTLET HI TEMP is received.
- 18022-C, "Loss of Auxiliary Component Cooling Water," is entered.

At time 1007:

- RCP #1 seal water inlet temperature is 220°F and rising at 1°F per minute.
- RCP #1 motor stator winding temperature is 307°F and rising at 1°F per minute.

Which one of the following completes the following statement?

To prevent damage to the RCP, the RCP must be stopped **no later** than time __(1)__,

and

after the reactor is tripped, the Shift Supervisor directs the OATC to stop the affected RCP per __(2)__ direction.

A✓ (1) 1012

(2) 18022-C, "Loss of Auxiliary Component Cooling Water"

B. (1) 1012

(2) 19000-C, "Reactor Trip or Safety Injection"

C. (1) 1016

(2) 18022-C, "Loss of Auxiliary Component Cooling Water"

D. (1) 1016

(2) 19000-C, "Reactor Trip or Safety Injection"

K/A

026 Loss of Component Cooling Water

AA2.06 Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water:

- The length of time after the loss of CCW flow to a component

before that component may be damaged.

K/A MATCH ANALYSIS

To answer this question, the applicant must know that loss of ACCW will lead to entry into 19000-C, requiring a manual trip (even though it does not approach or exceed any automatic setpoints), and which procedure provides the action. Since the Shift Supervisor is required to recall the specific strategy from the 18022-C procedure, the choices involve SRO only knowledge of the procedure flow path of the AOP (past the entry conditions and there are no immediate actions in 18022-C), and knowledge of the need to enter 19000-C. In addition, the question requires the candidate to recall not only how long the RCPs can operate without ACCW cooling but to evaluate other parameters that may require immediate stopping of the pumps.

EXPLANATION OF REQUIRED KNOWLEDGE

At 1005, annunciators ALB04-A03 and A04 indicate a complete loss of ACCW flow to RCP #1 only. AOP 18022-C is entered. Step 6 checks to see if the RCP should be stopped. At 1016, RCP #1 would have sustained a total loss of ACCW for >10 minutes and must be stopped. The stem also states that seal water inlet temperature is rising at a rate of 1F/min. At 1018, the RCP must be stopped due to seal water inlet temperature. Additionally, motor stator winding temperature is also rising at a rate f 1F/min. At 1012, 311F would have been exceeded and the RCP must be stopped. Therefore, the most limiting operationg condition would be the stator winding temperature and the the RCP must be stopped no later than 1012. The RCP operationg parameters in 18022-C are the same as those listed in SOP 13003-1 Limitation 2.2.10.

Since the reactor is greater than 15% RTP, the reactor must be tripped prior to tripping the RCP. Tripping the reactor is a direct entry condition into EOP 19000-C. On step 11 of 19000-C, a check of ACCW status is made and the RNO directs stopping the RCP. However, 18022 step 6 is specifically design to trip the reactor and stop the RCP prior to performing EOP 19000-C. This is done to ensure the RCP is not damaged in the time required to perform the initial steps of 19000-C before direction to stop the RCP is encountered.

ANSWER / DISTRACTOR ANALYSIS

A. Correct.

Part 1 is correct. Based of Westinghouse vendor requirements the RCPs must be stopped under normal conditions when ACCW cooling is lost to the motors to prevent potential damage. This is supported by SOP 13003-1, which establishes the ACCW operating time limit of 10 minutes and other trip parameters. In the case presented to the candidate the most limiting parameter is the stator limit of 311F which will require the pump to be stopped within 5 minutes of 10:07 (i.e. 10:12).

Part 2 is correct because 18022-C directs manual reactor trip, verify trip, the stop RCPs then perform 19000-C, therefore the correct direction to stop the RCP is 18022-C as opposed to 19000-C.

B. Incorrect. Plausible. Part 1 is correct. See Part 1 of choice A above.

Part 2 is incorrect but 'plausible' because the candidate may determine that the direction that is provided in 19000-C step 11 would be used to stop the RCPs as opposed to 18022-C step 1. The 10 minute time allowance could lead the candidate to determine there is no real pressure to speed up the action to stop the RCPs, prior to the initial steps in 19000-C.

C. Incorrect. Plausible. Part 1 is incorrect however is 'plausible' since the candidate may determine that none of the operating limits provided in the stem is above limits or will reach their limit within 10 minutes, therefore the 10 minutes becomes the most limiting factor.

Part 2 is correct. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice C above.

Part 2 is correct. See Part 2 of choice B above.

<u>SRO JUSTIFICATION</u>

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the system parameter needed to determine if the RCP should be stopped is system level knowledge. However, the direction to stop the RCP requires specific knowledge and priortization of steps in both the AOP and EOP.
-Can the question be answered *solely* by knowing immediate operator actions? No, there are no IOAs associated with the actions addressed in this question.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, stopping an RCP has nothing to do with the entry conditions for either AOP 18022-C or EOP 19000-C. 19000-C does have entry conditions associated with trip of the reactor, which would stem from stopping an RCP at 100% RTP. However, this would lead a candidate down the incorrect path.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, specific knowledge of individual procedure steps is required.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- **Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency**

contingency procedures **Yes, knowledge of specific diagnostics in step 6 of AOP 18022-C is required in contrast to step 11 of EOP 19000-C.**
• Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Level:	SRO
Tier # / Group #	T1 / G1
K/A#	026AA2.06
Importance Rating:	2.8 / 3.1

Technical Reference: EOP 19000-C Rev 37.1, page 21
AOP 18022-C Rev 15.2, page 6
SOP 13003-1, Rev 47.1, page 7

References provided: None

Learning Objective:
LO-TA-60005 Respond to a Loss of ACCW per 18022-C
LO-LP-60318-05 Describe the operator actions required during a loss of ACCW with the plant in operation and the RCP temperature time limits are exceeded.
LO-PP-16401-07 List the RCP components that are cooled by the ACCW system.

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 43.5

Comments:

You have completed the test!

OATC INITIAL ACTIONS

Sheet 5 of 4

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. Check ACCW Pumps - AT LEAST ONE RUNNING.	11. Try to start one ACCW Pump. IF an ACCW Pump can NOT be started within 10 minutes of loss of ACCW, THEN stop all RCPs. IF an ACCW Pump can NOT be started within 30 minutes of loss of ACCW, THEN close ACCW Containment isolation valves: • ACCW SPLY HDR ORC ISO VLV HV-1979 • ACCW SPLY HDR IRC ISO VLV HV-1978 • ACCW RTN HDR IRC ISO VLV HV-1974 • ACCW RTN HDR ORC ISO VLV HV-1975
12. Adjust Seal Injection flow to all RCPs 8 TO 13 GPM.	
13. Dispatch Operator to ensure one train of SPENT FUEL POOL COOLING in service per 13719, SPENT FUEL POOL COOLING AND PURIFICATION SYSTEM.	13. IF one train of SFP COOLING can NOT be restored to service, THEN initiate 18030-C, LOSS OF SPENT FUEL POOL LEVEL OR COOLING.

° END OF SUB-PROCEDURE TEXT

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***6. Check if RCPs should be stopped:**	
a. Check the following RCP parameters (using plant computer):	a. Perform the following:
	1) IF any parameter limit is exceeded, THEN perform Step 6.b.
• Motor bearing (upper or lower radial or thrust) - GREATER THAN 195°F.	2) Go to Step 7.
• Motor stator winding - GREATER THAN 311°F.	
• Seal water inlet - GREATER THAN 230°F.	
• Loss of ACCW - GREATER THAN 10 MINUTES.	
b. Perform the following:	
1) Trip the reactor.	
2) WHEN Reactor is verified tripped, THEN stop affected RCP(s).	
3) Initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.	

INITIALS

2.2.8 The following starting duty cycle for the RCP should be observed: _________

- Only one RCP shall be started at any one time.
- Two successive starts are permitted, provided the motor is permitted to coast to a stop between starts.
- A third start may be made when the winding and core have cooled by running for a period of 20 minutes, or by standing idle for a period of 45 minutes.

2.2.9 During RCS filling and venting, RCS pressure must be greater than 325 psig prior to starting an RCP to verify adequate seal D/P is maintained throughout RCS fill and vent. If necessary, the RCP should be stopped prior to seal D/P dropping less than 200 psid. If the seal D/P goes below 200 psid during pump operation or coast down, the RCP should be evaluated before restarting the RCP. _________

2.2.10 An RCP shall be stopped IF any of the following conditions exist: _________

- Motor bearing temperature exceeds 195°F.
- Motor stator winding temperature exceeds 311°F.
- Seal water inlet temperature exceeds 230°F
- Total loss of ACCW for a duration of 10 minutes.
- RCP shaft vibration of 20 mils or greater.
- RCP frame vibration of 5 mils or greater.
- Differential pressure across the number 1 seal of less than 200 psid.

2.2.11 If a loss of RCP seal cooling (Seal Injection and/or ACCW to Thermal barrier) occurs, resulting in RCP shutdown due to exceeding operating limits, then the unit should be cooled down to Mode 5 to facilitate recovery. Upon reaching Mode 5, ACCW to the Thermal barrier should be restored. Seal injection should then be returned to service. This sequence should prevent seal damage, RCP shaft bowing, ACCW System damage, etc. due to excessive thermal stresses. _________

1. 028AG2.4.8 001/LOIT AND LOCT/SRO/C/A 3.8/4.5/028AG2.4.8/LO-TA-37021///

Initial condition:

- Unit 1 reactor trip and SI occurred.

Current conditions:

- 19011-C, "SI Termination," Step 21, is in progress to evaluate if a bubble exists in the pressurizer.
- Controlling pressurizer level channel, 1LT-459, fails low.
- Actual pressurizer level is 92% and slowly rising.

Which one of the following completes the following statement?

Per 10020-C, "EOP and AOP Rules of Usage," the Shift Supervisor __(1)__ direct the use of 18001-C, "Systems Instrumentation Malfunction," guidance to restore pressurizer heaters to service while performing 19011-C actions,

and

in response to actual pressurizer level, the Shift Supervisor __(2)__ required to transition to 19261-C, "Response to High Pressurizer Level."

	__(1)__	__(2)__
A.✓	may	is NOT
B.	may	is
C.	may NOT	is NOT
D.	may NOT	is

K/A

028 **Pressurizer Level Malfunction**

G2.4.8 **Knowledge of how abnormal operating procedures are used in conjunction with EOPs.**

K/A MATCH ANALYSIS

The question is requires the candidate to determine if AOP 18001-C can be used in conjunction with EOP 19011-C to mitigate complications arising from a failed pressurizer level instrument. The question is elevated to the SRO level by requiring the candidate to make decision on whether to direct implementation of procedures during EOP implementation.

EXPLANATION OF REQUIRED KNOWLEDGE

Per EOP and AOP rules of usage 10020-C step 3.5.9, other procedures such as AOPs and ARPs may be performed in parallel with EOPs as long as their actions do not conflict with the EOP steps. EOP actions take priority. In both steps 21 and 26 of EOP 19011-C, pressurizer heaters are required to be energized to saturate the pressurizer. With pressurizer level at 92% and rising, this becomes a crucial action, however the step cannot be accomplished due to heaters being tripped off as a result of LT-459 failing low. The use of 18001-C to select away from the failed channel and restore pressurizer heaters is a prudent and necessary action.

With pressurizer level at 92% and rising, a yellow path on the CSFST for INVENTORY should exist. Per 19200-C step 1 RNO, if a yellow path exist, then initiate FRP based on plant conditions with Shift Supervisor approval. Per step 11 RNO, transition to FRP 19261-C should only be made if solid plant condtions exist, which are not currently present. Therefore, 19261-C is not required to be entered and should not be entered at this time. Instead, the priority should be placed on restoring pressurizer heaters and continuing efforts to lower pressurizer level via charging/letdown mismatch.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Part 1 is correct. Per 10020-C 'EOP and AOP Rules of Usage', Other procedures such as AOPs or ARPs may be performed in parallel with EOPs as long as their actions do not conflict with the EOP steps. EOP actions take priority. Additionally, performance of 18001-C to restore pressurizer heaters is prudent and necessary.

Part 2 is correct. 19200-C 'Critical Safety Function Status Trees', states IF a Yellow condition exists, THEN **initiate** FRP after evaluating plant conditions with Shift Supervisor's approval. 19011-C "SI Termination' states IF solid plant conditions are present, THEN **refer** to 19261 C, FR I.1 Response to High Pressurizer Level. Solid plant conditions are not present. Entry into 19261-C would actually impede the success path in 19011-C.

B. Incorrect. Plausible. Part 1 is correct. See Part 1 of choice A above.

Part 2 is incorrect but 'plausible' because the candidates know that EOP actions always take priority. The candidate may not realize 19261-C does not contain any steps that would be different from those already in progress in 19011-C and believe transition to 19261-C is necessary to prevent going solid in the pressurizer.

C. Incorrect. Plausible. Part 1 is incorrect but plausable because 10020-C 'EOP and AOP Rules of Usage', Other procedures such as AOPs or ARPs may be performed in parallel with EOPs as long as their actions do not conflict with the EOP steps. **EOP actions take priority.**

The candidate may not realize that the sucess path for current plant conditions resides in restoring pressuirzer heaters and deems use of 18001-C as unnecessarily slowing the progress of the EOP 19011-C.

Part 2 is correct. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice C above.

Part 2 is incorrect. See Part 2 of choice B above.

SRO JUSTIFICATION

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the question revolves around transition decisions and does not involve system knowledge.
-Can the question be answered *solely* by knowing immediate operator actions? No, the question does not address any IOAs.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, specific knowledge of procedure steps contained within the EOP and AOPs is needed in addition to entry conditions.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, specific knowledge of procedure steps contained within the EOP and AOPs is needed in addition to overall knowledge of the associated procedures.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- **Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures Yes, the question requires specific knowledge of the RNO of step 21 as well as the specifics of both 18001-C and 19261-C to determine if transition to these procedures would mitigate the challenges currently observed.**
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Level:	SRO	
Tier # / Group #	T1 / G2	
K/A#	028G2.4.8	
Importance Rating:	3.8 / 4.5	
Technical Reference:	EOP 10020-C, Rev 9.0, page 10 EOP 19011-C, Rev 29.2, page 12 & 18 EOP 19200-C, Rev 24.2, page 2, 3, & 10	
References provided:	None	
Learning Objective:	LO-TA-37021	Respond to High Pressurizer Level per 19261-C
	LO-TA-60030	Respond to a Failure of Pressurizer Level Instrumentation per 18001-C
	LO-TA-37005	Terminate Safety Injection per 19000-C or 19011-C
	LO-TA-05003	Respond to CSFST Trouble alarm and evaluate CSFSTs using SPDS and the PSMS per 17006-1/2, 13521-1/2, 13505-1/2, and 19200-C
	LO-LP-60301-12	Given that the pressurizer level control selector switch is in the NORMAL position (459/460), describe how and why the plant will respond to the following instrument failures. Consider each separately and include effects on pressurizer level control, alarms, RPS, and ESF actuations. b. 459 fails low
	LO-LP-37002-09	Using EOP 19200, as a guide, briefly describe how the steps are accomplished.
Question origin:	NEW	
Cognitive Level:	C/A	
10 CFR Part 55 Content:	43.5	
Comments:		

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

20. Align CCP suction to VCT:

    a. Open VCT OUTLET ISOLATION valves:

    - LV-0112B
    - LV-0112C

    b. Close RWST TO CCP A&B SUCTION valves:

    - LV-0112D
    - LV-0112E

*21. **Control PRZR pressure:**

a. Check Stub Busses - ENERGIZED:

- NB01
- NB10

RNO a. Energize Stub Busses by performing the following as necessary:

NB01	NB10
1) Open breaker NB01-01 2) Close breaker AA02-22 3) Close breaker NB01-01	1) Open breaker NB10-01 2) Close breaker BA03-18 3) Close breaker NB10-01

b. Check for a bubble in the PRZR to enhance RCS pressure control.

RNO b. Control charging and letdown flows to avoid sudden pressure changes.

Energize PRZR Heaters.

IF solid plant conditions are present,
THEN refer to 19261-C, FR-I.1 RESPONSE TO HIGH PRESSURIZER LEVEL.

Go to Step 22.

Step 21 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
26. Check RCP status:	
a. RCPs - ALL STOPPED.	a. Go to Step 27.
b. Check RVLIS full range indication - GREATER THAN 94%.	b. Perform the following: • Raise PRZR level greater than 90% [90% ADVERSE]. • Raise RCS Subcooling based on core exit TCs greater than 60°F [74°F ADVERSE]. • Use PRZR Heaters, as necessary to saturate the Pressurizer water.
c. Start an RCP using ATTACHMENT A. (RCP 4 or RCP 1 preferred)	c. IF an RCP can NOT be started, THEN verify natural circulation using ATTACHMENT B. IF natural circulation NOT established, THEN raise rate of dumping steam using Steam Dumps. After natural circulation is verified, maintain rate of dumping steam.

With LT-459 failed low, heaters cannot be energized. 18001-C would be used to select away from failed channel and energize heaters.

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

NOTES

- If SPDS display of the Plant Computer is not operable or questionable, manual monitoring of CSFSTs should be performed by a licensed operator.
- CSFSTs should be monitored continuously if a RED or ORANGE condition is present or each 10 to 15 minutes if the highest priority CSFST is no higher than YELLOW.
- CSFSTs should be checked in order listed.
- Priority of operator action is given by the following:
  - Red (Solid) Path - Extreme challenge, in Tree Order per Step 1.
  - Orange (Dashed) Path - Severe challenge, in the Tree Order per Step 1.
  - Yellow (Dotted) Path - Not satisfied, in Tree Order per Step 1.
  - Green (Outlined) Path - Satisfied.
- If using the Plant Computer (if available) to monitor CSFSTs:
  - The mode indication of the Plant Computer CSFSTs should be indicating zero.
  - RCP breakers should be opened for RCPs NOT running in order to provide proper RVLIS indication.
- If SPDS is operable, CSFSTs may be checked by scanning the display console for alarm conditions.
- Color status of CSFSTs will also be indicated by letter R for red, O for orange, Y for yellow, G for green, and M for magenta.
- CSFSTs will indicate active (alarming) paths as solid lines and non-active paths as empty or hollow lines.

1. Check CSFSTs- SATISFIED:

   a. Subcriticality (F-0.1)

   b. Core Cooling (F-0.2)

1. IF a Red condition exists, THEN immediately go to FRP.

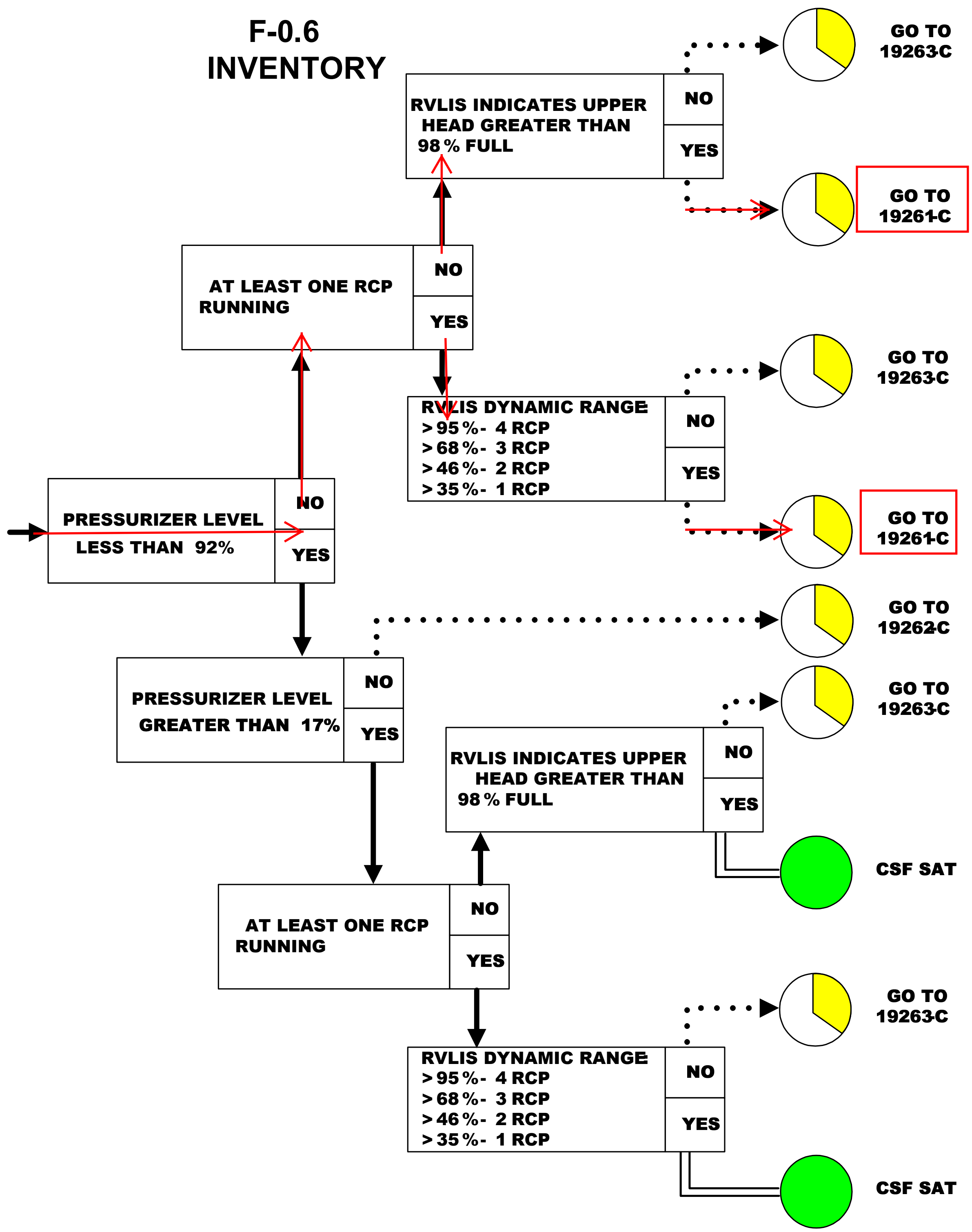
   IF an Orange condition exists, THEN go to FRP after completion of present pass thru CSFSTs.

° Step 1 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. Heat Sink (F-0.3)	IF a Yellow condition exists, THEN initiate FRP after evaluating plant conditions with Shift Supervisor's approval.
d. Integrity (F-0.4)	
e. Containment (F-0.5)	
f. Inventory (F-0.6)	
2. Report change of status of any CSFST to the Shift Supervisor, if necessary (i.e., change in status not understood).	
3. Check EOP usage - NO LONGER REQUIRED.	3. Return to Step 1.
4. Monitoring of CSFSTs is no longer required.	

END OF PROCEDURE TEXT

F-0.6 INVENTORY



3.5.8 ES-0.0, REDIAGNOSIS, may be entered any time based on operator judgement and may be entered as follows:

3.5.8.1 ES-0.0 may be entered when there is doubt in being in correct EOP.

3.5.8.2 Safety injection is in service or is required.

3.5.8.3 E-O, REACTOR TRIP OR SAFETY INJECTION, has been executed and a transition has been made to another EOP which is bounded by the ORGs only (not FRGs).

3.5.9 Other procedures such as AOPs or ARPs may be performed in parallel with EOPs as long as their actions do not conflict with the EOP steps. EOP actions take priority.

3.6 STEP PLACE-KEEPING

3.6.1 When exiting an EOP step it is necessary to track what procedure and step was exited such that when directed to "return to procedure step in affect", the correct procedure step may be re-entered. A red ribbon page marker has been provided in the Simulator and Main Control Room EOP sets for assistance in tracking such transitions.

3.6.2 Step by step place-keeping is a valuable human performance tool. It shall be performed in accordance with plant standard and management expectations.

3.7 NOTES AND CAUTIONS

All NOTES and CAUTIONS shall be reviewed by the Shift Supervisor. Those NOTES or CAUTIONS that are pertinent to the evolution in progress shall be read aloud to the operating crew.

3.8 MODES OF APPLICABILITY

19000-C	E-0	1,2,3	Assumes RHR system not in service and SI operable
19001-C	ES-01	1,2	Assumes trip from power
19002-C	ES-0.2	1,2.3	Assumes No-load conditions
19003-C	ES-0.3	1,2,3	Assumes No-load conditions
19004-C	ES-0.4	1,2,3	Assumes No-load conditions

1. 032AA2.03 001/LOIT AND LOCT/SRO/C/A 2.8/3.1/032AA2.03/LO-LP-39213-04///

Given the following conditions:

- Unit 1 is in Mode 6.
- Source Range N31/32 each indicate ~10 cpm.
- Source Range N31 is powered from 1AY1A.
- Source Range N32 is powered from 1NLP39.

Which one of the following completes the following statement?

Tech Spec LCO 3.9.3, "Nuclear Instrumentation," __(1)__ met,

If the Source Range N32 detector high voltage power supply breaker were to trip on overcurrent, the OATC would observe the QMCB N32 meter indicating __(2)__.

	__(1)__	__(2)__
A.	is	as-is
B.✓	is	bottom of scale
C.	is NOT	as-is
D.	is NOT	bottom of scale

K/A

032 **Loss of Source Range NI**

AA2.03 **Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation:**

- Expected values of source range indication when high voltage is automatically removed

K/A MATCH ANALYSIS

The question tests the candidate's ability to determine the indication of the Source Range NIs following the trip of the high voltage power supply breaker in Mode 6 with 10CPM indicated.

(Note: At Vogtle, Source Range NIs high voltage power supplies are no longer automatically removed - GAMAMETRICS now installed.)

EXPLANATION OF REQUIRED KNOWLEDGE

The candidate is required to know TS 3.9.3 Bases to determine the Operability of the Source Range NIs. In Mode 6, one SR NI can be powered from a Non-1E power supply provided the other SR NI is powered from it's normal 1E power supply. Additionally, the candidate is required to determine the SR NI indication following a loss of the high voltage power supply. In this situation, the NI detector would be down powered and the indication would drop to bottom of scale.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. When any of the safety-related busses supplying power to one of the detectors (NI-0031 or NI-0032) associated with the source range neutron flux monitors are taken out of service, the corresponding source range neutron flux monitor may be considered OPERABLE when its detector is powered from a temporary nonsafety-related power source, provided the detector for the opposite source range neutron flux monitor is powered from its normal source.

The second part is incorrect. Loss of the high voltage power supply will constitute a complete loss of power to the SR NI detectors, resulting in indication failing to bottom of scale. However, the NI's have 3 different power supplies that support various functions. If a loss of control power only occurs, all bistables trip, however NI indication is unchanged and remains "as-is". Therefore, this distractor is plausible.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. The loss of the High Voltage Power Supply will result in a complete loss of power to the SR NIs and indication will read bottom of scale.

C. Incorrect. Plausible. The first part is incorrect. Per TS 3.9.3 Bases, one SR NI can be powered from a Non-1E power supply provided the other SR NI is powered from it's normal 1E power supply in Mode 6. It is abnormal for a Safety-Related, Tech Spec required component to be powered from a non-1E power supply and be considered OPERABLE. It is reasonable for a candidate without knowledge of this specific exception to determine the LCO not met. Therefore, this distractor is plausible.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

SRO JUSTIFICATION (10CFR43(b))

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the knowledge required is not included in any TS or TRM action.

-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" No, the knowledge required does not exist above the line in any TS or TRM.

-Can question be answered *solely* by knowing the TS Safety Limits? No, SR NIs are not discussed in the TS Safety Limits.

-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- **Knowledge of TS bases that is required to analyze TS required actions and terminology. Yes, specific knowledge of the Mode 6 power supply alignments listed in TS Bases 3.9.3 are required to determine if the LCO is met.**

Level:	SRO
Tier # / Group #	T1 / G2
K/A#	032AA2.03
Importance Rating:	2.8 / 3.1
Technical Reference:	17010-1 Rev 50 , page 60 Tech Spec Bases 3.9.3 Rev 3-4/09, page B3.9.3-1&2
References provided:	None
Learning Objective:	LO-LP-60302-05 Describe how and why a reactor startup would be affected by a source range instrument failure when the reactor is at the following power levels: above and below P-6. LO-PP-17201-01 Discuss the operation of the Source & Intermediate Range Detectors to include: a. Type of detector b. Gamma compensation c. When they are used g. Power supplies (also including the effects on loss of instrument or control power) LO-LP-39213-04 Describe the bases for any given Tech Spec in section 3.9.
Question origin:	NEW
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.7 / 43.2
Comments:	

You have completed the test!

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND	The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors (NI-0031 and NI-0032) are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. Temporary neutron flux detectors which provide equivalent indication may be utilized in place of installed instrumentation. The installed source range neutron flux monitors are fission chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers seven decades of neutron flux (1E-1 cps to 1E +6 cps) with a 2% instrument accuracy. The detectors also provide continuous visual indication in the control room. The NIS is designed in accordance with the criteria presented in Reference 1.
APPLICABLE SAFETY ANALYSES	Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as an improperly loaded fuel assembly. The need for a safety analysis for an uncontrolled boron dilution accident is minimized by isolating all unborated water sources except as provided for by LCO 3.9.2, "Unborated Water Source Isolation Valves." The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).
LCO	This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE each monitor must provide visual indication. When any of the safety-related busses supplying power to one of the detectors (NI-0031 or NI-0032) associated with the source range neutron flux monitors are taken out of service, the corresponding source range neutron flux monitor may be considered OPERABLE when its detector is powered from a temporary nonsafety-related

(continued)

BASES

LCO (continued)

source of power, provided the detector for the opposite source range neutron flux monitor is powered from its normal source.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, the operability requirements for the installed source range detectors and circuitry are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

B.1

Condition B is modified by a Note to clarify the requirement that entry into or continued operation in accordance with Condition A is required for any entry into Condition B. The Note reinforces conventions of LCO applicability as stated in LCO 3.0.2 and as reflected in examples in 1.3, Completion Times.

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be

(continued)

ALB 10

	(1)	(2)	(3)	(4)	(5)	(6)
A	SR/IR SIG PROCESSOR TROUBLE	NIS SOURCE AND INTMD RANGE TRIP BYPASS	POWER RANGE HI NEUTRON FLX HI SETPOINT ALERT	REACTOR BYPASS BRKR BYA IN-OPERATE	REACTOR BYPASS BRKR BYA CLOSE	ROD CONTROL NON URGENT FAILURE
B	SOURCE RNG HI SHUTDOWN FLUX ALARM BLOCKED		POWER RANGE HI NEUTRON FLX LOW SETPOINT	REACTOR BYPASS BRKR BYB IN-OPERATE	REACTOR BYPASS BRKR BYB CLOSE	ROD CONTROL URGENT FAILURE
C	SOURCE RANGE HI FLUX LEVEL AT SHUTDOWN	POWER RANGE CHANNEL DEVIATION	OVERPOWER ΔT ROD BLOCK AND RUNBACK ALERT	ROD BANK LO LIMIT	RPI NON URGENT ALARM	NIS CHANNEL ON TEST
D	INTMD RANGE HI FLUX LEVEL ROD STOP	PWR RANGE UP DET HI FLX DEV	OVERPOWER ROD STOP	ROD BANK LO-LO LIMIT	RPI URGENT ALARM	ROD DEV
E	SR/IR REMOTE SIG PROCESSOR DPU-B TROUBLE	PWR RANGE LWR DET HI FLX DEV	OVERTEMP ΔT ROD BLOCK AND RUNBACK ALERT		ROD AT BOTTOM	RADIAL TILT
F	SR/IR AMPLIFIER TROUBLE	POWER RANGE HI NEUTRON FLX RATE ALERT		ROD DRIVE M-G SET TROUBLE	TWO OR MORE RODS AT BOTTOM	DELTA FLUX DEVIATION

WINDOW F01

ORIGIN	SETPOINT	
NC-35M NC-36M	Not Applicable	SR/IR AMPLIFIER TROUBLE

1.0 **PROBABLE CAUSE**

1. High Voltage Power Supply greater than 875V or less than 660V.
2. Loss or degraded +15v Power Supply in WR amplifier (or isolator for N36)
3. Loss or degraded –15v Power Supply in WR amplifier (or isolator for N36)
4. Degraded +5V Power Supply(s) in Isolator Assembly (N36 only)

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

Go to 18002-C, "Nuclear Instrumentation System Malfunction".

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCE: 1X6AS01-154

1. 033A2.03 001/LOIT AND LOCT/SRO/C/A 3.1/3.5/033A2.03/LO-TA-25010///

Procedure 13719-1, "Spent Fuel Pool Cooling and Purification," sections as follows:

- Section 4.2.2, "SFP Makeup from the RWST through the SFP Purification Loop"
- Section 4.2.4, "SFP Makeup from the RMWST"

Initial conditions:

- Unit 1 is defueled.
- Transfer canal is drained for transfer cart inspection.
- Spent fuel shuffle is in progress in the FHB.

Current conditions:

- ALB05-E02 SPENT FUEL PIT LO LEVEL is received.
- Personnel in the FHB report SFP level is slowly lowering.
- 18030-C, "Loss of Spent Fuel Pool Level or Cooling," is entered.

Which one of the following completes the following statement?

To mitigate the consequences of the event, the Shift Supervisor is required to direct makeup to the SFP using 13719-1, Section __(1)__,

and

per the Bases of Tech Spec 3.7.15, "Fuel Storage Pool Water Level," maintaining the required minimum water level in the SFP __(2)__ ensure adequate iodine decontamination factors are met for a fuel handling accident.

	__(1)__	__(2)__
A.✓	4.2.2	does
B.	4.2.2	does NOT
C.	4.2.4	does
D.	4.2.4	does NOT

K/A

033 **Spent Fuel Pool Cooling**

A2.03 **Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal spent fuel pool water level or loss of water level**

K/A MATCH ANALYSIS

The question tests the candidate's ability to predict the impact of low spent fuel pool level by having to recall the bases for TS 3.7.15 level. The candidate is also required to mitigate the consequence of the event through the selection of a makeup source by selecting the appropriate procedure section to perform.

EXPLANATION OF REQUIRED KNOWLEDGE

Per ARP 17005-1, ALB05-E02 alarms at a Spent Fuel Pool Level of 217'-0". Reports from the field have verified and low and decreasing pool level. AOP 18030-C is entered to mitigate the event. Per step 6 of this procedure, makeup to the SFP per SOP 13719-1 is directed. The specific procedure section is not specified. The candidate must recognize that makeup due to leakage will be from the RWST and not the RMWST. This ensures SFP boron concentration will be maintained. This requirement is stipulated in SOP 13719-1 Precaution and Limitation 2.1.7 and in a CAUTION at the beginning of sections 4.2.3 and 4.2.4. These state that non-borated makeup is usually **only** allowed for normal evaporative level losses. Borated water sources are the preferred maekup source for abnormal or unexplained level losses. Objective LO-PP-25102-11 is utilized during LOIT to re-enforce the use of borated water sources only during leakage.

With SFP level less than the Tech Spec limit of 217'-0", less than 23 feet of water exist over the spent fuel stored in the racks. Per TS 3.7.15 Bases, the minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The bases also discusses TS 3.7.15 water level as providing shielding to minimize general area dose and provide shielding during spent fuel movement.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. Per SOP 13719-1 'Spent Fuel Pool Cooling and Purification System', makeup due to leakage is from a borated source.

The second part is correct. Per Technical Specification 3.7.15 'Fuel Storage Pool Water Level' bases, the minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

B. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. Per Technical Specification 3.7.15 'Fuel Storage Pool Water Level' bases, the minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. However, the bases also discusses the shielding function of the water, which is a more commonly known benefit. A candidate without adequate knowledge of the TS Bases, may conclude that iodine decontamination is not a factor at all for water level,

or may assume the normal minimum water level of 218'-0" required. Therefore, this distractor is plausible.

C. Incorrect. Plausible. The first part is incorrect. Per SOP 13719-1 'Spent Fuel Pool Cooling and Purification System', makeup due to leakage is from a borated source. However, normally SFP makeup is made from either Demin Water or the RMWST since the level change is due to evaporative loss and the boron is left in solution. A candidate with inadequate knowledge of the purpose behind using the different sources would find it reasonable to makeup using a normal source. Therefore, this distractor is plausible.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

SRO JUSTIFICATION (10CFR43(b))

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the question requires specific knowledge of the bases for TS 3.7.15. The immediate action of the associated RAS does not address the purpose of the level or how to restore it.

-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" The information above the line deals with the required level to be maintained only. It does not address the reason for the level.

-Can question be answered *solely* by knowing the TS Safety Limits? No, Spent Fuel Pool level is not a Safety Limit.

-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- **Knowledge of TS bases that is required to analyze TS required actions and terminology. Yes, the reason SFP level is maintained >23 ft above the fuel is only specified in the Bases for TS 3.7.15.**

Level:	SRO
Tier # / Group #	T2 / G2
K/A#	033A2.03
Importance Rating:	3.1 / 3.5
Technical Reference:	SOP 13719-1 Rev 55.2, pages 6, 19, & 20 ARP 17005-1 Rev 34.2, pages 43-45 Tech Spec 3.7.15 Amendment No. 158, page 3.7.15-1 Tech Spec Bases 3.7.15 Rev 1-10/01, page B 3.7.5.15-1
References provided:	None
Learning Objective:	LO-PP-25102-12 Describe the minimum allowable water level over spent fuel and the basis for this level. LO-PP-25102-11 Describe when the different sources of makeup to the spent fuel pool would be used. For evaporation, For leakage LO-TA-25010 Makeup to the SFP per 13719-1/2, 13903-C, and 18030-C Attachment C
Question origin:	NEW
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.10 / 43.2
Comments:	**Early submittal 401-9 response:** -Need to make sure Section 4.2.4 cannot be argued as a correct answer. 13719 and EOP caution say borated water "should" rather than "shall" be used. -Having Keff requirements as a basis for the SFP water level does not seem plausible. However, I think this can be solved by reframing the second question to state "per Bases of Tech Spec 3.7.15", maintain the required minimum water level in the SFP does/does not ensure adequate idodine decontamination factors are met. - JAT 12/19/13 (Editorial) The new question incorporates the above suggestion, and the first concern with the original question has a learning objective to reinforce the use of borated water sources. Need to make sure this is definitive enough as a "technical source." - JAT 2/4/2014

You have completed the test!

3.7 PLANT SYSTEMS

3.7.15 Fuel Storage Pool Water Level

LCO 3.7.15 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1	-------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------ Suspend movement of irradiated fuel assemblies in the fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	In accordance with the Surveillance Frequency Control Program

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Subsection 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Subsection 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Subsection 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

INITIALS

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 The SFPCPS should be operated as necessary to maintain the SFP temperature below the high temperature alarm setpoint of 130°F. ________

2.1.2 The differential pressure across the SFP Skimmer Filter should not exceed 20 psid. ________

2.1.3 The differential pressure across the SFP Purification Loop cartridge filter should not exceed 70 psid. ________

2.1.4 The purification flow through the SFP Demineralizer System should not exceed 120 gpm. ________

2.1.5 Thoroughly fill and vent all applicable SFPCPS components prior to returning them to service after maintenance. This minimizes system performance degradation due to gas entrainment. ________

2.1.6 Any time that water is being removed from the RWST for makeup to the SFP or when the Refueling Water Purification Pump is taking suction from the RWST, the RWST level shall be maintained above the applicable Technical Specification low limit. ________

2.1.7 Non-borated makeup is usually only allowed for normal evaporative level losses. Borated water sources are the preferred makeup source for abnormal or unexplained level losses. ________

2.1.8 IF in Modes 1-4 opening of 1-1204-U4-158 must NOT be performed, since this would result in RWST declared inoperable ________

INITIALS

CAUTION

SFP boron concentration should be checked following makeup to assure a minimum boron concentration of 2000 ppm. ☐

4.2.3 **SFP Makeup from the Demineralized Water System** (SNC16999) (SNC12987)

CAUTION

Non-borated makeup is only allowed for normal evaporative level losses. Abnormal or unexplained level losses should be compensated for by using only borated sources. ☐

4.2.3.1 **Open** SFP DEMIN WTR SPLY ISO, 1-1213-U4-055. (RA53) ________

CAUTIONS

When gravity filling from the RWST, the Spent Fuel Pool level must be monitored continuously to prevent overflowing the SFP.

- Spent fuel Pool lighting receptacles are at the 218'9'" level. ☐
- Spent fuel Pool HI Level Alarm setpoint is at 219'. ☐

4.2.3.2 **Monitor** Spent Fuel Pool level (see Figure 1). ________

4.2.3.3 WHEN the required water level is reached, **close** and **lock** SFP DEMIN WTR SPLY ISO, 1-1213-U4-055, (RA53); (IV REQUIRED) ________

4.2.3.4 **Request** Chemistry **sample** the Spent Fuel Pool to determine the boron concentration. ________

INITIALS

CAUTION

SFP boron concentration should be checked following makeup to assure a minimum boron concentration of 2000 ppm. ☐

4.2.4 SFP Makeup from RMWST (SNC16999) (SNC12987)

CAUTION

Non-borated makeup is usually only allowed for normal evaporative level losses. Borated water sources are the preferred makeup source for abnormal or unexplained level losses. ☐

4.2.4.1 **Open** SFP CLG RMWST ISOLATION VALVE, 1-1213-U4-054. (RA53) ______

CAUTIONS

When gravity filling from the RWST, the Spent Fuel Pool level must be monitored continuously to prevent overflowing the SFP.

- Spent fuel Pool lighting receptacles are at the 218'9'" level. ☐
- Spent fuel Pool HI Level Alarm setpoint is at 219'. ☐

4.2.4.2 **Monitor** Spent Fuel Pool level (see Figure 1). ______

4.2.4.3 WHEN required water level is reached, **close** SFP CLG RMWST ISOLATION VALVE, 1-1213-U4-054, (RA53); (IV REQUIRED) ______

4.2.4.4 **Request** Chemistry **sample** the Spent Fuel Pool to determine the boron concentration. ______

WINDOW E02

ORIGIN	SETPOINT	
1-LSHL-625	217 feet elevation	SPENT FUEL PIT LO LEVEL

1.0 **PROBABLE CAUSE**

1. Insufficient inventory during filling or refueling operation.
2. Normal evaporation.
3. System leak.
4. Loss of air to the Fuel Transfer Canal and/or Cask Loading Pit Gate Seals.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

WINDOW E02
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Dispatch** an operator to determine actual level locally. (see Figure 1 in this procedure).

2. Notify the Security Alarm Station (CAS) to dispatch a security patrol to check for any indications of sabotage.

3. Refer to 13719-1, "Spent Fuel Pool Cooling And Purification" and **return** the Spent Fuel Pit to normal level (218.5 feet).

4. IF level cannot be maintained greater than 217 feet with fuel movement in containment in progress or 216.5 feet with the Spent Fuel Pool Gate Valve closed, THEN **suspend** movement of irradiated fuel assemblies in the Spent Fuel Pool and all crane operations over the Spent Fuel Pool. Initiate 18030-C, "Loss Of Spent Fuel Pool Level Or Cooling" and 18006-C **"**Fuel Handling Event.**"**

5. **Check** service air to gate seals and refer to 13710-1, "Service Air System" to restore service air if lost.

6. Refer to Technical Specification LCO 3.7.15.

5.0 COMPENSATORY OPERATOR ACTIONS

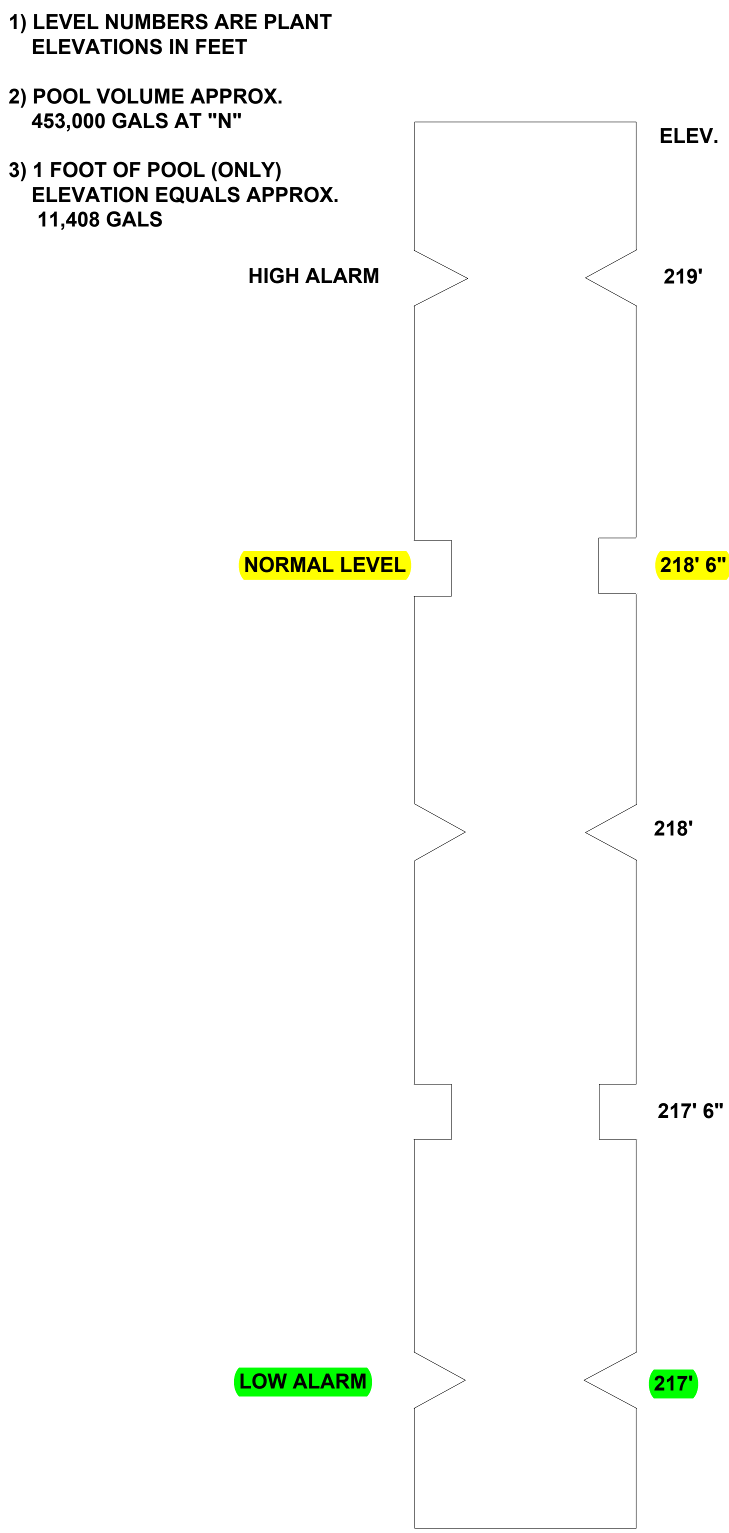
> NOTE
>
> If the East and West pools are connected through the cask loading pit, Unit 1 annunciator ALB05E02 will detect a low level condition for both pools. ☐

Verify Spent Fuel Pool Level every 6 hours per 11883-1, "Radwaste Rounds Sheets."

END OF SUB-PROCEDURE

REFERENCES: 1X4DB130, PLS, 1X5DT0037, Technical Specifications LCO 3.7.15
Commitments SNC11369, 1986308950; SNC4521, 1984301472;
SNC16061, 1996332947

Figure 1 - Spent Fuel Pool Local Water Level Indication



1. 055G2.2.42 001/LOIT/SRO/M/F 3.9/4.6/055G2.2.42/LO-TA-60020A///

At time 1000:

- Unit 1 is at 100% reactor power.
- 18009-C, "Steam Generator Tube Leak," is in progress.
- SG sample results indicate high activity on SG #1.

At time 1020:

- 1RE-0724, Steam Line Rad Monitor, indicates 105 gpd.
- 1RE-0810, SJAE Exhaust Rad Monitor, indicates 120 gpd.
- 1RE-0724 ROC is 55 gpd/hour.
- 1RE-0810 ROC is 60 gpd/hour.

Which one of the following completes the following statement?

Per Tech Spec 3.4.13, "RCS Operational Leakage," the primary to secondary leakage __(1)__ exceed the limit,

and

per 18009-C, the Shift Supervisor is required to initiate __(2)__ to lower reactor power.

	__(1)__	__(2)__
A.	does	18013-C, "Rapid Power Reduction"
B.	does	12004-C, "Power Operation (Mode 1)"
C.✓	does NOT	18013-C, "Rapid Power Reduction"
D.	does NOT	12004-C, "Power Operation (Mode 1)"

K/A

055 **Condenser Air Removal**

G2.2.42 **Ability to recognize system parameters that are entry-level conditions for Technical Specifications.**

K/A MATCH ANALYSIS

The question requires the candidate to utilize radiation monitors associated with Condenser Air Removal (1RE-0810) to recognize entry level for Tech Specs on RCS leakage. The candidate is then required to select which procedure will be utilzied based on the current primary to secondary leakage for plant shutdown.

EXPLANATION OF REQUIRED KNOWLEDGE

Per TS 3.4.13, 150 gpd primary to secondary LEAKAGE through any one steam generator exceeds allowable RCS operational LEAKAGE. In this condidtion, a shutdown to Mode 3 in 6 hrs and Mode 5 in 36 hrs is required. This shutdown can be accomplished using either the guidance of 12004-C or 18013-C. The decision is based on the characteristics of the leak and directed out of 18009-C.

Per 18009-C, if the tube leak is <5gpm and changing at a rate of < 30 gpd/hr, then a shutdown per 12004-C is sufficiently aggressive. If the tube leak is >5gpm or <5 gpm but changing at a rate >30 gpd/hr, then a more agressive shutdown utilizing 18013-C is necessary to ensure the plant is shutdown before the leak propogates into a rupture. The rate of change is determined using 1RE-0724, N-16 Rad Monitor and/or 1RE-0810, SJAE Exhaust Rad Monitor, whose rate of change indications become valid after 20 mintues.

This decission tree is a complex set encompassing steps 5 thru 10 of 18009-C and is often mis-navigated. A high level understanding of steps are required to ensure internal self-checking of this operational decision.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per TS 3.4.13, 150 gpd LEAKAGE in any steam generator exceeds the TS limit. Since both rad montiors have leakage below this value, the limit has not be exceeded. However, the threshold between a tube leak and a tube rupture is 120 gpm. Therefore, a candidate without sufficient knowledge of the TS limits could transpose in their minds the 120 and 150 values and conclude that the TS limit has been exceeded. Therefore, this distractor is plausible.

The second part is correct. Per 18009-C, with the leak rate <5 gpm and the rate of change > 30gpd/hr, a power reduction using 18013-C would be required.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per 18009-C, with the leak rate <5 gpm and the rate of change > 30gpd/hr, a power reduction using 18013-C would be required. However, a candidate without specific knowledge of the procedure decision tree values could conclude that the leak rate (120 gpd = 0.0833 gpm) is not sufficiently large to justify such an aggressive shutdown as 18013-C. Therefore, this distractor is plausible.

C. Correct. The first part is correct. Per TS 3.4.13, 150 gpd LEAKAGE in any steam generator exceeds the TS limit. Since both rad montiors have leakage below this value, the limit has not be exceeded.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

<u>SRO JUSTIFICATION (10CFR43(b))</u>

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the procedure direction decision is based out of NEI guidance on leak characteristics. It is not a logic decision, it is purely based on imperical data from the industry.
-Can the question be answered *solely* by knowing immediate operator actions? No, the procedure knowledge required is not an IOA.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, the procedure knowledge required is neither associated with entry conditions. It is specific to plant conditions the procedure utilizes to make operational decisions.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, detailed and not overall knowledge of steps and sequencing is required to answer the question.
-Does the question require one or more of the following?

- **Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed Yes, the question requires the candidate to have a high level of understanding of the operational goals associated with a decision tree encompassed by 5 steps in AOP 18009-C which then determine the mitigating strategy that will be utilized.**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Level:	SRO
Tier # / Group #	T2 / G2
K/A#	055G2.2.42
Importance Rating:	3.9 / 4.6

Technical Reference: TS 3.4.13, Rev Amendment No. 144, page 3.4.13-1
AOP 18009-C, Rev 29.2, page 6 & 7

References provided: None

Learning Objective:

LO-TA-60020A Respond to a Steam Generator Tube Leak per 18009-C

LO-TA-16010 RCS Leakage Calculation (Inventory Balance) using 14905-1/2

LO-LP-39208-05 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.4 are exceeded.
b. The required actions for all section 3.4 LCOs.

LO-LP-39208-04 Describe the bases for any given Tech Spec in section 3.4.

LO-LP-37311-02 Describe the response of the following parameters to a Steam Generator Tube Rupture: (include in the discussion the response at power, during a reactor startup, and after a reactor trip/safety injection)

k. Steam jet air ejector and steam packing exhauster radiation monitor

LO-LP-60309-10 Discuss how changes in the following affect radiation monitor response to a steam generator tube leak/rupture:

a. RCS activity
b. Power level
c. Process flow rate (i.e., SG blowdown)
d. Rupture size

Question origin: MODIFIED - HL17 NRC Question # 37AA2.10

Cognitive Level: M/F

10 CFR Part 55 Content: 41.5 / 43.5

Comments:

You have completed the test!

1. 037AA2.10 001/1/2/SGTL TECH SPEC/F 3.2/4.1/NEW/HL-17 NRC/SRO/EMT/GCW

Unit 2 is experiencing a Steam Generator Tube Leak on SG 4. The crew is performing 18009-C, "Steam Generator Tube Leak".

Current conditions:

- Reactor power is 100% and stable.
- 2RE-0724 N-16 Rad Monitor indicates 155 gpd.
- 2RE-0810 SJAE Exhaust Rad Monitor indicates 160 gpd.
- RCS specific activity is 1.31 X 10^{-3} micro Curies per gram DOSE EQUIVALENT I-131.

Based on these conditions, which one of the following correctly completes the following statement?

Per Tech Spec 3.4.13, "RCS Operational Leakage", the primary to secondary leakage is ___(1)___ the limit

and

per the Tech Spec Bases 3.4.17, "Steam Generator Tube Integrity," the limit ensures that under the stress of a LOCA or MSLB a single crack leaking this amount will not ___(2)___ .

A. (1) within

(2) exceed the limits for secondary coolant activity

B. (1) within

(2) propagate to a SGTR

C. (1) exceeding

(2) exceed the limits for secondary coolant activity

D.✓ (1) exceeding

(2) propagate to a SGTR

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

If available, both RE-0810 and RE-0724 should be used to determine the leakage rate of change in Step 7 RNO; however, if only one of the two radiation monitors is Functional, then the reading from the Functional monitor should be used to determine leakage rate of change.

6. Check Radiation monitors available:

   RE-810

   OR

   RE-724

   RNO 6. Go to Step 8

7. Check leakage rate of change:

   a. Greater than or equal to 30 GPD/HR based on a 20 minute trend:

   IPC Points:

   RE-0810: UR6810(GPD)
   UR6811(ROC)

   RE-0724: UR6724(GPD)
   UR6725(ROC)

   RNO a. Perform the following:

   1) After a 20 minute trend has elapsed, determine the leakage rate of change.

   IF leakage rate of change is greater than or equal to 30 gpd/hr, THEN go to Step 8.

   -OR-

   IF leakage rate of change is less than 30 gpd/hr, THEN go to Step 9.

Distractor flow path

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. Check leakage rate - LESS THAN 75 GPD.	8. Perform the following: a. Initiate 18013-C, RAPID POWER REDUCTION. b. Be less than 50% power within 1 hour. c. Be in Mode 3 within the next 2 hours. d. Go to Step 12.
9. Check leakage rate - LESS THAN 150 GPD.	9. Perform the following: a. Initiate 12004-C, POWER OPERATION (MODE 1). b. Be in Mode 3 within 6 hours. c. Go to Step 12.
10. Check leakage rate - LESS THAN 75 GPD.	10. IF leakrate has remained greater than or equal to 75 gpd for one hour, THEN perform the following: a. Initiate 12004-C, POWER OPERATION (MODE 1). b. Be in Mode 3 within 24 hours. c. Go to Step 12.

1. 058AA2.03 001/LOIT AND LOCT/SRO/C/A 3.5/3.9/058AA2.03/LO-TA-60040A//HL15/

Procedure titles are as follows:

- 18034-1, "Loss of Class 1E 125 VDC Power"
- 19000-C, "Reactor Trip or Safety Injection"

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- All Train 'A' MSIV red and green handswitch lights extinguish.
- RTB 'A' red and green lights extinguish.
- RCP #1 1E breaker red and green handswitch lights extinguish.
- Channel I TSLB bistable lights illuminate.

Which one of the following completes the following statement?

The Shift Supervisor __(1)__ required to enter 18034-1,

and

the Shift Supervisor __(2)__ required to enter 19000-C.

	__(1)__	__(2)__
A.✓	is	is
B.	is	is NOT
C.	is NOT	is
D.	is NOT	is NOT

K/A

058 Loss of DC Power

AA2.03 Ability to determine and interpret the following as they apply to the Loss of DC Power:

- DC loads lost; impact on ability to operate and monitor plant systems.

K/A MATCH ANALYSIS

The question tests the candidate’s ability to relate multiple indications and the immediate impact to plant operations. They must interpret these various indications

and make a decision on which procedures would address the problem.

EXPLANATION OF REQUIRED KNOWLEDGE

The indications given are the symptoms of a loss of power to 125VDC bus 1AD1. All 'A' train 1E switchgear breakers will loose control power. As such, the breaker will remain in its current state without electrical protection and all handswitch indication lights will be de-energized. The MSIVs and MFIVs will fail CLOSED as their solenoids are de-energized resulting in a Reactor Trip. All Channel I TSLBs lights will illuminate due to the loss of 1AY1A, which is normally feed from an inverter supplied by 1AD1.

Entry conditions for 18034-1 are met. Step 1 of 18034-1 directs a reactor trip and <u>INITIATION</u> of 19000-C. Per Admin procedure 10020-C step 3.3, "Initiate" means the referenced procedure will be used as a supplement to, and it will be performed concurrently with the one in effect. Therefore, 18034-C and 19000-C are expected to be worked in conjunction due to the complications resulting from a loss of DC. 18034-1 will address all issues required by the loss of the supported 120V Vital AC bus.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. Entry conditions for 18034-1 are met.

The second part is correct, Step 1 of 18034-1 directs a reactor trip and <u>INITIATION</u> of 19000-C.

B. Incorrect. Plausible. The first part is correct. Entry conditions for 18034-1 are met.

The second part is incorrect. Per step 1 RNO of 18034-1, a reactor trip should have occured and is required. However, step 1 is not an IOA. A candidate without specific knowledge of the procedure and who did not realize the MISIVs could conclude the reactor did not trip and entry into 19000-C is not required. Therefore, this distractor is plausible.

C. Incorrect. Plausible. The first part is incorrect. The loss of indication on the associated handswitches and TSLBs indicate a loss of 1AD1. However, a candidate with insufficient knowledge of power supplies could conclude that the handswitches are feed from 120V Vital AC and not the 125VDC supply. This is a common misconception. As such, it would be reasonable for the candidate to conclude a loss of 1AY1A to have occurred and entry into 18034-1 is not required. Therefore, this distractor is plausible.

The second part is correct. Step 1 of 18034-1 directs a reactor trip and <u>INITIATION</u> of 19000-C. However, a candidate that believes 18032-1 was entered instead of 18034-1, would still find it reasonable that the reactor would trip based on other trips. Therefore, this distractor is plausible.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. Step 1 of 18034-1 directs a reactor trip and INITIATION of 19000-C. However, a candidate that believes 18032-1 was entered instead of 18034-1 and has knowledge of 18032-1, would not expect the reactor to trip.

SRO JUSTIFICATION (10CFR43(b))

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, selection of the appropriate AOP/EOP combination is NOT associated with system knowledge.
-Can the question be answered *solely* by knowing immediate operator actions? No, the direction to initiate 19000-C concurrent with 18034-C is the RNO of step 1 of 18034-C and is not an IOA.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, the entry condition for each AOP/EOP is part of the question. However, the specific knowledge of 18034-C step 1 RNO is required. Normally, AOP use is discontinued upon entry into 19000-C. Most AOPs that direct tripping the reactor say "go to 19000-C", indicating that a transition to the EOP network is made. There are only a select few AOPs that required concurrent implementation of the AOP with the EOP. In most situation, the AOP is only performed at SS discretion and is typically not needed. 19000-C is designed to be successful with only one train of electrical power.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, specific knowledge of 18034-C step 1 RNO is required.
-Does the question require one or more of the following?

- **Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. Yes, the candidate is required to select the appropriate procedure/combination of procedures to mitigate the plant conditions.**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Level:	SRO
Tier # / Group #	T1 / G1
K/A#	058AA2.03
Importance Rating:	3.5 / 3.9

Technical Reference:	AOP 18034-1 Rev 13.1, page 4 Admin Proc 10020-C Rev 9.0, page 8
References provided:	None
Learning Objective:	LO-LP-60329-01 Given that a loss of power has occurred to any of the following 125VDC vital buses and given the appropriate plant procedures, describe the operator actions required and why these actions are taken. a. 1AD1 LO-LP-60329-04 Given conditions and/or indications, determine the required AOP to enter (including subsections, as applicable). LO-TA-60040A Respond to a Loss of Class 1E 125 VDC Power per 18034-1/2.
Question origin:	BANK - HL15 NRC Question # 058AA2.03
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.7 / 41.10 / 43.5

Comments:

Question appears to match the KA. Not sure if the procedure question is at the SRO-only level. The second question is systems knowledge and not at the SRO-only level.

Choices A-D have 4 different answers; the applicant does not need to know the answer to the second question to answer the question correctly.

The explanation states that an automatic reactor trip has occurred, however, the justification for the correct answer is that the RNO step for "Verify Reactor Trip," states to initiate 19000-C. Does this mean that, although it occurred automatically, Reactor Trip cannot be verified?

One possible fix could be dropping the second question (and anything in the stem that was solely used to answer the second question) and separate the first question to ask: "…the Shift Supervisor <is/is not> required to enter 18034-1 and the Shift Supervisor <is/is not> required to enter 19000-C."

(I still need to think about whether asking it this way is at the SRO-only level. I'm leaning towards it IS at the SRO-only level.)
- JAT 12/19/2013 (Editorial)

New question incorporates the above comments. SRO-only

appears to be met because the question requires knowledge of procedure rules-of-usage.
- JAT 2/4/14

You have completed the test!

ABNORMAL OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure provides operator actions to be followed in the event that power is lost to one of the 125V DC Vital Busses (1AD1, 1BD1, 1CD1, or 1DD1).

Specific instructional steps will be found in the following sections:

A. LOSS OF 125V DC BUS 1AD1

B. LOSS OF 125V DC BUS 1BD1

C. LOSS OF 125V DC BUS 1CD1

D. LOSS OF 125V DC BUS 1DD1

Handswitch lights for RCP#1 and RTB 'A' extinguished

SYMPTOMS

Channel 1 TSLB illuminated

SECTION A. LOSS OF 125V DC BUS 1AD1

- 125V DC Vital Bus 1AD1 voltage low.
- Loss of power to 1AY1A and 1AY2A 120V AC Vital Instrument Panels.
- Loss of indicating lights on 1AA02, 1AB04, 1AB05, and 1AB15 Switchgear Controls.
- Train A Main Steamline Isolation.
- Train A Main Feedwater Isolation.

No handswitch lights on MSIV 'A'

SECTION B. LOSS OF 125V DC BUS 1BD1

- 125V DC Vital Bus 1BD1 voltage low.
- Loss of power to 1BY1B and 1BY2B 120V AC Vital Instrument Panels.
- Loss of indicating lights on 1BA03, 1BB06, 1BB07, and 1BB16 Switchgear Controls.
- Train B Main Steamline Isolation.
- Train B Main Feedwater Isolation.

A. LOSS OF 125V DC BUS 1AD1

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- This procedure should be performed concurrent with 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
- RCP 1 undervoltage and underfrequency trips will NOT actuate.
- See ATTACHMENT A for equipment responses, breaker and valve control loss, valve failures from loss of instrument air, and annunciator failures.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__A1. Verify reactor trip.	A1. Perform the following: __a. Trip the reactor. __b. Initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
__A2. Initiate the Continuous Actions Page.	
__A3. Dispatch an operator to 1AA02 SWGR Room (CB-A48).	

NOTE

IF DG1A is NOT running, it can NOT be started.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__A4. Check DG1A – RUNNING.	__A4. Go to Step A7.

3.2 "GO TO" STEPS

To maintain consistency in referencing or branching to another procedure:

3.2.1 "Go to" is used when it is desired to branch to another procedure or to a preceding step in the procedure.

Example: IF the reactor trips,
THEN go to 19000-C, E-0
REACTOR TRIP OR SAFETY INJECTION.

Branching implies the procedure in use shall be exited and a new procedure entered.

3.2.2 "return to" is used when it is desired to branch to a previous step in the procedure.

3.3 "BY INITIATING" STEPS

When "by initiating" is used, the referenced procedure will be used as a supplement to, and it will be performed concurrently with the one in effect.

3.4 IMMEDIATE OPERATOR ACTIONS STEPS

3.4.1 These are actions that, for EOPs are to be committed to memory for immediate performance upon initiation of the procedure. These actions, which typically involve verification of automatic actions, are listed starting on top of the next page after the symptoms section with "IMMEDIATE OPERATOR ACTIONS" typed above Step 1.

3.4.2 Immediate Operator Action Steps shall be performed by memory by the operator. The Unit Shift Supervisor will state the high level steps as written in the procedure. Upon restatement the operator will repeat the step including all substeps to ensure completeness.

3.4.3 All EOP immediate actions must be completed prior to taking any early action or non-EOP action.

1. 062AG2.2.12 001/LOIT/SRO/C/A 3.7/2.7/062AG2.2.12/LO-LP-39204-04///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- The Shift Supervisor discovers the 24 hour Channel Check for Train 'A' NSCW basin level, 1LI-1606, was missed.
- The last performance of the 24 hour Channel Check was 0030 on 5-11-14.
- Time of discovery of the missed surveillance was 1700 on 5-12-14.

- A risk evaluation will NOT be performed.

Which one of the following completes the following statement?

To prevent declaring the LCO NOT met, the surveillance is required to be performed satisfactorily no later than ________.

A. 0030 on 5-12-14

B. 0630 on 5-12-14

C.✓ 1700 on 5-13-14

D. 0100 on 5-14-14

K/A

062 **Loss of NSCW**

G2.2.12 **Knowledge of surveillance procedures**.

K/A MATCH ANALYSIS

The question tests the candidates knowledge of generic TS survelliance SR 3.0.3 as specifically applied to a missed NSCW surveillance. If the missed surveillance results in an inoperable declaration, a loss of one train of NSCW would occur since the surveillance affects the Ultimate Heat Sink LCO 3.7.8.

EXPLANATION OF REQUIRED KNOWLEDGE

A Surveillance has been identified on an NSCW System as being missed and the candidate must use Tech Spec SR 3.0.3 to determine when the missed surveillance is to be performed by. If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow

performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Incorrect but plausible because the candidate may determine the surveillance must be perform within 24 hours of the time missed as opposed to time of discovery.

B. Incorrect. Plausible. Incorrect but plausible because the candidate may determine the surveillance must be perform from time missed within 24 hours plus the 125% grace period described in SR 3.0.2. The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

C. Correct. The answer is correct the missed surveillance must be completed within 24 hours of point of discovery. (e.g. If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.)

D. Incorrect. Plausible. Incorrect but plausible because the candidate may determine the surveillance must be perform from point of discovery within 24 hours plus the 125% grace period described in SR 3.0.2. The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

SRO JUSTIFICATION

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the question is not related to 1 hour action time requirements.
-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" No, the question is not related to above-the-line information.

-Can question be answered *solely* by knowing the TS Safety Limits? No, the question is not related to Safety Limits.

-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- **Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4) Yes, the required knowledge is application of SR 3.0.3 in Tech Spec.**
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Level:	SRO
Tier # / Group #	T1 / G1
K/A#	062G2.2.12
Importance Rating:	3.7 / 4.1

Technical Reference: OSP 14000-1, Rev 88.1, page 15
TS SR 3.0.3, Amendment No. 125, page 3.0-4
Surv Frequency Control Program, Rev 3, page 15

References provided: None

Learning Objective: LO-LP-39204-04 State the allowable time intervals for extension of surveillances. State the result of failure to perform surveillances within this period.
LO-LP-39204-06 In regard to surveillances, determine when time delay may be applied and the maximum time allowed to perform the surveillance.

Question origin: BANK

Cognitive Level: C/A

10 CFR Part 55 Content: 41.10 / 43.2

Comments:

You have completed the test!

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

Surveillance Requirement	Frequency	Notes	LDCR No.
SR 3.7.7.2	18 months on a STAGGERED TEST BASIS		2012-015
SR 3.7.8.1	31 days		N/A
SR 3.7.8.2	18 months	For the following components only: 1HV1668A 2HV1668A 1HV1668B 2HV1668B 1HV1669A 2HV1669A 1HV1669B 2HV1669B	N/A
	18 months on a STAGGERED TEST BASIS	For the following components only: 1HV1806 2HV1806 1HV1808 2HV1808 1HV1822 2HV1822 1HV1830 2HV1830 1HV2134 2HV2134 1HV2138 2HV2138 1HV1807 2HV1807 1HV1809 2HV1809 1HV1823 2HV1823 1HV1831 2HV1831 1HV2135 2HV2135 1HV2138 2HV2138	2012-015
SR 3.7.8.3	18 months on a STAGGERED TEST BASIS		2012-015
SR 3.7.9.1	24 hours		N/A
SR 3.7.9.2	24 hours		N/A
SR 3.7.9.3	31 days		N/A
SR 3.7.9.5	24 hours		N/A

DATA SHEET 1
MODE 1 & 2

MODE _______________

DATE _______________

LCO METHOD OF VERIFICATION	TECH SPEC SURV REQ	PARAMETER	INSTRUMENT	INDICATION DAY	INDICATION NIGHT	LIMIT(S) TOLERANCE	LCO/PROC
CREFS ACTUATION OPERABLE CHANNEL CHECK	SR 3.3.7.1 FCN 3	CR INTAKE RADIATION MONITORS (INIT)	1RE-12116			CHANNEL CHECK REQUIRED 2	3.3.7
			1RE-12117				
FHB ACTUATION OPERABLE CHANNEL CHECK	TRS 13.3.6.1	FHB EFFL RADIOGAS FHB ISO (INIT)	ARE-2532A			* REQUIRED 1	13.3.6
			ARE-2532B				
FHB ACTUATION OPERABLE CHANNEL CHECK	TRS 13.3.6.1	FHB EFFL RADIOGAS FHB ISO (INIT)	ARE-2533A			* REQUIRED 1	13.3.6
			ARE-2533B				
		*INDICATING NORMALLY. ALL STATUS AND ALARM LIGHTS EXTINGUISHED.					
DG1A FUEL OIL INVENTORY VERIFY FUEL OIL STORAGE TANK LEVEL	SR 3.8.3.1	DG 1A LEVEL (%)	1-LI-9024			≥82%	3.8.3
DG1B FUEL OIL INVENTORY VERIFY FUEL OIL STORAGE TANK LEVEL	SR 3.8.3.1	DG 1B LEVEL (%)	1-LI-9025			≥ 82%	3.8.3
TWO INDEPENDENT CONTROL ROOM EMERGENCY FILTRATION SYSTEMS SHALL BE OPERABLE VERIFY CONTROL ROOM TEMP	SR 3.7.10.1 SR 3.7.11.1	NOTE: TEMPERATURE INDICATION IS OBTAINED FROM HAND-HELD TEST EQUIPMENT. RECORD INSTRUMENT INFORMATION BELOW.					
		INSTRUMENT ID NO.				N/A	
		CAL DUE DATE					
		CONTROL ROOM TEMPERATURE (°F)	M&TE			≤85°F	3.7.10 3.7.11
THE RWST SHALL BE OPERABLE VERIFY TEMPERATURE	SR 3.5.4.1 TRS 13.1.7.1	RWST TEMPERATURE (°F)	1TIS-10980			≥51°F * ≤109°F *	3.5.4 13.1.7
		*WITH INDICATED RWST TEMPERATURE OUTSIDE THE LIMITS, THEN VERIFY RWST TEMPERATURE IS WITHIN TECHNICAL SPECIFICATION LIMITS BY PLACING THE RWST ON RECIRC USING SLUDGE MIXING PUMP WITH HEATER OFF AND OBSERVING 1-TI-10982 TO BE WITHIN ≥44°F AND ≤116°F.					
THE ULTIMATE HEAT SINK SHALL BE OPERABLE VERIFY WATER TEMPERATURE AND LEVEL	SR 3.7.9.2	TEMPERATURE (°F)	COMPUTER POINT T2601*			≤90°F	3.7.9
			-OR-				
			1TJI-1692 POINT 2*				
			COMPUTER POINT T2602*				
			-OR-				
			1TJI-1692 POINT 17*				
		*IF COMPUTER POINT AND RECORDER POINT ARE NOT AVAILABLE, TEMPERATURE READING MUST BE OBTAINED LOCALLY USING HAND-HELD TEST EQUIPMENT. RECORD INSTRUMENT INFORMATION BELOW.					
		INSTRUMENT ID NO.				N/A	
		CAL DUE DATE					
	SR 3.7.9.1	LEVEL (%)	1LI-1606			≥73%	
			1LI-1607				
CONTAINMENT AIR TEMPERATURE SHALL NOT EXCEED 120°F VERIFY AVERAGE AIR TEMPERATURE	SR 3.6.5.1	TEMPERATURE (°F)	COMPUTER POINT T2501			NA	3.6.5
			COMPUTER POINT T2502				
			COMPUTER POINT T2503				
			COMPUTER POINT UT2501 (AVG)			≤120°F	
		*IF COMPUTER POINT IS NOT AVAILABLE VERIFY CNMT HI TEMP ALARM ALB-01 (E06) IS NOT IN ALARM.				ALB-01 (E06) NOT IN ALARM	
		*IF COMPUTER POINT AND ALB-01 (E06) ARE NOT AVAILABLE, TEMPERATURE READING MUST BE OBTAINED LOCALLY USING HAND-HELD TEST EQUIPMENT FOR 1TE-2612 FOR POINT T2502 AND 1TE-2613. FOR POINT T2503 RECORD INSTRUMENT INFORMATION BELOW. USE MCB INDICATOR 1TI-2563 FOR POINT T2501 AND AVERAGE THE THREE.					
		INSTRUMENT ID NO.				≤120°F	
		CAL DUE DATE					

COMPLETED BY: DAY: _______________ TIME: ________ NIGHT: ________________ TIME: ________

SS REVIEW: DAY: _______________ TIME: ________ NIGHT: ________________ TIME: ________

1. 076A2.02 001/LOIT AND LOCT/SRO/C/A 2.7/3.1/076A2.02/LO-TA-60003/63013///

Initial conditions:

- Unit 1 is at 100% reactor power.
- SI Pump 'A' is tagged out.

Current conditions:

- Train ‘B’ NSCW supply header pressure is 75 psig and lowering.
- Train 'B' NSCW supply header flow is 25,000 gpm.
- Train 'B' NSCW return header flow is 10,000 gpm.
- 18021-C, "Loss of Nuclear Service Cooling Water System," is entered.

Which one of the following completes the following statement?

Per 18021-C, the crew is required to __(1)__ the standby Train ‘B’ NSCW pump,

and

after completing the actions of 18021-C, the Shift Supervisor will determine per 10008-C, "Recording Limiting Conditions for Operation," that a LOSF __(2)__ exist.

	__(1)__	__(2)__
A.	start	does
B.	start	does NOT
C✓	place in PTL	does
D.	place in PTL	does NOT

K/A

076 Service Water

A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

- Service water header pressure

K/A MATCH ANALYSIS

The question addresses a problem identified on an NSCW System to include low

header pressure and the candidate must determine the correct procedure action based on the indications provided. In addition, the candidates must determine the impact on plant operations using the information provided in the stem as related to LOSF evaluation. This concept brings the question to the SRO knowledge level.

EXPLANATION OF REQUIRED KNOWLEDGE

Per 18021-C symptoms, a drop in NSCW header pressure accompanied by a large difference between supply and return header flows indicates a large (catastrophic) leak. Per steps 1 and 2, all pumps in the affected train will be placed in PTL.

Since NSCW 'B' is a support system for all 'B' train ECCS pumps, a LOSF function exists with SIPs. With SIP 'A' tagged out in the stem and a subsequent loss of SIP 'B' due to loss of a support system, no medium head injection is available. Since TS 3.5.2 does not have a condition for two trains of ECCS inoperable, TS 3.0.3 must be entered. (Reference 10008-C for LOSF evaluation guidance.)

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Part 1 is incorrect however ‘plausible’ since the candidate may believe the stem indications are due a pump problem such as a broken coupling, as opposed to a leak since the symptoms would be the same with exception of the supply and return flow deviation. Per 18021-C step 6, the correct action for this condition would be to start the standby pump.

Part 2 is correct. The candidate should determine that with both Train ‘A’ SI Pump and Train ‘B’ NSCW System inoperable a LOSF exists per 10008-C 'Recording Limiting Conditions for Operation'. However, if the candidate misses the catastrophic leak in Part 1, it would still be plausible for them to incorrectly determine that the standby NSCW pump is inoperable due to the failure to start on low header pressure and determine a LOSF exists for the wrong reason.

B. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice A above.

Part 2 is incorrect. The candidate should determine that with both Train ‘A’ SI Pump and Train ‘B’ NSCW System inoperable a LOSF exists per 10008-C 'Recording Limiting Conditions for Operation'. However, if the candidate misses the catastrophic leak in Part 1 and recognizes that the failure of the start on low header pressure in not a required function, then determining that a LOSF does not exist would be correct for these erroneous conditions..

C. Correct. Part 1 is correct. The candidate should determine that from the stem information provided that a large leak has occurred in the NSCW piping and the correct actions per 18021-C ‘Loss of NSCW’, is to place the affected pumps in PTL.

Part 2 is correct. The candidate should determine that with both Train ‘A’ SI Pump and Train ‘B’ NSCW System inoperable a LOSF exists per 10008-C 'Recording Limiting Conditions for Operation'.

D. Incorrect. Plausible. Part 1 is correct. See Part 1 of choice C above.

Part 2 is incorrect however ‘plausible’ since the candidate may not make the operability connection between the two systems or assume that NSCW could be run in single pump operations to supply cooling and therefore believe a LOSF condition is not present.

SRO JUSTIFICATION

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the question only addresses TS action of 72 hrs in association with implementation of TS 3.0.3.
-Can question be answered *solely* by knowing the LCO/TRM information listed “above-the-line?” No, all necessary knowledge is below the line.
-Can question be answered *solely* by knowing the TS Safety Limits? No, the question does not address Safety Limits.
-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- **Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4) Yes, the candidate is required to determine the applicability of LCO 3.0.3 as applied to a LOSF.**
- Knowledge of TS bases that is required to analyze TS required actions and terminology

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? No, system knowledge will not address the operability/safety function determination.
-Can the question be answered *solely* by knowing immediate operator actions? No IOA's are addressed by the question.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, knowledge of an administrative process is required associated with Tech Specs.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, sequence or overall stategy of 18021-C will not answer either part of the question.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with

which to proceed

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- **Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures** **Yes, specific knowledge of Admin procedure 10008-C is required to perform a LOSF evaluation to determine which LCO(s) are not met based on the inoperability of support systems and the impact to pre-existing inoperabilities.**

Level:	SRO
Tier # / Group #	T2 / G1
K/A#	076A2.02
Importance Rating:	2.7 / 3.1

Technical Reference: ADMIN 10008-C, Rev 30.0, pages 1-6, 15-19, & 28
AOP 18021-C, Rev 19.0, pages 1-3
TS 3.5.2, Amendment No. 136, page 3.5.2-1

References provided: None

Learning Objective:

LO-LP-63508-04 Define the following terms per 10008-C.
a. LCO
d. Loss of Safety Function
e. Supported System
f. Support System

LO-PP-06101-04 Describe the indications of the following:
c. Supply header leak
d. Return header leak
e. NSCW pump trip
f. NSCW piping leak in a pump room

LO-TA-63013 Implement Technical Specification LCO using 10008-C (SRO Only)

LO-TA-60003 Respond to a Loss of NSCW per 18021-C

Question origin: MODIFIED - HL15 Question # 062AA2.02

Cognitive Level: C/A

10 CFR Part 55 Content: 41.4 / 41.10 / 43.2 / 43.5

Comments:

You have completed the test!

1. 062AA2.02 001/1/1/LOSS OF NSCW/C/A - 3.6/NEW/HL15/SRO/DS/TNT

Given the following conditions at 38% power:

- ACCW pump 2 is in service
- CCW pumps 2 & 4 are in service
- NSCW Pump 5 is danger tagged

Train A NSCW indications:	**Train B NSCW indications:**
- Supply header pressure 45 psig	- Supply header pressure 58 psig
- Supply header flow 8,000 gpm	- Supply header flow 25,000 gpm
- Return header flow 8,000 gpm	- Return header flow 10,000 gpm

Which of the following choices contains the correct procedural entry and actions?

A.✓ Enter AOP 18021-C, Loss of NSCW, due to loss of **both** NSCW Trains.

Place all NSCW pumps in PTL, trip the reactor and initiate EOP 19000-C. Trip the RCPs and isolate CVCS letdown.

B. Enter AOP 18021-C, Loss of NSCW, due to leakage on **Train A** NSCW.

Place all Train A NSCW pumps in PTL, trip the reactor and initiate EOP 19000-C. Trip the RCPs and isolate CVCS letdown if cooling not restored in 10 minutes.

C. Enter AOP 18021-C, Loss of NSCW, due to leakage on **Train B** NSCW.

Place all Train B NSCW pumps in PTL. Shift to Train A CCW pumps. Start ACCW pump #1 and remain in 18021-C.

D. Enter AOP 18021-C, Loss of NSCW, due to loss of **both** NSCW Trains.

Place NSCW Train B in single pump operation and all Train A NSCW pumps in PTL. Trip RCPs if seal temperatures exceed 230 F and remain in 18021-C.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

---NOTE--
In MODE 3, either residual heat removal pump to cold legs injection flow path may be isolated by closing the isolation valve to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

No condition exists for 2 trains inop - TS 3.0.3 must be entered.

ABNORMAL OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure addresses the loss or degraded operation of one or more trains of Nuclear Service Cooling Water.

SYMPTOMS

- Trip of operating NSCW pumps and failure of standby pump to start.
- Dropping NSCW Supply Header pressure.
- Large difference between Supply Header flow and Return Header flow, indicating a large leak.
- NSCW Tower Basin temperature rising above 90°F.
- High temperature or low flow alarms on any components or systems cooled by NSCW.

MAJOR ACTIONS

- ♦ Determine condition causing loss or degraded operation of NSCW.
- ♦ Transfer loads to unaffected train.
- ♦ Correct or repair condition causing loss or degraded operation of NSCW.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Check if catastrophic leakage from NSCW system – EXISTS.	1. Go to Step 6.
2. Place affected train NSCW pump handswitches in PULL-TO-LOCK.	
3. Depress both Emergency Stop pushbuttons for the affected DG.	
4. Verify proper operation of <u>UNAFFECTED</u> NSCW train:	4. <u>IF</u> neither NSCW train can be placed in normal, two pump operation, <u>THEN</u> perform the following:
• Two pumps running.	a. Trip the reactor.
• Supply header pressure greater than 70 psig: Train A: PI-1636 Train B: PI-1637	b. Initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION. c. Trip all reactor coolant pumps.
• Supply header temperature computer indication less than 90°F: Train A: T2601 Train B: T2602	d. Isolate letdown. e. Place one train of NSCW in single pump operation by initiating 13150, NUCLEAR SERVICE COOLING WATER SYSTEM.
• Supply header flow approximately 17,000 gpm: Train A: FI-1640B Train B: FI-1641B	f. Verify train-related CCP or NCP running and seal injection flow established using 13006, CHEMICAL AND VOLUME CONTROL SYSTEM.

Step 4 continued on next page

ABNORMAL OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure addresses the loss or degraded operation of one or more trains of Nuclear Service Cooling Water.

SYMPTOMS

- Trip of operating NSCW pumps and failure of standby pump to start.
- Dropping NSCW Supply Header pressure.
- Large difference between Supply Header flow and Return Header flow, indicating a large leak.
- NSCW Tower Basin temperature rising above 90°F.
- High temperature or low flow alarms on any components or systems cooled by NSCW.

MAJOR ACTIONS

- ♦ Determine condition causing loss or degraded operation of NSCW.
- ♦ Transfer loads to unaffected train.
- ♦ Correct or repair condition causing loss or degraded operation of NSCW.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Check if catastrophic leakage from NSCW system – EXISTS.	1. Go to Step 6.
2. Place affected train NSCW pump handswitches in PULL-TO-LOCK.	
3. Depress both Emergency Stop pushbuttons for the affected DG.	
4. Verify proper operation of <u>UNAFFECTED</u> NSCW train:	4. <u>IF</u> neither NSCW train can be placed in normal, two pump operation, <u>THEN</u> perform the following:
• Two pumps running.	a. Trip the reactor.
• Supply header pressure greater than 70 psig: Train A: PI-1636 Train B: PI-1637	b. Initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION. c. Trip all reactor coolant pumps.
• Supply header temperature computer indication less than 90°F: Train A: T2601 Train B: T2602	d. Isolate letdown. e. Place one train of NSCW in single pump operation by initiating 13150, NUCLEAR SERVICE COOLING WATER SYSTEM.
• Supply header flow approximately 17,000 gpm: Train A: FI-1640B Train B: FI-1641B	f. Verify train-related CCP or NCP running and seal injection flow established using 13006, CHEMICAL AND VOLUME CONTROL SYSTEM.

Step 4 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	g. Check RCP No. 1 seal temperatures less than 220°F.
	h. IF RCP No. 1 seal temperatures greater than 220°F, THEN do NOT attempt to restart RCPs prior to a status evaluation.
5. Go to Step 13.	
6. Verify two or more NSCW pumps on the affected train are operating properly by checking the following parameters exist:	6. Perform the following:
• Supply header pressure greater than 70 psig. Train A: PI-1636 Train B: PI-1637	a. Place affected train NSCW pump handswitches in PULL-TO-LOCK. b. Depress both Emergency Stop pushbuttons for the affected DG.
• Supply header flow approximately 17,000 gpm. Train A: FI-1640B Train B: FI-1641B	c. Investigate cause for trip of running pump(s).

Step 6 continued on next page

RECORDING LIMITING CONDITIONS FOR OPERATION

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	SECTIONS
Continuous:	NONE
Reference:	NONE
Information:	ALL

TABLE OF CONTENTS

1.0 PURPOSE

1.1 This procedure prescribes the method to record the failure to meet the Limiting Conditions for Operation (LCO), or Technical Requirement, the associated ACTION requirements, any change in status effecting the ACTION, and the return to compliance with LCO/TR.

1.2 This procedure also includes instructions for implementing Technical Specification 5.5.15, the Safety Function Determination Program. As required by LCO 3.0.6, this program ensures that proper actions are taken such that multiple inoperable Structures, Systems, or Components (SSC) do not result in an undetected LOSS OF SAFETY FUNCTION.

1.3 This procedure also ensures that the allowed out of service time of SUPPORTED SYSTEMS is not inappropriately extended as a result of multiple inoperable SUPPORT SYSTEMS.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Technical Specification LCO 3.0.2 states that the required Actions of an LCO MUST be **performed** when the requirements of the LCO are NOT met. LCO 3.0.6 provides an exception to LCO 3.0.2 for SUPPORTED SYSTEMS by NOT requiring the Required Actions for the SUPPORTED SYSTEMS to be performed WHEN the failure to meet an LCO is SOLELY due to the inoperability of a SUPPORT SYSTEM. In this situation, LCO 3.0.6 requires ONLY the Required Actions of the SUPPORT SYSTEM to be **performed**. Since “cascading” is NOT required in this case, a possibility exists that unrelated concurrent failures of more than one SUPPORT SYSTEM could result in the complete loss of both trains of a SUPPORTED SYSTEM. THEREFORE, upon a failure to meet two or more LCOs during the same time period, an **evaluation** SHALL be conducted to **determine** if a LOSS OF SAFETY FUNCTION (LOSF) exists. This LOSF Evaluation satisfies the criteria of Technical Specification Administrative Control 5.5.15, Safety Function Determination Program.

2.2 If the failure of a SSC not addressed by Technical Specification results in the inoperability of a required SUPPORT and/or SUPPORTED SYSTEM, then the LCO(s) for the required SUPPORT and/or SUPPORTED SYSTEM would be entered. Example: If 1NB10 is de-energized, the operability of D/G '1B' or the Pressurizer Heaters may be impacted.

2.3 A single component inoperability can result in multiple inoperabilities within a single train and affect multiple Technical Specification LCOs. LCO 3.0.6 limits the amount of "cascading" of actions that is required when an inoperable SSC renders a SUPPORT SYSTEM inoperable.

2.4 A single component inoperability CAN also impact operability on redundant trains. Example: IF 1HV-8716A is closed, both trains of ECCS may be impacted.

2.5 A Loss of Safety Function evaluation must be performed for each inoperability of a SCC impacting a required SUPPORT or SUPPORTED SYSTEM(s).

2.6 The LOSF Evaluation MUST be **reinitiated** whenever an additional required structure, system, or component (SSC) is declared inoperable. This includes LCOs with Required Actions that specify declaring additional components inoperable.

2.7 Alternating between LCO Conditions, in order to allow indefinite continued operation while not meeting the LCO, is not allowed.

3.0 DEFINITIONS

3.1 EXTENT OF CONDITION REVIEW

A review to determine the scope of SSCs (in other trains, units, or subcomponents) that are affected by a condition adverse to quality.

3.2 LIMITING CONDITION FOR OPERATION (LCO)

A condition specified in the plant Technical Specifications (TS) or Technical Requirements Manual (TRM) which limits unit operations. An LCO may be entered by an equipment malfunction or a change in a unit parameter. If an LCO is not met, the associated ACTION requirements shall be met.

3.3 TECHNICAL REQUIREMENT (TR)

A condition specified in the plant Technical Requirements Manual (TRM) which limits unit operations. A TR may be entered by an equipment malfunction OR a change in a unit parameter. If a TR is not met, the associated ACTION requirements SHALL be met. TR and Technical Requirement Surveillances (TRS) associated with each TR are implemented in the same way as Technical Specifications. However, TRs and TRSs are treated as plant procedures and are not part of the Technical Specification. Therefore exceptions apply (Reference TRM Section 11.5).

3.4 INFORMATION ONLY LIMITING CONDITION FOR OPERATION/TECHNICAL REQUIREMENT (Info LCO/TR)

A method of tracking an equipment malfunction or change in plant parameter which would restrict unit operation in another mode OR prevent a mode change in which it would become applicable, or may become an LCO/TR for the present mode should other Technical Specification related equipment or redundant safety related equipment become inoperable.

Information Only LCOs should NOT be prepared for conditions that are not applicable in the present operating mode unless used for tracking for entry into a mode in which a transition is to be directly made.

In addition, as an administrative tool to help track compliance with the ODCM, Information LCOs WILL be used when the requirements of the Offsite Dose Calculation Manual (ODCM) Sections 2.5 and 3.5 are not met.

Log entry (Electronic Log) of Information Only LCOs should NOT be made.

3.5 LOSS OF SAFETY FUNCTION (LOSF)

A LOSF exists WHEN; assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed.

3.6 SUPPORT SYSTEM

3.6.1 A SSC which is needed by another Technical Specification LCO required SSC to perform a safety function.

Example: The Component Cooling Water System (SUPPORT SYSTEM) is required by the Residual Heat Removal System (SUPPORTED SYSTEM) to fulfill its safety function. A SUPPORT SYSTEM may also be a SUPPORTED SYSTEM. Example: The Component Cooling Water System requires the Nuclear Service Cooling Water System to fulfill its safety function.

In the question the Train 'B' NSCW supports the Train 'B" SI Pump.

3.6.2 For the purpose of implementing LCO 3.0.6, a SSC which monitors or maintains a process parameter or operating limit is not a SUPPORT SYSTEM; however, specific functions of Technical Specification instrumentation required to fulfill a credited safety function, may be considered a SUPPORT SYSTEM.

Examples:

- The Digital Rod Position Indicators (DRPI) are used to monitor control rod insertion limits, however; inoperability of DRPI does not result in the control rods not being within insertion limits. Control rod insertion limits are monitored separately and actions taken as appropriate when insertion limits are not met or Surveillance Requirements not performed when required.
- Likewise, parameter limits that could affect other parameter limits if exceeded are also NOT considered SUPPORT SYSTEMS for the purposes of implementing LCO 3.0.6. Example: Exceeding control rod insertion limits could affect hot channel factors
- Auto Actuation Logic and Actuation Relays, although identified as Instrumentation, MAY be considered a SUPPORT SYSTEM.

3.7 SUPPORTED SYSTEM

A SSC, required by the Technical Specifications, which requires a SUPPORT SYSTEM to ensure its safety function can be performed. For the purposes of implementing LCO 3.0.6, process parameters, operating limits, or individual instrument channels are <u>NOT</u> SUPPORTED SYSTEMS; however, specific functions of Technical Specification instrumentation required to fulfill a credited safety function, <u>MAY</u> be considered a SUPPORTED SYSTEM.

4.4.3 Part III. Completed LCO/TR Status Sheets

Part III contains copies of LCO/TR Status Sheets for LCO/TRs that have been restored. Sheets are filed in order of their LCO/TR number. Copies of completed sheets SHOULD be **retained** in the binder for at least 30 days after they have been closed out.

4.5 LOSS OF SAFETY FUNCTION (LOSF) EVALUATION

4.5.1 **Review** Precautions and Limitations PRIOR to performing next step.

4.5.2 **Identify** the applicable Technical Specification conditions and required actions for the inoperable SSCs PRIOR to entering the LCO, IF possible.

NOTE

A flow chart of the LOSF Evaluation process is shown in Figure 5. ☐

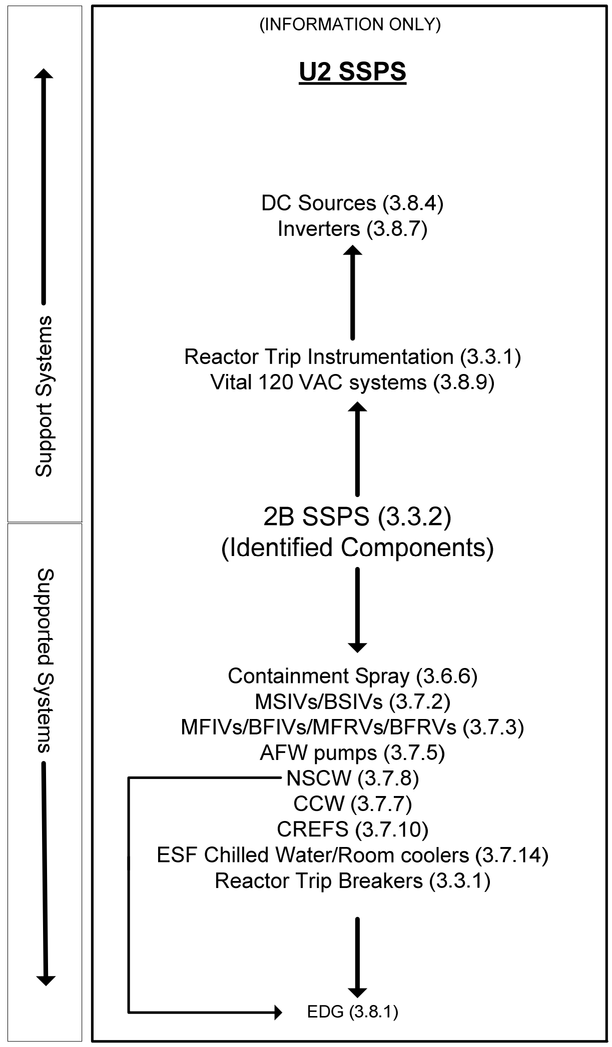
4.5.3 **Generate** a list of impacted SUPPORT/ SUPPORTED Systems.

4.5.3.1 Considering the Conditions identified in step 4.5.2 as well any LCO Condition(s) previously in effect, **determine** if required SUPPORT or SUPPORTED SYSTEM(s) are rendered inoperable on redundant safety-related trains.

	Train A	Train B
	System i	System i
	↓	↓
	System ii	System ii
	↓	↓
Inoperable system	System iii	System iii
	↓	↓
	System iv	System iv

For the above example, IF Train A System iii is inoperable THEN Train B Systems i, ii and iii (support systems) and System iv (supported system) MUST be verified operable.

IF all Conditions in effect are limited to a single train, THEN no LOSF exists. All applicable Conditions SHOULD be entered, the provisions of LCO 3.0.6 may be **applied**, and NO additional evaluation is required.
Below is an example of a list when Unit 2 SSPS is rendered inoperable while performing U2 RTB testing.



4.5.4 Procedure 10005-C SHALL be used to manually **ILLUMINATE** SSMP for the systems/components identified in steps 4.5.2 and 4.5.3.

4.5.5 Using flowchart (Figure 5) **determine** if a Loss of Safety Function will exist IF the component/system is rendered inoperable. A method of place keeping should be used ensuring correct flow path is used. A (SRO) SHALL conduct an independent peer check of flowchart.

4.5.6 **Determine** IF concurrent inoperable SUPPORT or SUPPORTED systems on required redundant train, results in the loss of a credited safety function.

4.5.6.1 Equipment supported by an inoperable Offsite Source OR Diesel Generator should NOT be considered inoperable for the purpose of this evaluation, UNLESS required by LCO 3.8.1 Required Action A.2 or B.3. IF LCO 3.8.1 Condition A OR Condition B is in effect AND implementation of Required Action A.2 or B.3 subsequently results in the inoperability of a required supported system, THEN a LOSF Evaluation MUST be **re-performed**.

4.5.6.2 The TS related systems that SHOULD be **evaluated** when determining if a potential loss of safety function exists are:

- Reactor Trip System Automatic Trip Logic
- Reactor Trip and Bypass Breakers
- ESFAS Automatic Actuation Logic & Actuation Relays:
    - Safety Injection
    - Containment Spray
    - Containment Isolation
    - Steamline Isolation
    - Turbine Trip and Feedwater Isolation
    - Auxiliary Feedwater
    - Containment Sump Semi-automatic Switchover
- LOSP Instrumentation - Loss of either Undervoltage or Degraded Voltage Functions
- CVI Automatic Actuation Logic & Actuation Relays
- CREFS Automatic Actuation Logic & Actuation Relays
- High Flux at Shutdown Alarm (HFASA)
- Decay Heat Removal (including refueling operations)
- Pressurizer PORVs and associated Block Valves
- Cold Overpressure Protection System
- ECCS (See Step 4.5.6.8)
- Containment Penetrations
- Containment Spray and Cooling Systems
- Main Steam Isolation Valves
- MFIVs, MFRVs, and associated Bypass Valves
- Atmospheric Relief Valves
- Auxiliary Feedwater System
- Component Cooling Water System
- Nuclear Service Cooling Water System
- Ultimate Heat Sink
- Control Room Emergency Filtration Systems
- Piping Penetration Area Filtration and Exhaust System
- ESF Room Cooler and Safety-Related Chiller System
- AC Sources (including Safety Systems Sequencer)
- Diesel Fuel Oil, Lube Oil, Starting Air, and Ventilation
- DC Sources
- Inverters
- Electrical Distribution Systems

4.5.6.3 A credited safety function is a function required to mitigate the consequences of a design basis event as described in the FSAR (reference FSAR Chapters 6 and 15), including all assumptions of the initiating event such as loss of offsite power.

NOTE

FSAR assumptions such as loss of offsite power are considered as part of the initiating event and should not be considered as an "additional concurrent failure" in the following step. ☐

4.5.6.4 A LOSF exists WHEN; assuming that with no additional concurrent failure during a design basis event, a required safety function assumed in the accident analysis CANNOT be performed.

4.5.6.5 If a LOSF is determined to exist, the appropriate Conditions and Required Actions of the LCO in which the LOSF exists SHALL be **entered**. IF no Condition within the LCO addresses the LOSF, THEN LCO 3.0.3 shall be entered. Results of the LOSF Evaluation SHOULD be **entered** in the Unit Control Log and/or by **initiation** of an LCO tracking sheet documenting the LCO in which the LOSF exists.

4.5.6.6 IF a LOSF does not exist, THEN the Required Actions for the LCO SUPPORT SYSTEM(s) address the condition AND the required actions of the SUPPORTED SYSTEM(s) do NOT have to be performed as permitted by LCO 3.0.6.

4.5.6.7 **Ensure** that the Completion Time of any SUPPORTED SYSTEM has not been inappropriately extended as shown in Figure 6. Completion Time Extensions are considered inappropriate if the SUPPORTED SYSTEM remains inoperable for longer than the allowed out of service time of the SUPPORT SYSTEM which caused the initial inoperability.

4.5.6.8 Technical Specification 3.5.2 Condition A, allows an ECCS train to be inoperable for up to 72 hours provided that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS Train remains available. Analyses have been performed for many of the potential flowpaths available that can be used to credit this allowance (reference 5.3). For cases where it is unclear if the flow equivalent of a single ECCS train remains operable, system engineering should be contacted for guidance.

LOSF EVALUATION FLOWCHART

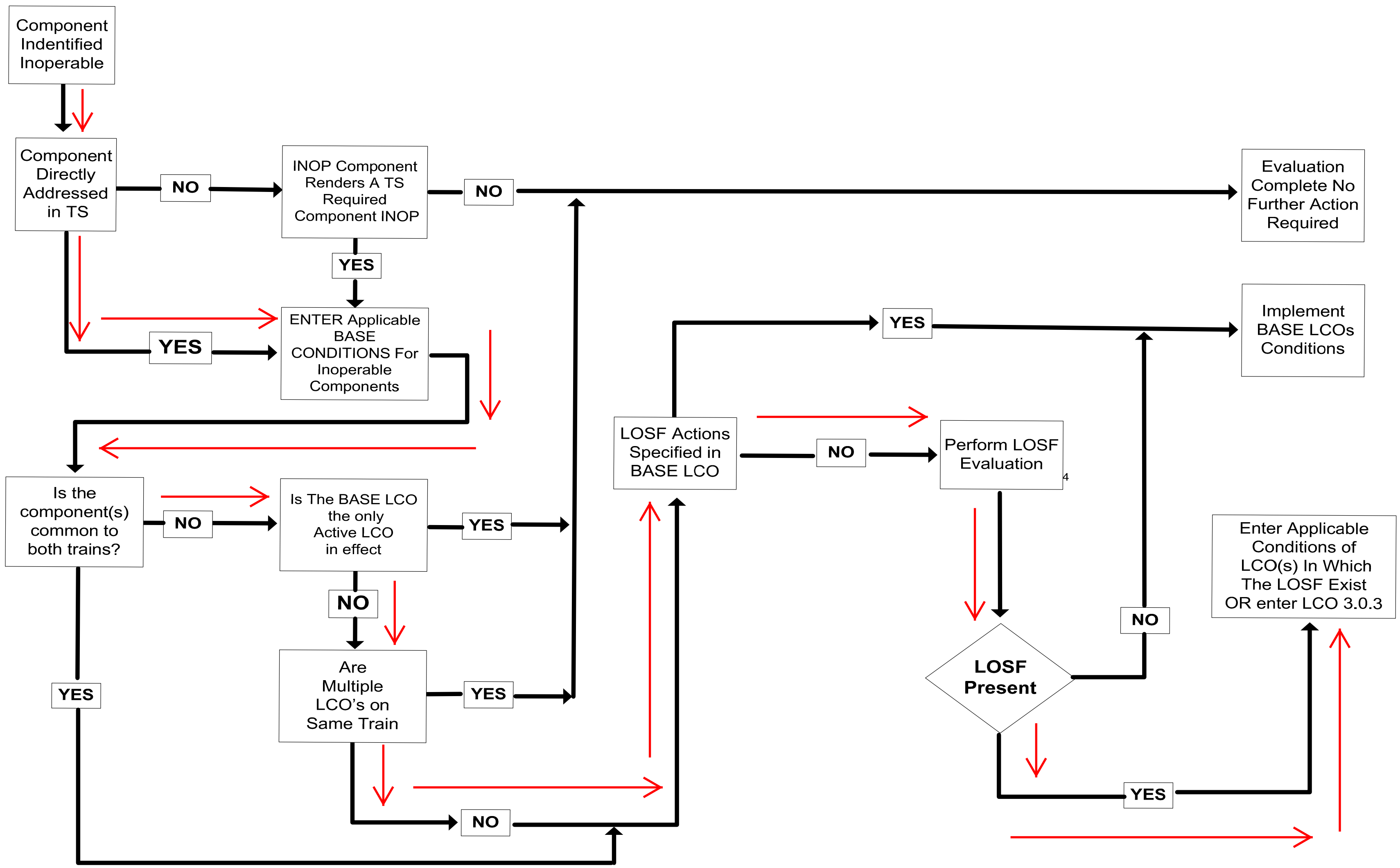


Figure 5

1. 076AG2.4.47 001/LOIT AND LOCT/SRO/C/A 4.2/4.2/076G2.4.47/LO-TA-63013///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- 18013-C, "Rapid Power Reduction," is entered due to a secondary transient.

- The following reactor power trends are recorded:

TIME	POWER
1100	100%
1115	97%
1130	93%
1145	89%
1200	86%
1215	81%
1230	79%

Which one of the following completes the following statement?

Chemistry sampling of the RCS __(1)__ required per Tech Spec 3.4.16, "RCS Specific Activity," Surveillance Requirements,

and

per the Bases of Tech Spec 3.4.16, "RCS Specific Activity," the required action to reduce RCS Tavg below 500°F if the gross specific activity is exceeded is to prevent opening of the __(2)__.

	__(1)__	__(2)__
A.	is	Atmospheric Relief Valves
B.✓	is	Main Steam Safety Valves
C.	is NOT	Atmospheric Relief Valves
D.	is NOT	Main Steam Safety Valves

K/A

076 **High Reactor Coolant Activity**

G2.4.47 **Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.**

K/A MATCH ANALYSIS

The question sets up a plausible scenario which includes all the required KA elements. First the SRO candidate must recognize in a timely manner the requirements for RCS sampling following the power reduction trend provided in the stem. Timely RCS activity sampling following a power reduction of more than 15% in one hour is required verify no fuel damage, which leads to high coolant activity. Then the candidate must determine the Tech Spec bases for lowering the energy level in the RCS if limits are exceeded.

EXPLANATION OF REQUIRED KNOWLEDGE

TS SR 3.4.16.2 requires verification of DOES EQUIVALENT I-131 specific activity less than or equal to 1.0 uCi/gm between 2 and 6 hours following a power change of greater than or equal to 15% RTP within a 1 hr period. By definition, the 1 hr period is a rolling hour. As stated in the stem, the power change between 1100 and 1200 is 14%RTP. The power change between 1115 and 1215 is 18%. Therefore, the 15% RTP in an hour has been exceeded.

Per TS 3.4.13, if I-131 is in excess of limits or LCO completion time of Cond A or B cannot be met, then the plant is required to be placed in Mode 3 with RCS Tavg <500F. Per TS 3.4.13 Bases, with RCS Tavg <500F, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valve lift setpoint.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Part 1 is correct the candidate should determine that from the stem information provided that a power reduction of more than 15% in one hour occured between 1115 and 1215 and require chemistry sampling of the RCS to verify no fuel damage.

Part 2 is incorrect however 'plausible' since the candidate may determine that the bases for the RCS temperature limit is to prevent the ARVs from lifting since they would normally open an a lower setpoint.

B. Correct. Part 1 is correct. See Part 1 of choice A above.

Part 2 is correct. Per Tech Spec 3.4.16 Bases, the purpose of lowering RCS temperature below 500°F is to prevent radioactive releases due to main stream safety valves lifting.

C. Incorrect. Plausible. Part 1 is incorrect however 'plausible' since the candidate may determine based on the time line given that RCS sampling is not required. The rolling hour starting at times 1100 and 1130 are <15% change.

Part 2 is incorrect. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice C above.

Part 2 is correct. See Part 2 of choice B above.

SRO JUSTIFICATION

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the question requires Bases knowledge.
-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" No, the applicability statement for TS 3.4.16 does state Mode 3 >500F Tavg. However, the Bases of this applicability and the Required Actions for both Conditon B & C are only listed in the Bases document.
-Can question be answered *solely* by knowing the TS Safety Limits? No, Safety Limits are not addressed by this question.
-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- **Knowledge of TS bases that is required to analyze TS required actions and terminology Yes, the question requires the candidate to specifically know the Bases for TS 3.4.13.**

Level:	SRO
Tier # / Group #	T1 / G2
K/A#	076G2.4.47
Importance Rating:	4.2 / 4.2

Technical Reference: TS 3.4.16, Ammendment No. 158, page 4.3.16-2
TS Bases 3.4.16, Rev 1-10/01, page B 3.4.16-3

References provided: None

Learning Objective:	LO-TA-63013	Implement Technical Specification LCO using 10008-C (SRO Only)
	LO-PP-16001-04	State the LCO, applicability, bases, and the 1 hr or less actions for each of the following: 3.4.16 RCS Specific Activity

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.5 / 43.2

Comments:

You have completed the test!

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	---------------------------NOTE---------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the exclusion area boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the exclusion area boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to exclusion area boundary doses that exceed the 10 CFR 100 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

1. G2.1.34 001/LOIT AND LOCT/SRO/M/F 2.7/3.5/G2.1.34/LO-TA-60024///

Given the following:

- Unit 1 is at 100% reactor power.

Which one of the following completes the following statement?

Per Tech Spec 3.7.16, "Secondary Specific Activity," the specific activity of the secondary coolant shall be ≤ __(1)__ μCi/gm Dose Equivalent I-131,

and

operating within this limit ensures that the off-site dose will be limited to within a small fraction of the 10 CFR 100 dose guideline values in the event of a __(2)__.

	__(1)__	__(2)__
A.✔	0.10	steam line break
B.	0.10	loss of all AC power
C.	1.0	steam line break
D.	1.0	loss of all AC power

K/A

G2.1.34 Knowledge of primary and secondary plant chemistry limits.

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of primary and secondary plant chemistry limits by requiring the student to select Tech Spec 3.7.16, "Secondary Specific Activity," specific activity limit for Dose Equivalent I-131 and the associated limiting design bases accident. This value is pitted against the limit for Tech Spec 3.4.16, "RCS Specific Activity".

EXPLANATION OF REQUIRED KNOWLEDGE

Per Tech Spec 3.7.16, "Secondary Specific Activity" bases, the accident analysis of the main steam line break (MSLB), assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits for whole body and thyroid dose rates.

Per Tech Spec 3.4.16, "RCS Specific Activity" bases, the limit of 1.0 μCi/gm DOSE EQUIVALENT I-131 on specific activity of ensures that the doses are held to a small fraction of the 10 CRF 100 limits during a steam generator tube rupture accident. The maximum dose to the whole body and the thyroid considers exposure to an individual at the exclusion boundary for 2 hours.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Part 1 is correct. Per Tech Spec 3.7.16, ‘Secondary Specific Activity’, the specific activity of the secondary coolant shall be < or = 0.10 μCi/gm DOSE EQUIVALENT I-131.

Part 2 is correct. Per Tech Spec 3.7.16, ‘Secondary Specific Activity’ bases, the accident analysis of the main steam line break (MSLB), assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits for whole body and thyroid dose rates.

B. Incorrect. Plausible. Part 1 is correct. See Part 1 of choice A above.

Part 2 is incorrect however ‘plausible’ since the candidate may

recall the LOSP as the most limiting accident not taking into account the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ARVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

C. Incorrect. Plausible. Part 1 is incorrect however ‘plausible’ since it’s reasonable to assume the candidate may confuse primary limit for specific activity with the secondary limit since the numbers are only distinguish by the movement of the decimal point.

Part 2 is correct. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice C above.

Part 2 is correct. See Part 2 of choice B above.

SRO JUSTIFICATION (10CFR43(b))

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the question is not addressing Tech Spec action times.
-Can question be answered *solely* by knowing the LCO/TRM information listed “above-the-line?” No, the question is not addressing Tech Spec above-the-line information.
-Can question be answered *solely* by knowing the TS Safety Limits? No, the question is not related to Tech Spec Safety Limits.
-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- **Knowledge of TS bases that is required to analyze TS required actions and terminology. Yes, the answer to the question is only found in Tech Spec bases.**

Level:	SRO	
Tier # / Group #	T3	
K/A#	G2.1.34	
Importance Rating:	2.7 / 3.5	
Technical Reference:	TS 3.4.16, Ammendment No. 137, page 3.4.16-1 TS 3.7.16, Ammendment No. 158, page 3.7.16-1 TS Bases 3.4.16, Rev 0, page B 3.4.16-1 TS Bases 3.7.16, Rev 1-10/01, page B 3.7.16-2	
References provided:	None	
Learning Objective:	LO-TA-63013	Implement Technical Specification LCO using 10008-C (SRO Only)
	LO-PP-16001-04	State the LCO, applicability, bases, and the 1 hr or less actions for each of the following: 3.4.16 RCS Specific Activity.
	LO-LP-39211-01	For any given item in section 3.7 of Tech Specs, be able to: a. State the LCO. b. State any one hour or less required actions.
	LO-TA-60024	Respond to Abnormal Plant Chemistry per 18014-C or 18015-C
Question origin:	NEW	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.5 / 41.10 / 43.2	
Comments:		

You have completed the test!

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

--NOTE--
LCO 3.0.4c is applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 µCi/gm.	A.1	Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	<u>AND</u>		
	A.2	Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1	Perform SR 3.4.16.2.	4 hours
	<u>AND</u>		
	B.2	Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

3.7 PLANT SYSTEMS

3.7.16 Secondary Specific Activity

LCO 3.7.16 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	In accordance with the Surveillance Frequency Control Program

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the exclusion area boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO limits specific activity for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the exclusion area boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes that the reactor has been operating at the maximum allowable Technical Specification limit for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ARVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ARVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner

(continued)

1. G2.1.37 001/LOIT AND LOCT/SRO/M/F 4.3/4.6/G2.1.37/LO-LP-63500-09///

Given the following:

- Unit 1 is at 3% reactor power and raising power following an outage.
- Unit 2 is at 100% reactor power.
- The Shift Manager and the following assigned personnel are in the control room:

Unit 1	**Unit 2**
Shift Supervisor	Shift Supervisor
Reactivity Management SRO	1 NPO*
2 NPOs*	

(*) NPO - Individual with a Reactor Operator License.

Which one of the following completes the following statement?

Per TRM 15.1, "Unit Staffing," the total number of NPOs assigned __(1)__ meet the minimum required for the given conditions,

and

per NMP-OS-001, "Reactivity Management Program," any changes to the Unit 1 **reactivity plan** must be approved by the __(2)__.

	__(1)__	__(2)__
A.	does	Reactivity Management SRO
B.✓	does	Shift Supervisor
C.	does NOT	Reactivity Management SRO
D.	does NOT	Shift Supervisor

K/A

Conduct of Operations

2.1.37 **Knowledge of procedures, guidelines, or limitations associated with reactivity management.**

K/A MATCH ANALYSIS

The question tests the candidate's knwoledge of administrative procedural requirements assoicated with the required approvals for changing reactivity plans during low power ascention following a reactor startup.

EXPLANATION OF REQUIRED KNOWLEDGE

The candidate is required to evaluate the minimum shift staffing for the given Modes per TRM 15.1 for the RO positions. Per Table 15.1.2-1, (3) ROs are required with both Units in Modes 1-4. (2) RO's must be assigned to the OATC position on each of the Units.

Per NMP-OS-001 step 6.1.2.1, the Shift Manager, who has ultimate responsibility for controlling the reactor core, approves formal Reactivity Management Plans as described in paragraph 6.6. The Shift Supervisor, or a designated Senior Reactor Operator, directly supervises reactivity changes. A written plan is developed by reactor engineering and approved for significant reactivity changes. The Shift Manager, Shift Supervisor, OATC, Shift Technical Advisor, and Reactor Engineer concur on Reactivity Management Plans and changes thereto, prior to implementation.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. Per TRM 15.1 Table 15.1.2-1, (3) ROs are required with both Units in Modes 1-4. (2) RO's must be assigned to the OATC position on each of the Units. This condition is met.

The second part is incorrect. Per NMP-OS-001 step 6.1.2.1, the Shift Manager, Shift Supervisor, OATC, Shift Technical Advisor, and Reactor Engineer concur on Reactivity Management Plans and changes thereto, prior to implementation. However, the Reactivity Management SRO has oversight of all reactivity changes. As such, it is reasonable for a candidate without specific knowledge of the procedural requirements to assume the Reactivity Management SRO would also have this authority. Therefore, this distractor is plausible.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. Per NMP-OS-001 step 6.1.2.1, the Shift Manager, Shift Supervisor, OATC, Shift Technical Advisor, and Reactor Engineer concur on Reactivity Management Plans and changes thereto, prior to implementation.

C. Incorrect. Plausible. The first part is incorrect. Per TRM 15.1 Table 15.1.2-1, (3) ROs are required with both Units in Modes 1-4. (2) RO's must be assigned to the OATC position on each of the Units. This condition is met. However, administrative procedure 00012-C ‘Shift Manning’ limits are not currently met. Per 00012-C, the ENN communication position is normally filled by each of the UOs which results in (4) NPOs being required. This is the normal shift alignment. It is reasonable for a candidate who does not have adequate knowledge of TRM 15.1 requirements to determine insufficient NPOs exist. Therefore, this distractor is plausible.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

SRO JUSTIFICATION (10CFR43(b))

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the question is not system related in any way.
-Can the question be answered *solely* by knowing immediate operator actions? No, the question does not involve any operator actions.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, the question does not involve an AOP or EOP.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, the question requires the knowledge of a specifc detail. Overall understanding of approvals actually directs the candidate to an incorrect answer.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- **Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. Yes, the question requires specific knowledge of the hierachy of an administrative procedure for approving Reactivity Plan changes once developed.**

(6) Procedures and limitations involved in initial core loading, alternations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Yes, the question requires the candidate to have specific knowledge of the approval process and responsibilities of control room personnel that approve Reactivity Plans and oversee Reactivity Manipulations.

Level:	SRO
Tier # / Group #	T3
K/A#	G2.1.37
Importance Rating:	4.3 / 4.6

Technical Reference: NMP-OS-001 Rev 17.0, page 9
00012-C Rev 17.2, pages 5&6
TRM 15.1, Table 15.1.2-1 Rev 0 12/26/96, page 15.1-2

References provided: None

Learning Objective:
LO-LP-36110-01 Per Technical Specifications, state the shift manning requirements. (SRO)
LO-LP-63503-02 State the requirements of the OATC with regards to shift manning when fuel is in either reactor.
LO-LP-63503-05 State the requirements of shift manning for the following conditions: (SRO only)
f. minimum shift crew
LO-LP-63510-04 Name the two site groups that have the most day-to-day effect on reactivity management.
LO-LP-63500-09 State reactivity manipulation expectations including: monitoring, briefing, pre-plans, peer checks, transient operation, pull-and-wait, load change concurrence, instrument response, changing temperature and power, and Reactivity Management SRO.

Question origin: BANK - Hatch 2011 NRC # G2.1.37

Cognitive Level: M/F

10 CFR Part 55 Content: 43.5 / 43.6

Comments:

You have completed the test!

Table 15.1.2-1

MINIMUM SHIFT CREW COMPOSITION
TWO UNITS WITH A COMMON CONTROL ROOM

Position	Number of Individuals Required to Fill Position		
	Both Units in MODE 1, 2, 3, or 4	Both Units in MODE 5 or 6 or DEFUELED	One Unit in MODE 1, 2, 3, or 4 and One Unit in MODE 5 or 6 or DEFUELED
SS	1	1	1
SRO	1	None(1)	1
RO	3(2)	2(2)	3(2)
NLO	3(2)	3(2)	3(2)
STA	1(3)	None	1(3)

(1) At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities must be present during CORE ALTERATIONS on either unit.

(2) At least one of the required individuals must be assigned to the designated position for each unit.

(3) See TS 5.2.2.g.

SS - Shift Superintendent with a Senior Operator License.
SRO - Individual with a Senior Operator License.
RO - Individual with an Operator License.
NLO - Non-licensed operator.
STA - Shift Technical Advisor.

DATA SHEET 1

Sheet 1 of 2

Minimum Shift Manning (Either Unit in Mode 1-4)

Date: ________________ Shift (Day/Night):_________________________

POSITION	UNIT #1	COMMON	UNIT #2
Shift Manager V-OPS-SS, V-ERO-CR01, and V-ERO-CR10		Also assigned as Emergency Director	
SS V-OPS-USS, V-ERO-CR02, AND V-ERO-CR10	Also assigned as ENS Communicator		Also assigned as ENS Communicator
OATC V-OPS-RO/BOP	1		2
UO V-OPS-RO/BOP and V-ERO-CR04	3 Also assigned as ENN Communicator		4 Also assigned as ENN Communicator
SO V-OPS-SO	SO/NPO		SO/NPO
STA (May be assigned other duties) V-OPS-STA		(SM, or SSS or SS not assigned to FB or ENN Communicator)	
Fire Team Captain V-FP-FIRE BRIGADE LEADER		SSS, or SS C&T	
FB Member V-FP-FIRE BRIGADE		1. SO(Also fulfills Common SO FSAR req)	
FB Member V-FP-FIRE BRIGADE		2. SO	
FB Member V-FP-FIRE BRIGADE		3. SO	
FB Member V-FP-FIRE BRIGADE		4. SO	
Security V-ERO-SEC or V-ERO-SEC02		Per Security Procedure 90101-C	
SAT Operator V-OPS-SO-OAO	Assigned per procedure 13419-C	5. SO/NPO/SRO	
Wilson Operator V-OPS-WILSON BLKSTRT	Assigned per procedure 13419-C	6. SO/NPO	

Emergency Plan

POSITION	UNIT #1	COMMON	UNIT #2
Emergency Director V-OPS-SS		Shift Manager	
ENN Communicator V-ERO-CR04 or V-ERO-CR10	UO Unaffected Unit		UO Unaffected Unit
ENS Communicator V-OPS-USS or V-OPS-STA	SS Unaffected Unit		SS Unaffected Unit

DATA SHEET 1
Emergency Plan (cont)

Sheet 2 of 2

POSITION	UNIT #1	COMMON	UNIT #2
Dose Assessment V-ERO-TSC09 OR V-ERO-TSC10		HP Foreman	
Field Monitoring Team (FMT) V-ERO-CR08 OR V-ERO-OSC16		1. HP Tech/Chem Tech/SO/I&C Tech 2. HP Tech/Chem Tech/SO/I&C Tech	
FMT Communicator V-ERO-TSC18		Chem Foreman/Chem Tech/HP Tech/ Maint. Shift ATL	
Chemistry Sampler V-ERO-OSC09		Chemistry Tech	
Mechanical Repair and Corrective Action V-ERO-OSC07		Mechanic	
Electrical Repair and Corrective Action V-ERO-OSC06		Electrician	
I & C Repair and Corrective Action V-ERO-OSC08		I & C Technician	
In Plant Monitors V-ERO-OSC17		1. HP Technician 2. HP Technician	
Search & Rescue/First Aid (May be assigned other Duties) V-ERO-OSC15		1. HP Technician 2. HP Technician	

Minimum Dual Unit Safe Shutdown

POSITION	UNIT #1	COMMON	UNIT #2
Emergency Director		SM	
ENN		UO Unit #1 or #2	
ENS		UO Unit #2 or #1	
Shutdown Panel "B"	SS		SS
Shutdown Panel "A"	OATC		OATC
Shutdown Panel "C"	SO		SO
Fire Brigade		Same as Normal Operations	

1. Personnel may NOT be assigned to more than one position unless specifically noted next to the position label.
2. If both units are in Modes 5, 6, or defueled, minimum shift manning for operations may be reduced per Operations Manager (not Emergency Plan or Fire Brigade staffing).

COMMENTS: ____

____

Approved by: ______________________ Date: __________ Time: __________
Shift Manager

6.1.2 Expectations

6.1.2.1 Reactivity Management Controls

NOTE:	**<u>Prior</u> to implementing any activity that has the potential to add positive reactivity; the plant shall be ramped down, as necessary, to ensure that the reactor does not exceed 100.0% of the unit's Rated Thermal Power limit.**

Reactivity management involves a systematic process of controlling evolutions with the potential to affect reactivity:

- Planned reactivity changes are conducted in a controlled and conservative manner
- Unexpected changes in reactivity are minimized
- Conservative actions are taken in response to unexpected reactivity changes
- Reactivity control systems, including reactivity monitoring instrumentation, are available and reliable
- Modifications, analyses, and predictions are correct and effectively implemented

The Shift Manager, who has ultimate responsibility for controlling the reactor core, approves formal Reactivity Management Plans as described in paragraph 6.6. The Shift Supervisor, or a designated Senior Reactor Operator, directly supervises reactivity changes.

A strong relationship exists between reactor engineering and operations. Reactor engineering is actively engaged in the planning of significant reactivity changes such as reactor startup, reactor shutdown, planned power changes, and special tests with the potential to affect reactivity. A written plan is developed by reactor engineering and approved for significant reactivity changes. The Shift Manager, Shift Supervisor, OATC, Shift Technical Advisor, and Reactor Engineer concur on Reactivity Management Plans and changes thereto, prior to implementation.

6.1.2.2 Control Room Operations

Only operators with an active license (NPO or SRO) manipulate the controls of the reactor. An individual in a NRC-approved license training program may manipulate reactor controls when under the direction and in the presence of a licensed operator.

The OATC obtains concurrence from the Shift Supervisor prior to allowing or performing planned reactivity manipulations. Directions that affect reactivity go through the Shift Supervisor.

A briefing is conducted prior to the start of a planned reactivity change. The reactor operator performing rod movement, the OATC and Shift Supervisor monitor reactivity manipulations and verify that the end state of the reactivity manipulation is as expected.

1. G2.2.20 001/LOCT AND LOIT/SRO/M/F 2.6/3.8/G2.2.20/LO-LP-63350-07//HL18 NRC/

Initial condition:

- ALB34-D01 125 VDC SWGR 1AD1 TROUBLE is received due to a bus ground.

Current conditions:

- Per NMP-AD-002, "'Problem Solving and Troubleshooting Guidelines," a troubleshooting plan has been written.

- As part of the plan, operations personnel will open various breakers and maintenance personnel will open links to measure resistance inside the 1AD1 panel.

Which one of the following completes the following statement?

This type of Troubleshooting Monitoring is called __(1)__,

and

the tracking of the equipment out-of-service time while troubleshooting 1AD1 is the responsibility of the __(2)__ Department.

A. (1) Intrusive

(2) Maintenance

B.✔ (1) Intrusive

(2) Operations

C. (1) Non-Intrusive

(2) Maintenance

D. (1) Non-Intrusive

(2) Operations

G2.2.20 Equipment Control

Knowledge of the process for managing troubleshooting activities.

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where Troubleshooting is to be performed in the 1AD1 Panel by both maintenance personnel and operations personnel. The activity involves opening links, breakers and measuring resistance to

resolve a ground problem. The candidate must determine if this is intrusive or non-intrusive trouble shooting and also has to determine the person responsible for maintaining the system status of the panel during Trouble Shooting activities.

EXPLANATION OF REQUIRED KNOWLEDGE

Per NMP-AD-002 definition 3.5, non-intrusive monitoring is defined as the act of monitoring a component or system by not affecting normal oepration of the conponent or system.

Per NMP-AD-002 definition 3.6, intrusive monitoring is defined as the act of temporarily altering the system to allow monitoring a component or system. This applies to electrical or mechanical testing methods.

Per NMP-AD-002 responsibility 4.2, the Operation Department is responsible for maintaining approved system status during troubleshooting activities (ie Out of Service).

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. Plausible. Part 1 is correct. Opening breakers by Operations and opening links by Maintenance is an intrusive troubleshooting per NMP-AD-002.

Part 2 is incorrect. Operations Department is responsible for tracking systems status, however the answer is 'plusable' because opening links, measuring resistance, etc. would all be maintenance activities and some amount of individual component status control is required. These components are typically tracked using a "lifted lead" sheet inside the MWO. Even though maintenance tracks status of some components, Operations still retains overall responsibility for ensuring equipment status and configuration control.

B. Correct. Part 1 is correct. See Part 1 of choice A above.

Part 2 is correct. Per NMP-AD-002 responsibility 4.2, the Operation Department is responsible for maintaining approved system status during troubleshooting activities.

C. Incorrect. Plausible. Part 1 is incorrect however 'plusable' because the word intrusive is very subjective and would require procedure knowledge to make this distinction. The candidate may determine that allowing intrusive troubleshooting on 1AD1 with the unit on line would imply too much risk to operations and thus eliminate this possibility.

Part 2 is incorrect. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice C above.

Part 2 is correct. See Part 2 of choice B above.

<u>SRO JUSTIFICATION</u>

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, system knowledge will not answer any part of this question.
-Can the question be answered *solely* by knowing immediate operator actions? No, IOAs are not addressed in any way in this question.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, this question does not pertain to an EOP or AOP.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, the question pertains to specific guidance in NMP-AD-002, which cannot be answered by broad knowledge of the procedure.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- **Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures Yes, the question requires specific knowledge of an administrative procedure that specifies the hiearchy by which plant configuration is statused and configuration control is maintained during normal plant operation.**

Level:	SRO
Tier # / Group #	T3
K/A#	G2.2.20
Importance Rating:	2.6 / 3.8

Technical Reference: NMP-AD-002, Rev 10.0, pages 4 & 5

References provided: None

Learning Objective: LO-LP-63350-07 Define the following terms:
f. Trouble shooting
LO-LP-63354-03 Describe the Shift Manager's responsibility concerning maintenance activities.

Question origin: BANK - HL18 NRC # G.2.2.20

Cognitive Level: M/F

10 CFR Part 55 Content: 41.10 / 43.5

Comments:

You have completed the test!

1.0 Purpose

The purpose of this procedure is to provide a process for performance of troubleshooting when required for plant problem resolution. These problems may include equipment failures, abnormal operating conditions, negative performance trends or recurring events.

2.0 Applicability

2.1 This procedure is applicable to troubleshooting activities at any of the SNC sites.

2.2 Entry into the formal troubleshooting process is not intended for simple problems where the cause appears straightforward or known. In these cases investigation will be controlled by the Work Order process.

2.3 Formal troubleshooting activities shall be performed in accordance with this procedure unless waived by plant management or management within the affected department. If waived, the justification shall be documented in the appropriate location (Condition Report, Work Order, etc).

2.4 This procedure does not apply to special tests.

2.5 All troubleshooting shall use high impedance M&TE and/or isolation transformers on the signal and AC power source unless low impedance is specifically called for in equipment procedure.

3.0 Definitions

3.1 **Troubleshooting** – A systematic approach to data collection, failure analysis, or a measurement plan that results in high confidence that the complete cause of system/equipment degradation has been determined. There may be potential personnel safety risk.

3.2 **High Risk Troubleshooting** - Potential impacts are assessed as high risk when evaluated per NMP-DP-001, Operational Risk Awareness.

3.3 **Medium Risk Troubleshooting** - Potential impacts are assessed as medium risk when evaluated per NMP-DP-001, Operational Risk Awareness.

3.4 **Low Risk Troubleshooting** - Potential impacts are assessed as low risk when evaluated per NMP-DP-001, Operational Risk Awareness.

3.5 **Non-Intrusive Monitoring** – The act of monitoring a component or system by not affecting normal operation of the component or system. Examples would be using "Voltage Test Jacks," monitoring voltages across relay contacts, power supplies, etc.

3.6 **Intrusive Monitoring** – The act of temporarily altering the system to allow monitoring a component or system. This applies to electrical or mechanical testing methods.

3.7 **Stop-Decision Points** – Administrative and Physical Hold Points within the Troubleshooting Plan to limit and control activities.

4.0 Responsibilities

4.1 **Troubleshooting Leader**- Individual assigned to develop the troubleshooting plan, coordinate work and team discussions, act as a single point of contact and/or obtain changes to the plan, as assigned by the responsible department manager. Perform Just In Time Risk Assessment as described in NMP-DP-001, Operational Risk Awareness.

4.2 **Operations Department**

The Operations Department, under the direction of the Operations Manager, is responsible for ensuring the troubleshooting activities are supported by:

- Approve the Troubleshooting plan where risk has been assessed as Medium or High
- Providing personnel in support of the Troubleshooting Team
- Maintain approved system status during troubleshooting activities (i.e. Out Of Service)

4.3 **Work Planning**

- The group developing the Troubleshooting plan will determine the level of risk associated with the Troubleshooting plan by using Procedure NMP-DP-001, Operational Risk Awareness.

- The plan should consider elimination of worst case, long lead time components early in the process, as potential causes.

- Troubleshooting plan steps that alter the configuration of the plant will be implemented and controlled by a planned work order or use of referenced instructions from an approved procedure. This requirement may be waived by the operations shift manager in which case configuration changes will be controlled using detailed instructions in the troubleshooting plan and will be approved by operations before implementation.

- Troubleshooting plan steps that do not affect plant configuration control, for example system walkdown, data collection and trending, field observation, and other similar fact finding steps may be implemented by the troubleshooting plan steps.

4.4 **Maintenance Department**

The Maintenance Department, under the direction of the Maintenance Manager, is responsible for:

- Maintenance Manager or his designee will approve High or medium risk Troubleshooting activities where personal safety or economic safety are assessed as Medium or High risk
- Providing personnel in support of the Troubleshooting Team
- Ensuring that Troubleshooting Plan is performed and documented in accordance with approved site procedures and safe work practices
- Determining the need for additional support for troubleshooting activities

1. G2.2.25 001/LOIT AND LOCT/SRO/M/F 3.2/4.2/G2.2.25/LO-LP-39209-02//HL-17 AUDIT/

Given the following:

- Unit 1 is at 100% reactor power.
- The following RWST parameters are recorded:

  Temperature is 47°F.
  Level is 93%.

Which one of the following completes the following statement?

Tech Spec action is required for RWST __(1)__,

and

the Tech Spec Basis for this parameter limit is to ensure __(2)__.

A. (1) Level

(2) sufficient borated water to support the ECCS during the injection phase of a design basis **main steam line break**

B.✔ (1) Level

(2) sufficient borated water to support the ECCS during the injection phase of a design basis **loss of coolant accident**

C. (1) Temperature

(2) that the amount of cooling provided from the RWST during the heatup phase of a **main steam line break** is consistent with safety analysis assumptions

D. (1) Temperature

(2) that the amount of cooling provided from the RWST during the heatup phase of a **main feed line break** is consistent with safety analysis assumptions

K/A

2.2.25 **Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.**

K/A MATCH ANALYSIS

The question asks the candidate straight forward and RO level question of which parameter will place the unit in a 1 hour shutdown LCO, RWST level or temperature. Once determining RWST level is the 1 hour shutdown, the SRO portion requires the candidate to know the correct bases for the RWST level.

EXPLANATION OF REQUIRED KNOWLEDGE

Per TS SR 3.5.4.1, the RWST borated water temperature must be maintained greater than or equal to 44°F and less than or equal to 116°F. Per TS 3.5.4 Bases, the maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with the safety analysis assumptions. The minimum temperature is an assumption in both the MSLB and inadvertent ECCS actuation. The inadvertent ECCS actuation is typically non-limiting.

Per TS SR 3.5.4.1, the RWST borated water must be maintained greater than or equal to 686,000 gallons. Per Tech Spec rounds OSP 14000-1, 686,000 gallons corresponds to an indicated level of 94%. Per TS 3.5.4 Bases, the RWST volume is an explicit assumption for LOCA events. The desired volume limit is set by LOCA and containment analyses. The volume is not an explicit assumption for non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be maintained than can be delivered.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. 93% level will place the unit in a 1 hour shutdown.

The second part is incorrect. RWST level ensures sufficient injection volume for a DBA LOCA, not a DBA MSLB. However, Safety Injection is required to mitigate a main steam line break and requires suction from the RWST, but is not a limiting factor since the required volume would be a small fraction of that available.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. Per TS 3.5.4 Bases, the RWST volume is an explicit assumption for LOCA events

C. Incorrect.Plausible. The first part is incorrect. Per TS SR 3.5.4.1, the RWST borated water temperature must be maintained greater than or equal to 44°F and less than or equal to 116°F.

The second part is incorrect. Per TS 3.5.4 Bases, the maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with the safety analysis assumptions, not a main steamline break.

D. Incorrect.Plausible. The first part is incorrect. See the first part of choice C above.

The second part is the correct for the first part. Per TS 3.5.4 Bases, the maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a

feedline break is consistent with the safety analysis assumptions, not a main steamline break.

<u>SRO-ONLY JUSTIFICATION</u>

(2) Facility operating limitations in the technical specifications and their bases.

-Can question be answered *solely* by knowing = 1 hour TS/TRM Action? No, the question requires knowledge of the TS Bases for the limit.
-Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" No, the question is not related to above-the-line Tech Spec. The information is found in a survelliance and TS Bases.
-Can question be answered *solely* by knowing the TS Safety Limits? No, the question is not related to any Tech Spec Safety Limit.
-Does the question involve one or more of the following for TS,TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- **Knowledge of TS bases that is required to analyze TS required actions and terminology. Yes, the question asks the bases for a Tech Spec survelliance.**

Level:	SRO
Tier # / Group #	T3
K/A#	G2.2.25
Importance Rating:	3.2 / 4.2

Technical Reference: TS 3.5.4, Ammendment No. 96, page 3.5.4-1 & 2
TS Bases 3.5.4, Rev 0, pages B3.5.4-3 & 4
OSP 14000-1, Rev 88.1, pages 8 & 15

References provided: None

Learning Objective:

LO-LP-39209-01 For any given item in section 3.5 of Tech Specs, be able to:

a. State the LCO.
b. State any one hour or less required actions.

LO-LP-39209-02 Given a set of the Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.5 are exceeded.
b. The required actions for all section 3.5 LCOs.

LO-LP-39209-03 Describe the bases for any given Tech Spec in section 3.5.

Question origin: BANK - LOIT Question # 006G2.2.39 001

Cognitive Level: M/F

10 CFR Part 55 Content: 41.8 / 43.2

Comments:

You have completed the test!

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. <u>OR</u> RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. One or more sludge mixing pump isolation valves inoperable.	B.1 Restore the valve(s) to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Isolate the sludge mixing system.	6 hours
D. RWST inoperable for reasons other than Condition A or B.	D.1 Restore RWST to OPERABLE status.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	-------------------------NOTE----------------------------- Only required to be performed when ambient air temperature is < 40°F. --- Verify RWST borated water temperature is ≥ 44°F and ≤ 116°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is ≥ 686,000 gallons. Level specified by % in TS Rounds.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2400 ppm and ≤ 2600 ppm.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.4	Verify each sludge mixing pump isolation valve automatically closes on an actual or simulated RWST Low-Level signal.	In accordance with the Surveillance Frequency Control Program

BASES

BACKGROUND (continued)

reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS — Operating"; B 3.5.3, "ECCS — Shutdown"; and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 39 seconds without offsite power (includes 12 seconds for the Emergency Diesel Generator). This response time includes an electronics delay, a stroke time for the RWST valves, and a stroke time for the VCT valves.

For a large break LOCA analysis, the minimum water volume limit of 499,091 gallons and the lower boron concentration limit of 2400 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2600 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 44°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. (The reduction in containment pressure correspondingly reduces the density of the vented steam. This reduces the flow of steam out of the core, which translates into a decrease in the ECCS flooding rate. This decrease in the flooding rate causes the increase in peak clad temperature.) The upper temperature limit of 116°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

(continued)

Sheet 2 of 10

DATA SHEET 1
MODE 1 & 2

MODE _______________

DATE _______________

LCO METHOD OF VERIFICATION	TECH SPEC SURV REQ	PARAMETER	INSTRUMENT	INDICATION DAY	INDICATION NIGHT	LIMIT(S) TOLERANCE	LCO/PROC
EACH ACCUMULATOR SHALL BE OPERABLE VERIFY DISCHARGE VALVE POSITION	SR 3.5.1.1	VALVE POSITION (INIT)	1HS-8808A MLB001 2.3			OPEN	3.5.1
			1HS-8808B MLB002 2.3				
			1HS-8808C MLB001 2.4				
			1HS-8808D MLB002 2.4				
TWO ECCS FLOW TRAINS SHALL BE OPERABLE VERIFY VALVES POSITIONED AND POWER REMOVED BY ASSOCIATED LOCKOUT SWITCH LIGHT EXTINGUISHED AND SWITCH IN LOCKOUT POSITION	SR 3.5.2.1	VALVE STATUS (INIT)	1HS-8806			OPEN AND POWER REMOVED	3.5.2
			1HS-8835			OPEN AND POWER REMOVED	
			1HS-8813			OPEN AND POWER REMOVED	
			1HS-8802A			CLOSED AND POWER REMOVED	
			1HS-8802B			CLOSED AND POWER REMOVED	
			1HS-8840			CLOSED AND POWER REMOVED	
			1HS-8809A			OPEN AND POWER REMOVED	
			1HS-8809B			OPEN AND POWER REMOVED	
ESFAS INSTRUMENTATION SHALL BE OPERABLE CHANNEL CHECK	SR 3.3.2.1 FCN 7B	RWST LEVEL (%)	1LI-0991A			CHANNEL CHECK	
			1LI-0993A			REQUIRED 4	3.3.2(K)
ACCIDENT MONITORING INSTRUMENTATION SHALL BE OPERABLE	SR 3.3.3.1 FCN 9		1LI-0990A			REQUIRED 2	3.3.3 (B,G,H,J)
CHANNEL CHECK	SR 3.5.4.2		1LI-0992A			≥94%	3.5.4

COMPLETED BY: DAY: _______________ TIME: ________ NIGHT: ________________ TIME: ________

SS REVIEW: DAY: _______________ TIME: ________ NIGHT: ________________ TIME: ________

Sheet 9 of 10

DATA SHEET 1
MODE 1 & 2

MODE ______________

DATE ______________

LCO METHOD OF VERIFICATION	TECH SPEC SURV REQ	PARAMETER	INSTRUMENT	INDICATION DAY	INDICATION NIGHT	LIMIT(S) TOLERANCE	LCO/PROC
CREFS ACTUATION OPERABLE CHANNEL CHECK	SR 3.3.7.1 FCN 3	CR INTAKE RADIATION MONITORS (INIT)	1RE-12116			CHANNEL CHECK REQUIRED 2	3.3.7
			1RE-12117				
FHB ACTUATION OPERABLE CHANNEL CHECK	TRS 13.3.6.1	FHB EFFL RADIOGAS	ARE-2532A			*	13.3.6
		FHB ISO (INIT)	ARE-2532B			REQUIRED 1	
FHB ACTUATION OPERABLE CHANNEL CHECK	TRS 13.3.6.1	FHB EFFL RADIOGAS	ARE-2533A			*	13.3.6
		FHB ISO (INIT)	ARE-2533B			REQUIRED 1	
		*INDICATING NORMALLY. ALL STATUS AND ALARM LIGHTS EXTINGUISHED.					
DG1A FUEL OIL INVENTORY VERIFY FUEL OIL STORAGE TANK LEVEL	SR 3.8.3.1	DG 1A LEVEL (%)	1-LI-9024			≥82%	3.8.3
DG1B FUEL OIL INVENTORY VERIFY FUEL OIL STORAGE TANK LEVEL	SR 3.8.3.1	DG 1B LEVEL (%)	1-LI-9025			≥ 82%	3.8.3
TWO INDEPENDENT CONTROL ROOM EMERGENCY FILTRATION SYSTEMS SHALL BE OPERABLE VERIFY CONTROL ROOM TEMP	SR 3.7.10.1 SR 3.7.11.1	NOTE: TEMPERATURE INDICATION IS OBTAINED FROM HAND-HELD TEST EQUIPMENT. RECORD INSTRUMENT INFORMATION BELOW.					
		INSTRUMENT ID NO.				N/A	
		CAL DUE DATE					
		CONTROL ROOM TEMPERATURE (°F)	M&TE			≤85°F	3.7.10 3.7.11
THE RWST SHALL BE OPERABLE VERIFY TEMPERATURE	SR 3.5.4.1 TRS 13.1.7.1	RWST TEMPERATURE (°F)	1TIS-10980			≥51°F * ≤109°F *	3.5.4 13.1.7
		*WITH INDICATED RWST TEMPERATURE OUTSIDE THE LIMITS, THEN VERIFY RWST TEMPERATURE IS WITHIN TECHNICAL SPECIFICATION LIMITS BY PLACING THE RWST ON RECIRC USING SLUDGE MIXING PUMP WITH HEATER OFF AND OBSERVING 1-TI-10982 TO BE WITHIN ≥44°F AND ≤116°F.					
THE ULTIMATE HEAT SINK SHALL BE OPERABLE VERIFY WATER TEMPERATURE AND LEVEL	SR 3.7.9.2	TEMPERATURE (°F)	COMPUTER POINT T2601*			≤90°F	3.7.9
			-OR-				
			1TJI-1692 POINT 2*				
			COMPUTER POINT T2602*				
			-OR-				
			1TJI-1692 POINT 17*				
		*IF COMPUTER POINT AND RECORDER POINT ARE NOT AVAILABLE, TEMPERATURE READING MUST BE OBTAINED LOCALLY USING HAND-HELD TEST EQUIPMENT. RECORD INSTRUMENT INFORMATION BELOW.					
		INSTRUMENT ID NO.				N/A	
		CAL DUE DATE					
	SR 3.7.9.1	LEVEL (%)	1LI-1606			≥73%	
			1LI-1607				
CONTAINMENT AIR TEMPERATURE SHALL NOT EXCEED 120°F VERIFY AVERAGE AIR TEMPERATURE	SR 3.6.5.1	TEMPERATURE (°F)	COMPUTER POINT T2501			NA	3.6.5
			COMPUTER POINT T2502				
			COMPUTER POINT T2503				
			COMPUTER POINT UT2501 (AVG)			≤120°F	
		*IF COMPUTER POINT IS NOT AVAILABLE VERIFY CNMT HI TEMP ALARM ALB-01 (E06) IS NOT IN ALARM.				ALB-01 (E06) NOT IN ALARM	
		*IF COMPUTER POINT AND ALB-01 (E06) ARE NOT AVAILABLE, TEMPERATURE READING MUST BE OBTAINED LOCALLY USING HAND-HELD TEST EQUIPMENT FOR 1TE-2612 FOR POINT T2502 AND 1TE-2613. FOR POINT T2503 RECORD INSTRUMENT INFORMATION BELOW. USE MCB INDICATOR 1TI-2563 FOR POINT T2501 AND AVERAGE THE THREE.					
		INSTRUMENT ID NO.				≤120°F	
		CAL DUE DATE					

COMPLETED BY: DAY: ______________ TIME: _______ NIGHT: ______________ TIME: _______

SS REVIEW: DAY: ______________ TIME: _______ NIGHT: ______________ TIME: _______

1. G2.3.4 001/LOCT AND LOIT/SRO/M/F 3.2/3.7/G2.3.4/LO-LP-40101-08///

Initial condition:

- General Emergency has been declared.

Current conditions:

- A first responder is briefed to rescue an injured worker.

- Health Physics estimates the first responder will receive 11 rem TEDE dose while performing the rescue.

Which one of the following completes the following statement?

Per 91301-C, "Emergency Exposure Guidelines," the dose received by the first responder during the rescue __(1)__ be added to the responder's occupational non-emergency exposure,

and

the LOWEST level of approval required to authorize the first responder's rescue exposure is the __(2)__.

	__(1)__	__(2)__
A.	will	Health Physics Supervisor
B.	will	Emergency Director
C.	will NOT	Health Physics Supervisor
D.✓	will NOT	Emergency Director

K/A

2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

K/A MATCH ANALYSIS

The question sets up a plausible scenario which includes all the required KA elements. First the SRO candidate must determine if the exposure in the General Emergency would be added to normal exposure already accumulated and the authorization level required for the given dose.

EXPLANATION OF REQUIRED KNOWLEDGE

Per Admin procedure 91301-C, "Emergency Exposure Guidelines" NOTE on TABLE 1, dose to workers performing emergency services may be treated as a once-in-a-lifetime

exposure and should not be added to occupational exposure accumulated under non-emergency conditions. Per Responsibilities 2.1, the Emergency Director (ED) has the sole authority to allow radiation exposures in excess of 10CFR20 limits. Per Responsibilities 2.2.4, the HP Supervisor or designedd can authorize individuals to recieve radiation exposures in excess of VEGP Admin Guidelines, but not in excess of 10CFR20 limits.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Part 1 is incorrect however 'plausible' since it's reasonable to assume the candidate may determine that all radiation exposure for the year would be additive to ensure accurate accounting for health reasons.

Part 2 is incorrect however 'plausible' since the candidate may determine, based on plant conditions, that with the exposure less than 25 Rem, that Emergency Director involvement would not be required and therefore the Health Physics Supervisor could authorizes this.

B. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice A above.

Part 2 is correct. Per 91301-C 'EMERGENCY EXPOSURE GUIDELINES', the Emergency Director (ED) has the sole authority to allow radiation exposures in excess of 10CFR20 limits.

C. Incorrect. Plausible. Part 1 is correct. Per 91301-C 'EMERGENCY EXPOSURE GUIDELINES', dose to workers performing emergency services may be treated as an once-in-a-lifetime exposure and should not be added to occupational exposure accumulated under non-emergency conditions.

Part 2 is incorrect. See Part 2 of choice A above.

D. Correct. Part 1 is correct. See Part 1 of choice C above..

Part 2 is correct. See part 2 of choice B abvoe.

SRO JUSTIFICATION (10CFR43(b))

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the knowledge required pertains to administrative guidance and ED non deligable duties.
-Can the question be answered *solely* by knowing immediate operator actions? No, information found in IOAs is not involved in the question.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, EOPs and AOPs

are not involved with this questioon.

-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, detailed knowledge of process and responsibilities within an admin procedure are required. Overall knowledge will not answer the question.

-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- **Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. Yes, specific knowledge of the duties of the ED and HP Supervisor as well as how emergency exposures are recorded for dose purposes is required.**

Level:	SRO	
Tier # / Group #	T3	
K/A#	G2.3.4	
Importance Rating:	3.2 / 3.7	
Technical Reference:	ADMIN 91301-C, Rev 12.1, pages 3 & 9	
References provided:	None	
Learning Objective:	LO-LP-40101-35	State what group of people should be first considered for emergency exposure, and what group should not be allowed to receive an emergency exposure (91301-C). (SRO only)
	LO-LP-40101-08	State from memory ED duties that cannot be delegated (SRO only).
Question origin:	MODIFIED - Turkey Point 2011 NRC Question # G.2.3.4	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	43.5	
Comments:		

You have completed the test!

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G3	2.3.4
	Importance Rating		3.7

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: SRO Question # 98

- The Shift Manager is the Emergency Coordinator.
- Medical Response Workers are briefed to rescue a worker who has suffered a broken leg, non-life threatening condition, during a radiological emergency.
- The worker is located where general area dose rates are 125 mrem/hr.
- The Medical Response Workers will pass through high dose rate areas to reach the worker.
- Health Physics estimates each worker will receive a total dose (TEDE) of 12 rem and a thyroid dose of 3.5 rem while performing this rescue.

Which ONE of the following correctly describes (1) whose approval is required to exceed dose in excess of the Annual Federal Limits and (2) the HP estimated dose (from above) as compared to the TEDE limit of 0-EPIP-20111, Re-entry?

A. (1) the Emergency Coordinator (EC) approval
(2) is within the TEDE limit

B. (1) the OSC Health Physics Supervisor approval
(2) is within the TEDE limit

C. (1) the Emergency Coordinator (EC) approval
(2) exceeds the TEDE limit

D. (1) the OSC Health Physics Supervisor approval
(2) exceeds the TEDE limit

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The EC is the correct position for authorization. This is plausible because the estimated total dose (12 rem) exceeds its limit (10 rem).

B. Incorrect. The OSC Health Physics Supervisor is the incorrect position for authorization. However, they will give radiological briefings, will issue emergency dosimetry with maximum exposure capabilities, will ensure the Nuclear Division Medical Review Officer notification, and will sign emergency exposure authorization form. Also, this is plausible because the estimated total dose (12 rem) exceeds its limit (10 rem).

C. Correct because the estimated total dose (12 rem) exceeds its limit (10 rem)

D. Incorrect. The OSC Health Physics Supervisor is the incorrect position for authorization. However, they will give radiological briefings, will issue emergency dosimetry with maximum exposure capabilities, will ensure the Nuclear Division Medical Review Officer notification, and will sign emergency exposure authorization form. Also, this is plausible because the estimated total dose (12 rem) exceeds its limit (10 rem).

Technical Reference(s): 0-EPIP-20111 Steps including Enclosure 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 3200001 Obj. 10 (As available)

Question Source: Bank #

Modified Bank # WTSI 66661 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty: Moderate (C)

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

Modified from Turkey Point 2009 NRC Exam. Changed dose rates to change correct answer.

SRO only because a decision must be made to minimize radiation exposure with an injured person in a high radiation area. This decision would be made by the SRO acting as the E-Plan Emergency Coordinator

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

[10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include:

- Process for gaseous/liquid release approvals, i.e., release permits.
- Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

Analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits.

SRO-only knowledge should not be claimed for questions that can be answered *solely* based on RO knowledge of radiological safety principles; e.g., RWP requirements, stay-time, DAC-hours, etc.

5.0 PROCEDURE

5.1 General

5.1.1 The following guidelines for emergency exposure of personnel shall be followed during the re-entry operation:

1. Re-entry personnel that have been authorized to exceed regulatory exposure limits should be volunteers, familiar with the risks involved (radiosensitivity of fetuses, effects of acute exposures, etc.), and whose normal duties have trained them for such missions.

2. Declared pregnant adults should not be used as on-site emergency workers.

3. Exposures to emergency workers shall be maintained as low as reasonably achievable (ALARA) and if possible be maintained within site specific radiological exposure guidelines and/or limits identified in 10 CFR 20.

4. Conditions may warrant re-entry into high radiation areas leading to exposure in excess of the regulatory limit. Except for rescue of personnel from a life threatening situation, authorization must be given in advance by the Emergency Coordinator (EC) in consultation with the TSC RP Supervisor (or alternate). If the EOF is operational and as time permits, the EC should obtain concurrence from the Recovery Manager (RM). In any case where regulatory limits have been exceeded, the EC shall notify the RM of the event.

5. If obtaining EC approval for exposure in excess of the regulatory limit will result in leaving the accident scene or decrease the victim(s) chance of survival, life-saving actions may be performed without obtaining EC approval. The EC shall be notified immediately following the rescue operation.

6. Emergency exposures requiring immediate action are not planned and are not controlled as a Planned Special Exposure. Dose received from exposure under emergency conditions will be added to the dose received during the current year, prior to the emergency, to determine compliance with the occupational dose limits in 10 CFR 20.

ENCLOSURE 1
(Page 1 of 3)

EMERGENCY WORKER EXPOSURE LIMITS AND GUIDANCE FOR POTASSIUM IODIDE USE

NOTE

Consult 0-EPIP-20129, Emergency Response Team, Radiological Monitoring for off-site monitoring exposure guidelines.

For the following missions, the exposure limits are (Note 1):	TOTAL DOSE(Note 2) (TEDE)	THYROID(Note 3) (CDE)
Performance of actions that would not directly mitigate the event, minimize escalation, or minimize effluent releases	5 REM	50 REM
Performance of actions that mitigate the escalation of the event, rescue persons from a non-life threatening situation, minimize exposures or minimize effluent releases.	10 REM	100 REM
Performance of actions that: decrease the severity of the event, or terminate the processes causing the event in an attempt to control effluent releases to avoid extensive exposure of large populations. Also rescue of persons from a life-threatening situation.	25 REM	250 REM
Rescue of persons from a life threatening situation. (Volunteers should be above the age of 45.) (Note 4)	(Note 5)	(Note 5)

NOTES

- *Both Total Dose (TEDE) and Thyroid Dose (CDE) should be used for purposes of controlling exposure.*
- *Protective clothing, including respirators, should be used where appropriate.*

(Note 1) Exposure limits to the lens of the eye are three times the Total Dose (TEDE) values listed.

(Note 2) Total Dose (TEDE) is the total dose from both external and internal (weighted) sources - Total Effective Dose Equivalent.

(Note 3) Thyroid dose (CDE) commitment from internal sources - Committed Dose Equivalent. The same dose limits also apply to other organs (CDE), skin (Shallow Dose Equivalent), and extremities (Extremity Dose Equivalent).

(Note 4) Volunteers with full awareness of risks involved, including numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

ATTACHMENT 2
(Page 1 of 1)

EMERGENCY EXPOSURE AUTHORIZATION FORM

Date:____________________

I have been briefed on the radiological consequences and hazards associated with the authorized emergency exposure, and I have volunteered to perform the task described below:

Name of Individual(s)	Social Security Number	TLD Number	Signature	Time

Brief Description of Task:____

Authorization Limit:____________________

Briefing Completed By:____________________________ Time:________
(Signature)

OSC Radiation
Protection Supervisor:____________________________ Time:________
(Signature)

OR

TSC Radiation
Protection Supervisor:____________________________ Time:________
(Signature)

Emergency Exposure Authorized by:

Emergency Coordinator:____________________________ Time:________
(Signature)

NOTE: Signatures required by TSC personnel may be authorized by phone or fax.

INFORMATION USE

1.0 PURPOSE

The purpose of this procedure is to provide instructions and controls for radiation exposures in excess of the Vogtle Electric Generating Plant (VEGP) Administrative Guidelines, or in excess of the 10CFR20 occupational limits during emergency conditions.

2.0 RESPONSIBILITIES

2.1 The Emergency Director (ED) has the sole authority to allow radiation exposures in excess of 10CFR20 limits in accordance with the provisions of this procedure.

2.2 The Health Physics (HP) Supervisor, or designee, shall have the following responsibilities:

2.2.1 Preparing Permits for Emergency Radiation Exposure (PERE). (1985304698)

2.2.2 Maintaining records of emergency exposures for each individual.

2.2.3 Providing recommendations to the ED on exposure control measures including issuance of Dosimetry, use of protective equipment and issuance of thyroid blocking agents such as potassium iodide (KI).

2.2.4 Authorizing individuals to receive radiation exposures in excess of the VEGP Administrative Guidelines, but which do not exceed the 10CFR20 limits.

3.0 PREREQUISITES

An emergency situation exists which results in a need to initiate corrective actions, protective actions, sampling activities, or lifesaving measures which might result in exposures greater than 10CFR20 limits. ☐

4.0 PRECAUTIONS

4.1 Personnel authorized to receive exposures in excess of 10CFR20 limits shall meet the following criteria: (1985305022) ☐

4.1.1 Personnel shall be familiar with the risks of exposure to the higher radiation levels which are likely during emergency conditions as outlined in Table 2 and Table 3. ☐

TABLE 1

EMERGENCY EXPOSURE GUIDELINES (1985305256) (1985305827)

NOTES

- Dose limits listed in this table apply to doses incurred over the duration of the emergency.
- Dose to workers performing emergency services may be treated as an once-in-a-lifetime exposure and should not be added to occupational exposure accumulated under non-emergency conditions.
- Workers performing services during emergencies shall limit dose to the lens of the eyes to three times the listed value and doses to any other organ (including skin and body extremities) to ten times the listed value.

Dose Limit (REM) Total Effective Dose Equivalent	Activity	Condition
5	All	
10	Protecting Valuable Property	Lower Dose not practicable
25	Lifesaving or protection of large population	Lower Dose not practicable
>25	Lifesaving or protection of large population	Only on a voluntary basis to persons fully aware of the risks involved

1. G2.3.7 001/LOIT AND LOCT/SRO/M/F 3.5/3.6/G2.3.7/LO-LP-63920-03///

Given the following:

- A Systems Operator (SO) will make multiple entries into AB-A-33 to place the CVCS cation demineralizer in service.
- The SO will use RWP 14-0108 (red RWP).
- The SO will exceed the Annual Administrative 4000 mrem per year TEDE limit during the task.

Which one of the following completes the following statement?

The SO __(1)__ required to receive an ALARA briefing prior to each AB-A-33 entry,

and

per NMP-HP-001, "Radiation Protection Standard Practices," the __(2)__ is the LOWEST level of approval required to exceed the Administrative dose limit.

	__(1)__	__(2)__
A.	is	HP Manager
B.✓	is	Plant Manager
C.	is NOT	HP Manager
D.	is NOT	Plant Manager

K/A

G2.3.7 Radiation Control

Ability to comply with radiation work permit requirements during normal or abnormal conditions:

K/A MATCH ANALYSIS:

The candidate is presented with a scenario where an Auxiliary Building Operator is required to enter several rooms with high dose rates in the area. The Operator is also on the verge of exceeding his Annual TEDE limits of 4000 mrem. The candidate has to determine the minimum level of authority that may approve exceeding the annual TEDE limits.

EXPLANATION OF REQUIRED KNOWLEDGE

Per NMP-HP206 step 5.3.31.2 states that RED RWPs are "Single Use" type briefings. Step 5.3.11.1.4 defines Single Use as "individuals must be authorized and the authorization is good for one entry only."

Per NMP-HP-001 step 6.2.3, there are 3 different administrative levels that require 3 different levels of approval:

1. 2000 mrem in a year requires HP Supervisor, Physicist, or Manager approval.
2. 4000 mrem in a year requres AGM or Plant General Manager approval.
3. 4500 mrem in a year requires Project Vice President approval.

Dose in excess of 5mrem requires special NRC approval for normal operation, ED approval during emergencies. These different admin limits do not have a noun description/name and therefore are generally referred to by the associated dose limit.

DISTRACTOR ANALYSIS:

A. Incorrect. Plausible. Part 1 is correct. RWP 14-0108 is a red RWP and therefore requires a Single Use briefing on each entry per NMP-HP-206.

Part 2 is incorrect. Per NMP-HP-001, exceeding the 4000mrem per year admin dose limit will required Plant General Manager approval. However, the HP Supervisor can approval all dose up to the 4000mrem limit. Therefore, this distractor is plausible.

B. Correct. Part 1 is correct.RED RWP and requires a briefing for each entry. The minimum authority level to approve the TEDE extension is the Plant Manager from the choices presented.

Part 2 is correct. Per NMP-HP-001, exceeding the 4000mrem per year admin dose limit will required Plant General Manager approval.

C. Incorrect. Plausible. Part 1 is incorrect. RWP 14-0108 is a red RWP and therefore requires a Single Use briefing on each entry per NMP-HP-206. However, both yellow and green RWPs allow re-entry into the room with a single HP brief. Therefore, this distractor is plausible.

Part 2 is incorrect. See Part 2 of choice A above.

D. Incorrect. Plausible. Part 1 is incorrect. See Part 1 of choice C above.

Part 2 is correct. See Part 2 of choice B above.

SRO JUSTIFICATION

(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
Yes, specific knowledge of administrative procedures associated with radiological safety and rad exposure authorization levels during normal plant conditions is tested.

Level:	SRO
Tier # / Group #	T3
K/A#	G2.3.7
Importance Rating:	3.5 / 3.6
Technical Reference:	NMP-HP-001, Rev 5.2, page 14 & 15 NMP-HP-206, Rev 3.0, pages 8 &12 V-LO-LP-63930, page 12
References provided:	None
Learning Objective:	LO-LP-63930-06 State the entry requirements applicable to each of the following: b. Radiation Control Area (RCA) c. Radiation Area d. High Radiation Area e. Locked High Radiation Area LO-LP-63920-03 State the plant administrative limits/guidelines for radiation dose. LO-LP-63920-04 State the actions to be taken if administrative dose limits are being approached.
Question origin:	MODIFIED - HL18 NRC - G2.3.7
Cognitive Level:	M/F
10 CFR Part 55 Content:	41.12 / 43.4

You have completed the test!

1. G2.3.7 003/LOCT AND LOIT/SRO/M/F 3.5/3.6/G2.3.7///HL18 NRC/079G2.1.27

(Original Question from HL18 NRC)

Given the following:

- A Fuel Handling Coordinator (FHC) is entering the Spent Fuel Pool area.
- The FHC is reviewing his RWP prior to beginning work and notices an ALARA briefing is required.
- The dose rate is 900 mrem/hour due to damaged fuel assemblies.
- The FHC will also exceed 2000 mrem Annual TEDE limits while in the area.

Which one of the following completes the following statement?

Based on the area dose rate, the FHC will be required to receive an ALARA briefing prior to __(1)__ entry,

and

per NMP-HP-001, "Radiation Protection Standard Practices", the ___(2)___ is the MINIMUM authority level required to exceed the Annual TEDE limit.

	__(1)__	__(2)__
A.	each	HP Manager
B.	each	Plant General Manager
C.✓	ONLY the first	HP Manager
D.	ONLY the first	Plant General Manager

6.1.55 20.2202 Notification of Incidents
Paragraph (a)(2) and (b)(2), for reporting the release of radioactive material inside or outside of a restricted area is caveated as not applicable to, "locations where personnel are not normally stationed during routine operations, such as hot cells or process enclosures." For purposes of this section, consistent with the answer to question 56 of the first NRC 10CFR20 Q & A document, locations where personnel are not normally stationed will be interpreted as areas, rooms and enclosures which are not normally occupied nor periodically patrolled during normal plant operations and maintenance.

6.1.56 20.2203 Reports of Exposures, Radiation Levels and Concentrations of Radioactive Material Exceeding the Limits
No fleet practices identified.

6.1.57 20.2204 Reports of Planned Special Exposures
No fleet practices identified.

6.1.58 20.2206 Reports of Individual Monitoring
The intent of Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," will be met in complying with this paragraph with NRC Form 5.

6.1.59 20.2301 Applications for Exemptions
No fleet practices identified.

6.1.60 20.2302 Additional Requirements
No fleet practices identified.

6.1.61 20.2401 Violations
No fleet practices identified.

6.2 Other Consensus Positions

6.2.1 Whole Body Count Performance Frequency
Monitored workers will be given an entrance and exit whole body count (WBC) or whole body scan (WBS). The exit WBC from another SNC site can be used in lieu of an entrance WBC if the SNC site was the last site the worker entered an RCA and/or monitored. Upon request from a worker, WBCs will be provided to the worker on a voluntary and reasonable basis.

6.2.2 Dose Limits for Workers Who Provide Outage Support at a SNC Plant Other Than Their Home Plant.
Workers should be limited to 500 mrem per visit, unless express consent is given by the home plants management to exceed that limit.

6.2.3 Administrative Annual TEDE Dose Limits and the Approval Authority Necessary to Exceed Limits

6.2.3.1 2000 mrem in a year requires HP Support Supervisor, Plant Health Physicist, or HP Manager approval.

6.2.3.2 4000 mrem in a year requires AGM or Plant General Manager approval.

6.2.3.3 4500 mrem in a year requires Project Vice President approval.

6.2.4 Discrepant Dosimeter Investigation Criteria

An assessment of worker's dose should be initiated for discrepant dosimetry results when the following criteria are met: the primary or secondary dosimeter dose exceeds 100 mrem; and the secondary dosimeter reading differs by more than 25% from the primary dosimeter.

6.2.5 Dose Monitoring Threshold

All individuals entering an RCA will be monitored for radiation exposure. A single dosimeter suffices for Visitors or Radiation Workers whose annual dose from sources external to the body is not expected to exceed 100 mrem at a particular station. An Optically Stimulated Luminescent Dosimeter (OSLD) and a self-reading dosimeter, such as an Electronic Dosimeter (ED), will be provided to all other individuals.

6.2.6 Training Requirements for Visitors or Temporary Radiation Workers Who Enter RCAs

All SNC plants will administer training to all who must enter the RCAs in the following manner:

6.2.6.1 If the individual is expected to receive < 100 mrem in a year, the individual will be escorted by a GET qualified worker and will be provided instructions.

6.2.6.2 If the individual is expected to receive ≥100 and <500 mrem in a year, the individual will receive a visitor handout (containing all of the instruction elements required by 10CFR19.12) and will acknowledge receipt of the instructions or handout by signing a form. The individual will have a GET-trained radiological escort. If special circumstances dictate that entries into contaminated areas are required, the individual will receive dress-out training (if he has no history of such training at our plants). Similarly, if the individual requires entry to high radiation areas, special training or instructions may be required.

6.2.6.3 If the individual is expected to receive >=500 mrem in a year, the individual is required to complete GET which includes testing. If the individual has previous GET training at a nuclear facility within the past two years or as allowed by the Training Department, then exemption GET and the exemption GET test can be administered.

6.2.7 Air Flow for Fume Hoods Containing Radioactive Materials/Fluids

Sample station fume hoods containing radioactive materials/fluids will meet a minimum air flow acceptance criteria of 100 LFPM, and will be labeled to ensure that 100 LFPM is maintained or exceeded.

6.2.8 Respirator Training and Fit Test Frequencies

Sites will apply a program whereby classroom or Computer-based training with an examination will be conducted annually. For individuals required to utilize respirator

5.3.11 Enter the End Date in the same manner as step 5.3.10. When the End Date is entered, press the "Tab" key to advance to the "Authorization Type" field.

5.3.11.1 The type of worker authorization is based upon the type of RWP being written. Worker authorizations fall into 4 categories:

5.3.11.1.1 All – Anyone can use this RWP.

5.3.11.1.2 Work Group – Work Group must be authorized.

5.3.11.1.3 Individual – Individuals must be authorized.

5.3.11.1.4 Single Use – Individuals must be authorized and the authorization is good for one entry only.

5.3.12 From the pull-down menu, select "All" or "Work Group" for a Green RWP. Select "Individual" for a Yellow RWP. Select "Single Use" for a Red RWP.

5.3.13 Press the "Tab" key to advance to the "RWP Type" field.

5.3.14 With the cursor in the "Type" field, select the appropriate RWP type (General or Specific) from the pull-down table. Press the "Tab" key to advance to the "Principle Work Document" field.

5.3.15 With the cursor in the "Principle Work Document" field, type the activity or MWO number, if applicable. If the RWP is not written for a specific activity or MWO, this field may be either left blank or type N/A in the field.

5.3.16 Press the "Tab" key and advance past the ALARA Review Number field. Tab to the HP Job Coverage field.

5.3.17 In the HP Coverage field, enter None, Intermittent, or Continuous as appropriate for the RWP. Table 1 defines the type of HP coverage used.

5.3.18 Press the "Tab" key to advance to the "Job Description" field.

5.3.19 Type a short general description of the work to be performed in the "Job Description" field. Press "Tab" to advance to the "Location" field.

NOTE

For an RWP, Location is the area where the majority of the work should be performed.

5.3.20 Type the work location code or select a location from the pull-down table in the "Location" field.

5.3.21 When the work location is selected, press "Tab" to advance to the "Area" field.

5.3.22 The Area field is normally left blank or N/A is typed in this field. Press "Tab" to advance to the "Comments" field. Enter comments as needed or leave blank.

5.3.29.2 When "Detail" is selected, a Dosimetry Types dialog box will appear. Check the box next to the type of dosimetry required for the RWP. Click on the "OK" button.

5.3.29.3 Click on the "Apply" button at the bottom of the screen.

5.3.30 Select the "Worker Instructions" tab in the "Maintain RWP" screen.

5.3.30.1 With the cursor in the blank Worker Instructions field, click on the right mouse button and select "Detail" from the pop-up table.

5.3.30.2 When Detail is selected, a "Worker Instructions" dialogue box will appear. Place a check mark next to the applicable work instructions, and select "OK".

5.3.30.3 If working from a model, "Rem_occ" may be used to remove any instructions that may be no longer necessary. To remove an instruction, place the cursor in the line of the instructions and click on the "Rem_occ" button.

5.3.30.4 Worker Instructions may be added as free form text.

5.3.30.5 Additional Worker Instructions may be added by using the "Add_occ" button.

5.3.30.6 When all Worker Instructions are entered, click on "Apply" at the bottom of the screen.

5.3.31 Select the "Briefing" tab in the "Maintain RWP" screen.

5.3.31.1 With the cursor in the "Briefing" field, right click on the mouse and select "Detail" from the pop-up table.

5.3.31.2 When Detail is selected, a dialogue box will open. Check the appropriate type of briefing for the RWP and select "OK".

- For Red RWPs, use "Single Use" briefing type.
- For Yellow RWPs, the briefing type is conditional based on the activity.
- For Green RWPs, no briefing is required.

5.3.31.3 Select "Yes" or "No" in the "Required" field if applicable.

5.3.31.4 After entering the briefing type, click on the "Apply" button at the bottom of the screen.

5.3.32 Select the Supervisors tab in the Maintain RWP screen.

5.3.32.1 Type the name of the job supervisor in the "Job Supervisor" field. Press the "Tab" key.

5.3.32.2 In the "Department" field, select the appropriate department from the pull-down table. Press the "Tab" key.

5.3.32.3 In the "Phone/Ext" field, type the phone number of the job supervisor. Press the "Tab" key.

III. LESSON OUTLINE	NOTES

Risk-Based RWP Format and Requirements

Color Code	***Radiological Significance***	***Types of RWPs***	*Type of Briefing Required*	***General Radiological Conditions***
Green	**Low**	**All *General* RWPs** And ***Specific* RWPs with low radiological risk**	**No ALARA briefing required.**	**Dose Rate: < 100 mrem/hr** **Contamination Levels: < 200,000 dpm/100** ***Airborne Levels: < 0.3 DAC*** * Workers should always refer to the most recent survey information for the area(s) being worked in.
Yellow	**Moderate**	***Specific* RWPs that are tied to unique *Work Groups* -** ***Specific* RWPs covering work in areas with intermediate levels of radiological risk.**	• **Initial ALARA briefing required prior to *first* entry.** • **Additional ALARA briefing required when specified rad conditions are exceeded.** ***A pre-job ALARA briefing will be required if:*** • **Radiological conditions that are addressed in the Worker Instructions section may be exceeded, or** • **If the RWP default settings for the accumulated dose or dose rate alarms may be exceeded, or** • **Breach of a contaminated system**	**Dose Rate: < 1000 mrem/hr** **Contamination Levels: < 500,000 dpm/100 cm^2** **Airborne Levels: < 0.3 DAC** * Workers should always refer to the most recent survey information for the area(s) being worked in.
Red	**High**	***Specific* RWPs** covering work in areas with high levels of radiological risk.	***ALARA Briefing required prior to each entry.*** • **Radiological conditions on the RWP will be based on actual, projected or historical survey information.** • **Latest rad conditions and specific instructions will be covered in the pre-job ALARA briefing**	**Dose Rate: > 1000 mrem/hr** **Contamination Levels: > 500,000 dpm/100 cm^2** **Airborne Levels: > 0.3 DAC** * Workers should always refer to the most recent survey information for the area(s) being worked in.

5. HP reviews active RWPs on a routine basis
6. Normally, HP will survey the work area prior to issuing an RWP
   a) In high radiation areas, survey performance may not be consistent with ALARA

1. G2.4.46 001/LOIT AND LOCT/SRO/C/A 4.2/4.2/G2.4.46/LO-TA-40002///056AG2.4.45

At time 1000:

- Unit 1 is in Mode 6.

At time 1005 the following alarms illuminate:

- ALB32-D02 RESV AUX XFMR 1NXRA HI SIDE PHOC LOR TRIP
- ALB32-E02 RESV AUX XFMR 1NXRB HI SIDE PHOC LOR TRIP

- ALB35-A10 DG1A TRIP OVERSPEED
- ALB35-F10 DG1A EMERGENCY START
- ALB36-A01 4160V SWGR 1AA02 TROUBLE

- ALB37-A01 4160V SWGR 1BA03 TROUBLE alarms, then subsequently clears.
- ALB38-F10 DG1B EMERGENCY START

Current time is 1025:

Based on the current time, which one of the following is the correct Emergency Classification required to be declared?

REFERENCE PROVIDED

A. Alert Emergency (CA3)

B. Alert Emergency (SA5)

C. Notification of Unusual Event (SU1)

D✓ Notification of Unusual Event (CU3)

G2.4.46 Emergency Procedures / Plan

Ability to verify that the alarms are consistent with the plant conditions.

K/A MATCH ANALYSIS:

The candidate must analyze various alarms and indications associated with the electrical distribution system to determine plant status and the correct emergency classification. The event initiation time will also affect the classification.

EXPLANATION OF REQUIRED KNOWLEDGE

The following annunciators are symptomatic of both train RATs being de-energized.

- ALB32-D02 RESV AUX XFMR 1NXRA HI SIDE PHOC LOR TRIP

- ALB32-E02 RESV AUX XFMR 1NXRB HI SIDE PHOC LOR TRIP

The following annunciators are symptomatic of 1AA02 being de-energized.

- ALB35-A10 DG1A TRIP OVERSPEED
- ALB35-F10 DG1A EMERGENCY START
- ALB36-A01 4160V SWGR 1AA02 TROUBLE

The following annunciators are symptomatic of 1BA03 de-energizing and then being re-energized by the 1B DG.

- ALB37-A01 4160V SWGR 1BA03 TROUBLE alarms, then subsequently clears.
- ALB38-F10 DG1B EMERGENCY START

The question stem states that the plant is in Mode 6 and 20 minutes has elapsed since the loss of power event occured. Per NMP-EP-110 Figure 3, an NOUE should be classified based on Loss of All Offsite Power to Essential Buses for GREATER THAN 15 minutes and one EDG is supplying the 4160VAC bus.

An upgrade to an ALERT would occur if at any time the 1B DG fails to keep 1BA03 energized.

On NMP-EP-110 Figure 2, there are two similar classifications. The ALERT threshold is the same as the NOUE threshold in Mode 6. The difference arise out of a requirement to maintain 1 train in Mode 6 and 2 trains in Modes 1-4. A similar NOUE also exist for a Loss of All Offsite Power to Essential Buses for GREATER THAN 15 minutes and one EDG is supplying each of the 4160VAC buses.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. Plausible. CU3 is the correct classification. However, if the candidate does not recognize that 1BA03 is energized by the 1B DG and believes both 4160V buses are de-energized, then CA3 would be the correct classificaton.

B. Incorrect. Plausible. CU3 is the correct classification. However, if the candidate incorrectly utilizes NMP-EP-110 Figure 2 instead of Figure 3, then SA5 would match the conditions of the stem for the, but for the incorrect mode.

C. Incorrect. Plausible. CU3 is the correct classification. However, if the candidate incorrect diagnoses 1AA02 and believes it is energized by the 1A DG and also incorrect utilizes NMP-EP-110 Figure 2 instead of Figure 3, then SU1 would be the correct threshold.

D. Correct. CU3 is the correct classification. Loss of All Offsite Power to Essential Buses for GREATER THAN 15 minutes and one EDG is supplying the 4160VAC bus in Mode 6.

ANSWER / DISTRACTOR ANALYSIS

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, the answer requires specific knowledge of emergency classification thresholds.
-Can the question be answered *solely* by knowing immediate operator actions? No, IOAs are not addressed by this question.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, the question does not address AOP or EOP entry conditions.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, the answer requires specific knowledge of emergency classification thresholds.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures
- **Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. Yes, the answer requires specific knowledge of emergency classification thresholds and determination of the specific classification based on current plant conditions. This is an SRO ONLY job link associated with an SRO ONLY objective. [LO-LP-40101-13 Given an emergency scenario, and the procedure, classify the emergency (SRO only).]**

Level:	SRO	
Tier # / Group #	T3	
K/A#	G2.4.46	
Importance Rating:	4.2 / 4.2	
Technical Reference:	NMP-EP-110-GL03, Figure 2, Rev 3.0, page 122 NMP-EP-110-GL03, Figure 3, Rev 3.0, page 123	
References provided:	NMP-EP-110-GL03, Figure 1, Rev 3.0, page 121 NMP-EP-110-GL03, Figure 2, Rev 3.0, page 122 NMP-EP-110-GL03, Figure 3, Rev 3.0, page 123	
Learning Objective:	LO-TA-40002	Emergency Classification and Implementing Instructions using NMP-EP-110 (SRO Only)
	LO-LP-40101-13	Given an emergency scenario, and the procedure, classify the emergency (SRO only).
	LO-PP-11101-56	Predict the possible consequences of paralleling and loading the EDG when a loss of offsite power is anticipated.
	LO-PP-11101-30	Describe in general terms the actions that occur on a normal or emergency start of a diesel engine up to and including the final condition of the diesel and the differences between a normal and emergency start.
	LO-LP-60323-05	Given the entire AOP, describe: a. Purpose of selected steps b. How and why the step is being performed c. Expected response of the plant/parameter(s) for the step
	LO-TA-60009A	Respond to a Loss of Class 1E Electrical Systems per 18031-1/2
	LO-TA-37018	Respond to a Loss of All AC Power per 19100-C
	LO-TA-11021	Respond to Diesel Generator Alarms Using Procedure 17035-1/2 or 17038-1/2
Question origin:	MODIFIED - HL18 NRC Question # 056AG2.4.45	
Cognitive Level:	C/A	
10 CFR Part 55 Content:	43.5	
Comments:		

You have completed the test!

1. 056AG2.4.45 001/1/1/LOSP- EP/C/A-4.1/4.3/NEW/HL-18 NRC/SRO/

At 10:00:

- Unit 1 is in Mode 4.

At 10:05 the following alarms illuminate:

- ALB32-D02, RESV AUX XFMR 1NXRA HI SIDE PHOC LOR TRIP
- ALB32-E02, RESV AUX XFMR 1NXRB HI SIDE PHOC LOR TRIP

- ALB35-A10, DG1A TRIP OVERSPEED
- ALB35-F10, DG1A EMERGENCY START
- ALB36-A01, 4160V SWGR 1AA02 TROUBLE

- ALB37-A01, 4160V SWGR 1BA03 TROUBLE alarms, then subsequently clears.
- ALB38-F10, DG1B EMERGENCY START

Current time is 10:25:

Based on the current time, which one of the following is the correct Emergency Classification required to be declared?

REFERENCE PROVIDED

A. Alert Emergency (CA3)

B.✓ Alert Emergency (SA5)

C. Notification of Unusual Event (SU1)

D. Notification of Unusual Event (CU3)

056AG2.4.45 Loss of Offsite Power

Ability to prioritize and interpret the significance of each annunciator or alarm:
(CFR: 41.10 / 43.5 / 45.3 / 45.12)

<u>K/A MATCH ANALYSIS:</u>

The candidate is given various alarms and indications associated with the electrical distribution system. The candidate has to analyze the alarms to determine the plant status and determine the correct emergency classification, there is a time given when the event occurred that will also play into the classification.

The question is SRO only due to the Vogtle specific objective for Classification of an Emergency is an SRO only objective.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. CA3 is a Cold Matrix classification, the plant is in Mode 4, not Mode 5 or 6. The mode was NOT stated in the question but just an RCS temperature given to increase the plausibility the candidate may select the wrong classification matrix. If the candidate selects the wrong matrix with the given alarms, it is plausible he could misinterpret the event and classify wrong. With the multiple alarms and indications, this can easily occur.

B. Correct. SA5 is the correct classification using the Hot Matrix, this is still a difficult determination with the multiple annunciator windows illuminated. The plant is only one failure away from a total plant blackout in this condition but the candidate has to determine this and correlate the event has been ongoing for > 15 minutes.

C. Incorrect. SU1 is a Hot Matrix classification. The plant is only one failure away from a total plant blackout in this condition but the candidate has to determine this and correlate the event has been ongoing for > 15 minutes. This choice is very plausible as the only difference between this and SA5 is that both diesels have to be carrying the buses to classify as SU1 versus 1 DG as in the correct choice. This is a difficult determination with the mulitiple annuciators illuminated.

D. Incorrect. CA3 is a Cold Matrix classification, the plant is in Mode 4, not Mode 5 or 6. The mode was NOT stated in the question but just an RCS temperature given to increase the plausiblity the candidate may select the wrong classification matrix. IF, the plant were in Mode 5 and the Cold Matrix required to be used, this choice would then be correct.

REFERENCES:

The following references will be provided to the candidates during the exam.

NMP-EP-110, GL03, Figure 3, Cold Initiating Condition Emergency Action Level Matrix - Modes 5, 6, and Defueled Only

NMP-EP-110, GL03, Figure 2, Hot Initiating Condition Emergency Action Level Matrix - Modes 1, 2, 3, and 4 Only

NMP-EP-110, GL03, Figure 1, Fission Product Barrier Evaluation

VEGP learning objectives:

LO-LP-40101-13 Given an emergency scenario, and the procedure, classify the emergency (SRO only).

This question is SRO only because the Emergency Plan is linked to a learning objective that is specifically labeled in the lesson plan as SRO Only.

You have completed the test!

FIGURE 2

Vogtle Electric Generating Plant

Procedure Number **NMP-EP-110-GL03** Rev 3.0

EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS

Figure 2 - HOT INITIATING CONDITION EMERGENCY ACTION LEVEL MATRIX - MODES 1, 2, 3, and 4 ONLY

RADIOLOGICAL (EFFLUENTS; RAD LEVELS) | FISSION PRODUCT BARRIERS | SYSTEM MALFUNCTIONS (AC/DC POWER; RX and CORE; ALARMS / COMMUNICATIONS) | HAZARDS (NATURAL / DESTRUCTIVE; FIRE / EXPLOSION; TOXIC / FLAMMABLE; SECURITY; CR EVACUATION; ED DISCRETION)

GENERAL EMERGENCY

EFFLUENTS

RG1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mR TEDE OR 5000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology (pg. [illegible])

See Notes 1 and 2

1. VALID reading on any of the following radiation monitors that exceeds OR expected to exceed the reading shown for 15 minutes OR longer:
   - RE-12839 1.5 x 10^[illegible] µCi/cc
   - RE-12444 7.0 x 10^[illegible] µCi/cc
   - RE-13119 thru 13122 3.1 x 10^[illegible] µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 1000 mR TEDE OR 5000 mR thyroid CDE at OR beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates exceeding 1000 mR/hr expected to continue for more than one hour; OR analyses of field survey samples indicate thyroid CDE of 5000 mR for one hour of inhalation, at OR beyond the site boundary.

RAD LEVELS

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note 2: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

FISSION PRODUCT BARRIERS

FG1 - Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier (pg. 39)

See Fission Product Barrier Matrix

AC/DC POWER

SG1 - Prolonged Loss of All Offsite Power AND Prolonged Loss of All Onsite AC Power to essential busses (pg. [illegible])

1. Loss of ALL AC power indicated by:
   a. Loss of power to OR from transformers 1(2)NXRA AND 1(2)NXRB resulting in loss of all off-site electrical power to BOTH 1(2)AA02 AND 1(2)BA03 for greater than 15 minutes
   AND
   b. Failure of emergency diesel generators to supply power to emergency busses

AND EITHER

2. Restoration of at least one 4160V ESF bus, AA02 OR BA03, within 4 hr. of time of loss is NOT likely

OR

3. Fuel Clad Barrier Evaluation indicates continuing degradation (Loss or Potential Loss) of core cooling

RX and CORE

SG2 - Failure of the RPS to complete an Automatic Trip and Manual Trip was NOT successful AND there is Indication of an Extreme Challenge to the ability to Cool the Core **Modes 1, 2 Only** (pg. 73)

NOTE Heat Sink CSF should not be considered – RED if total AFW flow is less than 570 gpm due to operator action.

1. Subcriticality CSF - RED
   AND
   Core Cooling CSF - RED

OR

2. Subcriticality CSF - RED
   AND
   Heat Sink CSF - RED

SECURITY

HG1 – HOSTILE ACTION resulting in loss of physical control of the facility (pg. 119)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

ED DISCRETION

HG2 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency (pg. 120)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity
   OR
   HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

EFFLUENTS

RS1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE OR 500 mR Thyroid CDE for the Actual or Projected Duration of the Release (pg. [illegible])

See Notes 3 and 4

1. VALID reading on any of the following radiation monitors that exceeds OR is expected to exceed the reading shown for 15 minutes OR longer:
   - RE-12839 1.5 x 10^[illegible] µCi/cc
   - RE-12444 7.0 x 10^[illegible] µCi/cc
   - RE-13119 thru 13122 3.1 x 10^[illegible] µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE OR 500 mR thyroid CDE at OR beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; OR analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at OR beyond the site boundary.

RAD LEVELS

Note 3: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note 4: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

FISSION PRODUCT BARRIERS

FS1 - Loss or Potential Loss of ANY Two Barriers (pg. 39)

See Fission Product Barrier Matrix

AC/DC POWER

SS1 - Loss of All Offsite Power AND Loss of All Onsite AC Power to Essential Busses (pg. 65)

1. Loss of all AC power indicated by:
   a. Loss of power to OR from transformers 1(2)NXRA AND 1(2)NXRB resulting in loss of all off-site electrical power to BOTH 1(2)AA02 AND 1(2)BA03 for greater than 15 minutes
   AND
   b. Failure of emergency generators to supply power to emergency busses.
   AND
   c. Restoration of at least one 4160V ESF bus, AA02 OR BA03, has NOT occurred within 15 minutes of time of loss of all AC power.

SS3 - Loss of All Vital DC Power (pg. 67)

1. Loss of Vital DC power to 125 VDC Buses AD1, BD1, CD1, AND DD1 indicated by bus voltage indications less than 105 VDC for greater than 15 minutes.

RX and CORE

SS2 - Failure of RPS Instrumentation to complete or Initiate an Automatic Reactor Trip Once a RPS Setpoint Has Been Exceeded AND Manual Trip Was NOT Successful. **Modes 1, 2 Only** (pg. 66)

NOTE A successful manual trip for purposes of declaration is any action taken from the Main Control Board (MCB) that rapidly inserts the control rods, this can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB.

1. Automatic AND manual trip were NOT successful as indicated by:
   a. An automatic reactor trip setpoint was exceeded
   AND
   b. An automatic reactor trip did NOT occur
   AND
   c. A successful manual trip did NOT occur from the Control Room (NOTE)

SS4 - Complete Loss of Heat Removal Capability (pg. 68)

NOTE: Heat Sink CSF should NOT be considered RED if total AFW flow is less than 570 gpm due to operator action.

1. Complete Loss of Heat Removal Capability as indicated by:
   a. Core Cooling CSF - ORANGE
   AND
   b. Heat Sink CSF - RED

ALARMS / COMMUNICATIONS

SS6 - Inability to Monitor a SIGNIFICANT TRANSIENT in Progress (pg. 69)

NOTE: SIGNIFICANT TRANSIENT: *is an* UNPLANNED *event involving one or more of the following: (1) automatic turbine runback >25% thermal reactor power, (2) electrical load rejection >25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations >10%.*

1. Inability to monitor a SIGNIFICANT TRANSIENT in progress as indicated by:
   a. Loss of most OR all of the MCB annunciators OR indicators associated with safety systems
   AND
   b. Compensatory non-alarming indications are NOT available
   AND
   c. Indications needed to monitor the Critical Safety Function Status Tree parameters are NOT available.

SECURITY

HS4 – HOSTILE ACTION within the PROTECTED AREA (pg. 118)

1. A HOSITLE ACTION is occurring or has occurred within the PROTECTED AREA as reported by security shift supervision.

CR EVACUATION

HS2 - Control Room Evacuation Has Been Initiated AND Plant Control Cannot Be Established (pg. 116)

1. a. Control Room evacuation has been initiated
   AND
   b. Control of the plant CANNOT be established per 18038 – Operation From the Remote Shutdown Panels, within 15 minutes.

ED DISCRETION

HS3 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency (pg. 117)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public
   OR
   HOSTILE ACTION that results in intentional damage OR malicious acts;
   (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels that exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

ALERT EMERGENCY

EFFLUENTS

RA1 - Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer. (pg. 27)

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

1. VALID reading on any of the following effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes OR longer.

Monitor	200 X Setpoint Value
RE-0018	2.8 x 10^[illegible] µCi/cc
RE-0021	4.0 x 10^[illegible] µCi/cc
RE-0848	3.5 x 10^[illegible] µCi/cc
ARE-0014	2.4 x 10^[illegible] µCi/cc
RE-12839	7.9 x 10^[illegible] µCi/cc
RE-12442C/12444C	1.3 x 10^[illegible] µCi/cc

OR

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes OR longer.
   Main Steam RE-13119 thru 13122 2.5 µCi/cc

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations OR release rates in excess of 200 times Technical Specification 5.5.4 as confirmed by the ODCM, with a release duration of 15 minutes OR longer.

RAD LEVELS

RA2 - Damage to Irradiated Fuel OR Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel (pg. 30)

1. UNPLANNED VALID alarm on any of the following radiation monitors:
   - Fuel Handling Building RE-0008
   - Fuel Handling BLDG EFFL. ARE-2532 A/B
   - Fuel Handling BLDG EFFL. ARE-2533 A/B

OR

2. Loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel as indicated by ANY of the following:
   Personnel report during fuel assembly movements, Spent Fuel Pool Storage Less than El 193'-5", Transfer Canal Transit El Less than El 186'10", Reactor Core El Less than El 181'-10" (63% on Full Range RVLIS)

RA3 - Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown (pg. 32)

1. VALID radiation monitor readings greater than 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:
   Control Room radiation monitor RE-001 OR Central Alarm Station (by survey)

OR

2. UNPLANNED VALID radiation readings greater than 1 R/hr values in areas requiring infrequent access to maintain plant safety functions:
   Turbine Building, South & North Main Steam Valve Rooms, Control Building Fire Pump House, Aux Feed Water Pump House, Low Voltage Switchyard, Diesel Generator Buildings, NSCW Towers

FISSION PRODUCT BARRIERS

FA1 - ANY Loss or ANY Potential Loss of EITHER Fuel Clad Barrier OR RCS Barrier (pg. 39)

See Fission Product Barrier Matrix

AC/DC POWER

SA5 - AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout (pg. 64)

1. a. AC power capability to 1(2)AA02 AND 1(2)BA03 reduced to a single power source for greater than 15 minutes
   AND
   b. ANY additional single failure will result in station blackout.

Distractor

RX and CORE

SA2 - Failure of RPS Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a RPS System Setpoint Has Been Exceeded AND Manual Trip was Successful. Modes 1, 2, 3 Only (pg. 61)

NOTE A successful manual trip for purposes of declaration is any action taken from the Main Control Board (MCB) that rapidly inserts the control rods, this can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB.

1. Automatic trip was NOT successful but manual trip was successful as indicated by:
   a. An automatic Reactor Protection System trip setpoint exceeded
   AND
   b. An automatic reactor trip did NOT occur
   AND
   c. A successful manual trip occurred from the Control Room (NOTE)

ALARMS / COMMUNICATIONS

SA4 - UNPLANNED Loss of Most or All Annunciation or Indication With EITHER a SIGNIFICANT TRANSIENT in Progress, OR Compensatory Non-Alarming Indicators are Unavailable (pg. 62)

NOTE: SIGNIFICANT TRANSIENT: *is an* UNPLANNED *event involving one or more of the following: (1) automatic turbine runback >25% thermal reactor power, (2) electrical load rejection >25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations >10%.*

"Compensatory non-alarming indications" in this context includes computer based information such as SPDS. The indications needed to monitor safety functions necessary for protection of the public must include control room indications, computer generated indications and dedicated annunciation capability. The specific indications are those used to determine such functions as the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

1. UNPLANNED loss of most OR all MCB annunciators OR indicators associated with safety systems for greater than 15 minutes

AND EITHER

2. a. A SIGNIFICANT TRANSIENT is in progress
   OR
   b. Compensatory non-alarming indications are NOT available

NATURAL / DESTRUCTIVE

HA1 - Natural and Destructive Phenomena Affecting the Plant VITAL AREA (pg. 105)

1. Seismic monitoring system confirms seismic event greater than OBE Earthquake acceleration of 0.12 g.

OR

2. Tornado OR high winds greater than 100 mph within the PROTECTED AREA boundary AND resulting in VISIBLE DAMAGE to any of the following plant structures/equipment (Table 1) OR the Control Room has indication of degraded performance of any listed systems.

OR

3. Vehicle crash within PROTECTED AREA boundary AND resulting in VISIBLE DAMAGE to any of the following plant structures (Table 1) OR equipment therein OR control indication of degraded performance of those systems.

OR

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to OR penetration of any area containing safety-related equipment, their controls OR their power supplies.

OR

6. Sustained hurricane winds onsite resulting in VISIBLE DAMAGE to plant structures within the PROTECTED AREA boundary containing equipment necessary for safe shutdown.

TABLE 1

- Containment Building
- Fuel Handling Building
- Diesel Generator Building
- Diesel Fuel Oil Storage Tank Pumphouse
- Auxiliary Building
- Auxiliary Feedwater Pumphouse
- Control Building
- NSCW CoolingTowers

FIRE / EXPLOSION

HA2 - FIRE OR EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown (pg. 108)

1. FIRE OR EXPLOSION in a plant VITAL AREA AND affected system parameter indications show degraded performance OR plant personnel report VISIBLE DAMAGE to permanent structures OR safety related equipment in any of the following:
   - Containment Building
   - Fuel Handling Building
   - NSCW Cooling Towers
   - Control Building
   - Diesel Generator Building
   - Auxiliary Building
   - Diesel Fuel Oil Storage Tank Pumphouse
   - Auxiliary Feedwater Pumphouse

TOXIC / FLAMMABLE

HA3 - Release of Toxic, Asphyxiant, or Flammable Gases Within or contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown (pg. 110)

1. Report OR detection of toxic OR asphyxiant gas within OR contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report OR detection of flammable gases in concentration greater than the LOWER FLAMMABILITY LIMIT within OR contiguous to a VITAL AREA.

SECURITY

HA4 – HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat. (pg. 112)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by security shift supervision.

OR

2. A validated notification from the NRC of an airliner attack threat within 30 minutes of the site.

CR EVACUATION

HA5 - Control Room Evacuation Has Been Initiated. (pg. 114)

1. Entry into 18038 – Operation From the Remote Shutdown Panels, procedure for Control Room evacuation.

ED DISCRETION

HA6 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. (pg. 115)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant
   OR
   a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

NOTIFICATION OF UNUSUAL EVENT

EFFLUENTS

RU1 - Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. (pg. 23)

Note: The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes OR longer.

Monitor	2 X Setpoint Value
RE-0018	2.8 x 10^[illegible] µCi/cc
RE-0021	4.0 x 10^[illegible] µCi/cc
RE-0848	3.5 x 10^[illegible] µCi/cc
ARE-0014	2.4 µCi/cc
RE-12839	7.9 x 10^[illegible] µCi/cc
RE-12442C/12444C	1.3 x 10^[illegible] µCi/cc

OR

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes OR longer.
   Main Steam (RE-13119 thru 13122) 2.5 x 10^[illegible] µCi/cc

OR

3. Confirmed sample analyses for gaseous OR liquid releases indicates concentrations OR release rates, with a release duration of 60 minutes OR longer, in excess of two times Technical Specification 5.5.4, as confirmed by the ODCM.

RAD LEVELS

RU2 - Unexpected Rise in Plant Radiation (pg. 25)

1. a. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, OR fuel transfer canal with all irradiated fuel assemblies remaining covered by water by any of the following:
   - Personnel report of low water level, OR
   - LSHL-0623 officials low, OR
   - Personnel report of cavitation OR low discharge pressure for SFP (1/2-1213-P6-002 OR -005) AND/OR RHR (1/2-1205-P6-001 OR -002) Pumps

   AND

   b. UNPLANNED VALID Direct Area Radiation Monitor readings rise on any of the following:
   - RE-0008 in the fuel building
   - RE-0002, -0003, -0004 in containment
   - RE-0011 at the seal table
   - RE-0005, -0006 in containment.

OR

2. UNPLANNED VALID Direct Area Radiation Monitor readings rise by a factor of 1000 over normal* levels.
   - RE-0001 Control Room
   - RE-0002/0003 Cntmt Bldg Low Range
   - RE-0004 Cntmt Bldg Pers Hatch
   - RE-0005/0006 Cntmt Bldg High Range
   - RE-0007B Sampling Room
   - RE-0008 FHB
   - RE-0011 Cntmt Bldg Incore Instr
   - ARE-16971 RPF North Wall General Area
   - ARE-16972 RPF East Wall General Area
   - ARE-16973 RPF Dress Out Area
   - *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Distractor

FISSION PRODUCT BARRIERS

FU1 - ANY Loss OR ANY Potential Loss of Containment Barrier (pg. 39)

See Fission Product Barrier Matrix

AC/DC POWER

SU1 - Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes (pg. 54)

1. a. Loss of power to OR from transformers 1(2)NXRA AND 1(2)NXRB resulting in loss of all offsite electrical power to BOTH 1(2)AA02 AND 1(2)BA03 for greater than 15 minutes
   AND
   b. Emergency diesel generators supplying power to BOTH 1(2)AA02 AND 1(2)BA03.

RX and CORE

SU2 - Inability to Reach Required Shutdown Within Technical Specification Limits (pg. 55)

1. Plant is NOT brought to required operating mode within Technical Specifications LCO Action Statement Time limit.

SU5 - RCS Leakage (pg. 58)

1. Unidentified OR pressure boundary leakage greater than 10 gpm.

OR

2. Identified leakage greater than 25 gpm.

SU4 - Fuel Clad Degradation (pg. 57)

1. RE48000 radiation monitor reading greater than 5.55µCi/gm indicating fuel clad degradation greater than Technical Specification allowable limits.

OR

2. RCS coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits as indicated by ANY of the following:
   - Dose Equivalent I-131 greater than 1 µCi/gm for greater than 48 hours
   - Dose Equivalent I-131 greater than Technical Specification figure 3.4.16-1 limits
   - RCS specific activity greater than 100/E µCi/gm gross radioactivity.

SU8 - Inadvertent Criticality Modes 3, 4 Only (pg. 60)

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

ALARMS / COMMUNICATIONS

SU3 - UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes (pg. 56)

1. UNPLANNED loss of most OR all of the MCB annunciators OR indicators associated with safety systems for greater than 15 minutes.

SU6 - UNPLANNED Loss of All Onsite OR Offsite Communications Capabilities (pg. 59)

1. UNPLANNED loss of ALL of the following on-site communications capability affecting the ability to perform routine operations:
   - In plant telephones
   - Public address system
   - Plant radio systems

OR

2. UNPLANNED loss of ALL of the following off-site communications capability:
   - ENN (Emergency Notification Network)
   - ENS (Emergency Notification System)
   - Commercial phones (Radio, PBX, Satellite, Wireless)
   - VOIP (Voice Over Internet Protocol)
   - OPX (Off Premise Extension)

NATURAL / DESTRUCTIVE

HU1 - Natural and Destructive Phenomena Affecting the PROTECTED AREA (pg. 97)

1. Seismic monitoring system indicates seismic event OR plant operators report an earthquake was "felt".

OR

2. Report by plant personnel of tornado OR high winds greater than 100 mph striking within PROTECTED AREA.

OR

3. Crash of vehicle, large enough to cause significant damage, into plant structures containing functions or systems required for safe shutdown within the PROTECTED AREA boundary.

OR

4. Report by plant personnel of an unanticipated EXPLOSION within the PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure OR equipment.

OR

5. Report of turbine failure resulting in casing penetration OR damage to turbine OR generator seals.

OR

7. Sustained hurricane force winds greater than 73 mph forecast to be at the plant site in the next four hours in accordance with 11889-C, Severe Weather Checklist.

FIRE / EXPLOSION

HU2 - FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection (pg. 99)

1. FIRE in buildings OR areas contiguous to any of the following areas NOT extinguished within 15 minutes of control room notification OR verification of a control room alarm unless disproved by personnel observation:

Containment Building
Fuel Handling Building
NSCW Cooling Towers
Control Building
Diesel Generator Building
Auxiliary Building
Diesel Fuel Oil Storage Tank Pumphouse
Auxiliary Feedwater Pumphouse

TOXIC / FLAMMABLE

HU3 - Release of Toxic, Asphyxiant, OR Flammable Gases Deemed Detrimental to Normal Operation of the Plant (pg. 101)

1. Report OR detection of toxic, asphyxiant, OR flammable gas that has OR could enter the Owner Controlled Area in amounts greater than life threatening OR flammable concentrations that can affect NORMAL PLANT OPERATIONS.

OR

2. Report by Local, County, OR State Officials for evacuation OR sheltering of site personnel based on an offsite toxic, asphyxiant, OR flammable gas event.

SECURITY

HU4 - Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant (pg. 102)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by security shift supervision.

OR

2. A CREDIBLE VEGP security THREAT notification.

OR

3. A validated notification from NRC providing information of an aircraft threat.

ED DISCRETION

HU5 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE. (pg. 104)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

FIGURE 3

Vogtle Electric Generating Plant — Procedure Number: NMP-EP-110-GL03 Rev 4.0

EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS

Figure 3 - COLD INITIATING CONDITION EMERGENCY ACTION LEVEL MATRIX - MODES 5, 6 AND DEFUELED ONLY

Categories: **RADIOLOGICAL** (EFFLUENTS, RAD LEVELS) | **System Malfunctions** (AC / DC POWER, RX and CORE, HEAT REMOVAL, COMMUNICATIONS) | **HAZARDS** (NATURAL / DESTRUCTIVE, FIRE / EXPLOSION, TOXIC / FLAMMABLE, SECURITY, CR EVACUATION, ED DISCRETION)

GENERAL EMERGENCY

EFFLUENTS

RG1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mR TEDE **OR** 5000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. (pg. 36)

See Notes 1 and 2 →

1. VALID reading on any of the following radiation monitors that exceeds **OR** expected to exceed the reading shown for 15 minutes **OR** longer:

RE-12839	2.5 x 10^{3} µCi/cc
RE-12444	1.2 x 10^{1} µCi/cc
RE-13119 thru 13122	5.2 x 10^{3} µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 1000 mR TEDE **OR** 5000 mR thyroid CDE at **OR** beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates exceeding 1000 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 5000 mR for one hour of inhalation, at **OR** beyond the site boundary.

RAD LEVELS

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note 2: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

RX and CORE

CG1 - Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV. **Modes 5 and 6 Only**. (pg. 93)

1. Loss of RPV inventory as indicated by unexplained Containment sump level rise, , Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

AND

2. RPV Level:
a. Less than 63% on RVLIS full range for greater than 30 minutes
OR
b. RPV level **CANNOT** be monitored **WITH** indication of core uncovery for greater than 30 minutes as evidenced by ANY of the following:
- RE-005 **OR** 006 greater than162 R/hr
- RE-011 greater than 15 mR/hr
- Erratic Source Range Monitor Indication.

AND

3. Containment challenged as indicated by ANY of the following:
- Explosive mixture inside containment greater than **OR** equal to 6% H_2
- Containment Pressure greater than **OR** equal to 3 psig **WITH** CONTAINMENT CLOSURE established
- Containment Pressure greater than **OR** equal to 52 psig **WITH** Tech Spec containment integrity intact
- CONTAINMENT CLOSURE **NOT** established.

SECURITY

HG1 – HOSTILE ACTION resulting in loss of physical control of the facility (pg. 119)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

ED DISCRETION

HG2 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency. (pg. 120)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation ormelting with potential for loss of containment integrity
OR
HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

EFFLUENTS

RS1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE **OR** 500 mR Thyroid CDE for the Actual or Projected Duration of the Release. (pg. 34)

See Notes 3 and 4 →

1. VALID reading on any of the following radiation monitors that exceeds **OR** is expected to exceed the reading shown for 15 minutes OR longer:

RE-12839	2.5 x 10^{2} µCi/cc
RE-12444	1.2 µCi/cc
RE-13119 thru 13122	5.2 x 10^{2} µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE **OR** 500 mR thyroid CDE at **OR** beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at **OR** beyond the site boundary.

RAD LEVELS

Note 3: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note 4: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

Distractor

RX and CORE

CS1 - Loss of RPV Inventory Affecting Core Decay Heat Removal Capability **Mode 5 Only** (pg. 89)

1. Loss of Reactor Pressure Vessel (RPV) inventory affecting core decay heat removal capability with CONTAINMENT CLOSURE NOT established as indicated by:
a. RPV level less than 72% on RVLIS full range [6" below Bottom ID of loop]
OR
b. RPV level CANNOT be monitored for greater than 30 minutes with a possible loss of RPV inventory as indicated by unexplained level rise in Containment sump, Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

OR

2. Loss of RPV inventory affecting core decay heat removal capability with CONTAINMENT CLOSURE established as indicated by:
a. RPV level less than 63% on RVLIS full range [TOAF]
OR
b. RPV level CANNOT be monitored for greater than 30 minutes with a possible loss of RPV inventory as indicated by EITHER of the following:
Unexplained Containment sump level rise or
Erratic Source Range Monitor Indication.

HEAT REMOVAL

CS2 - Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV. **Mode 6 Only** (pg. 91)

1. **WITH** CONTAINMENT CLOSURE **NOT** established:
a. RPV level less than 72% on RVLIS full range [6" below Bottom ID of loop]
OR
b. RPV level CANNOT be monitored WITH indication of core uncovery as evidenced by ANY of the following:
RE-002, 003, OR 004 greater than 3.1 mR/hr or
Erratic Source Range Monitor Indication.

OR

2. **WITH** CONTAINMENT CLOSURE established:
a. RPV level less than 63% on RVLIS full range [TOAF]
OR
b. RPV level CANNOT be monitored WITH indication of core uncovery as evidenced by ANY of the following:
RE-005 OR 006 greater than 162 R/hr,
RE-011 greater than 15 mR/hr,
Erratic Source Range Monitor Indication.

SECURITY

HS4 – HOSTILE ACTION within the PROTECTED AREA (pg. 118)

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by security shift supervision.

CR EVACUATION

HS2 – Control Room Evacuation Has Been Initiated AND Plant Control Cannot Be Established (pg. 116)

1. a. Control Room evacuation has been initiated
AND
b. Control of the plant **CANNOT** be established per 18038 – Operation From the Remote Shutdown Panels, within 15 minutes.

ED DISCRETION

HS3- Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency (pg. 117)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual **or** likely major failures of plant functions needed for protection of the public
OR
HOSTILE ACTION that results in intentional damage **or** malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels that exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

ALERT EMERGENCY

EFFLUENTS

RA1 - Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer. (pg. 27)

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

1. VALID reading on any of the following effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes OR longer.

Monitor	200 X Setpoint Value
RE-0018	2.8 x 10^{-3} µCi/cc
RE-0021	4.0 x 10^{-3} µCi/cc
RE-0848	3.5 x 10^{-3} µCi/cc
ARE-0014	2.4 x 10^{2} µCi/cc
RE-12839	7.9 x 10^{-1} µCi/cc
RE-12444C	1.3 x 10^{-1} µCi/cc

OR

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes **OR** longer:
Main Steam RE-13119 thru 13122 2.5 µCi/cc

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations **OR** release rates in excess of 200 times Technical Specification 5.5.4 as confirmed by the ODCM, with a release duration of 15 minutes **OR** longer.

RAD LEVELS

RA2 - Damage to Irradiated Fuel **OR** Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. (pg. 30)

1. UNPLANNED VALID alarm on any of the following radiation monitors:
- Fuel Handling Building RE-008
- CNMT BLDG Low Range** RE-002/003
** Mode 6 during fuel transfer
- Fuel Handling BLDG EFFL. ARE-2532 A/B
- Fuel Handling BLDG EFFL. ARE-2533 A/B

OR

2. Loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel as indicated by ANY of the following:
- Personnel report during fuel assembly movements.
- Spent Fuel Pool Storage Less than El 193'-5"
- Transfer Canal Transit El Less than El 186'10"
- Reactor Core El Less than El 181'-10" (63% on RVLIS full range)

RA3 Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown (pg. 32)

1. VALID radiation monitor readings greater than 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:
Control Room radiation monitor RE-001 OR Central Alarm Station (by survey)

OR

2. UNPLANNED VALID radiation readings greater than 1 R/hr values in areas requiring infrequent access to maintain plant safety functions.

South & North Main Steam Valve Rooms, Turbine Building, Control Building, Fire Pump House, Aux Feed Water Pump House, Low Voltage Switchyard, Diesel Generator Buildings, NSCW Towers

AC / DC POWER

CA3 - Loss of All Offsite Power **AND** Loss of All Onsite AC Power to Essential Busses. (pg. 86)

1. a. Loss of power to **OR** from transformers 1(2)NXRA **AND** 1(2)NXRB resulting in loss of all off-site electrical power to **BOTH** 1(2)AA02 **AND** 1(2)BA03 for greater than 15 minutes.
AND
b. Failure of emergency generators to supply power to emergency busses.
AND
c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

RX and CORE

CA1 - Loss of RCS Inventory. **Mode 5 Only** (pg. 84)

1. Loss of RCS inventory as indicated by RPV level less than elevation 185'-9.5" (185.79') (73% on RVLIS full range)

OR

2. a. RCS level **CANNOT** be monitored for greater than 15 minutes
AND
b. A possible loss of RCS inventory may be occurring as indicated by unexplained level rise in Containment sump, Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

CA2 - Loss of RPV Inventory with Irradiated Fuel in the RPV. **Mode 6 Only** (pg. 85)

1. Loss of inventory as indicated by RPV level less than 185'-9.5" (185.79') (73% on RVLIS full range)

OR

2. a. RPV level **CANNOT** be monitored for greater than 15 minutes
AND
b. A possible loss of RCS inventory may be occurring as indicated by unexplained level rise in Containment sump, , Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

HEAT REMOVAL

CA4 - Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV. **Modes 5 and 6 Only.** (pg. 87)

Note The Emergency Director should not wait until 20 minutes has elapsed, but should declare the event as soon as it is determined that the duration has or will likely exceed 20 minutes.

1. An UNPLANNED event results in RCS temperature exceeding 200°F with:
a. CONTAINMENT CLOSURE **NOT** established
AND
b. RCS integrity **NOT** established.

OR

NOTE: If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this Threshold Value is not applicable.

2. An UNPLANNED event results in RCS temperature exceeding 200°F for greater than 20 minutes (Note) with:
a. CONTAINMENT CLOSURE established
AND
b. RCS integrity NOT established
OR
c. RCS inventory reduced.

OR

3. An UNPLANNED event results in:
a. RCS temperature exceeding 200°F for greater than 60 minutes (Note)
OR
b. RCS pressure increasing greater than 10 psig.

NATURAL / DESTRUCTIVE

HA1 - Natural and Destructive Phenomena Affecting the Plant VITAL AREA (pg. 105)

1. Seismic monitoring system confirms seismic event greater than OBE Earthquake acceleration of 0.12 g.

OR

2. Tornado **OR** high winds greater than 100 mph within the PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures/equipment (Table 1) **OR** the Control Room has indication of degraded performance of any listed systems: (Table 1)

OR

3. Vehicle crash within PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures **OR** equipment therein (Table 1) **OR** control indication of degraded performance of those systems (Table 1)

OR

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to **OR** penetration of any area containing safety-related equipment, their controls **OR** their power supplies. (Table 1)

OR

6. Sustained hurricane winds onsite resulting in VISIBLE DAMAGE to plant structures within the PROTECTED AREA boundary containing equipment necessary for safe shutdown. (Table 1)

Table 1
- Containment Building
- Fuel Hand Building
- NSCW Cooling Towers
- Control Building
- Diesel Generator Building
- Auxiliary Building
- Diesel Fuel Oil Storage Tank Pumphouse
- Auxiliary Feedwater Pumphouse

FIRE / EXPLOSION

HA2 - FIRE **OR** EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown (pg. 108)

1. FIRE **OR** EXPLOSION in a plant VITAL AREA **AND** affected system parameter indications show degraded performance **OR** plant personnel report VISIBLE DAMAGE to permanent structures **OR** safety related equipment in any of the following:
- Containment Building
- Fuel Hand Building
- NSCW Cooling Towers
- Control Building
- Diesel Generator Building
- Auxiliary Building
- Diesel Fuel Oil Storage Tank Pumphouse
- Auxiliary Feedwater Pumphouse

TOXIC / FLAMMABLE

HA3 - Release of Toxic, Asphyxiant, or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown (pg. 110)

1. Report **OR** detection of toxic **OR** asphyxiant gas within **OR** contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report **OR** detection of flammable gases in concentration greater than the LOWER FLAMMABILITY LIMIT within **OR** contiguous to a VITAL AREA.

SECURITY

HA4 – HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat. (pg. 112)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by security shift supervision.

OR

2. A validated notification from the NRC of an airliner attack threat within 30 minutes of the site.

CR EVACUATION

HA5 - Control Room Evacuation Has Been Initiated. (pg. 114)

1. Entry into 18038 – Operation From the Remote Shutdown Panels, procedure for Control Room evacuation.

ED DISCRETION

HA6 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. (pg. 115)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant
OR
a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

NOTIFICATION OF UNUSUAL EVENT

EFFLUENTS

RU1 – Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. (pg. 23)

Note: The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes **OR** longer.

Monitor	2 X Setpoint Value
RE-0018	2.8 x 10^{-5} µCi/cc
RE-0021	4.0 x 10^{-5} µCi/cc
RE-0848)	3.5 x 10^{-5} µCi/cc
ARE-0014	2.4 µCi/cc
RE-12839	7.9 x 10^{-3} µCi/cc
RE-12442C or 1RE-12444C	1.3 x 10^{-3} µCi/cc

OR

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes **OR** longer:
Main Steam (RE-13119-RE-13122) 2.5 x 10^{-2} µCi/cc

OR

3. Confirmed sample analyses for gaseous **OR** liquid releases indicates concentrations **OR** release rates, with a release duration of 60 minutes **OR** longer, in excess of two times Technical Specification 5.5.4, as confirmed by the ODCM.

RAD LEVELS

RU2 – Unexpected Rise in Plant Radiation (pg. 25)

1. a. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, **OR** fuel transfer canal with all irradiated fuel assemblies remaining covered by water indicated by any of the following:
- Personnel report of low water level
- LSHL-0625 offscale low
- Personnel report of cavitation OR low discharge pressure for SFP (1/2-1213-P6-002 OR -005) AND/ OR RHR (1/2-1205-P6-001 OR -002) Pumps

AND

b. UNPLANNED VALID Direct Area Radiation Monitor readings rise on any of the following:
- RE-0008 in the fuel building
- RE-0002, -0003, -0004 in containment
- RE-0011 at the seal table
- RE-0005, -0006 in containment

OR

2. UNPLANNED VALID Direct Area Radiation Monitor readings rise by a factor of 1000 over normal* levels.
- RE-0001 Control Room
- RE-0002 Cnmt Bldg Low Range
- RE-0003 Cnmt Bldg Low Range
- RE-0004 Cnmt Bldg Per. Hatch
- RE-0005 Cnmt Bldg High Range
- RE-0006 Cnmt Bldg High Range
- RE-0007B Sampling Room
- RE-0008 FHB
- RE-0011 Cnmt Bldg Incore Instr
- ARE-16971 RPF North Wall General Area
- ARE-16972 RPF East Wall General Area
- ARE-16973 RPF Dress Out Area

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

AC / DC POWER

CU3 - Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes. (pg. 78)

1. a. Loss of power to **OR** from transformers 1(2)NXRA **AND** 1(2)NXRB resulting in loss of all off-site electrical power to **BOTH** 1(2)AA02 **AND** 1(2)BA03 for greater than 15 minutes
AND
b. One emergency diesel generators supplying power to **EITHER** 1(2)AA02 **OR** 1(2)BA03.

Answer

CU7 - UNPLANNED Loss of Required DC Power for Greater Than 15 Minutes. **Modes 5 and 6 Only.** (pg. 82)

1. a. UNPLANNED loss of Vital DC power to 125 VDC Buses AD1, BD1, CD1, AND DD1 indicated by bus voltage indications less than 105 VDC
AND
b. Failure to restore power to at least one DC bus within 15 minutes from the time of loss.

RX and CORE

CU8 - Inadvertent Criticality **Modes 5 and 6 Only**. (pg. 83)

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

CU1 - RCS Leakage **Mode 5 Only.** (pg. 76)

1. Unable to establish or maintain pressurizer level greater than 17%.

CU2 - Unplanned Loss of RCS Inventory with Irradiated Fuel in the RPV **Mode 6 Only** (pg. 77)

1. UNPLANNED RCS level lowering below 194' (RPV flange) for greater than 15 minutes

OR

2. a. RPV level **CANNOT** be monitored
AND
b. A possible loss of RPV inventory may be occurring as indicated by unexplained level rise in Containment sump, Reactor Coolant Drain Tank (RCDT) or Waste Holdup Tank (WHT).

HEAT REMOVAL

CU4 - Unplanned Loss of Decay Heat Removal Capability with Irradiated Fuel in the RPV. **Modes 5 and 6 Only.** (pg. 79)

NOTE The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the duration has or will likely exceed 15 minutes.

1. An UNPLANNED event results in RCS temperature exceeding 200°F.

OR

2. Loss of all RCS temperature **AND** RPV level indication for greater than 15 minutes.

COMMUNICATIONS

CU6 - UNPLANNED Loss of All Onsite **OR** Offsite Communications Capabilities (pg. 81)

1. UNPLANNED loss of ALL of the following on-site communications capability affecting the ability to perform routine operations:
- In plant telephones
- Public address system
- Plant radio systems

OR

2. UNPLANNED loss of ALL of the following off-site communications capability:
- ENN (Emergency Notification Network)
- ENS (Emergency Notification System)
- Commercial phones (Radio, PBX, Satellite, Wireless)
- VOIP (Voice Over Internet Protocol)
- OPX (Off Premise Extension)

NATURAL / DESTRUCTIVE

HU1 - Natural and Destructive Phenomena Affecting the PROTECTED AREA (pg. 97)

1. Seismic monitoring system indicates seismic event OR plant operators report an earthquake was "felt".

OR

2. Report by plant personnel of tornado **OR** high winds greater than 100 mph striking within PROTECTED AREA.

OR

3. Crash of vehicle, large enough to cause significant damage, into plant structures containing functions or systems required for safe shutdown within the PROTECTED AREA boundary.

OR

4. Report by plant personnel of an unanticipated EXPLOSION within the PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure **OR** equipment.

OR

5. Report of turbine failure resulting in casing penetration **OR** damage to turbine **OR** generator seals.

OR

7. Sustained hurricane force winds greater than 73 mph forecast to be at the plant site in the next four hours in accordance with 11889-C, Severe Weather Checklist.

E-HU1 - Damage to a loaded cask CONFINEMENT BOUNDARY (pg. 122)

1. Damage to a loaded dry fuel storage cask CONFINEMENT BOUNDARY due to natural phenomena events, accident conditions **OR** any condition in the opinion of the Emergency Director that affects **OR** causes a loss of loaded dry fuel storage cask CONFINEMENT BOUNDARY

FIRE / EXPLOSION

HU2 - FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection (pg. 99)

1. FIRE in buildings **OR** areas contiguous to any of the following areas **NOT** extinguished within 15 minutes of control room notification **OR** verification of a control room alarm unless disproved by personnel observation:
- Containment Building
- Fuel Hand Building
- NSCW Cooling Towers
- Control Building
- Diesel Generator Building
- Auxiliary Building
- Diesel Fuel Oil Storage Tank Pumphouse
- Auxiliary Feedwater Pumphouse

TOXIC / FLAMMABLE

HU3 - Release of Toxic, Asphyxiant, or Flammable Gases Deemed Detrimental to Normal Operation of the Plant (pg. 101)

1. Report **OR** detection of toxic, asphyxiant, **OR** flammable gas that has **OR** could enter the Owner Controlled Area in amounts greater than life threatening **OR** flammable concentrations that can affect NORMAL PLANT OPERATIONS.

OR

2. Report by Local, County, **OR** State Officials for evacuation **OR** sheltering of site personnel based on an offsite toxic, asphyxiant, **OR** flammable gas event.

SECURITY

HU4 - Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant (pg. 102)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by security shift supervision.

OR

2. A CREDIBLE VEGP security THREAT notification.

OR

3. A validated notification from NRC providing information of an aircraft threat.

ED DISCRETION

HU5 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE (pg. 104)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

1. WE11EA2.02 001/LOIT/SRO/M/F 3.4/3.9/WE11EA2.02/LO-TA-37020//HL15 NRC/

Initial conditions:

- Unit 1 experienced a LOCA.
- 19111-C, "Loss of Emergency Coolant Recirculation," was entered.
- RCS cooldown to cold shutdown has been initiated.
- RWST level is 8% and slowly lowering.

Current condition:

- Critical Safety Function Status Tree (CSFST) is ORANGE on Integrity.

Which one of the following completes the following statement?

The crew is required to __(1)__,

and

then the Shift Supervisor __(2)__ required to transition to 19241-C, "Response to Imminent Pressurized Thermal Shock Condition."

A.✓ (1) stop all pumps taking suction from the RWST

(2) is

B. (1) reduce ECCS flow from the RWST to one running train

(2) is

C. (1) stop all pumps taking suction from the RWST

(2) is NOT

D. (1) reduce ECCS flow from the RWST to one running train

(2) is NOT

K/A

W/E11 Loss of Emergency Coolant Recirc. / 4

EA2.02 -Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation):

- Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

K/A MATCH ANALYSIS

The question requires the candidate to make two decisions based on plant conditions while in 19111-C, "Loss of Emergency Coolant Recirculation" - stopping all ECCS pumps and transitioning out of 19111-C. Both decisions challenge the candidate's ability to adhere to the rules for using EOPs in compliance with WOG and facility requirements. Actions taken in accordance with these EOPs are part of the bases for granting the facility's license.

EXPLANATION OF REQUIRED KNOWLEDGE

Per 19111-C steps 6 and 33 if RWST level lowers to <8%, all ECCS pumps taking suction from the RWST are to be placed in Pull-to-Lock (PTL).

Transition to any ORANGE or RED CSFST will be made when conditions are met. Per the rules of EOP usage, CSFSTs are initiated when either step 22 of 19000-C is reached, or a transition out of 19000-C is made. CSFSTs remain in effect during the entire EOP network unless otherwise directed. 19111-C does NOT contain any exceptional guidance on CSFST implementation.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. Per continuous actions step 6 and steps 33 and 34, if RWST lowers to <8%, all ECCS pumps taking suction from the RWST are placed in PTL.

The second part is correct. CSFST monitoring was initiated on transition out of 19000-C. There is no guidance in 19111-C that prohibits actions based on CSFSTs. Therefore, a transition to 19241-C will be made as soon as ORANGE path conditions are verified.

B. Incorrect. Plausible. The first part is incorrect. Per continuous actions step 6 and steps 33 and 34, if RWST lowers to <8%, all ECCS pumps taking suction from the RWST are to be placed in PTL. However, step 15 does reduce ECCS flow to only one train. This is a mitigation strategy that prolongs RWST inventory. A candidate who does not possess the knowledge of the overall mitigating strategy of 19111-C could find it unreasonable to stop all ECCs pumps with a LOCA in progress and then transition to another procedure that does not address loss of injection flow.

The second part is correct. See the second part of choice A above.

C. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. There is no guidance in 19111-C that prohibits actions based on CSFSTs. Therefore, a transition to 19241-C would be made as soon as conditions for the ORANGE path are verified to exist. However, step 1 of EOP 19113-C, "Recirculation Sump Blockage" directs the operator to "initiate monitoring CSFSTs for information only. Function

Restoration Procedures (FRP) should NOT be implemented." A candidate may confuse the two EOPs and believe transition out of 19111-C on CSFSTs is not allowed.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice B above.

The second part is incorrect. See the second part of choice C above.

SRO JUSTIFICATION (10CFR43(b))

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

-Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? No, CSFSTs are symptom based, not system based.
-Can the question be answered *solely* by knowing immediate operator actions? No, stopping the ECCS pump and transitioning to FRPs is not governed by IOA's.
-Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? No, the decisions made in the question require specific step knowledge.
-Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? No, the decision to transition is based on specific parameters and direction.
-Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- **Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures. Yes, the question requires the SRO to make a transition decision to an FRP after stopping all ECCS pumps. The decision requires application of EOP rules of usage, knowledge of specific limitations on FRP implementation, and a high level knowledge of ECA and FRP mitigations strategy and interrelationships.**
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Level:	SRO
Tier # / Group #	T1 / G1
K/A#	WE11EA2.02
Importance Rating:	3.4 / 3.9
Technical Reference:	19111-C REV 33.2, pages 6, 11, & 21
References provided:	None
Learning Objective:	LO-LP-37114-12 State the intent of EOP 19111, Loss of Emergency Coolant Recirculation. LO-PP-37117-04 Describe the differences between the actions for 19113-C and 19111-C and the reason for the differences. LO-TA-37020 Respond to a Loss of Emergency Coolant Recirculation Capability per 19111-C
Question origin:	BANK - HL15 Question # WE11EG2.4.2
Cognitive Level:	M/F
10 CFR Part 55 Content:	43.5 / 45.13
Comments:	Question appears to match the KA. Transitioning to the FRG's on an Orange path is not SRO-only knowledge. Knowledge of what to do when RWST reaches 8% may be SRO-only knowledge. Since the 8% RWST does not govern a procedure transition, please make sure this is required knowledge of the operators (i.e., not minutia). - JAT 12/19/2013 (SAT)

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

If offsite power is lost after SI reset, action is required to restart the following ESF equipment if plant conditions require their operation:

- RHR Pumps
- SI Pumps
- Post-LOCA Cavity Purge Unit
- Containment Coolers in low speed (Started in high speed on a UV signal).
- ESF Chilled Water Pumps (IF CRI is reset).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. Reset SI if necessary.	4. IF SI will NOT reset, THEN initiate ATTACHMENT E.
5. Check Containment Cooling Units - RUNNING IN LOW SPEED.	5. Start Cooling Units in low speed.
*6. **Check RWST level - GREATER THAN 8%.**	*6. Go to Step 33.
7. Determine Containment Spray requirements:	
a. Check CS Pump suction - FROM RWST:	a. IF CS Pump suction from Sump, THEN go to Step 9.
• HV-9017A - CNMT SPRAY PMP-A RWST SUCT ISO VLV - OPEN	
• HV-9017B - CNMT SPRAY PMP-B RWST SUCT ISO VLV - OPEN	

° Step 7 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. Check if ECCS is in service: CCPs - ANY RUNNING. -OR- BIT NOT ISOLATED. -OR- RHR Pumps - ANY RUNNING IN INJECTION MODE.	14. Go to Step 24.
15. Establish one train of ECCS flow:	
a. CCP - ONLY ONE RUNNING.	a. Start or stop a CCP to establish only one Pump running.
b. SI Pump - ONLY ONE RUNNING.	b. Start or stop an SI Pump to establish only one Pump running.
c. RCS pressure - LESS THAN 300 PSIG.	c. Stop RHR Pumps. Go to Step 16.
d. RHR Pump - ONLY ONE RUNNING.	d. Start or stop an RHR Pump to establish only one Pump running.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***30. Check if RCPs must be stopped:**	
a. Check the following: Seal number 1 differential pressure - LESS THAN 200 PSID. -OR- Seal number 1 leakoff flow - LESS THAN 0.2 GPM.	a. IF neither condition satisfied, THEN go to Step 31.
b. Stop affected RCPs.	
c. Close Spray Valve for idle RCP: RCP 1: PIC-0455C RCP 4: PIC-0455B	
31. Check RCS WR Hot Leg temperature - GREATER THAN 200°F.	31. Go to Step 45.
32. Check RWST level - LESS THAN 8%.	32. Return to Step 2.
33. Stop Pumps taking suction from RWST and place switches in PULL-TO-LOCK positions: • RHR Pumps • SI Pumps • CCPs • CS Pumps	