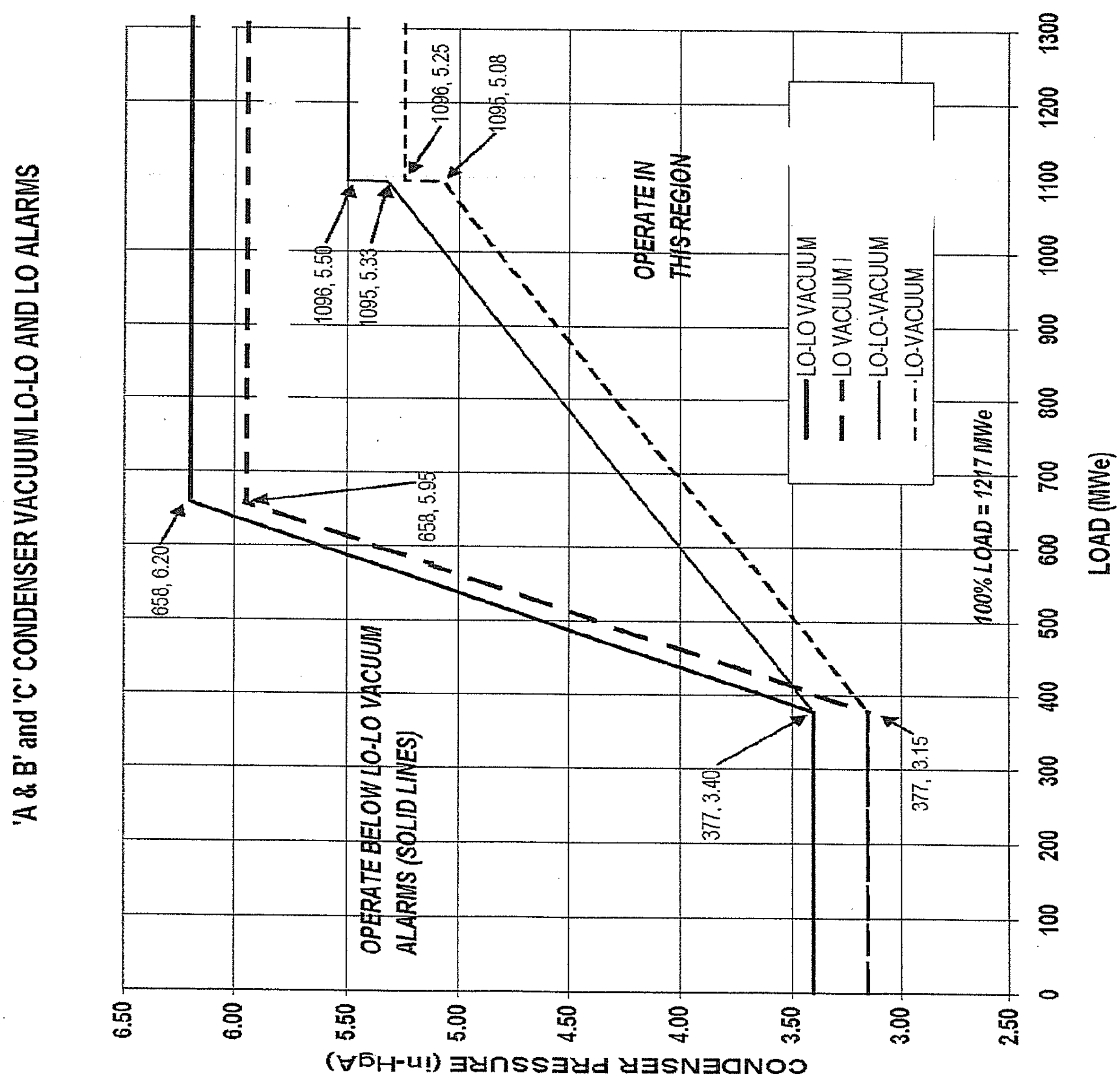
WBN 10-2011 NRC SRO EXAM as Submitted
<u>08/15/2011</u>

REFERENCE PACKAGE

1. Steam Tables
2. AOI-11, Appendix A, Condenser Vacuum ICS Graph, (1 page)
3. 0-SI-0-3, Weekly Log, Appendix A, (2 pages)
4. ICS 'AFD TARGET DISPLAY', (1 page)
5. ECA-1-1, Loss of RHR Sump Recirculation, (2 pages)
6. Tech Spec 3.6.12, Ice Condenser Doors (5 pages)
7. EPIP-1, "Emergency Plan Classification Logic," (1 page)
8. AOI-30.2 APP B, Fire Safe Shutdown Elevation Diagrams 2.0 AB EL 772.0, 776.0 & 763.5 ELEVATION DIAGRAM (1 page)

Appendix A
(Page 1 of 1)

Condenser Vacuum ICS Graph



NOTE: This Appendix is an ICS Controlled Graph and should **NOT** be modified without Corporate Computer Engineering acknowledgement.

Appendix A
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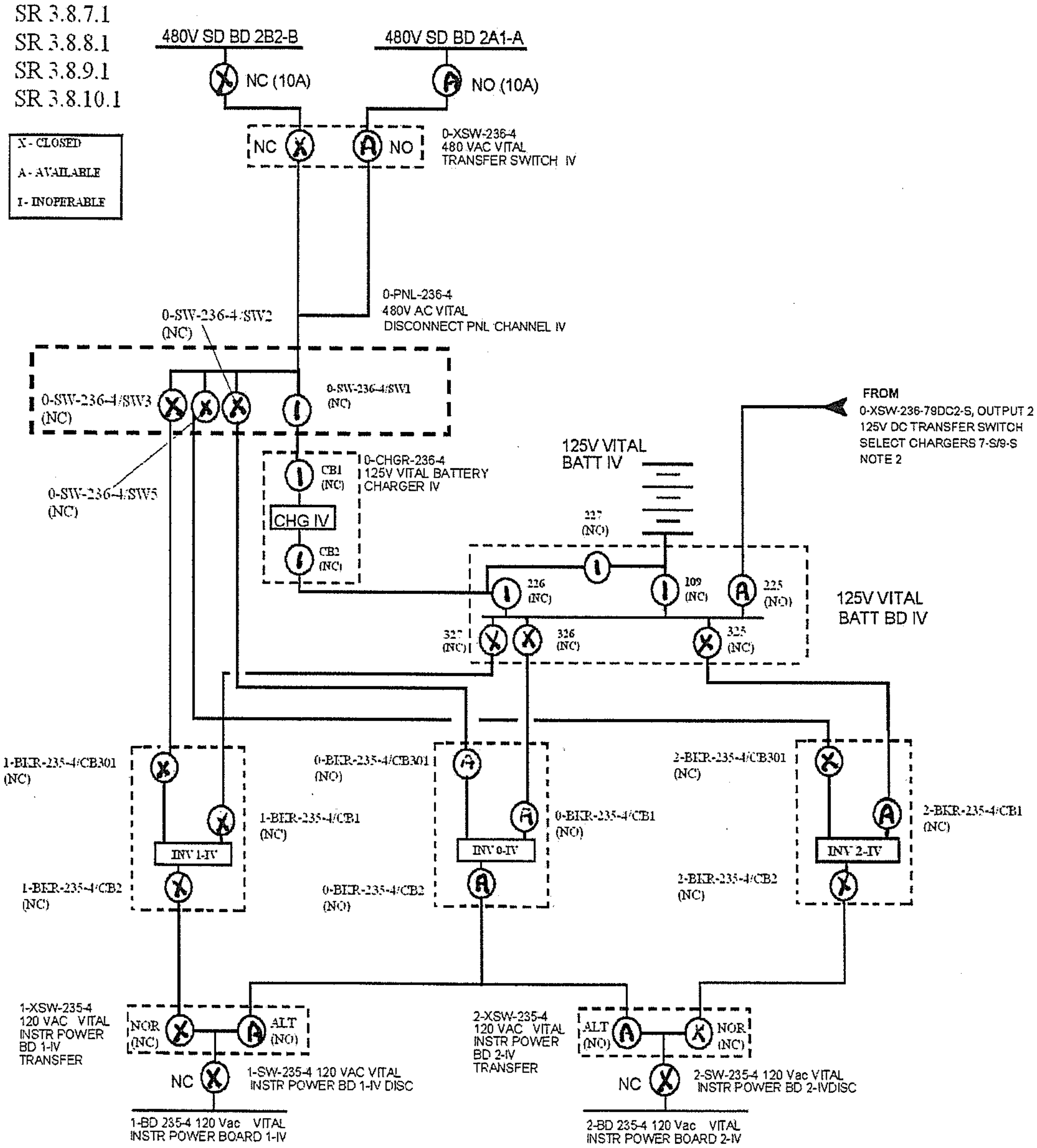
SR 3.8.4.3
SR 3.8.5.1
SR 3.8.7.1
SR 3.8.8.1
SR 3.8.9.1
SR 3.8.10.1

X - CLOSED
A - AVAILABLE
I - INOPERABLE



NOTE (1) In Modes 5&6 only one train of ac /dc PWR is required, if this train is not required this page may be N/A.
(2) When 7-S or 9-S charger is connected to Batt Bd then verify assoc. train Transfer Switches are closed and Alt bkrs are open.

INITIALS OF DATA COLLECTOR: ____ DATE ______

REMARKS: ______

Appendix A
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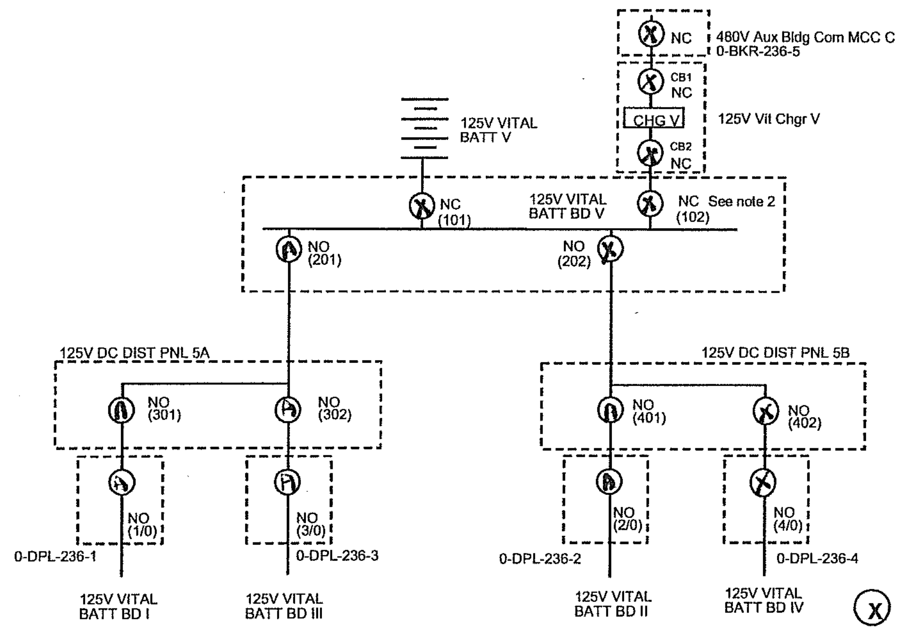
SR 3.8.9.1
SR 3.8.10.1



(X) CLOSED

(A) AVAILABLE

(I) INOPERABLE

NOTE 1: Only one battery board may be feed from the 5th battery board at any one time.

NOTE 2: 0-BKR-236-5/102 must be OPEN when 5th battery is inservice for battery bd I, II, III, or IV

INITIALS OF DATA COLLECTOR: ____________ DATE ____________

REMARKS: ____________

TVA AFD TARGET DISPLAY

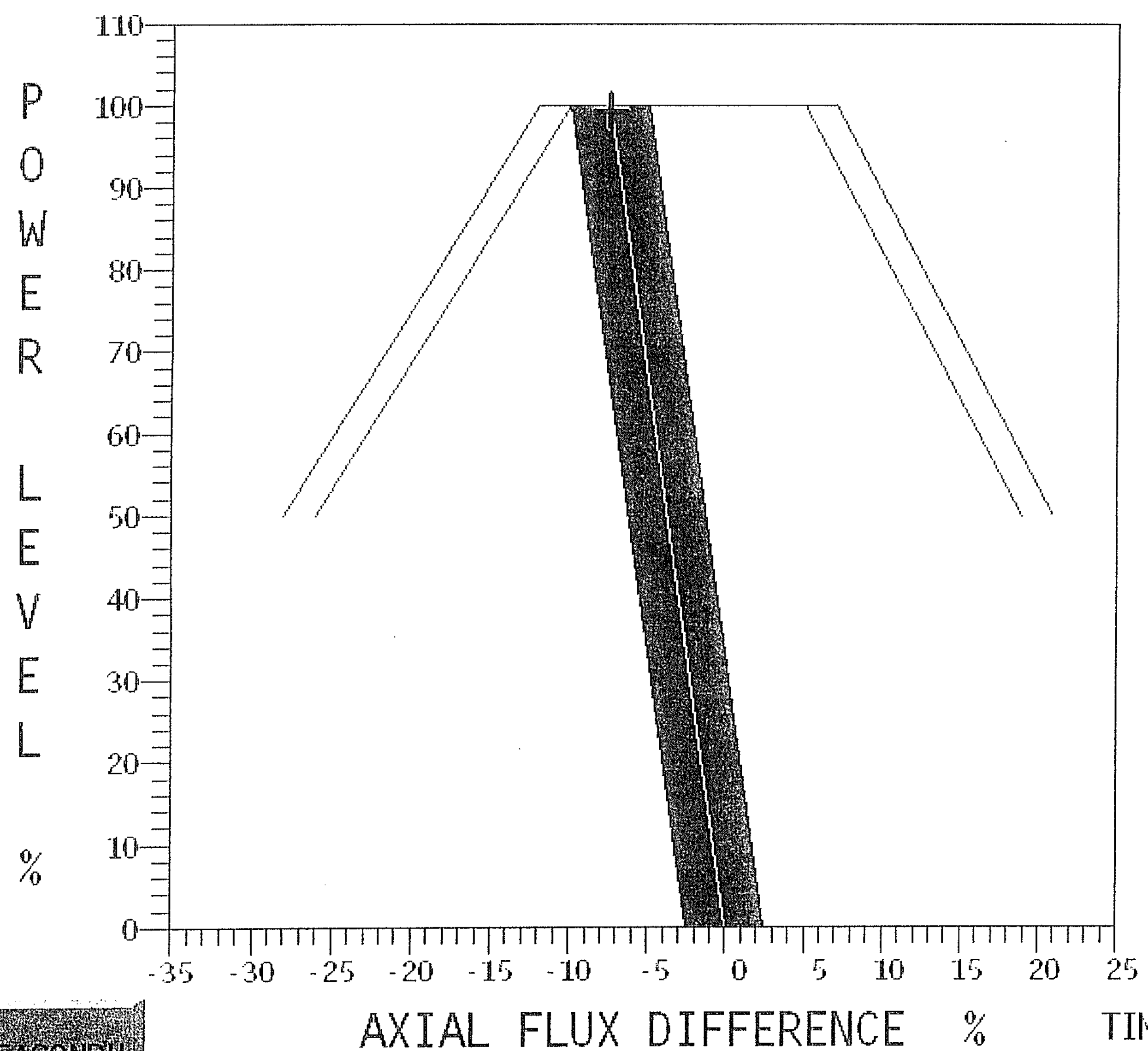
Main Alarms Graphics Trends Points Zoom/Layers Print Help

06-JUL-2011 07:46:16

SELECT FUNC. KEY OR TURN-ON CODE DOGHOUSE>

S C H P I

M



POWER LEVEL	99.4 %
CTRL BANK D (STEPS)	220.0
AFD NIS CHANNEL 41	-7.6 %
AFD NIS CHANNEL 42	-7.4 %
AFD NIS CHANNEL 43	-7.7 %
AFD NIS CHANNEL 44	-7.4 %
NIS ACTUAL AFD	-7.5 %
NIS TARGET AFD	-7.5 %
AFD LOW LIMIT	%
AFD HIGH LIMIT	%
CONTROL BAND LOW LIM	%
CONTROL BAND HIGH LIM	%
TIME OUT OF BAND ACCUM (MIN)	0.0

BEACONDH

PREVIOUS (F7) CANCEL (ESC) F1=CLEAR F2= F3= F4= F5= F6=

PG UP PG DN TT059 WK=ANONYMOUS SEC LVL= 15 PRIM/BACK CPU I MODE 1

19. **CHECK** SI termination criteria:

a. RVLIS greater than 60% with NO RCP running,

OR

RVLIS greater than 63% with ANY RCP running.

a. **IF** RVLIS is less than or equal to setpoint, **THEN**

** **GO TO** Step 25.

b. RCS subcooling greater than required from table:

b. **ESTABLISH** minimum ECCS flow for decay heat removal:

1) **REFER TO** Figure 1, Minimum SI Flow For Decay Heat Versus Time After Trip.

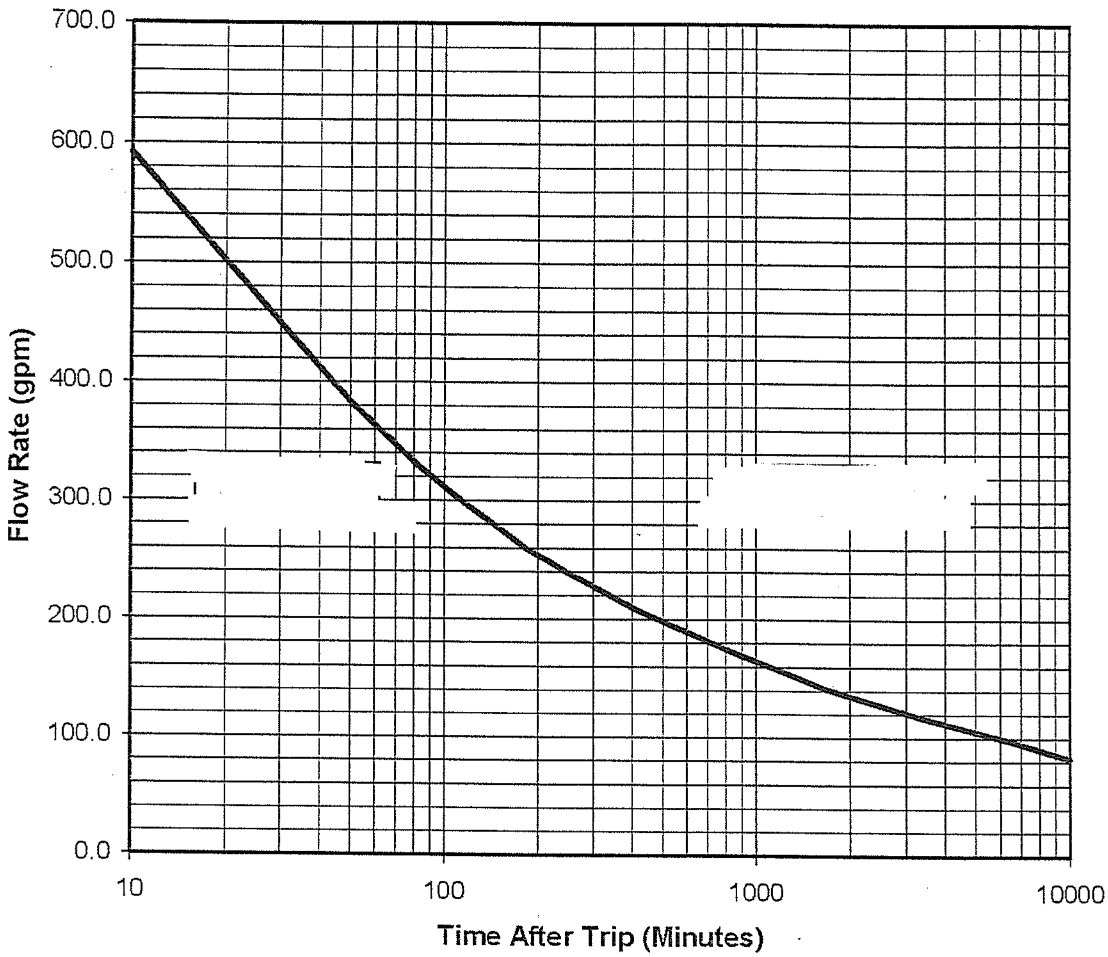
2) ** **GO TO** Step 25.

RCS PRESS BETWEEN	REQUIRED SUBCOOLING
285 AND 585 psig	115°F [135°F ADV]
585 AND 1085 psig	102°F [123°F ADV]
1085 AND 1885 psig	97°F [117°F ADV]
Greater than 1885 psig	94°F [114°F ADV]

20. **RESET** Phase A and Phase B.

Figure 1
(Page 1 of 1)

Minimum SI Flow for Decay Heat vs. Time After Trip



3.6 CONTAINMENT SYSTEMS

3.6.12 Ice Condenser Doors

LCO 3.6.12 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be OPERABLE and closed.

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

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each ice condenser door.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ice condenser inlet doors inoperable due to being physically restrained from opening.	A.1 Restore inlet door to OPERABLE status.	1 hour
B. One or more ice condenser doors inoperable for reasons other than Condition A or not closed.	B.1 Verify maximum ice bed temperature is $\leq 27°F$. AND B.2 Restore ice condenser door to OPERABLE status and closed positions.	Once per 4 hours 14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Restore ice condenser door to OPERABLE status and closed positions.	48 hours
D. Required Action and associated Completion Time of Condition A or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.12.1	Verify all inlet doors indicate closed by the Inlet Door Position Monitoring System.	12 hours
SR 3.6.12.2	Verify, by visual inspection, each intermediate deck door is closed and not impaired by ice, frost, or debris.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.12.3	Verify, by visual inspection, each inlet door is not impaired by ice, frost, or debris.	-----NOTE------ The 3 month performance due September 9, 1996 (per SR 3.0.2) may be extended until October 21, 1996. --------------- 3 months during first year after receipt of license <u>AND</u> 18 months
SR 3.6.12.4	Verify torque required to cause each inlet door to begin to open is ≤ 675 in-lb.	-----NOTE------ The 3 month performance due September 9, 1996 (per SR 3.0.2) may be extended until October 21, 1996. -------------- 3 months during first year after receipt of license <u>AND</u> 18 months

(continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.6.12.5	Perform a torque test on a sampling of ≥ 50% of the inlet doors.	-----NOTE------ The 3 month performance due September 9, 1996 (per SR 3.0.2) may be extended until October 21, 1996. -------------- 3 months during first year after receipt of license AND 18 months
SR 3.6.12.6	Verify for each intermediate deck door: a. No visual evidence of structural deterioration; b. Free movement of the vent assemblies; and c. Free movement of the door.	3 months during first year after receipt of license AND 18 months

(continued)

SURVEILLANCE REQUIREMENTS (Continued)

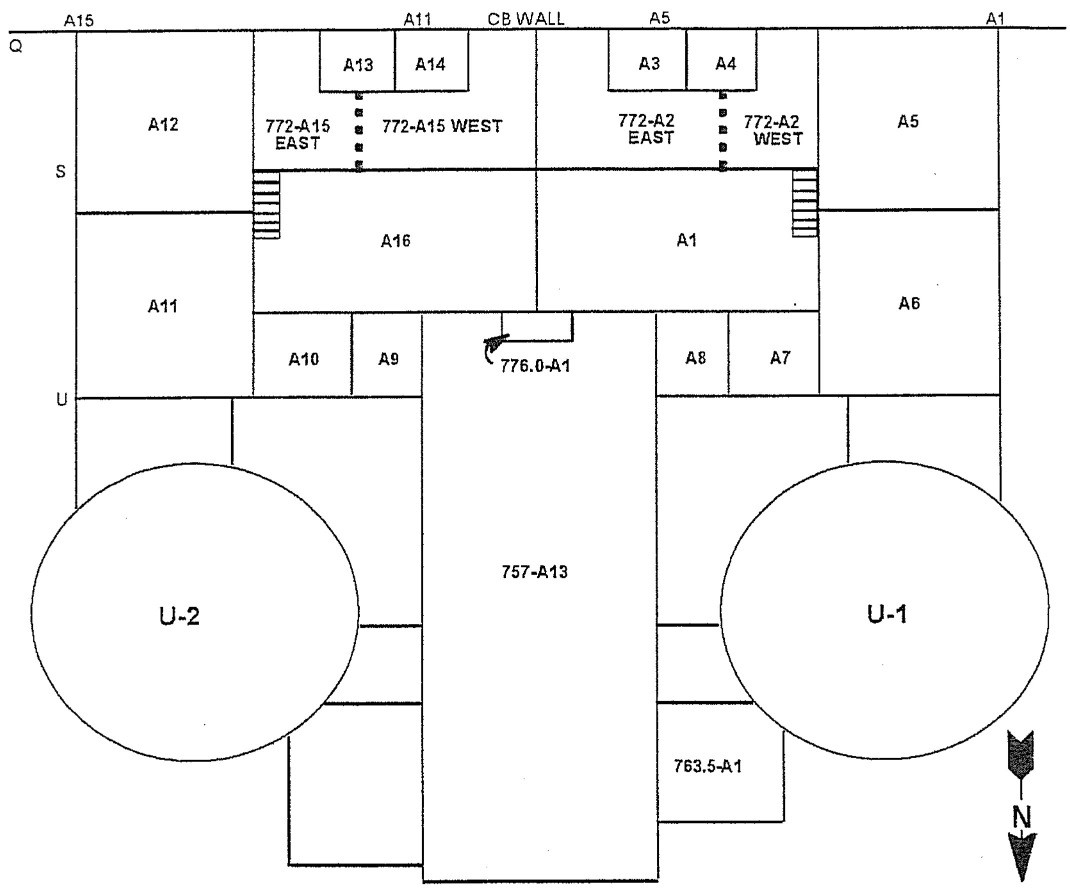
SURVEILLANCE		FREQUENCY
SR 3.6.12.7	Verify, by visual inspection, each top deck door: a. Is in place; b. Free movement of top deck vent assembly; and c. Has no condensation, frost, or ice formed on the door that would restrict its opening.	92 days

Attachment 7
(Page 6 of 7)

	7.3 Radiation Levels		7.4 Fuel Handling	
	Mode	**Initiating/Condition**	**Mode**	**Initiating/Condition**
GENERAL		*Refer to "Fission Product Barrier Matrix" or "Gaseous Effluents" (7.1)*		*Refer to "Gaseous Effluents" (7.1)*
SITE		*Refer to "Fission Product Barrier Matrix" or "Gaseous Effluents" (7.1)*		*Refer to "Gaseous Effluents" (7.1)*
ALERT	All	**UNPLANNED increases in Radiation levels within the Facility that impedes Safe Operations <u>or</u> establishment <u>or</u> maintenance of Cold Shutdown** *(1 or 2)* 1. **VALID** area Radiation Monitor readings <u>or</u> survey results exceed 15 mrem/hr in the Control Room <u>or</u> CAS 2. *(a and b)* a. **VALID** area radiation monitor readings exceed values listed in Table 7-2 b. Access restrictions impede operation of systems necessary for Safe Operation <u>or</u> the ability to establish Cold Shutdown *See UNUSUAL EVENT Note Below*	All	**Major damage to Irradiated Fuel, <u>or</u> Loss of water level that has <u>or</u> will uncover Irradiated Fuel outside the Reactor Vessel** *(1 and 2)* 1. VALID alarm on 0-RE-90-101B <u>or</u> 0-RE-90-102 <u>or</u> 0-RE-90-103 <u>or</u> 1-RE-90-130/131 <u>or</u> 1-RE-90-112 <u>or</u> 1-RE-90-400 <u>or</u> 2-RE-90-400 2. *(a or b)* a. Plant personnel report damage of Irradiated Fuel sufficient to rupture Fuel Rods b. Plant personnel report water level drop has <u>or</u> will exceed makeup capacity such that Irradiated Fuel will be uncovered
UNUSUAL EVENT	All	**UNPLANNED increase in Radiation levels within the Facility** 1. **VALID** area Radiation Monitor readings increase by a factor 1000 over normal levels *Note: In Either the UE or ALERT EAL, the SED must determine the cause of Increase in Radiation Levels and Review Other INITIATING/CONDITIONS for Applicability (e.g., a dose rate of 15 mrem/hr in the Control Room could be caused by a release associated with a DBA).*	All	**UNPLANNED loss of water level in Spent Fuel Pool <u>or</u> Reactor Cavity <u>or</u> Transfer Canal with fuel remaining covered** *(1 and 2 and 3)* 1. Plant personnel report water level drop in Spent Fuel Pool, <u>or</u> Reactor Cavity, <u>or</u> Transfer Canal 2. **VALID** alarm on 0-RE-90-102 <u>or</u> 0-RE-90-103 <u>or</u> 1-RE-90-59 <u>or</u> 1-RE-90-60 3. Fuel remains covered with water.

2.0 AB EL 772.0, 776.0 & 763.5 ELEVATION DIAGRAM

Auxiliary Building El 772.0, 776.0 & 763.5 Diagram



ROOM	ROOM NAME	PROCEDURE
772.0–A1	480V Rx MOV Bd Rm 1A	AOI-30.2 C.2
772.0–A2–E	480V Rx MOV Bd Rm 1B (East)	AOI-30.2 C.3
772.0–A2–W	480V Rx MOV Bd Rm 1B (West)	AOI-30.2 C.4
772.0–A3	125V Vital Batt Rm II	AOI-30.2 C.5
772.0–A4	125V Vital Batt Rm I	AOI-30.2 C.6
772.0–A5	480V Xfmer Rm 1B	AOI-30.2 C.7
772.0–A6	480V Xfmer Rm 1A	AOI-30.2 C.8
772.0–A7	Mech Equip Rm	AOI-30.2 C.9
772.0–A8	5th Vit Batt & Bd Rm	AOI-30.2 C.10
772.0–A9	HEPA Filter Plenum Rm	AOI-30.2 C.44

ROOM	ROOM NAME	PROCEDURE
772.0–A10	Mech Equip Rm	AOI-30.2 C.11
772.0–A11	480V Xfmer Rm 2B	AOI-30.2 C.12
772.0–A12	480V Xfmer Rm 2A	AOI-30.2 C.13
772.0–A13	125V Vital Batt Rm IV	AOI-30.2 C.14
772.0–A14	125V Vital Batt Rm III	AOI-30.2 C.15
772.0–A15–E	480V Rx MOV Bd Rm 2B (East)	AOI-30.2 C.16
772.0–A15–W	480V Rx MOV Bd Rm 2B (West)	AOI-30.2 C.17
772.0–A16	480V Rx MOV Bd Rm 2A	AOI-30.2 C.18
776.0–A1	Elev Mach Rm	AOI-30.2 C.60
763.5–A1	Ice Equip Rm	AOI-30.2 C.45
757.0–A13	(Next Page)	

76. 011 EG 2.4.41 076

During a LOCA, which ONE of the following identifies...

(1) how many of the EPIP-1, "Emergency Plan Classification Logic," Fission Product Barrier Matrix contain a decision point based directly on RVLIS

and

(2) the RVLIS threshold level that first requires a classification declaration be made?

	Barrier Matrix	Threshold level
A.	1	<44%
B.	2	<44%
C.	1	<33%
D.✓	2	<33%

DISTRACTOR ANALYSIS:

A. Incorrect, Plausible because many of the criteria only appear in one of the barriers and 44% is the void content of the RCS while the RCPs are in service that would result in a vessel level of less than 33% during a LOCA if the RCPs were to trip.

B. Incorrect, Plausible because RVLIS level appears in both Fuel Clad Barrier (as a Potential LOSS) and in the RCS Barrier (as a LOSS) and 44% is the void content of the RCS while the RCPs are in service that would result in a vessel level of less than 33% during a LOCA if the RCPs were to trip.

C. Incorrect, Plausible because many of the criteria only appear in one of the barriers and less than 33% being the RVLIS value during a LOCA that will require a declaration is correct.

D. Correct, Valid RVLIS level less than 33% appears in both 1.1 Fuel Clad Barrier and in 1.2 RCS Barrier. RVLIS appears in the Fuel Clad Barrier as a Potential LOSS and in the RCS Barrier as a LOSS.

Question Number: 76

Tier: 1 **Group** 1

K/A: 011 EG2.4.41
Large Break LOCA
Emergency Procedures / Plan
Knowledge of the emergency action level thresholds and classifications.

Importance Rating: 2.9 / 4.6

10 CFR Part 55: 41.10 / 43.5 / 45.11

10CFR55.43.b: 6,7

K/A Match: K/A is matched because the question requires knowledge of the emergency action level threshold value for reactor vessel level and the barriers that are affected by the Reactor Vessel Level Indicating System indicating below the minimum required.
Question is SRO because it requires detailed knowledge of the procedures used to evaluate plant conditions to determine Emergency Classifications and this determination is an SRO function.

Technical Reference: EPIP-1, "Emergency Plan Classification Logic," Fission Product Barrier Matrix, Revision 0035
FR-0, Status Trees, Revision 0014

Proposed references to be provided: None

Learning Objective: 3-OT-PCD048C
1. Classify emergency events
16. Recognize conditions which constitute activation of the emergency response facilities regardless of the time of day when an emergency has been declared.

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the WBN 10/2011 NRC exam

Comments:

Attachment 1
(Page 3 of 4)

1.1. _ Fuel Clad Barrier	
1. Critical Safety Function Status	
LOSS	**Potential LOSS**
Core Cooling Red (FR-C.1)	Core Cooling Orange (FR-C.2) **OR** Heat Sink Red (FR-H.1) (RHR Not in Service)
-OR-	
2. Primary Coolant Activity Level	
LOSS	**Potential LOSS**
RCS sample activity is Greater Than 300 µCi/gm dose equivalent iodine-131	Not applicable
-OR-	
3. Incore TCs Hi Quad Average	
LOSS	**Potential LOSS**
Greater Than 1200°F	Greater Than 727°F
-OR-	
4. Reactor Vessel Water Level	
LOSS	**Potential LOSS**
Not Applicable	VALID RVLIS level <33% (No RCP running)
-OR-	
5. Containment Radiation Monitors	
LOSS	**Potential LOSS**
VALID reading increase of Greater Than: 293 R/hr On 1-RM-90-271 and 272 **OR** 261 R/hr On 1-RM-90-273 and 274 **(see instruction note 5)**	Not Applicable
-OR-	
6. Site Emergency Director Judgment	
Any condition that, in the Judgment of the SM/SED, Indicates Loss or Potential Loss of the Fuel Clad Barrier Comparable to the Conditions Listed Above.	

1.2. _ RCS Barrier	
1. Critical Safety Function Status	
LOSS	**Potential LOSS**
Not Applicable	Pressurized Thermal Shock Red (FR-P.1) **OR** Heat Sink Red (FR-H.1) (RHR **Not** in Service)
-OR-	
2. RCS Leakage/LOCA	
LOSS	**Potential LOSS**
RCS Leak results in Loss of subcooling (<65°F Indicated), [85°F ADV]	Non Isolatable RCS Leak Exceeding The Capacity of One Charging Pump (CCP) In the Normal Charging Alignment. **OR** RCS Leakage Results In Entry Into E-1
-OR-	
3. Steam Generator Tube Rupture	
LOSS	**Potential LOSS**
SGTR that results in a safety injection actuation **OR** Entry into E-3	Not Applicable
-OR-	
4. Reactor Vessel Water Level	
LOSS	**Potential LOSS**
VALID RVLIS level <33% (No RCP Running)	Not Applicable
-OR-	
5. Site Emergency Director Judgment	
Any condition that, in the Judgment of the SM/SED, Indicates Loss or Potential Loss of the RCS Barrier Comparable to the Conditions Listed Above.	

Attachment 1
(Page 4 of 4)

1.3. _ CNTMT Barrier	
1. Critical Safety Function Status	
LOSS	**Potential LOSS**
Not Applicable	Containment (FR-Z.1) Red **OR** Actions of FR-C.1 (Red Path) are INEFFECTIVE (i.e.: core TCs trending up)
-OR-	
2. Containment Pressure/Hydrogen	
LOSS	**Potential LOSS**
Rapid unexplained decrease following initial increase **OR** Containment pressure or Sump level **Not** increasing (with LOCA in progress)	Containment Hydrogen Increases to >4% by volume **OR** Pressure >2.8 PSIG (Phase B) with < One full train of Containment spray
-OR-	
3. Containment Isolation Status	
LOSS	**Potential LOSS**
Containment Isolation is Incomplete (when required) **AND** a Release Path to the Environment Exists	Not Applicable
-OR-	
4. Containment Bypass	
LOSS	**Potential LOSS**
RUPTURED S/G is also FAULTED outside CNTMT **OR** Prolonged (>4 Hours) Secondary Side release outside CNTMT from a S/G with a SGTL > T/S Limits	Unexplained VALID increase in area or ventilation RAD monitors in areas adjacent to CNTMT (with LOCA in progress)
-OR-	
5. Significant Radioactivity in Containment	
LOSS	**Potential LOSS**
Not Applicable	VALID Reading increase of Greater Than: 5290 R/hr on 1-RM-90-271 and 1-RM-90-272 **OR** 4710 R/hr on 1-RM-90-273 and 1-RM-90-274 **(see instruction note 5)**
-OR-	
6. Site Emergency Director Judgment	
Any condition that, in the Judgment of the SM/SED, Indicates Loss or Potential Loss of the CNTMT Barrier Comparable to the Conditions Listed Above.	

Modes: 1, 2, 3, 4

INSTRUCTIONS

NOTE:
A condition is considered to be MET if, in the judgment of the Site Emergency Director, the condition will be MET imminently (i.e., within 1 to 2 hours, in the absence of a viable success path). The classification shall be made a soon as this determination is made.

1. In the matrix to the left, review the **INITIATING CONDITIONS** in all columns and identify which, if any, **INITIATING CONDITIONS** are MET. Circle these **CONDITIONS**.
2. For each of the three barriers, identify if any **LOSS** or Potential **LOSS INITIATING CONDITIONS** have been MET.
3. If a CSF is listed as an **INITIATING CONDITION**; the respective status tree criteria will be monitored and used to determine the **EVENT** classification for the Modes listed on the classification flowchart.
4. Compare the barrier losses and potential losses to the **EVENTS** below and make the appropriate declaration.
5. Containment High Range Radiation Monitors (HRRMs) are temperature sensitive and can be affected by both temperature induced currents and insulation resistance temperature effects. Following the initial increase in containment temperature the HRRM monitors can give erratic indication for up to 1 minute. Steady state temperature effects on cable insulation resistance for the HRRM signal cable is dependent on containment temperature and could result in a shift in monitor output indication. With a containment excursion temperature to 327 °F (HELB), the output of the HRRMs could potentially have up to a 25 R/hr indicated offset for duration of 10 minutes until the containment air return fans are started and temperature starts to reduce. (**Caution: Should the containment air return fans not start, containment temperatures could remain elevated resulting in potential false HRRM indicated readings).**

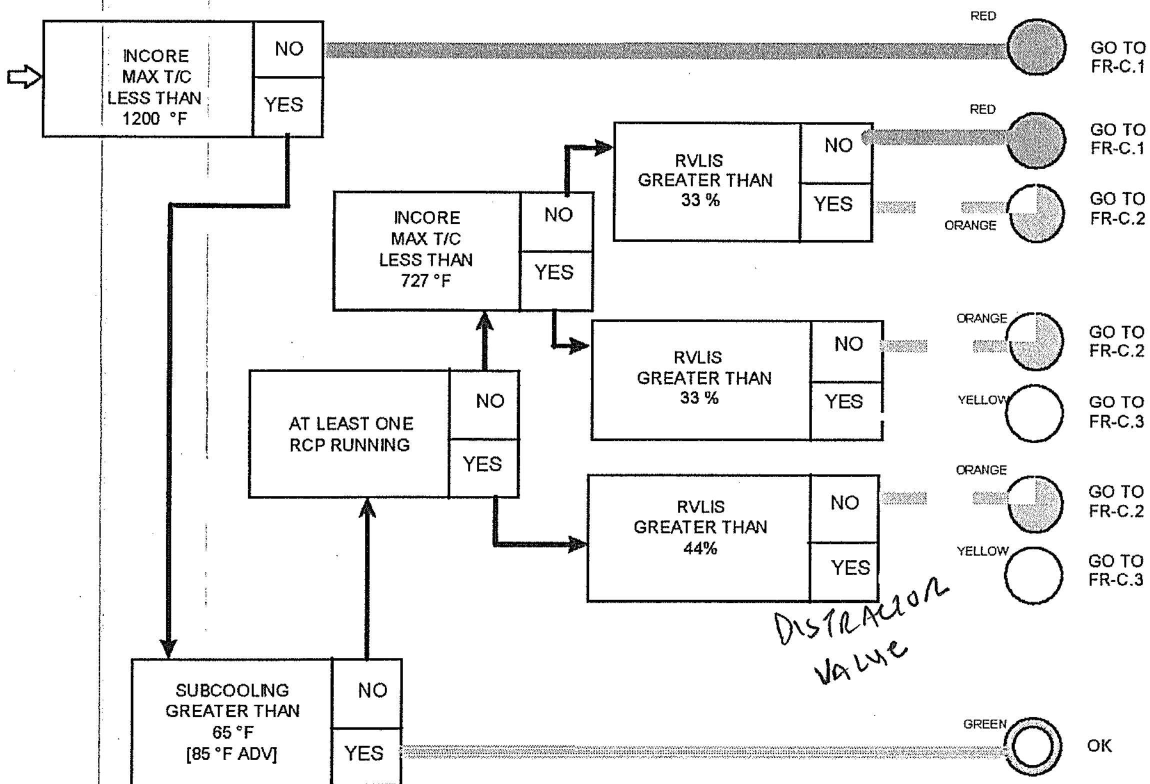
EVENTS

UNUSUAL EVENT	ALERT
Loss **or** Potential LOSS of Containment Barrier	Any LOSS **or** Potential LOSS of Fuel Clad barrier **OR** Any LOSS **or** Potential LOSS of RCS barrier
SITE AREA EMERGENCY	**GENERAL EMERGENCY**
LOSS **or** Potential LOSS of any two barriers	LOSS of any two barriers **and** Potential LOSS of third barrier

F I S S I O N P R O D U C T B A R R I E R M A T R I X U 1

Attachment 1
(Page 2 of 8)
Monitoring Critical Safety Functions

CORE COOLING
FR-C



COLOR	PROC

V. TRAINING OBJECTIVES

1. Classify emergency events.
2. Recognize the reasons for having the Radiological Emergency Plan (REP).
3. Identify the functions of the onsite emergency response facilities.
4. Formulate Protective Action Recommendations (PARs).
5. Use the WBN Emergency Plan Implementing Procedures (EPIPs).
6. State three Site Emergency Director responsibilities that cannot be delegated.
7. Identify Operation's responsibilities for the following emergency response positions:
   - Site Emergency Director (who is initially the SM)
   - Operations Manager in the TSC
   - Control Room Communicator in the Control Room
   - Operations Communicator in the TSC
   - OSC Operations Advisor
   - Operation's emergency response team assignments
   - NOMS Logkeeper in the Control Room (when available)
   - Technical Advisor
   - Designated Phone Talker
8. Recognize how AUOs are dispatched and controlled during radiological emergencies.
9. Recognize REP communications guidelines (OPDP-1).
10. Demonstrate effective communication techniques used in emergency response.
11. Identify lessons learned from TVA/industry events, drills and exercises.
12. Recall where radios can and cannot be used at WBN (BP-364).
13. Use the Integrated Computer System (ICS).
14. Identify <u>all locations</u> where the Emergency Paging System (EPS) may be activated from and demonstrate the use of the EPS to include the printed report from the TSC.
15. Using WBN EPIPs 2, 3, 4, and 5, recognize who is responsible to activate the Emergency Paging System.
16. Recognize conditions which constitute activation of the emergency response facilities regardless of the time of day when an emergency has been declared.
17. Identify and use the back-up Emergency Response Organization call lists used when the Emergency Paging System has failed.

V. TRAINING OBJECTIVES (continued)

18. Recognize entry conditions for Severe Accident Management Guidelines (SAMGs).
19. Use the Radiological Emergency Notification Directory (REND).
20. Use the Satellite Phone to make calls during emergencies.
21. Identify the WBN REP procedure addressing MERT responsibilities, offsite agreement support, and emergency phone numbers.
22. Review Operations drill critique items.
23. Perform dose assessments using ICS for WBN EPIP-13.
24. Interpret MET data obtained in the TSC from the CECC computer.
25. Identify specific actions of OSC Emergency Responders in the OSC team's staging area (EPT 309.000).
26. Understand the critical times associated with
    - Event Declaration
    - Offsite Notification
    - Facility Staffing
    - Printed EPS Report

F. <u>Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.</u> [10 CFR 55.43(b)(6)]

Some examples of SRO exam items for this topic include:

- Evaluating core conditions and emergency classifications based on core conditions.
- Administrative requirements associated with low power physics testing processes.
- Administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths and/or activities.
- Administrative controls associated with the installation of neutron sources.
- Knowledge of TS bases for reactivity controls.

G. <u>Fuel handling facilities and procedures.</u> [10 CFR 55.43(b)(7)]

Some examples of SRO exam items for this topic include:

- Refuel floor SRO responsibilities.
- Assessment of fuel handling equipment surveillance requirement acceptance criteria.
- Prerequisites for vessel disassembly and reassembly.
- Decay heat assessment.
- Assessment of surveillance requirements for the refueling mode.
- Reporting requirements.
- Emergency classifications.

This does not include items that the RO may be responsible for at some sites such as fuel handling equipment and refueling related control room instrumentation operability requirements, abnormal operating procedure immediate actions, etc. For example, an RO is required to stop the refueling process when communication is lost between the control room and the refueling floor, therefore, this is a task that is both an RO and SRO responsibility and is not SRO-only.

77. 025 AG2.4.21 077

Given the following:

- Unit 1 was shutdown 24 hours ago for a refueling outage.
- RHR pump 1B-B is tagged for maintenance.
- Current RCS temperature is 135°F and 80 psig.
- RHR pump 1A-A trips and cannot be restarted.
- RCS temperature begins to rise.
- The crew is performing AOI-14, "Loss of RHR Shutdown Cooling," and Section 3.9, "RCS Alternate Cooling Method with RX Vessel Head Installed," has been initiated.

Which ONE of the following identifies...

(1) the cooling method that is directed to be attempted <u>first</u> after AOI-14 Section 3.9 is implemented

and

(2) the condition that will result in an REP declaration being required?

A. (1) Establish RCS feed and bleed using a CCP and a pressurizer PORV.
(2) RCS incore temperature > 200°F

B✓ (1) Establish natural circulation in the RCS using the steam generators.
(2) RCS incore temperature > 200°F

C. (1) Establish RCS feed and bleed using a CCP and a pressurizer PORV.
(2) RHR not established for > 15 minutes

D. (1) Establish natural circulation in the RCS using the steam generators.
(2) RHR not established for > 15 minutes

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because RCS feed and bleed with a CCP and a pressurizer PORV is a cooling process implemented by Section 3.9 of the AOI and because EPIP-1 requiring a declaration following the loss of RHR when the incore temperatures exceed 200°F is correct.*

B. *Correct, Section 3.9 will check for conditions to establish cooling by natural circulation in the RCS and EPIP-1 will require an ALERT to be declared following the loss of RHR when the incore temperatures exceed 200°F.*

C. *Incorrect, Plausible because RCS feed and bleed with a CCP and a pressurizer PORV is a cooling process implemented by Section 3.9 of the AOI and because "exceeding 15 minutes" is a time frame used in several conditions for requiring a declaration of the REP (e.g. electrical board not available for > 15 minutes, fire lasting >15 minutes, rad assessments not completed within 15 minutes, etc.)*

D. *Incorrect, Plausible because establishing natural circulation is correct and because "exceeding 15 minutes" is a time frame used in several conditions for requiring a declaration of the REP (e.g. electrical board not available for > 15 minutes, fire lasting >15 minutes, rad assessments not completed within 15 minutes, etc.)*

Question Number: 77

Tier: 1 **Group** 1

K/A: 025 AG2.4.21
Loss of Residual Heat Removal System
Emergency Procedures / Plan
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Importance Rating: 4.0 / 4.6

10 CFR Part 55: 41.7 / 43.5 / 45.12

10CFR55.43.b: 5, 6

K/A Match: K/A is matched because the question requires knowledge of core cooling and heat removal processes directed by the AOI following a loss of the RHR system and is SRO because of requiring detailed knowledge of the procedure content (including flowpath through the procedure) to prevent radioactive releases and the requirements for implementation of the Radiological Emergency Plan following a loss

K/A Match: K/A is matched because the question requires knowledge of core cooling and heat removal processes directed by the AOI following a loss of the RHR system and is SRO because of requiring detailed knowledge of the procedure content (including flowpath through the procedure) to prevent radioactive releases and the requirements for implementation of the Radiological Emergency Plan following a loss of the RHR system.

Technical Reference: AOI-14, Loss of RHR Shutdown Cooling, Revision 0037
EPIP-1, Emergency Plan Classification Logic, Revision 0035

Proposed references to be provided: None

Learning Objective: 3-OT-AOI400
5. Explain Alternate RHR Cooling methods.
3-OT-PCD048C
1. Classify emergency events.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: WBN bank question AOI1400.05 001 modified

Comments:

Step	Action/Expected Response	Response Not Obtained
3.9	**RCS Alternate Cooling Method With Rx Vessel Head Installed** [C.4, C.7]	
1.	**CONTINUE** attempts to restore RHR Shutdown Cooling flow.	
2.	**COMPLETE** the following: [C.2] • **EVACUATE** non-essential personnel from cntmt. • **NOTIFY** STA to implement TI-68.002 for cntmt closure. • **IF** any RCP shaft uncoupled, **THEN** **NOTIFY** STA to install RCP shaft restraint device(s) to limit pump shaft leakage. • **NOTIFY** RP to provide monitoring and Rad protection guidance for workers involved in cntmt closure.	
3.	**IF** at least one RCP is running, **THEN** **USE** steam dumps or SG PORV operation to restore cooling, **AND** ** **GO TO** Step 18. NO	
4.	**CHECK** at least TWO S/Gs narrow range levels greater than 29%.	**IF** S/G narrow range **NOT** available, **THEN** ** **GO TO** Note prior to Step 12. - one path to distraction

Step	Action/Expected Response	Response Not Obtained

3.9 RCS Alternate Cooling Method With Rx Vessel Head Installed [C.4, C.7] **(continued)**

5.	**CHECK** RCS intact and capable of being pressurized & transferring heat: • Pzr COLD CAL level indicator 1-LI-68-321 on scale. • RCP shafts coupled. • Pzr PORVs or associated block valves capable of being CLOSED. • Pzr safeties installed. • S/G primary manways installed. • RCS intact with SG tubes filled.	**IF** RCS **NOT** capable of being pressurized, **THEN** ** **GO TO** Note prior to Step 12.
6.	**CHECK** RCS press less than COPS: • **REFER TO** Appendix D.	**ENSURE** PORV and associated block valve OPEN. **WHEN** press less than COPS, **THEN** **ENSURE** PORV or associated block valve CLOSED. ** **GO TO** Caution prior to Step 8.
7.	**ENSURE** pzr PORVs CLOSED and HSs in AUTO: • 1-HS-68-334A. • 1-HS-68-340A.	**IF** any PORV can **NOT** be closed, **THEN** **CLOSE** associated block valve.

Step	Action/Expected Response	Response Not Obtained

3.9 RCS Alternate Cooling Method With Rx Vessel Head Installed [C.4, C.7] **(continued)**

CAUTION **RCS pressure control must be established prior to the onset of core boiling to avoid the potential loss of natural circulation due to steam binding of the S/G U-tubes.**

8. **ESTABLISH** RCS press greater than 100 psig and less than COPS:
   - **REFER TO** Appendix D.
   - **IF** pzr bubble exists,
     THEN
     USE heaters and sprays.
   - **IF** water solid,
     THEN
     USE charging and letdown.
   - **IF** less than water solid **AND** pzr subcooled,
     THEN
     USE charging and letdown to raise level and maintain press control.

3.9 RCS Alternate Cooling Method With Rx Vessel Head Installed [C.4, C.7] **(continued)**

NOTE When initially establishing natural circulation, RCS temperature response will be delayed until the necessary delta-T is established.

Step	Action/Expected Response	Response Not Obtained
9.	**ESTABLISH** natural circulation: a. **MAINTAIN** RCS press greater than 100 psig and less than COPS. b. **MAINTAIN** at least two SGs NR level greater than 29%. c. **OPEN** S/G PORVs for the selected S/Gs. d. **USE** AFW and S/G blowdown for feed and bleed of S/G. e. **CHECK** RCS temp stable or dropping.	**IF** natural circulation can **NOT** be established, **THEN** ** **GO TO** Note prior to Step 12. — Another path to distractor
10.	**IF** RCP support conditions exist, **OR** can be established, **THEN** **START** one RCP: • **REFER TO** SOI-68.02, Reactor Coolant Pumps.	
11.	** **GO TO** Step 18.	

3.9 RCS Alternate Cooling Method With Rx Vessel Head Installed [C.4, C.7] (continued)

NOTE The following steps only apply if natural circulation can NOT be established and Reactor Vessel head is installed.

12. **ENSURE** both pocket sump pumps STOPPED [M-15]:
    - 1-HS-77-410.
    - 1-HS-77-411.

13. **DETERMINE** appropriate step to initiate a feed and bleed cooling method:

Distractor

IF the following feed and bleed path is to be used:	**THEN**
Normal charging to RCS; bleed through manual pzr PORV control with RCS intact,	**GO TO** Step 17.
Normal charging to RCS; bleed through S/G HL manway with nozzle dam removed,	**GO TO** Step 15.
Normal charging to RCS; bleed through pzr manway or removed PORV/Safety valve flanges,	**GO TO** Step 16.
Gravity feed to RCS; bleed through S/G CL manway with nozzle dam removed,	**GO TO** Step 14.
Gravity feed to RCS; bleed through RCP shaft leakage (shaft uncoupled and NOT blocked),	**GO TO** Step 14.

3.9 RCS Alternate Cooling Method With Rx Vessel Head Installed [C.4, C.7] (continued)

NOTE Core boiling is necessary for long-term core heat removal during the loss of RHR cooling. RCS inventory must be maintained.

Step	Action/Expected Response	Response Not Obtained
14.	**PERFORM** the following to cool RCS via the S/G CL manway or RCP shaft:	
	a. **DISPATCH** Operator to operate 1-FCV-63-1 locally OR electrically [480V Rx MOV Bd 1A1-A].	
	b. **OPEN** the following valves: • 1-FCV-74-1, Loop 4 Hot Leg To RHR Suction. • 1-FCV-74-2, Loop 4 Hot Leg To RHR Suction.	b. **ENSURE** 1-FCV-74-1 and 1-FCV-74-2 CLOSED. **RESTORE** power, and **OPEN** bypass valves: 1) 1-FCV-74-9, RHR System Isol Bypass. 2) 1-FCV-74-8, RHR System Isol Bypass.

Step continued on next page.

Step	Action/Expected Response	Response Not Obtained

3.9 RCS Alternate Cooling Method With Rx Vessel Head Installed [c.4, c.7] **(continued)**

NOTE 1-FCV-63-1 should be slowly throttled open and flow monitored with communications maintained between MCR and local Operator throughout this evolution.

Action/Expected Response	Response Not Obtained
c. **CONTROL** 1-FCV-63-1 electrically [480V Rx MOV Bd 1A1-A, c/10A] using breaker operation.	c. Locally **CONTROL** 1-FCV-63-1.
d. **MONITOR** RCS level stable or rising. • RVLIS • 1-LT-68-399A, NR Level • 1-LT-68-399B, WR Level	d. **THROTTLE OPEN** 1-FCV-63-1.
e. **MONITOR** RCS temp stable or dropping.	e. **IF** RCS temp control can **NOT** be established, **THEN** ** **GO TO** Step 13.
f. ** **GO TO** Step 18.	

Attachment 6
(Page 3 of 4)

	6.1 Loss of Shutdown Systems		6.2 Loss of AC (Shutdown)	
	Mode	**Initiating/Condition**	**Mode**	**Initiating/Condition**
GENERAL	5,6	Note: Additional information will be provided later pending NRC Guidance on Shutdown EALs *Refer to "Gaseous Effluents" (7.1)*		*Not Applicable*
SITE	5,6	**Loss of water level in the Rx vessel that has <u>or</u> will uncover fuel in the Rx vessel** *(1 and 2 and 3 and 4)* 1. Loss of RHR capability 2. Rx vessel water level < el. 718' 3. Incore TCs (if available) indicate RCS temp. >200° F 4. RCS is vented/open to CNTMT *Note: If CNTMT open, refer to "Gaseous Effluents" (7.1)*		*Not Applicable*
ALERT	5,6	**Inability to maintain Unit in Cold** Shutdown *(1 and 2)* 1. RHR capability is **<u>not</u>** available for RCS Cooling 2. Incore TCs (if available) indicate RCS temp. >200° F *Note: If CNTMT open, refer to "Gaseous Effluents" (7.1)*	5,6 or De-Fuel	**UNPLANNED loss of Offsite <u>and</u> Onsite AC Power for >15 minutes** 1. 1A **<u>and</u>** 1B 6.9 KV Shutdown Bds de-energized for >15 minutes
UNUSUAL EVENT	5,6	Note: Additional information will be provided later pending NRC Guidance on Shutdown EALs	5,6 or De-Fuel	**UNPLANNED loss of All Offsite Power for >15 minutes** *(1 and 2)* 1. C **<u>and</u>** D CSSTS not available For >15 minutes. 2. Either Diesel Generator is supplying power to its respective Shutdown Board

DISTRACTOR
15 minutes appears here & in several other EALs

WBN BANK QUESTION

AOI1400.05 001

Given the following plant conditions:

- The Unit is being cooled down and depressurized for an outage.
- 1B-B RHR pump is tagged for required repairs.
- RCS temperature is 100°F.
- RCS pressure is 225 psig.
- All RCPs are off.
- S/G levels are all 38%.
- The running 1A-A RHR pump trips.

Which of the following is required per AOI-14, Loss of RHR Shutdown Cooling?

a. Maintain RCS pressure < 100 psig, and establish natural circulation using AFW and Steam Dumps.

b.✓ Maintain RCS pressure > 100 psig, and less than COPS, and open S/G PORVs to establish natural circulation using AFW and S/G blowdown.

c. Establish RCS feed and bleed using normal Charging and a Pzr PORV.

d. Immediately start an RCP and establish a secondary heat sink by steaming the S/Gs.

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. LICENSE TRAINING
B. NON-LICENSE TRAINING

III. TITLE

AOI-14, LOSS OF RHR SHUTDOWN COOLING

IV. LENGTH OF LESSON

A.	License Training	2 Hours
B.	NOTP	1 Hour

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. Demonstrate knowledge of the Purpose/Goal of AOI-14, Loss of RHR Shutdown Cooling.
X	X	X	X	2. Identify 3 factors that effect Severity of the Loss of RHR Shutdown Cooling.
	X	X	X	3. Explain possible Alarms for Loss of RHR Shutdown Cooling.
X	X	X	X	4. Describe 5 ways that RHR Cooling can be lost.
	X	X	X	5. Explain Alternate RHR Cooling methods.
X	X	X	X	6. Identify 5 Causes for Loss/Degradation of RHR capability in PWRs in the industry, per SOER 85-4.
	X	X	X	7. Demonstrate ability/knowledge of AOI, to correctly: a. Recognize Entry conditions. b. Respond to Action steps. c. Respond to Contingencies (RNO column). d. Respond to Notes & Cautions.
X	X	X	X	8. Describe AUO actions for venting a RHR pump when it becomes air bound.

V. TRAINING OBJECTIVES

1. Classify emergency events.
2. Recognize the reasons for having the Radiological Emergency Plan (REP).
3. Identify the functions of the onsite emergency response facilities.
4. Formulate Protective Action Recommendations (PARs).
5. Use the WBN Emergency Plan Implementing Procedures (EPIPs).
6. State three Site Emergency Director responsibilities that cannot be delegated.
7. Identify Operation's responsibilities for the following emergency response positions:
   - Site Emergency Director (who is initially the SM)
   - Operations Manager in the TSC
   - Control Room Communicator in the Control Room
   - Operations Communicator in the TSC
   - OSC Operations Advisor
   - Operation's emergency response team assignments
   - NOMS Logkeeper in the Control Room (when available)
   - Technical Advisor
   - Designated Phone Talker
8. Recognize how AUOs are dispatched and controlled during radiological emergencies.
9. Recognize REP communications guidelines (OPDP-1).
10. Demonstrate effective communication techniques used in emergency response.
11. Identify lessons learned from TVA/industry events, drills and exercises.
12. Recall where radios can and cannot be used at WBN (BP-364).
13. Use the Integrated Computer System (ICS).
14. Identify <u>all locations</u> where the Emergency Paging System (EPS) may be activated from and demonstrate the use of the EPS to include the printed report from the TSC.
15. Using WBN EPIPs 2, 3, 4, and 5, recognize who is responsible to activate the Emergency Paging System.
16. Recognize conditions which constitute activation of the emergency response facilities regardless of the time of day when an emergency has been declared.
17. Identify and use the back-up Emergency Response Organization call lists used when the Emergency Paging System has failed.

V. TRAINING OBJECTIVES (continued)

18. Recognize entry conditions for Severe Accident Management Guidelines (SAMGs).
19. Use the Radiological Emergency Notification Directory (REND).
20. Use the Satellite Phone to make calls during emergencies.
21. Identify the WBN REP procedure addressing MERT responsibilities, offsite agreement support, and emergency phone numbers.
22. Review Operations drill critique items.
23. Perform dose assessments using ICS for WBN EPIP-13.
24. Interpret MET data obtained in the TSC from the CECC computer.
25. Identify specific actions of OSC Emergency Responders in the OSC team's staging area (EPT 309.000).
26. Understand the critical times associated with
    - Event Declaration
    - Offsite Notification
    - Facility Staffing
    - Printed EPS Report

F. <u>Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.</u> [10 CFR 55.43(b)(6)]

Some examples of SRO exam items for this topic include:

- Evaluating core conditions and emergency classifications based on core conditions.
- Administrative requirements associated with low power physics testing processes.
- Administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths and/or activities.
- Administrative controls associated with the installation of neutron sources.
- Knowledge of TS bases for reactivity controls.

G. <u>Fuel handling facilities and procedures.</u> [10 CFR 55.43(b)(7)]

Some examples of SRO exam items for this topic include:

- Refuel floor SRO responsibilities.
- Assessment of fuel handling equipment surveillance requirement acceptance criteria.
- Prerequisites for vessel disassembly and reassembly.
- Decay heat assessment.
- Assessment of surveillance requirements for the refueling mode.
- Reporting requirements.
- Emergency classifications.

This does not include items that the RO may be responsible for at some sites such as fuel handling equipment and refueling related control room instrumentation operability requirements, abnormal operating procedure immediate actions, etc. For example, an RO is required to stop the refueling process when communication is lost between the control room and the refueling floor, therefore, this is a task that is both an RO and SRO responsibility and is not SRO-only.

C. Facility licensee procedures required to obtain authority for design and operating changes in the facility. [10 CFR 55.43(b)(3)]

Some examples of SRO exam items for this topic include:

- 10 CFR 50.59 screening and evaluation processes.
- Administrative processes for temporary modifications.
- Administrative processes for disabling annunciators.
- Administrative processes for the installation of temporary instrumentation.
- Processes for changing the plant or plant procedures.

Section IV provides an example of a satisfactory SRO-only question related to this topic.

D. 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.

E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. One area of SRO level knowledge (with respect to selecting a procedure) is knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item, for example:

78. 026 AA2.06 078

Given the following:

- Unit 1 is operating at 100% power with the two components listed below out of service and tagged for maintenance:
  - CCP 1B-B
  - CCS pump 1B-B
- Component Cooling Water pump 1A-A trips due to motor failure.
- AOI-15, "Loss of Component Cooling Water (CCS)," is implemented.

In accordance with AOI-15, which ONE of the following identifies...

(1) the maximum time the Reactor Coolant Pumps can be allowed to remain in service

and

(2) if the AOI-15 Attachments listed below require implementation?

Attachment 1 - Alignment of ERCW to CCP 1A-A Lube Oil Coolers
Attachment 2 - Alignment of CCS Train B to SFP HX B

	Max time	Attachments
A.✓	10 minutes	Only Attachment 1 performance is required.
B.	10 minutes	Performance of both Attachments is required.
C.	12 minutes	Only Attachment 1 performance is required.
D.	12 minutes	Performance of both Attachments is required.

DISTRACTOR ANALYSIS:

A. *Correct, AOI-15 has a Caution stating "RCPs can be operated for up to 10 minutes after loss of CCS flow" and during performance of the AOI, a step will direct the performance of Attachment 1, but the step directing the performance of Attachment 2 will not be performed because the 'if...then" condition is not met due to the 2A header being available.*

B. *Incorrect, Plausible because the time being 10 minutes is correct and both Attachments would be performed if the 2A header was not available.*

C. *Incorrect, Plausible because 12 minutes is a time in the procedure section being performed but it is the time that a CCP may survive (not the time required to remove the RCPs) and only Attachment 1 being performed is correct.*

D. *Incorrect, Plausible because 12 minutes is a time in the procedure section being performed but it is the time that a CCP may survive (not the time required to remove the RCPs) and both Attachments would be performed if the 2A header was not available.*

Question Number: 78

Tier: 1 **Group** 1

K/A: 026 AA2.06
Loss of Component Cooling Water
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

Importance Rating: 2.8* / 3.1*

10 CFR Part 55: 43.5 / 45.13

10CFR55.43.b: 5

K/A Match: The K/A is matched because the question requires knowledge of the length of time after the loss of CCW flow to the RCPs before they are required to be removed to prevent being damaged and is SRO because it requires the knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps to mitigate the damage due to the loss of Component Cooling Water.

Technical Reference: AOI-15, Loss of Component Cooling Water (CCS),

Technical Reference: AOI-15, Loss of Component Cooling Water (CCS), Revision 0032

Proposed references to be provided: None

Learning Objective: 3-OT-AOI1500

11. Demonstrate ability/knowledge of AOI, by:
    a. Recognizing Entry conditions.
    b. Responding to Actions.
    c. Responding to Contingencies (RNO).
    d. Responding to Notes/Cautions.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: WBN bank question AOI1500 002 modified for the WBN 10/2011 NRC exam.

Comments:

3.0 OPERATOR ACTIONS

3.1 Diagnostics

NOTE The loss of CCS heat sink (e.g., loss of ERCW to CCS heat exchanger) should be evaluated as a loss of CCS flow (Subsection 3.2) with equipment and CCS temperatures monitored closely.

IF	GO TO	Page
Loss of CCS flow, **OR** Surge Tank level less than 60% or dropping uncontrolled.	Subsection 3.2	6
Surge Tank level greater than 72% or rising uncontrolled, **OR** CCS Rad Monitor alarm.	Subsection 3.3	20
Loss of CCS due to loss of AC power train.	AOI-35	

Step	Action/Expected Response	Response Not Obtained

3.2 Loss of CCS Flow/Out-leakage

1. **CHECK** CCS pumps status:

Action/Expected Response	Response Not Obtained
a. **CHECK** any CCS pump TRIPPED or running pump **NOT** pumping forward: • ERCW/CCS Motor tripout alarm, • Low header pressure (Train A or B), • Multiple low flow alarms.	a. ** **GO TO** CAUTION prior to Step 2.
b. **CHECK** at least one U-1 Train A header supply pump RUNNING AND pumping forward: • 1A-A • 1B-B	b. **START** available U-1 Train A CCS pump. NO pump NONE Available
c. **CHECK** any Train B header supply pump RUNNING AND pumping forward: • C-S • 2B-B	c. **START** available Train B CCS pump **OR** **REFER TO** SOI-70.01, Component Cooling Water (CCS), to align CCS pump 1B to Train B header as necessary.
d. **PLACE** any non-operable or tripped CCS pump in STOP/PULL-TO-LOCK.	

Step continued on next page

3.2 Loss of CCS Flow/Out-leakage (continued)

1. (continued from previous page)

	Action/Expected Response		Response Not Obtained
e.	**CHECK** TWO U-1 Train A header supply pumps RUNNING: • 1A-A • 1B-B	e.	**ENSURE** at least one of the following CLOSED to avoid excessive flow: • RHR HX A, 1-FCV-70-156, OR • SFP HX A, 0-FCV-70-197.
f.	**CHECK** flows returned to NORMAL.	f.	** **GO TO** CAUTION prior to Step 2.

g.	**CHECK** A and B side Surge Tank levels between 57% and 85%.	g.	**IF** Surge Tank level less than 57%, **THEN**
			** **GO TO** CAUTION prior to Step 2.
			IF Surge Tank level greater than 85%, **THEN**
			** **GO TO** Subsection 3.3.

h.	** **GO TO** Step 15.		

Step	Action/Expected Response	Response Not Obtained
3.2	**Loss of CCS Flow/Out-leakage (continued)**	
	CAUTION A closed Surge Tank vent valve may cause a positive or negative tank pressure, giving an erroneous level indication.	
2.	**CHECK** 1-FCV-70-66, U1 Surge Tank Vent, OPEN.	**OPEN** 1-FCV-70-55, U1 Surge Tank Vent.
3.	**IF** Surge Tank level less than 57%, **THEN** **ENSURE** 1-LCV-70-63, U1 Surge Tank Makeup LCV, OPEN (Refer to SOI-70.01 as required if makeup **NOT** available).	
4.	**MONITOR** A and B side Surge Tank levels greater than 10%.	**STOP** affected CCS pumps.
5.	**IF** RHR Shutdown Cooling is in service, **THEN** ** **GO TO** AOI-14, Loss of RHR Shutdown Cooling.	

Step	Action/Expected Response	Response Not Obtained

3.2 Loss of CCS Flow/Out-leakage (continued)

DISTRACTOR.

CAUTION CCP may survive for only 10 to 12 minutes after loss of CCS to lube oil cooler.

6. **MONITOR** the following for Unit 1 CCS Train A:

- U-1 CCS Train A level
- ERCW flow to CCS HX A

IF loss of either is imminent, **THEN** — ** **GO TO** Step 7.

PERFORM the following:

a. **ENSURE** CCP 1B-B is RUNNING.

   RNO: **INITIATE** alignment of ERCW to CCP 1A-A lube oil heat exchanger USING Attachment 1 (may use placard posted locally in CCP room 1A-A). CORRECT

   ** **GO TO** Substep c.

b. **ENSURE** CCP 1A-A is STOPPED.

c. **ISOLATE** charging and letdown.

Step continued on next page

3.2 Loss of CCS Flow/Out-leakage (continued)

6. (continued from previous page)

   d. **STOP** and **LOCKOUT** the following pumps:

      - TBBPs 1-A & 1-B,
      - CCS pumps 1A-A & 1B-B,
      - CS pump 1A-A,
      - RHR pump 1A-A,
      - SI pump 1A-A,
      - CCP 1A-A.

CORRECT ANS.

CAUTION RCPs can be operated for up to 10 minutes after loss of CCS flow.

   e. **TRIP** Reactor.

   f. **STOP** RCPs.

   g. ** **GO TO** E-0, Reactor Trip or Safety Injection, WHILE continuing this Instruction.

   h. **INITIATE** alignment of ERCW to CCP 1A-A lube oil heat exchanger USING Attachment 1 (may use placard placed locally in CCP room 1A-A).

Step continued on next page

Step	Action/Expected Response	Response Not Obtained

3.2 Loss of CCS Flow/Out-leakage (continued)

6. (continued from previous page)

CAUTION CCS should **NOT** be reestablished to RCP seals on a total loss of cooling due to probable damage to the seals. ECA-0.0, Loss of Shutdown Power, has guidance to isolate RCP seals.

Step	Action/Expected Response	Response Not Obtained
i.	**IF** CCS Train B is available AND CCP 1B-B is in service, **THEN** ** **GO TO** Substep k.	**WHEN** ERCW cooling is aligned to CCP 1A-A, **THEN** **EVALUATE** performing the following based on time thermal barrier and RCP seal injection flow lost: a. **START** CCP 1A-A. b. **STOP** CCP 1B-B.
j.	**IF** thermal barrier flow lost and RCP seal injection flow **NOT** reestablished, **THEN** **REFER TO** ECA-0.0, Loss of Shutdown Power, to isolate RCP seals.	
k.	**IF** CCS Train A, Unit 1 & 2, is **NOT** available, **THEN** **ALIGN** CCS Train B to SFP HX B USING Attachment 2.	DISTRACTOR

AOI1500 002

WBN BANK QUESTION

Given the following plant conditions:

- AOI-15, Loss of Component Cooling Water (CCS), is in progress.
- BOTH trains of CCS flow indicate 0 gpm.

Per AOI-15, All RCPs MUST be stopped...?

a. immediately upon loss of CCS flow to motor oil coolers.

b. immediately upon loss of CCS flow to RCP thermal barriers.

c. ✓ within 10 minutes of a total loss of CCS flow to motor oil coolers.

d. within 10 minutes of a total loss of CCS flow to RCP thermal barriers.

I. PROGRAM

Watts Bar Operator Training

II. COURSE

A. License Training
B. Non-License Training

III. TITLE

AOI-15, Loss of Component Cooling Water (CCS)

IV. LENGTH OF LESSON

A. License Training 1.5 Hours
B. Non-License Training 1.5 Hours

V.TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. Describe the Purpose/Goal of this AOI.
	X	X	X	2. Identify Alarms associated with Loss of CCS.
	X	X	X	3. Describe Auto Actions designed to compensate for loss of CCS.
	X	X	X	4. Describe Action if Surge Tank Level is not maintained.
X	X	X	X	5. Describe affect on plant operation if Surge Tank Level is not maintained.
X	X	X	X	6. Determine the Purpose of AOI Appendix A.
	X	X	X	7. Determine Action for Loss of an ESF Equipment header.
X	X	X	X	8. Given a Rx Bldg hdr leak, determine components affected and actions to take upon header isolation.

V. TRAINING OBJECTIVES (continued)

AUO	RO	SRO	STA	
X	X	X	X	9. Given indications for a CCS Hx, determine if the Hx has a tube leak.
	X	X	X	10. Give 3 sources of potential In-leakage to the CCS.
	X	X	X	11. Demonstrate ability/knowledge of AOI, by: a. Recognizing Entry conditions. b. Responding to Actions. c. Responding to Contingencies (RNO). d. Responding to Notes/Cautions.
X	X	X	X	12. Given a loss of Component Cooling Water is in progress demonstrate the process for performing NAUO actions for AOI-15 Attachment associated with"ALIGNMENT OF ERCW TO CCP 1A-A LUBE OIL COOLERS."

VI. TRAINING AIDS

A. Marker board & markers
B. Multimedia/Overhead Projector(s)

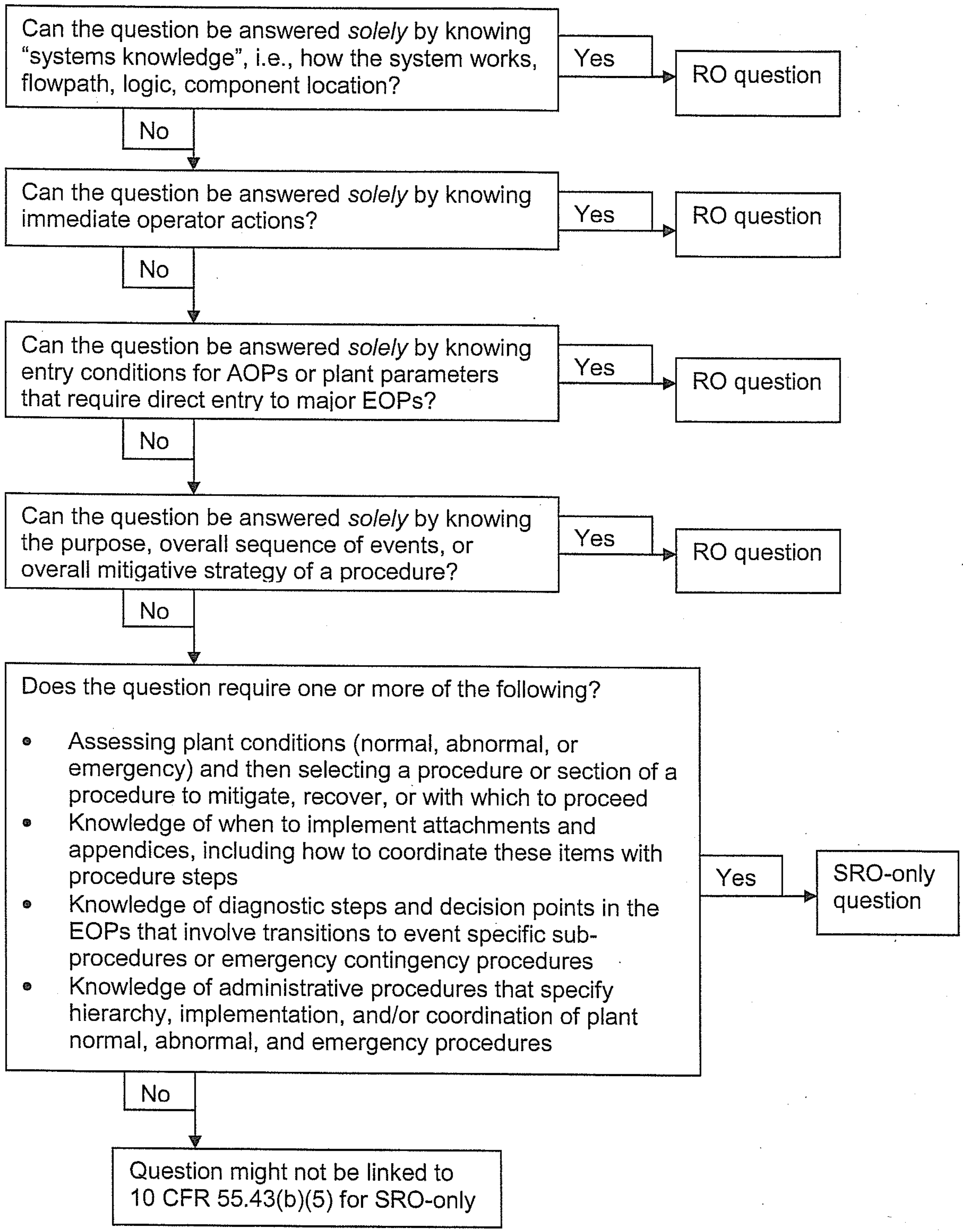
VII. MATERIALS

Attachments:

VIII. REFERENCES

ENGINEERING SYSTEM DESCRIPTION(S)		
Number	**Title**	**Rev.**
N3-70-4002	Component Cooling System (CCS)	15
WBN FSAR		
Section	**Title**	**Amend.**
9.2.2	Component Cooling System	7

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



79. 038 EG2.4.18 079
Given the following:

- Unit 1 was operating at 100% power when a Safety Injection occurred due to a tube rupture in SG #2.
- The crew determined a Target Incore Temperature of 466°F and has initiated the rapid RCS cooldown in accordance with E-3, "Steam Generator Tube Rupture."
- Before reaching the Target Incore Temperature, SG #2 pressure begins dropping in a uncontrolled manner.

Which ONE of the following identifies...

(1) the basis of cooling the RCS to a Target Incore Temperature of 466°F

and

(2) the action the SRO is required to take in accordance with the emergency instructions?

A. (1) Ensures adequate RCS subcooling is maintained after the subsequent RCS depressurization to the ruptured SG pressure determined from the E-3 table.
(2) Immediately transition to ECA-3.1, "SGTR and LOCA Subcooled Recovery."

B✓ (1) Ensures adequate RCS subcooling is maintained after the subsequent RCS depressurization to the ruptured SG pressure determined from the E-3 table.
(2) Continue in E-3 until the Target Incore Temperature is reached, then transition to ECA-3.1, "SGTR and LOCA Subcooled Recovery."

C. (1) Provides the maximum amount of RCS temperature reduction without exceeding Pressurized Thermal Shock limits of the RCS prior to the RCS depressurization.
(2) Immediately transition to ECA-3.1, "SGTR and LOCA Subcooled Recovery."

D. (1) Provides the maximum amount of RCS temperature reduction without exceeding Pressurized Thermal Shock limits of the RCS prior to the RCS depressurization.
(2) Continue in E-3 until the Target Incore Temperature is reached, then transition to ECA-3.1, "SGTR and LOCA Subcooled Recovery."

82. 003 AA2.02 082

Given the following:

- The Unit 1 reactor was at 85% when one Control Bank D, Group 2 rod dropped into the core.
- The crew has taken the actions in accordance with AOI-2, "Malfunction of Reactor Control System," and reduced reactor power to 73%.
- The control rod has been repaired.
- During recovery of the dropped rod, reactor power increases to the maximum allowed before the rod is fully recovered.
- The operators reconnect the lift coils for the appropriate rods in order to reduce reactor power.
- Annunciator 86-A, CONTROL ROD URGENT FAILURE, has **NOT** been reset.
- OAC postions 1-RBSS, ROD BANK SELECT, from the CBD to the MAN position.

Which ONE of the following identifies...

(1) which control rods, if any, will move when the IN-HOLD-OUT switch lever is placed to the IN position

and

(2) the basis for the reactor power reduction required by Tech Spec 3.1.5, Group Rod alignment Limits?

A. (1) No rod motion will occur.
(2) To ensure AFD remains within limits to prevent exceeding core design limits for hot channel factors.

B. (1) Bank D Group 2 rods ONLY.
(2) To ensure core design limits for local LHR are not exceeded due to the misaligned rod.

C✓ (1) No rod motion will occur.
(2) To ensure core design limits for local LHR are not exceeded due to the misaligned rod.

D. (1) Bank D Group 2 rods ONLY.
(2) To ensure AFD remains within limits to prevent exceeding core design limits for hot channel factors.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible since the first part is correct, with a Rod Urgent Failure alarm in all rod motion is blocked while in Auto or Manual. Also the second part is not correct for a reason in Tech Spec basis for a misaligned rod, but plausible because AFD would be changing as the rod is being withdrawn during recovery of the rod*

B. *Incorrect, Plausible since during the recovery of the dropped rod an Urgent Failure alarm is generated in power cabinet 2 BD due to having all lift coils disconnected but demanding a signal for movement while withdrawing the dropped rod. This will not prevent rod movement in the other group due to the position of the selector switch being in the CBD position. However with the selector switch in the MAN position all rod motion is stopped. Also the reason for 75% power is correct according to Tech Spec 3.1.5 bases.*

C. *Correct, With the rod control selector switch in the MAN position and Rod Control Urgent Failure alarms will stop/prevent all rod motion. The candidate will have to determine the difference between system response based on the input signals from the bank selector switch. If the switch is positioned to CBD as is the case when the rod in recovered the power cabinet for the other group would receive an Urgent Failure Alarm and block rod movement for that group, however the other group would still function and allow the rod to be recovered. The candidate must recognize the position of the selector switch, and its affect on rod motion. Also per Tech Spec 3.1.5 bases, "The reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded."*

D. *Incorrect, Plausible since during the recovery of the dropped rod an Urgent Failure alarm is generated in power cabinet 2 BD due to having all lift coils disconnected but demanding a signal for movement while withdrawing the dropped rod. This will not prevent rod movement in the other group due to the position of the selector switch being in the CBD position. Also the second part is not correct for a reason in Tech Spec basis for a misaligned rod, but plausible because AFD would be changing as the rod is being withdrawn during recovery of the rod.*

Question Number: 82

Tier: 1 **Group** 2

K/A: 003 AA2.02
Dropped Control Rod
Ability to determine and interpret the following as they apply to the Dropped Control Rod:
Signal inputs to rod control system

Importance Rating: 2.7 / 2.8

10 CFR Part 55: 43.5 / 45.13

10CFR55.43.b: 2

K/A Match: This question matches the K/A by having the candidate determine the Urgent Failure Signals affect on rod control depending on the position of the rod bank selector switch during a dropped rod recovery. SRO by having the candidate recall from the Tech Spec bases the reason for reducing reactor power to 75% during a dropped rod recovery.

Technical Reference: Tech Spec 3.1.5 and bases
AOI-2, Malfunction of Reactor Control System, Revision 0038

Proposed references to be provided: None

Learning Objective: 3-OT-SYS085A

20. Differentiate between the Rod Urgent Failure and Non-Urgent Failure alarms. Explain the cause and effect of the alarms and how resetting of alarms is accomplished.
26. Discuss applicable Technical Specifications, Technical Requirements, and Bases.

Cognitive Level:

Higher X
Lower ____

Question Source:

New ____
Modified Bank X
Bank ____

Question History: WBN bank question and a McGuire question combined and modified for use at WBN

Comments:

Step	Action/Expected Response	Response Not Obtained

3.3 Dropped RCCA (continued)

NOTE Boric acid requirements can be determined using Reactivity Briefing Sheet.

24. **ALIGN** RCCA to affected bank position:
    - (ρ) **USING** rod control in Bank Select, position affected RCCA to bank affected position recorded in Step 18
    - (ρ) **BORATE** RCS to maintain T-ave and T-ref within 3°F.
    - **MAINTAIN** less than or equal to 75% Reactor power.

    IF RCCA can **NOT** be aligned, **THEN**:

    a. **RECONNECT** lift coils (toggle down) of bank.

    b. **RESET** CONTROL ROD URGENT FAILURE alarm [86-A] with 1-RCAR.

    c. **SET** affected group step counter to original value.

    d. **RESET** the computer to its original value USING the UPDATE function.

    e. **REFER TO** Tech Specs
    - 3.1.5, Rod Group Alignment Limits.
    - 3.1.6, Shutdown Bank Insertion Limits.
    - 3.1.7, Control Bank Insertion Limits.
    - 3.1.8, Rod Position Indication.

    f. **NOTIFY** Plant Management and Reactor Engineering.

    g. **RETURN TO** Instruction in effect.

25. **RECONNECT** lift coils (toggle down) disconnected in Step 19.

Step	Action/Expected Response	Response Not Obtained

3.3 Dropped RCCA (continued)

26. **ENSURE** the following agree with values recorded in Step 18:
    - Bank overlap counter
    - Group step counters
    - Computer points

27. **RESET** CONTROL ROD URGENT FAILURE alarm [86-A] using ROD CONTROL ALARM RESET pushbutton 1-RCAR.

28. **PLACE** control rods in MAN.

29. (ρ) **RESTORE** T-ave and T-ref to within 3°F.

NOTE Computer constant K0015 contains the current monthly full out rod position for all rod banks.

30. **WHEN** plant stabilized,
    THEN
    REFER TO 1-SI-85-2, Reactivity Control Systems Movable Control Assemblies, for affected bank. (Modes 1 and 2)

3.3 Dropped RCCA (continued)

CAUTION **Allowing at least 5 minutes between any rod control input (i.e., T-ave, T-ref, or NIS) changes and placing rods in AUTO, will help prevent undesired control rod movement.**

31. **WHEN** auto rod control desired,
THEN:
    a. **ENSURE** T-ave and T-ref within 1°F.
    b. **ENSURE** zero demand on control rod position indication [1-M-4].
    c. **PLACE** rods in AUTO.

32. **RETURN TO** Instruction in effect.

End of Subsection

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Rod Group Alignment Limits

LCO 3.1.5 All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) untrippable.	A.1.1 Verify SDM is ≥ 1.6% -k/k.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	<u>OR</u>	
	B.2.1.1 Verify SDM is ≥ 1.6% -k/k.	1 hour
	<u>OR</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	B.2.2 Reduce THERMAL POWER to $\leq$ 75% RTP.	2 hours
	<u>AND</u>	
	B.2.3 Verify SDM is $\geq$ 1.6% –k/k	Once per 12 hours
	<u>AND</u>	
	B.2.4 Perform SR 3.2.1.1.	72 hours
	<u>AND</u>	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	<u>AND</u>	
	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is $\geq$ 1.6% Δk/k.	1 hour
	OR	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify individual rod positions within alignment limit.	12 hours AND Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable
SR 3.1.5.2	Verify rod freedom of movement (tripability) by moving each rod not fully inserted in the core $\geq$ 10 steps in either direction.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.5.3	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 551°F$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after initial fuel loading and each removal of the reactor head

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy and Capability," (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each). A group consists

(continued)

BASES

BACKGROUND (continued)

of two or more RCCAs that are electrically paralleled to step simultaneously. Except for Shutdown Banks C and D, a bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Analog Rod Position Indication (ARPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The ARPI System provides an accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six

(continued)

BASES

BACKGROUND (continued)	steps. The normal indication accuracy of the ARPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and ARPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

   1. specified acceptable fuel design limits, or
   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients other than a main steam line break (MSLB).

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Three types of analysis are performed in regard to static rod misalignment (Ref. 4). The first type of analysis considers the case where any one rod is completely inserted into the core with all other rods completely withdrawn. With control banks at their insertion limits, the second type of analysis considers the case when any one rod is completely inserted into the core. The third type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip in response to a main steam pipe rupture and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5). The reactor is shutdown by the boric acid injection delivered by the ECCS.

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly using incore power distribution measurements. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

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 also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

(continued)

BASES

LCO (continued)

some cases a total misalignment from fully withdrawn to fully inserted is assumed.

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 neutron (or fission) power is generated, 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 bottomed and the reactor is shut down and not producing fission power. 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) - $T_{avg} > 200°F$," for SDM in MODES 3 and 4, LCO 3.1.2, "Shutdown Margin (SDM)-$T_{avg} \leq 200°F$" for SDM in MODE 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating boration to restore SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

(continued)

BASES

ACTIONS (continued)

A.2

If the untrippable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

(continued)

BASES

ACTIONS

B.2.1.1 and B.2.1.2 (continued)

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour.

The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 6). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain an incore power distribution measurement and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

(continued)

BASES

ACTIONS

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 (continued)

to obtain the required input data and to perform the analysis.

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable.

(continued)

BASES

ACTIONS

D.2 (continued)

To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.5.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.5.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.5.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.2 (continued)

control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.5.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after initial fuel loading and reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature $\geq$ 551°F to simulate a reactor trip under actual conditions.

This Surveillance is performed prior to initial criticality and during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," and General Design Criterion 26, "Reactivity Control System Redundancy and Capability."
2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
3. Watts Bar FSAR, Section 15.0, "Accident Analyses."
4. Watts Bar FSAR, Section 15.2.3, "Rod Cluster Control Assembly Misalignment."
5. Watts Bar FSAR, Section 15.4.2, "Major Secondary System Pipe Rupture."

(continued)

WBN BANK QUESTION

2. AOI0200.05 013

Given the following plant conditions:

- The Unit is at 98% power.
- One Control Bank D, Group 1 rod dropped fully into the core.
- The dropped rod recovery is in progress per AOI-2, Malfunction of Reactor Control System.
- The dropped rod is being withdrawn, resulting in reactor power increasing to 74%.
- To address the above conditions, operators have reconnected the lift coil(s) for the appropriate rod(s), per AOI-2.
- Annunciator 86-A, CONTROL ROD URGENT FAILURE, has <u>not</u> been RESET.

With 1-RBSS, Rod Bank Select, in the CBD position, which control rods, if any, will move if the In-Hold-Out switch lever is placed to the IN position prior to resetting the Urgent Failure?

and T.S. basis for reducing Rx Power to < 75 %

a.✓ Group 1 rods, only.

b. Group 2 rods, only.

c. No rod motion will occur.

d. All Bank D rods.

a. CORRECT. When the rod was being recovered an Urgent Failure was generated in Bank D Group 2 because motion was demanded, and with all Group 2 lift coils disconnected no motion was sensed. Thus, after reconnecting the lift coils, the Urgent Failure is still present and prevents rod motion in Group 2.

b. Incorrect. In Individual Bank Select for Bank D, Group 2 rods are NOT capable of motion, since the Control Rod Urgent Failure alarm affected Power Cabinet 1BD (the power cabinet for Group 2 rods). The Control Rod Urgent Failure alarm originated from the 1BD power cabinet. With all of the Group 2 lift coils disconnected, the 1BD sensed an Urgent Failure when the Group 1 rod was withdrawn. Reconnecting the lift coils will not reset the Control Rod Urgent Failure alarm on Power Cabinet 1BD.

c. Incorrect. Plausible, since the Control Rod Urgent Failure alarm <u>does</u> block rod movement for one of the groups in Bank D (Group 2), but Group 1 Control Bank D rods will move on demand.

d. Incorrect. Plausible, since applicant may fail to recall that the effect of the standing Control Rod Urgent Failure alarm is group specific - until this alarm is reset Group 2 rods are blocked from movement. Group 1 rods WILL move, since they are on a separate power cabinet. An URGENT FAILURE exists on the 1BD power cabinet due to the previous rod withdrawal.

McGuire 2008 draft SRO exam

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	003 AA2.02	
	Importance Rating		2.8

Ability to determine and interpret the following as they apply to the Dropped Control Rod: Signal inputs to rod control system

Proposed Question: SRO 82

Given the following Unit 1 initial conditions:

- Reactor power is at 40%
- Power range NIS indicate:
  - 40% (N41), 41% (N42), 41% (N43), 41% (N44)
- Tave for each loop indicates:
  - 567°F ('A'), 567°F ('B'), 568°F ('C'), 568°F ('D')
- Turbine power is at 481 MWe
- Rod control is in automatic
- Group demand counters and DRPI indicate Control Bank 'D' at 140 steps.

Control Bank 'D' Rod L-12 drops fully into the core and the following conditions now exist:

- Power range NIS indicate:
  - 40% (N41), 40% (N42), 42% (N43), 38% (N44)
- Tave for each loop indicates:
  - 564°F ('A'), 564°F ('B'), 563°F ('C'), 564°F ('D')
- Turbine power is 478 MWe

Assuming NO operator action, which ONE of the following describes the effect on the rod control system, and the technical specification action required?

A. Rods withdraw due to the Tave-Tref mismatch. Verify Shutdown Margin requirements are met or initiate boration to ensure Shutdown Margin is met, to ensure accident analysis assumptions remain valid.

B. Rods withdraw due to the Power Range NIS Mismatch Rate signal. Verify Shutdown Margin requirements are met or initiate boration to ensure Shutdown Margin is met, to ensure accident analysis assumptions remain valid.

C. Rods withdraw due to Power Range NIS Mismatch Rate signal. Verify AFD requirements are met to ensure that fuel design limits and hot channel factors are maintained within limits.

D. Rods withdraw due to the Tave –Tref mismatch. Verify AFD requirements are met to ensure that fuel design limits and hot channel factors are maintained within limits.

Proposed Answer: A

Explanation (Optional):

A. Correct. Tave deviation is higher than 1.5 degrees F and rods will withdraw. TS action is correct.

B. Incorrect. Power mismatch is not high enough to overcome the Tave mismatch, and power mismatch is based on rate of change with turbine power, which is minimal

C. Incorrect. Incorrect bases and also incorrect reason for rod withdrawal. Plausible because power mismatch is an input and AFD would be a concern above 50% power

D. Incorrect. Incorrect basis but AFD would be a concern at higher power, as well as action required (>50%)

Technical Reference(s)	OP-MC-IC-IRX, Rev 23	(Attach if not previously provided)
	AP/14 Rev 10	
	AP-14 Basis Document Rev 6	
	TS 3.1.4	

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-MC-IRX-Obj 5 (As available)

Question Source:	Bank #	X	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam 2002 McGuire

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:
Stem not modified but distractors all different from original

KA met because inputs to rod control are the evaluated parameters.
SRO level because the effect of the failure has implications in TS basis that the applicant must determine

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training

B. Non-License Training

III. TITLE

Rod Control and Motor Generator Sets

IV. LENGTH OF LESSON

A. Licensed Training 6 hour

B. Non-Licensed Training 6 hours

V. TRAINING OBJECTIVES

NOTE: Objectives will be identified in the text as "Objective RC1" etc.

AUO	RO	SRO	STA	
X	X	X	X	1. State the design basis (purpose) of the control rod drive system.
X	X	X	X	2. State the number of RCCAs and their compositions.
X	X	X	X	3. Identify the number of banks, groups per bank, and rods per group for the shutdown control rods.
X	X	X	X	4. Describe the sequence of shutdown bank withdrawal or insertion including mode of control and speed.
X	X	X	X	5. Identify the number of banks, groups per bank, and rods per group for the control banks.
X	X	X	X	6. Describe how the rod drive mechanism moves rods on withdrawal, rest, or insertion.
X	X	X	X	7. Describe the effects of normal control rod motion on RCS T_{avg}.
X	X	X	X	8. Describe the controls for the control rods, including mode selector switch, speeds, and bank overlap.
	X	X	X	9. Sketch the control rod drive control logic from the input signals to the cyclers.
X	X	X	X	10. Identify and explain the input channels to the automatic rod control system.
	X	X	X	11. Explain how the rod control inputs serve to position the control rods on a given change in any one.
X	X	X	X	12. Describe the operation of the rate comparator circuit.
X	X	X	X	13. Discuss the purpose of the non-linear gain circuit.
X	X	X	X	14. Discuss the purpose of the variable gain circuit.

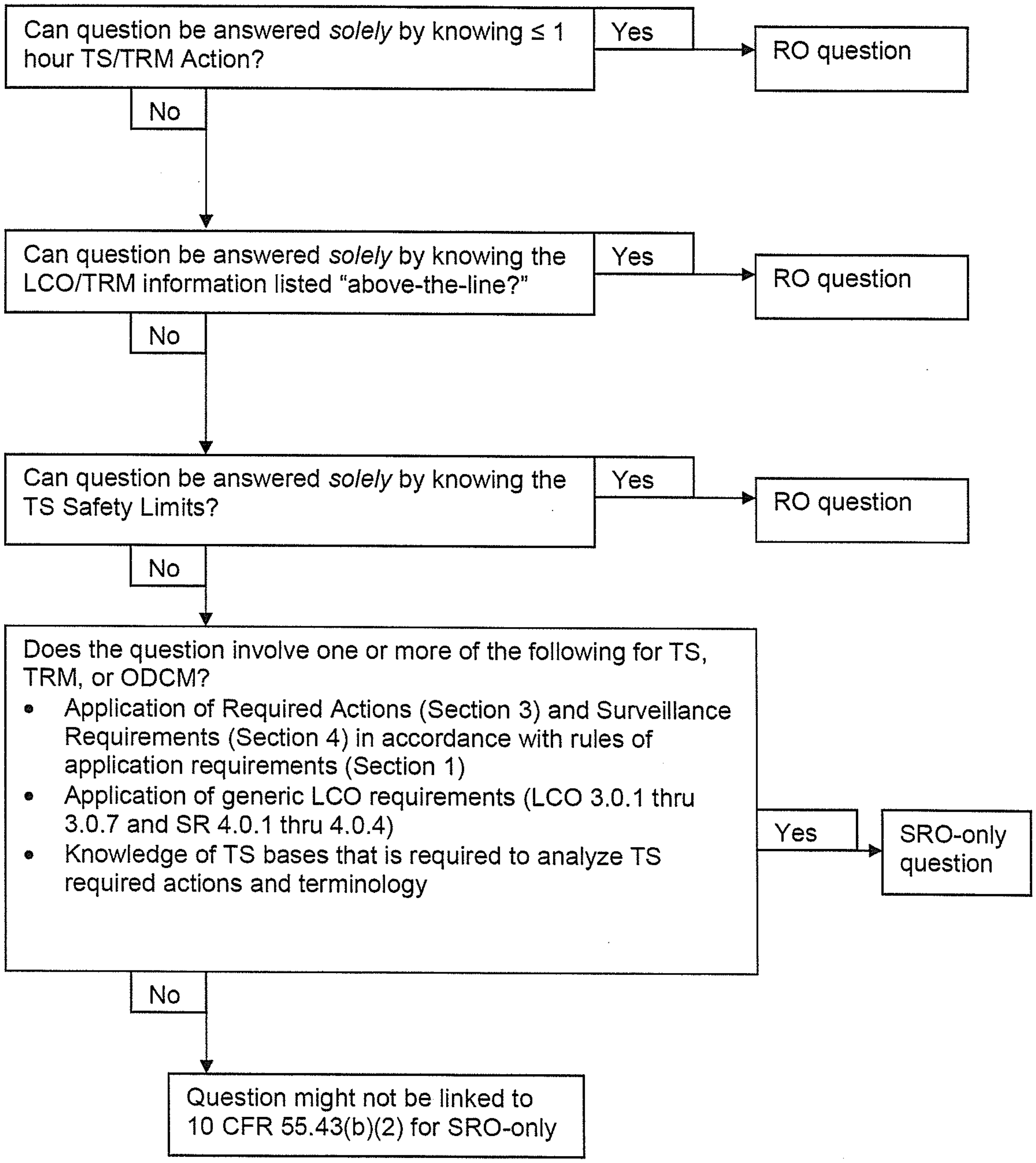
A U O	R O	S R O	S T A	
X	X	X	X	15. Draw and explain the "gull wing" program.
X	X	X	X	16. Briefly describe the purpose of each type of control rod system cabinet.
X	X	X	X	17. Briefly explain how to start up the motor generator sets.
X	X	X	X	18. Explain the purpose of the maintenance hold system for the control rod system.
X	X	X	X	19. Describe the power supplies for the control rod drive system.
	X	X	X	20. Differentiate between the Rod Urgent Failure and Non-Urgent Failure alarms. Explain the cause and effect of the alarms and how resetting of alarms is accomplished.
X	X	X	X	21. List each of the rod control stops/interlocks and give its purpose.
	X	X	X	22. For the rod position indicators, state the sources of signals, type of indication, and all alarms generated by each circuit.
	X	X	X	23. Given a failure of the controlling input instrumentation for rod control and no operator action, describe the effects of rod motion on the plant, if any.
X	X	X	X	24. Explain how a normal reactor trip occurs and how to perform an emergency reactor trip from outside the main control room.
	X	X	X	25. Explain the bases, input, alarms, and operator actions relative to the rod insertion limits.
	X	X	X	26. Discuss applicable Technical Specifications, Technical Requirements, and Bases.

TRAINING OBJECTIVES MG SETS

NOTE: Objectives will be identified in the text as "Objective MG1" etc.

AUO	RO	SRO	STA	
X	X	X	X	1. Describe the power supply to the Control Rod Drive Mechanisms.
X	X	X	X	2. Identify the power supply to the MG Sets.
X	X	X	X	3. Explain what the bypass breakers are used for.
X	X	X	X	4. Explain the function of the protective relaying equipment provided to each MG Set.
X	X	X	X	5. Describe the position indication or annunciation's the Operator has in the Main Control Room for the reactor trip and bypass breakers.
X	X	X	X	6. Describe the daily routine checks an AUO makes on the MG Sets and CRD Equipment Room as specified in "Electronic Logs".
X	X	X	X	7. Explain how to place a MG Set in service.
X	X	X	X	8. Explain how to take a MG Set out of service normally.
X	X	X	X	9. Explain how to locally trip the reactor in the event of an ATWS.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



83. 028 AA2.07 083
Given the following:

- Unit 1 was operating at 100% power with 1-RBSS, ROD BANK SELECT, in MAN.

- 1-XS-68-339E, PZR LEVEL CONTROL CHANNEL SELECT, is selected to LI-68-339 & 335.

- Performance of 1-SI-68-33, "Measurement Of Reactor Coolant Pump Seal Injection Flow," Section 6.2, "Determination of Seal Leakage," is in progress.

- The CRO has adjusted 1-HIC-62-89A, CHRG HDR-RCP SEAL FLOW CONTROL, as required and is now ready to record the seal injection flow rates for each of the RCPs.

- The 'Auctioneered High Tavg' signal fails LOW.

Which ONE of the following identifies...

(1) how the RCP seal injection flow indication will respond due to the 'Auctioneered High Tavg' signal failure

and

(2) the Bases for Tech Spec LCO 3.5.5 requiring performance of the test?

A.✓ (1) Remain the same
(2) To ensure sufficient CCP flow to the RCS through ECCS injection lines during an accident.

B. (1) Remain the same
(2) To ensure CCP flow to the RCP seals remains within 8-13 gpm after an actuation of the ECCS during an accident.

C. (1) Decrease
(2) To ensure sufficient CCP flow to the RCS through ECCS injection lines during an accident.

D. (1) Decrease
(2) To ensure CCP flow to the RCP seals remains within 8-13 gpm after an actuation of the ECCS during an accident.

DISTRACTOR ANALYSIS:

A. *Correct, As identified in LCO 3.5.5 the surveillance requirement requires the pressurizer level control valve to be fully open to perform the test. This condition is established in the Surveillance Instruction by taking manual control of the level control valve and positioning it fully open for the test. While the Tavg signal (used to determine pressurizer program level setpoint) failure would normally cause the valve to close, the valve remain full open due to being in manual leaving RCP seal injection flow unaffected. Also, the bases background for T/S 3.5.5 Seal Injection Flow states "The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident." (Also, see below)*

B. *Incorrect, Plausible because the seal injection flow remaining the same is correct and while the RCPs seal flow is designed to be maintained during an accident the bases is to limit the flow to the seals not to ensure the seals have flow.*

C. *Incorrect, Plausible because the seal injection flow dropping would be correct if the level control valve had been in automatic and the bases is correct.*

D. *Incorrect, Plausible because the seal injection flow dropping would be correct if the level control valve had been in automatic and while the RCPs seal flow is designed to be maintained during an accident the bases is to limit the flow to the seals not to ensure the seals have flow.*

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).
The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the charging pump discharge pressure is greater than or equal to the value specified in this LCO. The charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the pressurizer level control valve being full open, is required since the valve is designed to fail open for the accident condition. With the discharge pressure and control valve position as specified by the LCO, a flow limit is established. It is this flow limit that is used in the accident analyses.
The limit on seal injection flow, combined with the charging pump discharge header pressure limit and an open wide condition of the pressurizer level control valve, must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

Question Number: 83

Tier: 1 **Group** 2

K/A: 028 AA2.07
Pressurizer Level Control Malfunction
Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions:
Seal water flow indicator for RCP

Importance Rating: 2.6 / 2.9

10 CFR Part 55: 43.5 / 45.13

10CFR55.43.b: 2

K/A Match: The question matches the K/A because it requires the ability to determine how a pressurizer level control system malfunction will affect the RCP seal water flow indications for the RCPs while the plant is in alignment to perform a Surveillance Requirement. SRO because it requires knowledge of the plant alignment requirements for performance of the Surveillance Requirement and also the bases of the applicable Tech Spec.

Technical Reference: 1-SI-68-33, Measurement Of Reactor Coolant Pump Seal Injection Flow, Revision 0012
Tech Spec 3.5.5 Bases

Proposed references to be provided: None

Learning Objective: 3-OT-T/S0305
2. Determine the bases for each specification, as applicable, to the ECCS.

Cognitive Level:
Higher X
Lower ____

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the WBN 10/2011 NRC exam.

Comments:

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because establishing sufficient subcooling of the RCS so that the primary system will remain subcooled after the RCS pressure is decreased to stop the primary to secondary leakage is correct. Procedure transition plausible because a transition to ECA-3.1 will be made but not until after the cooldown is complete and if any steam generator other than the ruptured steam generator had faulted, then an immediate transition would be required but the transition would be to E-2.*

B. *Correct, The step is to establish sufficient subcooling of the RCS so that the primary system will remain subcooled after the RCS pressure is decreased to stop the primary to secondary leakage. Procedurally if the ruptured steam generator pressure starts to drop uncontrolled during the cooldown, E-3 will be continued complete the cooldown to Target incore temperature and after the target temperature is reached step will address the need to make the transition to ECA-3.1.*

C. *Incorrect, Plausible because the RCS is being rapidly cooled and the bases discusses the concern for a PTS condition and how the target temperature table is built to preclude a PTS condition. Procedure transition plausible because a transition to ECA-3.1 will be made but not until after the cooldown is complete and if any steam generator other than the ruptured steam generator had faulted, then an immediate transition would be required but the transition would be to E-2.*

D. *Incorrect, Plausible because the RCS is being rapidly cooled and the bases discusses the concern for a PTS condition and how the target temperature table is built to preclude a PTS condition. The second part is plausible because the correct procedure path is to continue the cooldown in E-3 and make the transition to ECA-3.1 after the cooldown is complete.*

Question Number: 79

Tier: 1 **Group** 1

K/A: 038 EG2.4.18
Steam Generator Tube Rupture
Knowledge of the specific bases for EOP's

Importance Rating: 3.3 / 4.0

10 CFR Part 55: 41.10 / 43.1 / 45.13

10CFR55.43.b: 5

K/A Match: This question matches the K/A by requiring the candidate to apply the basis for the step in E-3 to perform a rapid cooldown of the RCS. SRO by requiring the knowledge of specific EOP step basis, and applying the information to make the correct procedure selection.

Technical Reference: WOG E-3 Background HP-Rev 2, Step 5
E-3, Steam Generator Tube Rupture, Revision 0023

Proposed references to be provided: None

Learning Objective: 3-OT-EOP0300
5. Given a set of plant conditions, use E-3, ES-3.1, ES-3.2, and ES-3.3 to correctly diagnose and implement: Action Steps, RNOs, Foldout Pages, Notes and Cautions.

Cognitive Level:
Higher X
Lower ____

Question Source:
New ____
Modified Bank X
Bank ____

Question History: WBN bank question EOP0300 010 modified for the for the WBN 10/2011 NRC exam

Comments:

Step	Action/Expected Response	Response Not Obtained

15. **ENSURE** major steam flowpaths from the ruptured S/G isolated:

    a. TD AFW pump steam supply from Ruptured S/G CLOSED (if applicable).

    b. Ruptured S/G MSIV and bypass valve CLOSED,

    OR

    Intact S/G MSIVs and bypass valves CLOSED.

    RNO: **ISOLATE** secondary pathways to limit depressurization and contamination by INITIATING Attachment 3 (E-3), Steamline Isolation (MCR), and Attachment 4 (E-3), Steamline Isolation (Local).

16. **CHECK** Ruptured S/G pressure greater than 690 psig.

    RNO: ** **GO TO** ECA-3.1, SGTR and LOCA - Subcooled Recovery.

    Transition Before Cooldown

17. **DETERMINE** target incore temp for RCS cooldown:

    - **IF** Ruptured S/G pressure is between listed values, **THEN**

      USE lower value:

RUPTURED S/G PRESSURE (PSIG)	TARGET INCORE TEMP (°F)
1100	491°F [471°F ADV]
1000	479°F [459°F ADV]
900	466°F [446°F ADV]
800	451°F [431°F ADV]
700	434°F [414°F ADV]
690	433°F [413°F ADV]

Step	Action/Expected Response	Response Not Obtained

CAUTION

- The 1500 psig RCP trip criteria is **NOT** applicable during or after a controlled RCS cooldown and depressurization.
- If total feed flow CAPABILITY of 410 gpm is AVAILABLE, FR-H.1, Loss of Secondary Heat Sink, should **NOT** be implemented.
- Excessive steam dump cooldown rate will cause MSIV isolation due to the rate sensitive signal.
- If RCPs are **NOT** running, a false red or orange path may be indicated for FR-P.1 during the following steps. T-cold in the ruptured loop should be disregarded until Step 43.

18. **INITIATE** RCS cooldown to target incore temp, determined from Step 17.

a. **DUMP** steam to condenser from Intact S/G(s) at maximum achievable rate:

IF dumps are in Tavg mode, **THEN**:

1) **PLACE** steam dump controls OFF.
2) **PLACE** steam dump mode switch in STEAM PRESSURE.
3) **ENSURE** steam dump demand indicator 1-XI-1-33 reading zero.
4) **PLACE** steam dump controls ON.
5) **PLACE** steam dump controller in MAN, **AND FULLY OPEN** three cooldown valves (≤ 25% demand).

RESPONSE NOT OBTAINED:

a. **IF** condenser steam dumps **NOT** available, **THEN**

USE Intact S/G PORVs at maximum achievable cooldown rate.

IF an Intact S/G is **NOT** available, **THEN**

PERFORM one BUT **NOT** BOTH of the following:

- **USE** Faulted S/G,

  OR

- ** **GO TO** ECA-3.1, SGTR LOCA - Subcooled Recovery.

Transition During cooldown But not because the Ruptured SG is faulted

Step continued on the next page

Step	Action/Expected Response	Response Not Obtained

18. (continued)

b. **WHEN** RCS pressure is less than 1962 psig (P-11), **THEN**:

- **BLOCK** low pzr pressure SI.
- **BLOCK** low steam pressure SI.

c. **WHEN** Tavg is less than 550°F (P-12), **THEN**

BYPASS Lo-Lo Tavg interlock.

d. **WHEN** incore temp is less than target temp, **THEN**

STOP RCS cooldown, **AND**

MAINTAIN incore temperature less than or equal to target.

e. **CONTINUE** with Step 19 of this Instruction.

19. **MONITOR** Intact S/G levels:

a. At least one S/G NR level greater than 29% [39% ADV].

Response Not Obtained: a. **ENSURE** feed flow greater than 410 gpm.

b. S/G NR levels less than 50% and controlled.

Response Not Obtained: b. **IF** NR level in any unisolated S/G continues to rise with no feed flow, **THEN**

STOP RCS cooldown, **AND**

** **GO TO** Step 2.

Step	Action/Expected Response	Response Not Obtained
20.	**CONTROL** Intact S/G NR levels between 29% and 50% [39% and 50% ADV].	
21.	**MONITOR** pzr PORVs and block valves:	
	a. Pzr PORVs CLOSED.	a. **WHEN** RCS pressure less than 2335 psig, **THEN** **ENSURE** pzr PORV or associated block valve CLOSED. **IF** PORV fails open **AND** associated block valve can **NOT** be closed, **THEN** ** **GO TO** ECA-3.1, SGTR and LOCA - Subcooled Recovery.
	b. At least one block valve OPEN.	b. **OPEN** one block valve UNLESS it was closed to isolate an open PORV.
22.	**CHECK** pzr safety valves CLOSED: • **EVALUATE** tailpipe temperatures and acoustic monitors.	**IF** RCS pressure is less than 2485 psig, and pzr safety valve failed open, **THEN** ** **GO TO** ECA-3.1, SGTR and LOCA - Subcooled Recovery.

Step	Action/Expected Response	Response Not Obtained
	CAUTION If offsite power is lost after SI reset, manual action will be required to restart the SI pumps and RHR pumps due to loss of SI start signal.	
23.	**RESET** SI, **AND** **CHECK** the following: • SI ACTUATED permissive DARK. • AUTO SI BLOCKED permissive LIT.	**NOTIFY** IMs to block auto SI USING IMI-99.040, Auto SI Block.
24.	**RESET** Phase A and Phase B.	
25.	**ENSURE** cntmt air in service:	
	a. Aux air pressure greater than 75 psig [M-15].	a. **DISPATCH** Operator to aux air compressors: 1) **ENSURE** affected compressor(s) running. 2) **ENSURE** affected train isolation valve CLOSED: • Train A, 0-FCV-32-82. • Train B, 0-FCV-32-85.
	b. Cntmt air supply valves OPEN [M-15]: • 1-FCV-32-80. • 1-FCV-32-102. • 1-FCV-32-110.	

Step	Action/Expected Response	Response Not Obtained
26.	**DETERMINE** if RHR pumps should be stopped:	
	a. **CHECK** RHR suction aligned from RWST.	a. ** **GO TO** Step 27.
	b. **CHECK** RCS pressure greater than 150 psig.	b. **ENSURE** RHR pumps RUNNING. ** **GO TO** Step 27.
	c. **CHECK** RCS pressure stable or rising.	c. **ENSURE** CCS aligned to RHR heat exchanger: • 1-FCV-70-153 OPEN • 1-FCV-70-156 OPEN. **CLOSE** SFP heat exchanger A CCS supply 0-FCV-70-197. ** **GO TO** Step 27.
	d. **STOP** RHR pumps **AND** **PLACE** in A-AUTO.	
	e. **MONITOR** RCS pressure greater than 150 psig.	e. Manually **RESTART** RHR pumps.
27.	**CHECK** target incore temperature:	
	a. **VERIFY** incore temperature less than target temperature.	a. **DO NOT CONTINUE** this instruction **UNTIL** incore temperature less than target temperature.
	b. **STOP** RCS cooldown.	
	c. **MAINTAIN** incore temperature less than target temperature.	

Step	Action/Expected Response	Response Not Obtained
28.	**MONITOR** Ruptured S/G pressure stable or rising.	**MAINTAIN** Ruptured S/G at least 250 psig greater than the pressure of the S/G(s) used for cooldown: • Slowly **DUMP** steam from S/G(s) used for cooldown. • **MAINTAIN** RCS cooldown rate less than 100° F in one hour. **IF** the Ruptured S/G depressurizes to less than 250 psig above the pressure of the S/G(s) used for cooldown, **THEN** ** **GO TO** ECA-3.1, SGTR and LOCA - Subcooled Recovery.

29.	**CHECK** RCS subcooling greater than 85°F [105°F ADV].	**IF** subcooling is less than 65°F [85°F ADV], **THEN**
		
		** **GO TO** ECA-3.1, SGTR and LOCA - Subcooled Recovery.
		
		IF subcooling is STABLE **OR** DROPPING, **THEN**
		
		** **GO TO** ECA-3.1, SGTR and LOCA - Subcooled Recovery.
		
		DO NOT CONTINUE this instruction UNTIL subcooling is greater than 85°F [105°F ADV].

TRANSITION AFTER cooldown

STEP: Initiate RCS Cooldown

PURPOSE: To establish sufficient subcooling in the RCS so that the primary system will remain subcooled after pressure is decreased to stop primary-to-secondary leakage.

BASIS:

The principal goal of the E-3 guideline is to stop primary-to-secondary leakage and to establish and maintain sufficient indications of adequate coolant inventory. These indications include a pressurizer level indication to trend coolant inventory and RCS subcooling to ensure that the indicated pressurizer level is reliable. This step is designed to establish sufficient subcooling in the RCS so that the primary system will remain subcooled after RCS pressure is decreased in subsequent steps to stop primary-to-secondary leakage.

Since, in order to stop this leakage, the RCS pressure must be decreased to a value equal to the ruptured steam generator pressure, the temperature at which this cooldown is terminated is dependent upon the ruptured steam generator pressure. A table should be constructed for various ruptured steam generator pressures showing the fluid temperature corresponding to 20°F subcooling at each of these pressures, including allowances for subcooling uncertainties with normal or adverse containment conditions. The cooldown should be based on the core exit TCs since these also provide the input for SI termination and reinitiation. The 20°F subcooling is provided as operating margin to accommodate fluctuations in RCS temperature, perturbations in ruptured steam generator pressure, interpolation between listed ruptured steam generator pressures, and overshoot during RCS depressurization.

As previously demonstrated (see Step 3), the pressure of the intact steam generators must be maintained less than the pressure of the ruptured steam generators in order to maintain RCS subcooling. Since flow from the ruptured steam generator should be isolated, this pressure differential is established by dumping steam only from the intact steam generators. Steam dump to the condenser is preferred to minimize radiological releases and conserve feedwater supply. However, the PORVs on the intact steam generators provide an alternative steam release path. If no intact steam generator is available, RCS temperature should be controlled by adjusting feed flow to a faulted steam generator or by releasing steam from a ruptured steam generator. This latter method will result in continued primary-to-secondary leakage and is best handled in ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED.

ACTIONS:

- o Determine required core exit temperature
- o Dump steam to condenser at maximum rate
- o Dump steam from intact SG PORVs at maximum rate
- o Control feed flow to faulted SG to cooldown RCS
- o Control steam release and feed flow to stabilize RCS temperature when required temperatures are reached
- o Transfer to ECA-3.1

INSTRUMENTATION:

- o SG pressure indication
- o Core exit TCs
- o Main steamline isolation and bypass valve position indications
- o Condenser status indications
- o Steam dump valve position indication
- o SG PORVs position indication

CONTROL/EQUIPMENT:

- o Steam dump valves
- o SG PORVs
- o Feed flow control valve
- o Plant specific controls to dump steam from intact SGs by other means

KNOWLEDGE:

- It is not intended for the operator to reevaluate the required core exit temperature or precisely interpolate between values listed in the table.
- When the required core exit temperature is reached, the intact steam generator pressure (or feed flow to a faulted steam generator) should be controlled to maintain that temperature.
- Cooldown of the RCS should be completed before continuing in the guideline.
- Natural circulation flow in the ruptured loops may stagnate during this cooldown. The hot leg temperature in that loop may remain significantly greater than the intact loops. In addition, safety injection flow into the cold leg may cause the cold leg fluid temperature to decrease rapidly in that same loop. Steps to depressurize the RCS and terminate SI should be performed as quickly as possible after the cooldown has been completed to minimize possible pressurized-thermal shock of the reactor vessel.
- RCS cooldown should proceed as quickly as possible and should not be limited by the 100°F/hr Technical Specification limit. Integrity limits should not be exceeded since the final temperature will remain above 350°F.
- The RCP trip criteria (Step 1) does not apply after a controlled cooldown is initiated.
- If more than one steam generator is ruptured, the lowest ruptured steam generator pressure should be used to determine the required core exit temperature. If cooldown to a target core exit temperature is already in progress when a subsequent SGTR is diagnosed the operator should stop the cooldown until the subsequent ruptured steam generator is isolated since continuing the cooldown would lower the pressure in the newest ruptured steam generator and result in unnecessary releases prior to its isolation from the intact steam generators. The target core exit temperature should be reexamined to determine if the temperature should be reduced based on the subsequent ruptured steam generator pressure. If a RCS depressurization is in progress, although it does not impact the pressure in the newest ruptured steam generator, for the sake of simplicity it should be stopped and the plant stabilized by the operator until the newest ruptured steam generator is isolated.

PLANT-SPECIFIC INFORMATION:

- o If no intact SG is available, the operator must decide between feeding a faulted SG or steaming a ruptured SG for RCS cooldown to RHR conditions. One must weigh the concerns of reactor vessel thermal stresses, increased discharge to containment, and stresses on the SG tubes against increased radiological releases from the ruptured SG and the potential for SG overfill on an event specific basis. Refer to Section 3.2, Key Utility Decision Points.
- o For some plant designs, the probability of having no intact steam generators available may be sufficiently small to warrant removal of the associated contingency actions. The benefit of these actions should be weighed against the increased burden on operator training and complexity of the E-3 guideline.
- o Alternative means of dumping steam from the intact steam generators, such as steam flow to the turbine-driven AFW pumps, should be evaluated on plant specific basis.
- o (0.05) SG saturation pressure to preclude a PTS condition, including allowances for normal channel accuracy and post accident transmitter errors for pressure instrument. Refer to Background Document for E-3.
- o (G.01) Temperature corresponding to 20°F subcooling at the ruptured steam generator pressure, including allowances for normal channel accuracy.

  Allowances for normal channel accuracy should be based on RCS subcooling uncertainty.
- o (G.02) Temperature corresponding to 20°F subcooling at the ruptured steam generator pressure, including allowances for normal channel accuracy and post-accident transmitter errors, not to exceed 100°F.

  Allowances for normal channel accuracy and post accident transmitter errors should be based on RCS subcooling uncertainty.

STEP: Check Ruptured SG(s) Pressure - GREATER THAN (0.05) PSIG

PURPOSE:
- o To identify a secondary side break in the ruptured steam generator and transfer the operator to the appropriate contingency guideline
- o To minimize possible pressurized thermal shock of the reactor vessel due to rapid cooldown below 350°F in subsequent steps

BASIS:

Subsequent steps direct the operator to dump steam from the intact steam generators to cool the RCS as rapidly as possibly in order to establish adequate subcooling margin. The temperature at which this cooldown is terminated depends on the pressure in the ruptured steam generators. If this pressure is less than (0.05) psig, this cooldown could result in an ORANGE priority on the Integrity Status Tree. To avoid this condition the operator is transferred to ECA-3.1, SGTR WITH LOSS OF COOLANT-SUBCOOLED RECOVERY DESIRED, which limits the cooldown rate to less than 100°F/hr.

A ruptured steam generator pressure less than the saturation pressure corresponding to a temperature for precluding pressurized thermal shock (PTS) conditions is also a possible indication of a steam break associated with the affected steam generator. For such an event, the ECA-3.1 guideline is more appropriate since primary-to-secondary leakage cannot be terminated until cold shutdown.

DISTRACTOR

The basis for determining footnote (0.05) starts with determining a temperature that will preclude PTS since the cooldown rate of 100°F can be exceeded in E-3. Violating the PTS limitation would transition the operator to FR-P.2 and stop the cooldown, which contradicts the instructions provided in Step 6b of E-3. To prevent this occurrence, footnote (I.02) should be selected as the starting temperature. To this value, an assumed 40°F temperature rise across the core and the uncertainty of the core exit temperature indication should be added to the value. If this value exceeds 350°F, it should be used as the initial temperature input. If this value does not exceed 350°F, then 350°F should be used as the minimum initial temperature input. A 20°F margin should then be added to this initial temperature value, along with the uncertainty in the RCS subcooling indication. This temperature should then be converted to a saturation pressure, and the uncertainty in SG pressure indication considered.

A pressure based on this temperature input was chosen to prevent unnecessary transitions from E-3 at higher pressures when it is still desirable to continue with E-3 and to minimize possible pressurized thermal shock of the reactor vessel. Since there is no check on the reactivity condition, there is no guarantee that return to criticality will not occur during RCS cooldown for plants with BIT removed or boron concentration reduced. A plant specific evaluation may be required to determine the optimized RCS temperature used as the basis for Footnote (0.05) for plants with BIT removed or reduced BIT boron concentration.

Under the unlikely case that recriticality occurs, the RCS cooldown would result in a challenge to the Critical Safety Functions, i.e., a criticality condition on the Subcriticality Status Tree. The operator will be directed to the Guideline FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, or the Guideline FR-S.2, RESPONSE TO LOSS OF CORE SHUTDOWN, to initiate emergency boration of the RCS and obtain adequate shutdown margin. After the adequate shutdown margin is assured, the operator will be directed to go back to E-3 Guideline to continue the recovery actions.

ACTIONS:

o Check ruptured SG pressure
o Transfer to ECA-3.1

INSTRUMENTATION:

SG pressure indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

(0.05) SG saturation pressure to preclude a PTS condition, including allowances for normal channel accuracy and post accident transmitter errors for pressure instrument. Refer to Background Document for E-3.

WBN BANK QUESTION

EOP0300 010

After a ruptured S/G is isolated, E-3, Steam Generator Tube Rupture, directs an RCS cooldown to a specific target Incore TC temperature, derived from a table in E-3.

Which of the following is the BASIS for the target temperature from the E-3 table?

a. Allows a maximum amount of RCS temperature reduction without exceeding the Pressurized Thermal Shock limits.

b. Minimizes back leakage from the ruptured S/G until the subsequent RCS depressurization can be initiated.

c.✔ Ensures adequate subcooling (including instrument inaccuracies) is maintained during the subsequent RCS depressurization.

d. Prevents void formation in the S/G tubes when depressurizing the RCS with Aux Spray or Prz PORVs.

I. **PROGRAM:**

Watts Bar Operator Training

II. **COURSE:**

A. License Training

B. License Operator Requalification

III. **TITLE:**

E-3, Steam Generator Tube Rupture

IV. **LENGTH OF LESSON:**

A. License training 3 Hours

License operator REQUAL time will be determined after objectives are identified.

V. **TRAINING OBJECTIVES:**

AUO	RO	SRO	STA	
	X	X	X	1. Explain why timely operator response is important in mitigating the effects of a SGTR accident.
	X	X	X	2. Given a set of plant conditions, the operator will be able to identify which SGs, if any, are ruptured by evaluating the symptoms of a ruptured SG.
	X	X	X	3. Describe the major actions of E-3.
	X	X	X	4. Explain the basis for controlling the ruptured SG NR level greater than 29%.
	X	X	X	5. Given a set of plant conditions, use E-3, ES-3.1, ES-3.2, and ES-3.3 to correctly diagnose and implement: Action Steps, RNOs, Foldout Pages, Notes and Cautions.
	X	X	X	6. Explain the basis for cooling the RCS to a <u>target incore temp prior</u> to depressurization of the RCS.

V. **TRAINING OBJECTIVES:** (continued)

AUO	RO	SRO	STA	
	X	X	X	7. Given a set of plant conditions, including ruptured SG press, determine the target incore temp for RCS cooldown.
	X	X	X	8. Given a set of plant conditions, evaluate the conditions to determine if natural circulation exists and take appropriate action to initiate, restore, or maintain natural circulation.
	X	X	X	9. Describe the most effective method of collapsing a steam bubble in the reactor vessel head, SOER 83-02, recommendations 13c and 13b.
	X	X	X	10. Describe the action(s) taken if a RCP cannot be restarted to help cooldown and depressurize, SOER 83-02, recommendation 13d.
	X	X	X	11. Explain why it is especially important to monitor Shutdown Margin while cooling down using procedure ES-3.1.
	X	X	X	12. Explain why it is undesirable for the safety valves on a ruptured steam generator to open during a tube rupture event and explain how the possibility of their opening is reduced, SOER 83-2, recommendation 15.
	X	X	X	13. Explain why the cold leg accumulators are isolated when RCS press drops to less than 1000 psig (assuming RCS subcooling and inventory requirements are met).
	X	X	X	14. Describe the advantages and disadvantages of ES-3.1, Post SGTR Cooldown Using Backfill, (SOER 83-02, recommendation 14).

V. **TRAINING OBJECTIVES:** (continued)

AUO	RO	SRO	STA	
	X	X	X	15. Describe the consequences of letting the RCS go solid (i.e., excessive use of Safety Injection) during a steam generator tube rupture (SOER 83-02, recommendation 13a).
	X	X	X	16. Explain why it is important to cooldown to Cold Shutdown as quickly as possible (<100°F/hr) when performing procedure ES-3.2 or ES-3.3.
	X	X	X	17. Deleted.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

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 sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

80. 056 AA2.74 080

Given the following:

- Unit 1 is operating at 100% power with 1-FCV-68-332, BLOCK VALVE FOR PORV 334, closed as required by Tech Specs due to PZR PORV 334 being inoperable but capable of being cycled.

- The following sequence of events occur:

  1300 - Both 161kV Offsite power supplies are lost.

  1400 - Annunciator 91-A, PZR PORV/SAFETY OPEN, alarms due to PORV 340A opening and the PORV sticks open in mid-position during a pressure transient.

  1401 - The OAC reports:

  1-TI-68-331, PORV 340A & 334 TAILPIPE TEMP, rising, PORV 340A GREEN and RED indicating lights DARK, and 1-FCV-68-333A, BLOCK VALVE FOR PORV 340A has been closed.

  1500 - Both offsite power supplies are restored.

Which ONE of the following identifies...

(1) why the indicating lights on 1-HS-68-340A, PZR PORV 340AA, are DARK at 1401

and

(2) if Tech Specs allow continued operation in Mode 1 for an unlimited period of time with the current status of the pressurizer PORVs?

A. (1) Due to the loss of offsite power.
(2) Continued operation allowed.

B. (1) Due to the loss of offsite power.
(2) Continued operation **NOT** allowed.

C. (1) Due to the valve being at mid-position.
(2) Continued operation allowed.

D✓ (1) Due to the valve being at mid-position.
(2) Continued operation **NOT** allowed.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because there are circuits that would not have power while offsite power was lost. Also, plausible because there are conditions with both PORVs failed and isolated that will allow the unit to continue to operate in Mode 1 for an unlimited time period.*

B. *Incorrect, Plausible because there are circuits that would not have power while offsite power was lost. Also plausible because the unit being required to be placed in MODE 3 within 78 hours of the PORV 340A failure is correct.*

C. *Incorrect, Plausible because neither the RED nor the GREEN indicating light being lit is due to the PORV being stuck in the mid position and there are conditions (both PROVs inoperable but capable of being cycled) with both PORVs isolated that will allow the unit to continue to operate in Mode 1 for an unlimited time period.*

D. *Correct, with PORV 340A stuck in the mid position neither the RED nor the GREEN indicating light will be lit and the status of PORV 340A requires the plant be placed in MODE 3 within 78 hours of the failure.*

Question Number: 80

Tier: 1 **Group** 1

K/A: 056 AA2.74
Loss of Off-Site Power
Ability to determine and interpret the following as they apply to the Loss of Offsite Power:
PORV position

Importance Rating: 3.6 / 3.7

10 CFR Part 55: 43.5 / 45.13

10CFR55.43.b: 2

K/A Match: K/A is matched because the question requires the ability to determine the status of PORV indications during a loss of offsite power and is SRO because the questions requires knowledge of Tech Spec information below the line.

Technical Reference: Tech Spec LCO 3.4.11, Pressurizer PORVs, Amendment 55
1-45W600-68-1 R12

Proposed references None

to be provided:

Learning Objective: 3-OT-T/S0304

4. Given plant conditions and parameters correctly determine the applicable Limiting Conditions for Operations or Technical Requirements for the various components of the RCS.

3-OT-SYS068C

11. Describe the indication an operator has that a PORV is open or leaking through.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	X
Modified Bank	
Bank	

Question History: New question for the WBN 10/2011 NRC exam.

Comments:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each PORV.

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or more PORVs inoperable and capable of being manually cycled.	A.1	Close and maintain power to associated block valve.	1 hour
B.	One PORV inoperable and not capable of being manually cycled.	B.1	Close associated block valve.	1 hour
		<u>AND</u>		
		B.2	Remove power from associated block valve.	1 hour
		<u>AND</u>		
		B.3	Restore PORV to OPERABLE status.	72 hours

(continued)

DISTRACTOR
If both PORVs had been failed but capable of being cycled, then after Block va[lve] closed, continued operation is permitt[ed]

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One block valve inoperable.	C.1 Place associated PORV in manual control.	1 hour
	AND	
	C.2 Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 4.	12 hours
E. Two PORVs inoperable and not capable of being manually cycled.	E.1 Close associated block valves.	1 hour
	AND	
	E.2 Remove power from associated block valves.	1 hour
	AND	
	E.3 Be in MODE 3.	6 hours
	AND	
	E.4 Be in MODE 4.	12 hours
F. Two block valves inoperable.	F.1 Place associated PORVs in manual control.	1 hour
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. (continued)	F.2 Restore one block valve to OPERABLE status.	2 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-------------------------------NOTE------------------------------- Not required to be met with block valve closed in accordance with the Required Action of Condition B or E. -- Perform a complete cycle of each block valve.	92 days
SR 3.4.11.2	Perform a complete cycle of each PORV.	18 months

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. License Training
B. Licensed Requalification

III. TITLE

T/S 3.4, "Reactor Coolant System," Bases, and Technical Requirements Manual

IV. LENGTH OF LESSON

A. License Training 1 Hour

Licensed Requalification time will be determined after objectives are identified.

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	00. Demonstrate an understanding of NUREG 1122 knowledge's and abilities associated with the Reactor Vessel that are rated ≥2.5 during Initial License Training and ≥3.0 during License Operator Requalification Training for the appropriate license position as identified in Appendix A.
	X	X	X	1. Demonstrate the ability to extract specific information from the Technical Specifications and Technical Requirements, as they pertain to RCS.
		X	X	2. Determine the bases for each specification, as applicable, to the RCS.
		X	X	3. Given plant conditions/parameters correctly determine the OPERABILITY of components associated with RCS.
	X	X	X	4. Given plant conditions and parameters correctly determine the applicable Limiting Conditions for Operations or Technical Requirements for the various components of the RCS.

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training

B. . Non-License Training

III. TITLE

PZR, PZR Pressure Control System/ PZR Level Control System, and PRT

IV. LENGTH OF LESSON

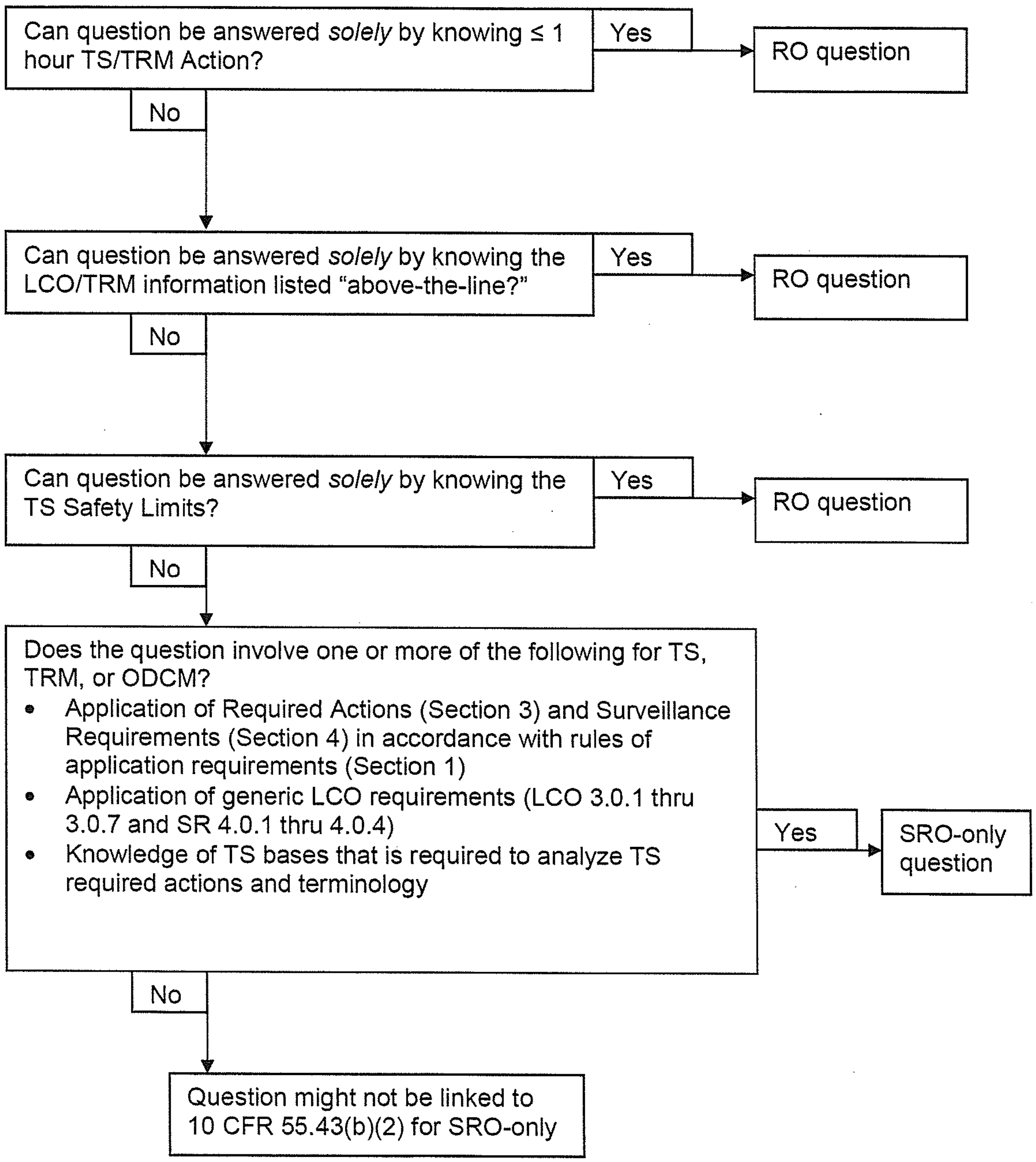
A. License Training 4 Hours
B. Non-License 6 Hours

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. Identify the three (3) main purposes of the Pressurizer.
X	X	X	X	2. Describe the major components of the Pressurizer.
X	X	X	X	3. Describe the purposes of the Manual Bypass Pressurizer Spray Throttle Valves.
X	X	X	X	4. Identify the normal setpoint required to auto open the PZR Relief Valves (PORVs).
X	X	X	X	5. Identify each setpoint and resulting automatic action for the Pressurizer Pressure Program.
	X	X	X	6. State the basis for the low pressure reactor trip, as stated in Tech Specs Section 2.1.1.
	X	X	X	7. State the basis for the high pressure reactor trip, as stated in Tech Specs Section 2.1.1.
	X	X	X	8. Describe the operation of the master pressure controller.
	X	X	X	9. Describe what control room indication would alert the operator that the pressurizer spray valves were open.
	X	X	X	10. Describe the method of control for the power operated relief valves.
	X	X	X	11. Describe the indication an operator has that a PORV is open or leaking through.

AUO	RO	SRO	STA	
X	X	X	X	12. Identify the program setpoints, and describe any automatic actions relative to the pressurizer level program.
X	X	X	X	13. Describe the basis for the program setpoints of the pressurizer level program circuit.
X	X	X	X	14. Explain the basis for programming the level vs. maintaining the level constant in the pressurizer.
X	X	X	X	15. Describe the response to a deviation from pressurizer level program.
X	X	X	X	16. Explain the purpose of the PRT.
X	X	X	X	17. Identify the components which drain into the Pressurizer Relief Tank.
	X	X		18. Deleted.
	X	X		19. Deleted
X	X	X	X	20. Describe the in-plant location of major system components, instrumentation, controls, and piping/header arrangements.
X	X	X	X	21. Describe the flow path of sources of supply, discharges, vents, drains, leakoff, and connections/penetrations that intertie this system to other systems.
X	X	X	X	22. Explain the operation of major system components.
X	X	X	X	23. Deleted
	X	X	X	24. Deleted

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



81. W/E04 EA2.1 081

Given the following:

- During performance of ECA-1.2, "LOCA Outside Containment," the crew determines RCS pressure is rising after RHR Train B cold leg injection valve 1-FCV-63-94 is closed.

- The crew then stops and locks out RHR pump 1B-B and closes its suction valve.

Which ONE of the following identifies the required procedure transition?

A. ES-1.1, "SI Termination"

B.✓ E-1, "Loss of Reactor or Secondary Coolant"

C. ECA-1.1, "Loss of Emergency Coolant Recirculation"

D. ES-1.2, "Post LOCA Cooldown and Depressurization"

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because ES-1.1 is a sub-procedure in the LOCA series of emergency procedures and would be a transition that could be required subsequent to the E-1 transition depending on the RCS pressure trend.*

B. *Correct, the RCS pressure rising indicates that the leak has been terminated and with the RCS pressure rising the transition to E-1 is directed by the step in ECA-1.1.*

C. *Incorrect, Plausible because if the RCS pressure had been dropping after the valve closure , then the transition would be to ECA-1.1*

D. *Incorrect, Plausible because ES-1.2 is a sub-procedure in the LOCA series of emergency procedures and would be a transition that could be required subsequent to the E-1 transition depending on the RCS pressure trend.*

Question Number: 81

Tier: 1 **Group** 1

K/A: W/E04 EA2.1
LOCA Outside Containment
Ability to determine and interpret the following as they apply to

K/A: W/E04 EA2.1
LOCA Outside Containment
Ability to determine and interpret the following as they apply to the (LOCA Outside Containment)
Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Importance Rating: 3.4 / 4.3

10 CFR Part 55: 43.5 / 45.13

10CFR55.43.b: 5

K/A Match: K/A is matched because the question requires the ability to assess plant conditions to determine the proper procedure transition during a LOCA outside containment event. The question is SRO because it requires '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.'

Technical Reference: ECA-1.2, LOCA Outside Containment, Revision 0005
WOG ECA-1.2 Background, Revision 2
E-1, Loss of Reactor or Secondary coolant, Revision 0016

Proposed references to be provided: None

Learning Objective: 3-OT-ECA0101
08. Given a set of plant conditions, use procedures ECA-1.1 and 1.2 to identify any required procedure transition.

Cognitive Level:
Higher X
Lower ___

Question Source:
New ___
Modified Bank ___
Bank X

Question History: VOGTLE 2010 bank question WE04EA2.1 01 used on the VOGTLE 2010 exam with wording changes in stem and to allow use at WBN. Stem conditions modified but no choices changed.

Comments:

Step	Action/Expected Response	Response Not Obtained
3.0	**OPERATOR ACTIONS**	
1.	**ENSURE** RHR suction from RCS CLOSED: • 1-FCV-74-1 and 1-FCV-74-9. **AND** • 1-FCV-74-2 and 1-FCV-74-8.	
2.	**ENSURE** SI pumps hot leg injection 1-FCV-63-156 and 1-FCV-63-157 CLOSED.	
3.	**ENSURE** RCS letdown isolated: • Letdown isolation 1-FCV-62-69 and 1-FCV-62-70 CLOSED. • Excess letdown isolation 1-FCV-62-54 and 1-FCV-62-55 CLOSED.	
4.	**ENSURE** RHR hot leg injection 1-FCV-63-172 CLOSED.	
5.	**CHECK** RCS press DROPPING or stable.	** **GO TO** Step 14.
6.	**CLOSE** RHR crosstie valve 1-FCV-74-33 or 1-FCV-74-35.	
7.	**CLOSE** RHR Train A cold leg injection valve 1-FCV-63-93.	**IF** 1-FCV-63-93 failed OPEN, **THEN** ** **GO TO** Step 10.

Step	Action/Expected Response	Response Not Obtained
8.	**CHECK** LOCA isolated: • RCS press rising.	**OPEN** 1-FCV-63-93. ** **GO TO** Step 10.
9.	**ISOLATE** RHR Train A: a. **STOP** RHR pump A-A, **AND** **PLACE** in PULL TO LOCK. b. **CLOSE** RHR suction valve 1-FCV-74-3. c. ** **GO TO** Step 15.	
10.	**CLOSE** RHR Train B cold leg injection valve 1-FCV-63-94.	**IF** 1-FCV-63-94 failed OPEN, **THEN** ** **GO TO** Step 13.
11.	**CHECK** LOCA isolated: • RCS press rising.	**OPEN** 1-FCV-63-94. ** **GO TO** Step 13.
12.	**ISOLATE** RHR Train B: a. **STOP** RHR pump B-B, **AND** **PLACE** in PULL TO LOCK. b. **CLOSE** RHR suction valve 1-FCV-74-21. c. ** **GO TO** Step 15.	

Step	Action/Expected Response	Response Not Obtained
13.	**ENSURE** RHR crosstie valves 1-FCV-74-33 and 1-FCV-74-35 OPEN.	
14.	**IDENTIFY** break location: • Radiation Protection surveys. • RHR pipe break lights [M-6]. • ECCS pump flows. • Aux bldg flood alarms [M-15; light panel, Aux Bldg 757]. • Radiation area monitor recorders 1-RR-90-1 and 0-RR-90-12A.	**NOTIFY** TSC of failure to identify break location.
15.	**DETERMINE** appropriate Instruction: • **IF** LOCA outside cntmt isolated, **THEN** ** **GO TO** E-1, Loss of Reactor or Secondary Coolant. CORRECT	**NOTIFY** TSC of failure to isolate break. ** **GO TO** ECA-1.1, Loss of RHR Sump Recirculation. DISTRACTOR

End of Section

Step	Action/Expected Response	Response Not Obtained
11.	**CHECK** SI termination criteria:	
	a. **CHECK** RCS subcooling greater than 65°F [85°F ADV].	a. ** **GO TO** Caution prior to Step 12.
	b. **CHECK** secondary heat sink available with either: • Total feed flow to Intact S/Gs greater than 410 gpm, **OR** • At least one Intact S/G NR level greater than 29% [39% ADV].	b. **ENSURE** no higher priority exists, **THEN** ** **GO TO** FR-H.1, Loss of Secondary Heat Sink.
	c. **CHECK** RCS pressure stable or rising.	c. ** **GO TO** Caution prior to Step 12.
	d. **CHECK** pzr level greater than 15% [33% ADV].	d. **RESTORE** pzr level: 1) **ATTEMPT** to stabilize RCS pressure with normal pzr sprays. 2) ** **GO TO** Caution prior to Step 12.
DISTRACTOR	e. ** **GO TO** ES-1.1, SI Termination.	

Step	Action/Expected Response	Response Not Obtained
23.	**DETERMINE** if RCS cooldown and depressurization is required:	
	a. **CHECK** RCS pressure greater than 150 psig.	a. **IF** RHR pump injecting to RCS, **THEN** ** **GO TO** Step 24.
	Distraction b. ** **GO TO** ES-1.2, Post LOCA Cooldown and Depressurization.	
24.	**PREPARE** for switchover to RHR cntmt sump:	
	a. **ENSURE** power restored to 1-FCV-63-1 USING Appendix B (E-1), 1-FCV-63-1 Breaker Operation.	
	b. **CHECK** RWST level less than 34%.	b. ** **GO TO** Step 19.
	c. ** **GO TO** ES-1.3, Transfer to Containment Sump.	

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline ECA-1.2 is to provide actions to identify and isolate a LOCA outside containment.

The following subsection provides a summary of the major categories of operator actions for guideline ECA-1.2, LOCA OUTSIDE CONTAINMENT.

3.1 High Level Action Summary

A high level summary of the actions performed in ECA-1.2 is given on the following page in the form of major action categories. These are discussed below in more detail.

o Verify Proper Valve Alignment

The first instruction given to the operator is to verify that all normally closed valves in lines that penetrate containment are closed. If a normally closed valve is open, this action may isolate the break.

o Identify and Isolate Break

The operator then attempts to identify and isolate the break by sequentially closing all normally open valves in paths that penetrate containment.

o Check If Break Is Isolated

RCS pressure is monitored to determine if the break has been isolated. A significant increase in RCS pressure indicates the break is isolated and the operator is sent to guideline E-1, LOSS OF REACTOR OR SECONDARY COOLANT. If the break is not isolated, the operator transfers to guideline ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions.

STEP: Check If Break Is Isolated

PURPOSE: To determine if the LOCA outside containment has been isolated from previous actions

BASIS:

This step instructs the operator to check RCS pressure to determine if the break has been isolated by previous actions. If the break is isolated in Step 2, a significant RCS pressure increase will occur due to the SI flow filling up the RCS with break flow stopped.

The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. If the break has not been isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there will be no inventory in the sump.

ACTIONS:

- Determine if RCS pressure is increasing
- Transfer to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1
- Transfer to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1

INSTRUMENTATION:

RCS pressure indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

It should be noted that for some breaks SI flow may cause an RCS pressure increase independent of break isolation. It should also be noted that for larger breaks, RCS repressurization may be delayed following break isolation. Additionally, if the RCS is saturated or a cooldown is in progress, RCS repressurization will proceed more slowly. Other means of verifying break isolation should be checked. For example, increasing RVLIS trend due to injection flow, decreasing trends in local abnormal conditions and local observation (if practical) may be useful.

VOGTLE BANK QUESTION

99.

The crew is implementing EOP 19112-C, "ECA-1.2 LOCA Outside Containment".

- The COLD LEG INJECTION FROM SIS HV-8835 has been closed.
- The SS determines the leak is now isolated.

The SS will transition to....

A. 19011-C, "ES-1.1 SI Termination".

B. 19010-C, "E-1 Loss of Reactor or Secondary Coolant".

C. 19111-C, "ECA-1.1 Loss of Emergency Coolant Recirculation".

D. 19012-C, "ES-1.2 Post-LOCA Cooldown and Depressurization".

I. **PROGRAM:**

Watts Bar Operator Training

II. **COURSE:**

A. License Training

B. License Requalification

III. **TITLE:**

Emergency Contingency Actions, ECA-1.1 and 1.2

IV. **LENGTH OF LESSON:**

License Certification 2 Hours

License operator REQUAL time will be determined after objectives are identified.

Non-License operator REQUAL time will be determined after objectives are identified.

V. **TRAINING OBJECTIVES:**

AUO	RO	SRO	STA	
	X	X	X	00. Deleted.
	X	X	X	01. Identify and explain the major actions of procedures ECA-1.1 and 1.2.
	X	X	X	02. Given the time of reactor trip, be able to use ECA-1.1 figure 1 to identify the minimum required SI flow.
	X	X	X	03. Explain the purpose of establishing minimum SI flow as determined by figure 1.
	X	X	X	04. Identify problems which might result from dropping RCS press to less than 250 psig prior to CLA isolation.

V. **TRAINING OBJECTIVES:** (continued)

AUO	RO	SRO	STA	
	X	X	X	05. Explain the action taken and the basis for that action if RWST level decreases to 8%.
	X	X	X	06. Identify and explain the limitation on charging flow after the RWST is empty and the CCP suction is aligned to the VCT.
	X	X	X	07. Identify the limitations for continued RCP operation at low RCS pressures.
	X	X	X	08. Given a set of plant conditions, use procedures ECA-1.1 and 1.2 to identify any required procedure transition.
	X	X	X	09. Discuss the reasons that ECA-1.1, Loss of RHR Sump Recirculation, is given priority over FR-Z.1, High Containment Pressure for directing Containment Spray Operation.
	X	X	X	10. Determine appropriate operator actions/system response for the Containment Spray System with SI actuated under each of the following conditions: • RWST LEVEL LO RECIRC INTLK Alarm • RWST LEVEL LO-LO Alarm • Containment Pressure > 13.5 psig • Containment Pressure between 2.0 & 13.5 psig • Containment Pressure <2.0 psig

V. **TRAINING OBJECTIVES:** (continued)

AUO	RO	SRO	STA	
	X	X	X	11. Given a set of plant conditions, use ECA-1.1 and ECA-1.2 to correctly diagnose and implement: Action Steps, RNOs, Notes and Cautions.

VI. **TRAINING AIDS:**

A. White Marker Board.
B. Classroom PC
C. Projector.

VII. **MATERIALS:**

A. Appendix A - Instructor Guide for Fundamental Overview of ECA 1-1.

B. Attachments, Handouts: One copy of each of the following for each participant :

1. Attachment 1- WBN Emergency Contingency Actions, ECA-1.1
2. Attachment 2- WBN Emergency Contingency Actions, ECA-1.2
3. Attachment 3- Background Information for ECA-1.1
4. Attachment 4- Background Information for ECA-1.2
5. Attachment 5- ECA 1.1 and ECA 1.2 Power Point presentation
6. Attachment 6- Operating Experience - OE23154 - Watts Bar - Unplanned Loss of Reactor Coolant

X. LESSON BODY — INSTRUCTOR NOTES

Purpose: To check if an excessive containment hydrogen concentration is present.

Basis: This step instructs the operator to obtain a current hydrogen concentration measurement. Depending upon the magnitude of the hydrogen concentration, the operator will either continue with ECA-1.1, turn on the hydrogen recombiners or notify the TSC to determine additional recovery actions before continuing with the instruction.

47) **CONSULT** TSC for long term plant operation.

Purpose: To consult with the plant engineering staff for further actions.

Basis: This ECA provides generic steps for cooldown and depressurization of the plant to atmospheric conditions following a loss of emergency containment recirculation capabilities.
Subsequent actions are plant specific and plant operators, TSC personnel and plant management need to make decisions about long term plant operation and any repairs necessary for plant restart.
The presence of acidic water from the LOCA may lead to chloride induced stress corrosion of the recirculation loop piping.

B. Discussion of ECA-1.2,.LOCA Outside Containment

Instructor note: Use latest revision of Emergency Instructions.

1. Purpose

This Instruction provides actions to identify and isolate a LOCA outside containment.

Instructor note: **Procedure Use and Adherence:** Reinforce procedure usage and placekeeping standards during presentation of LP.

2. Symptoms and Entry Conditions

a. Symptoms

X. LESSON BODY | INSTRUCTOR NOTES

Abnormal Auxiliary Building radiation due to LOCA outside containment.

b. Transitions

1) E-0, Reactor Trip or Safety Injection.

2) E-1, Loss Of Reactor Or Secondary Coolant.

3. Major Action Categories

Objective 1

a. Verify proper valve alignment.

b. Identify and isolate break.

c. Check if break is isolated.

4. Steps, Purposes, and Bases

NOTE: Outline is changed to correspond to procedure step numbers.

1) **ENSURE** RHR suction from RCS CLOSED.

Purpose: To ensure that normally closed valves are closed.

Basis: This step instructs the operator to verify that all normally closed valves in low pressure lines and other plant specific lines that penetrate containment are closed. The valving connecting the RHR system to the RCS is of particular interest since the RHR system is a low pressure system connected to the high pressure RCS. Therefore a rupture or break outside containment is most probable to occur in the low pressure RHR piping.

Operator Fundamentals: Understanding plant design.

2) **ENSURE** SI pumps hot leg injection 1-FCV-63-156 and 1-FCV-63-157 CLOSED.

Purpose: To ensure that normally closed valves are closed.

Basis: Same as Step 1.

3) **ENSURE** RCS letdown isolated.

X. LESSON BODY | INSTRUCTOR NOTES

Purpose: To ensure that normally closed valves are closed.

Basis: Same as Step 1.

4) **ENSURE** RHR hot leg injection 1-FCV-63-172 CLOSED.

Purpose: To ensure that normally closed valves are closed.

Basis: Same as Step 1.

5) **CHECK** RCS press DROPPING or stable.

Purpose: To determine if actions performed to this point isolated the break.

Basis: Plant specific step added to address the possibility that prior actions may have isolated the break.

6) **CLOSE** RHR crosstie valve 1-FCV-74-33 or 1-FCV-74-35.

Purpose: To attempt to identify and isolate a LOCA outside containment.

Basis: This step instructs the operator to sequentially close and open all normally opened valves in paths that penetrate containment to identify and isolate the break outside containment.
Again as in the previous steps, the valving connecting the RHR system to the RCS is of primary interest since the probability of a break occurring outside containment is most probable to occur in the low pressure RHR system piping.

Procedure use and adherence:

Reinforce procedure usage standard.

7) **CLOSE** RHR Train A cold leg injection valve 1-FCV-63-93.

Purpose: To attempt to identify and isolate a LOCA outside containment.

Basis: Same as Step 6.

8) **CHECK** LOCA isolated:

X. LESSON BODY | INSTRUCTOR NOTES

Purpose: To determine if the LOCA outside containment has been isolated by previous actions.

Basis: This step instructs the operator to check RCS pressure to determine if the break has been isolated by a previous action. If the break is isolated, a significant RCS pressure rise will be observed due to the SI flow filling up the RCS with break flow stopped.

9) **ISOLATE** RHR Train A:

Purpose: To attempt to identify and isolate a LOCA outside containment.

Basis: Same as Step 6.

10) **CLOSE** RHR Train B cold leg injection valve 1-FCV-63-94.

Purpose: To attempt to identify and isolate a LOCA outside containment.

Basis: Same as Step 6.

11) **CHECK** LOCA isolated:

Purpose: To determine if the LOCA outside containment has been isolated by previous actions.

Basis: Same as Step 8

12) **ISOLATE** RHR Train B:

Purpose: To attempt to identify and isolate a LOCA outside containment.

Basis: Same as Step 6.

13) **ENSURE** RHR crosstie valves 1-FCV-74-33 and 1-FCV-74-35 OPEN.

Purpose: To properly realign RHR if not the source of leakage.

X. LESSON BODY	INSTRUCTOR NOTES
Basis: Plant specific step added to reopen the crosstie valves if leak path is not found in RHR system.	
14) **IDENTIFY** break location:	
Purpose: To attempt to identify and isolate a LOCA outside containment.	
Basis: Same as Step 6. Plant specific guidance on indications/actions for leak identification have been added to this step.	
15) **DETERMINE** appropriate instruction:	
Purpose: To direct the operator to the proper instruction after leakage has either been isolated or cannot be isolated.	
Basis: If the break can be identified and isolated, then the operator is directed to E-1, for further recovery actions. If the break cannot be identified and isolated, then the operator is directed to ECA-1.1, where actions are taken to minimize break flow and initiate makeup to the RWST.	**Objective 8**
C. Operating Experience	
On 7/7/06 at **Watts Bar**, an unplanned loss of Volume Control Tank inventory occurred during preparations for the transfer of resins from the mixed bed demineralizer to the spent resin storage tank.	
One of two possible valves was leaking through to the in-service mixed bed vessel, and upon venting the cation bed to the Tritiated Drain Collector Tank, the loss of inventory occurred.	

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**