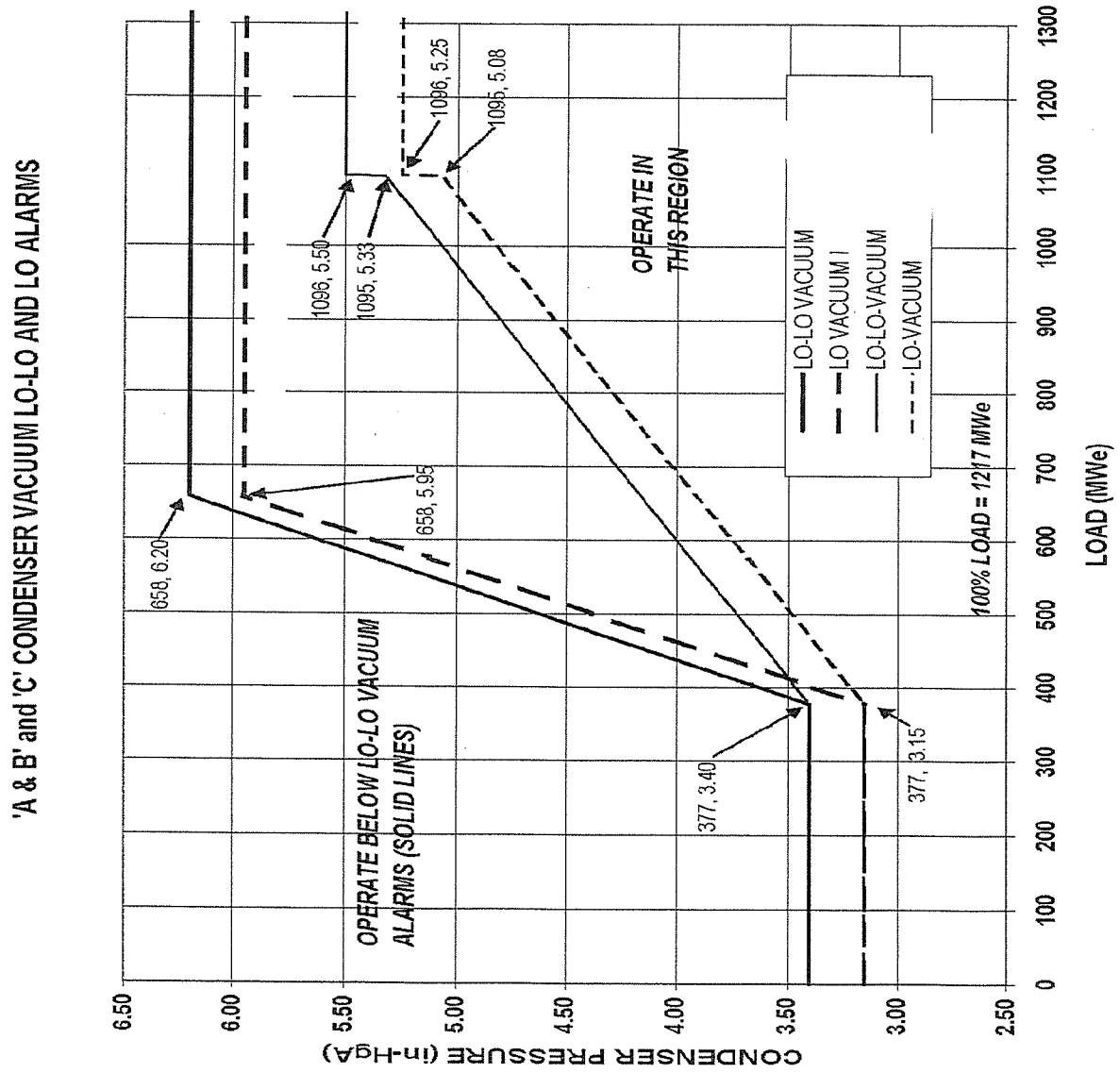
WBN 10-2011 NRC RO EXAM as Submitted
08/15/2011

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)

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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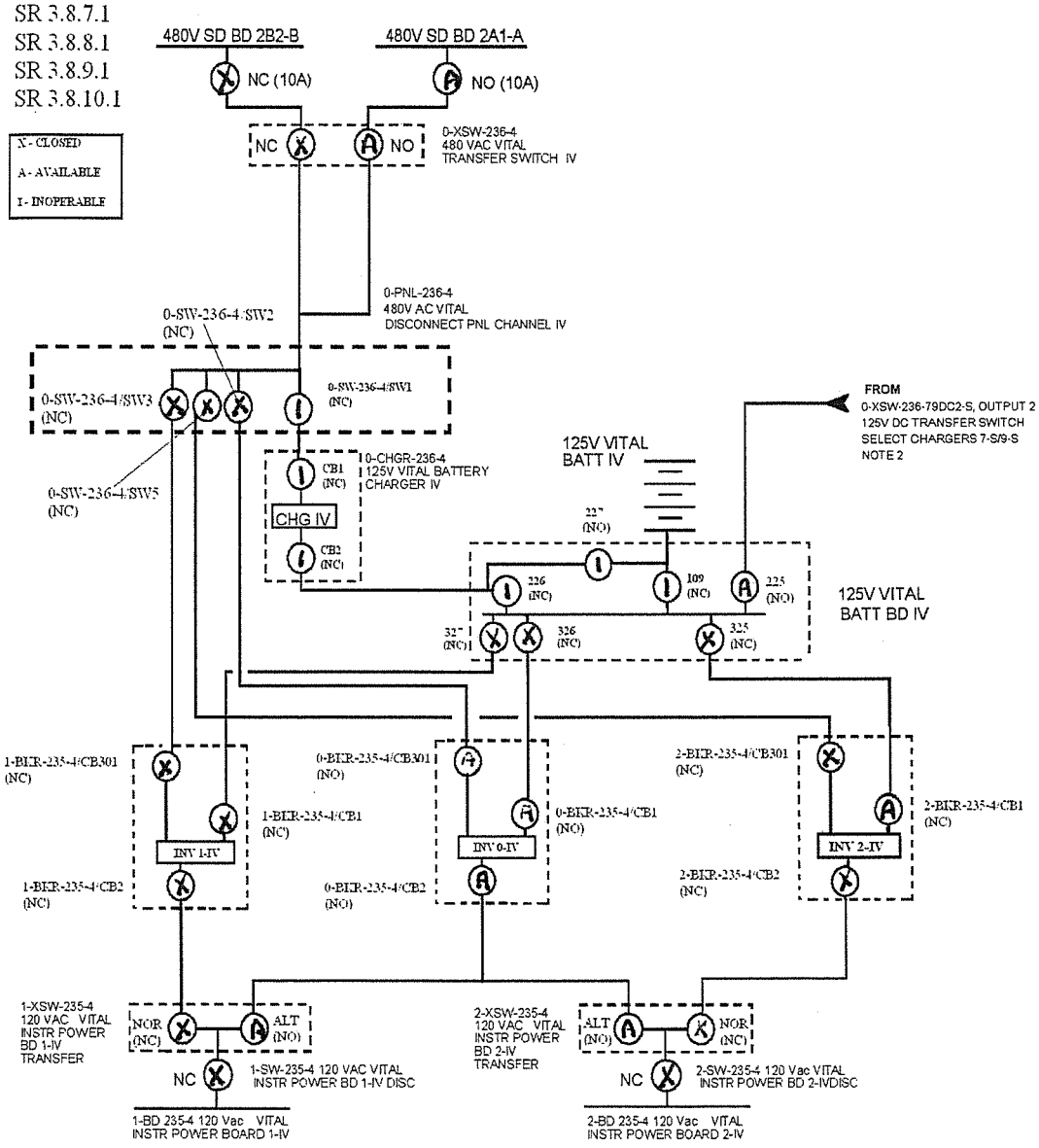
SR 3.8.4.3
SR 3.8.5.1
SR 3.8.7.1
SR 3.8.8.1
SR 3.8.9.1
SR 3.8.10.1



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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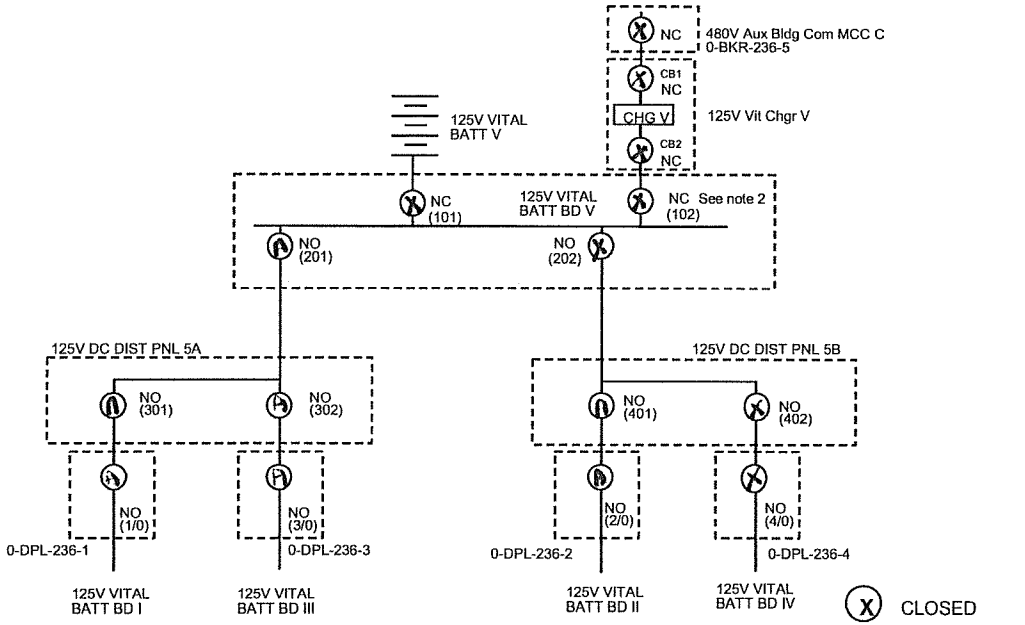
SR 3.8.9.1
SR 3.8.10.1



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: _____

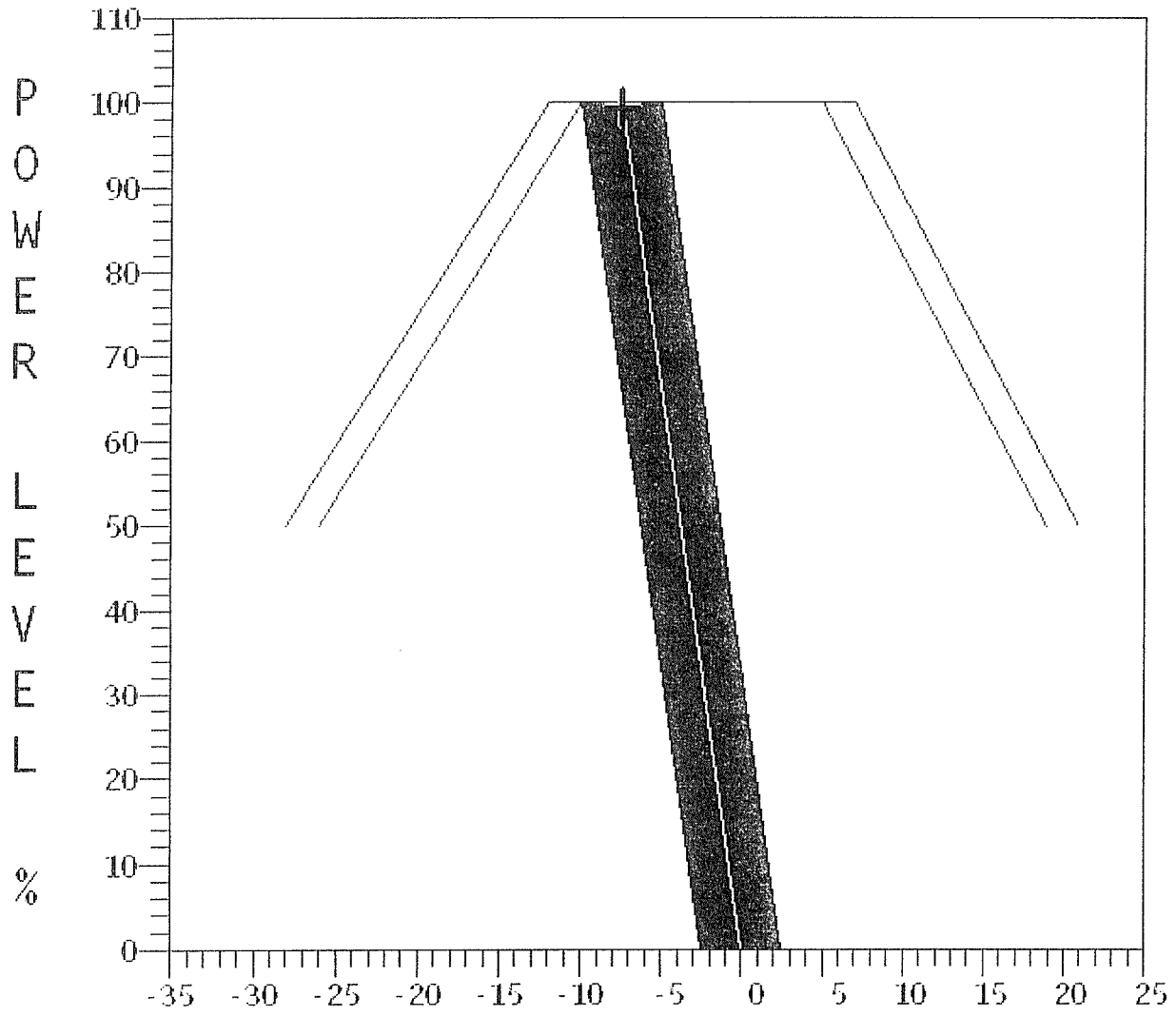
TVA AFD TARGET DISPLAY

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

Step	Action/Expected Response	Response Not Obtained

19. **CHECK** SI termination criteria:

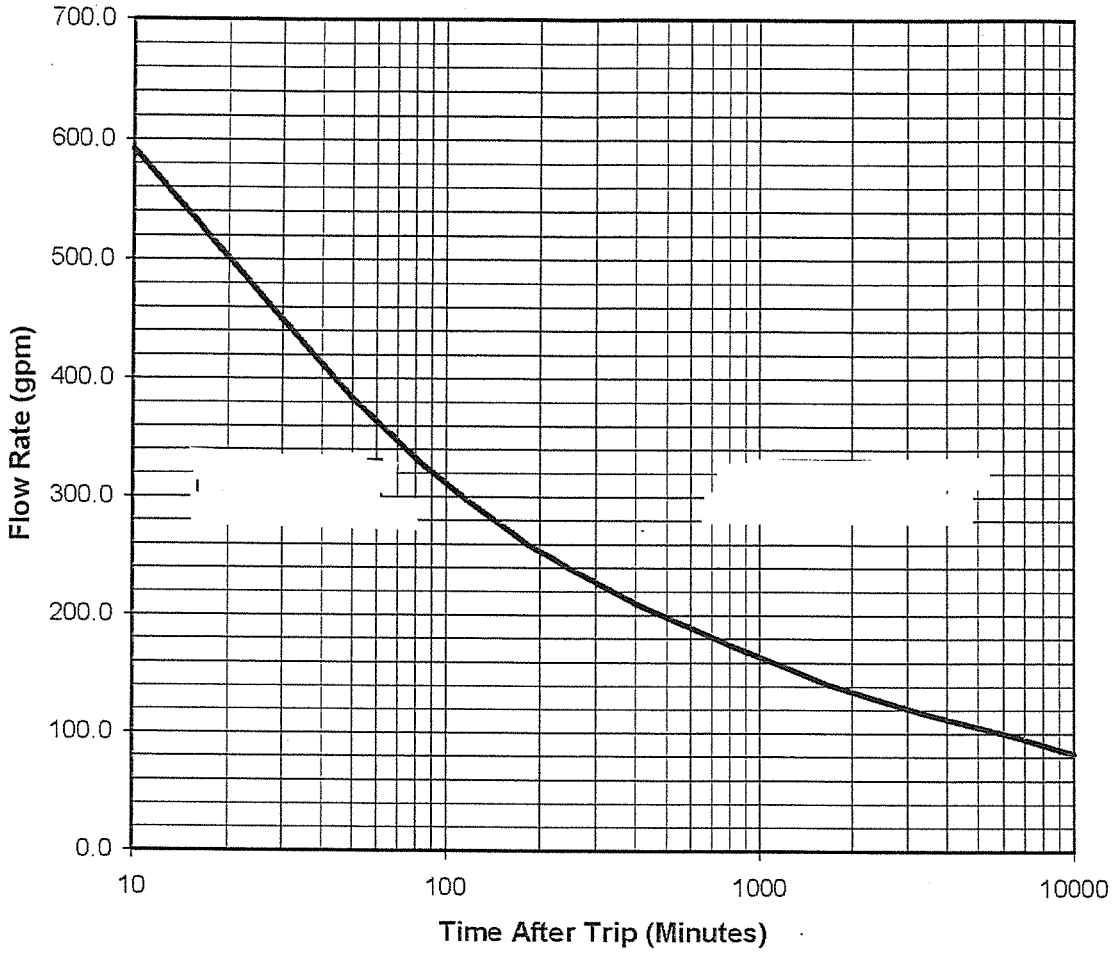
a.	RVLIS greater than 60% with NO RCP running, **OR** RVLIS greater than 63% with ANY RCP running.	a. **IF** RVLIS is less than or equal to setpoint, **THEN** ** **GO TO** Step 25.
b.	RCS subcooling greater than required from table:	b. **ESTABLISH** minimum ECCS flow for decay heat removal: 1) **REFER 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



1. 007 EK1.02 001

Given the following plant conditions:

- A reactor trip occurs on Unit 1.
- ES-0.1, "Reactor Trip Response" has been implemented.
- One control rod failed to insert.
- Tavg dropped to 549°F before stabilizing.

Which ONE of the following identifies the status of the reactor?

Conditions indicate adequate Shutdown Margin is...

A. present and Immediate Boration is **NOT** required.

B. present but Immediate Boration is required due to the stuck control rod.

C. **NOT** present and Immediate Boration is required due to the stuck control rod.

D. **NOT** present and Immediate Boration is required due to the RCS temperature.

DISTRACTOR ANALYSIS:

A. Correct, Shutdown margin is assured following a reactor trip even if 1 control rod is stuck in the fully withdrawn position and the RCS was cooled down to 549°F, thus emergency boration is NOT required.

B. Incorrect, Plausible because conditions indicate adequate shutdown margin does exist and if two rods had failed to insert, then immediate Boration is required.

C. Incorrect, Plausible because Immediate Boration would be required if two of more control rods were stuck out due to the concern for maintaining adequate shutdown margin.

D. Incorrect, Plausible because Immediate Boration would be required if the RCS Temperature had dropped to less than 547°F due to the concern for maintaining adequate shutdown margin.

Question Number: 1

Tier: 1 **Group:** 1

K/A: 007 EK1.02
Reactor Trip
Knowledge of the operational implications of the following concepts as they apply to the reactor trip:
Shutdown Margin

Importance Rating: 3.4 / 3.8

10 CFR Part 55: 41.8 / 41.10

10CFR55.43.b: Not applicable

K/A Match: This question matches the K/A by having the candidate recall the operational effect of a stuck rod following a reactor trip on shutdown margin.

Technical Reference: ES-0.1, Reactor Trip Response, Revision 0023

Proposed references to be provided: None

Learning Objective: 3-OT-EOP0000
26. Given a set of plant conditions, use E-0, ES-0.0, ES-0.1, ES-0.2, ES-0.3, ES0.4 and the Critical Safety Function Status Trees 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: SQN bank question E-0-B.3.A 010 modified

Comments:

Step	Action/Expected Response	Response Not Obtained

CAUTION Plant conditions, AFW pump start signals and flow requirements should be evaluated as time allows.

1. **MONITOR** SI actuation criteria:
   - **IF** SI actuation occurs during the performance of this Instruction, **THEN**
     ** **GO TO** E-0, Reactor Trip or Safety Injection.

2. **CHECK** Generator PCBs OPEN. — **OPEN** manually.

Step	Action/Expected Response	Response Not Obtained
3.	**MONITOR** RCS temperature stable at or trending to 557°F using: • RCS Loop T-avg with any RCP running. OR • RCS Loop T-cold with RCPs out-of-service.	**IF** temperature is less than 557°F, **THEN** **ENSURE** steam dumps, S/G PORVs, and blowdown isolation valves CLOSED. **IF** cooldown continues, **THEN** **ENSURE** total feed flow is less than or equal to 500 gpm: • **REFER TO** SOI-3.02, Auxiliary Feedwater System, for manual control of TDAFWP. • **MAINTAIN** at least one S/G NR level greater than 29%, or total feed flow between 410 and 500 gpm for heat sink. **IF** cooldown continues after AFW flow is controlled, **THEN:** • **CLOSE** MSIVs. • **ENSURE** MSIV bypasses CLOSED. • **PLACE** steam dump controls OFF. **IF** temperature is less than 547°F after AFW is controlled, **THEN** **INITIATE** boration: • **REFER TO** AOI-34, Immediate Boration. **IF** temperature is greater than 564°F, **THEN** **ENSURE** either steam dumps, or S/G PORVs OPEN. **WHEN** cooldown is controlled, **THEN** **RETURN** AFW to AUTO as desired.

Step	Action/Expected Response	Response Not Obtained
4.	**ENSURE** AFW operation:	
	a. AFW established: • Both MD AFW pumps RUNNING. • TD AFW pump RUNNING. • LCVs in AUTO or controlled in MANUAL.	a. **ESTABLISH** feed flow from AFW or MFW as necessary.
	b. Heat sink available: • Total feed flow greater than 410 gpm, OR • At least one S/G NR level greater than 29%.	b. **IF** heat sink can **NOT** be established, **THEN** ** **GO TO** FR-H.1, Loss Of Secondary Heat Sink.

Step	Action/Expected Response	Response Not Obtained
5.	**CHECK** MFW status:	
	a. **CHECK** RCS T-avg less than 564°F.	a. **WHEN** T-avg is less than 564°F, **THEN** **PERFORM** Substep 5b. ** **GO TO** Step 6.
	b. **ENSURE** MFW isolation: • MFW isolation and bypass isolation valves CLOSED. • MFW reg and bypass reg valves CLOSED. • MFP A and B TRIPPED. • Standby MFP STOPPED. • Cond demin pumps TRIPPED. • Cond booster pumps TRIPPED. • #3 HDT Pumps TRIPPED. • #7 HDT Pumps TRIPPED.	b. Manually **CLOSE** valves, and **STOP** pumps, as necessary. **IF** valves can **NOT** be closed, **THEN** **CLOSE** #1 heater outlet valves.

Step	Action/Expected Response	Response Not Obtained
6.	**ENSURE** all control and shutdown rods fully inserted: • RPIs at bottom scale.	**IF** two or more rods are **NOT** fully inserted, **THEN** **INITIATE** boration of 3250 gals of greater than or equal to 6120 ppm boron for each rod not fully inserted: • **REFER TO** AOI-34, Immediate Boration.
7.	**ANNOUNCE** reactor trip over PA system.	
8.	**MONITOR** S/G levels:	
	a. At least one S/G NR level greater than 29%.	a. **ENSURE** feed flow greater than 410 gpm.
	b. S/G NR levels less than 50% and controlled.	b. **IF** any S/G NR level continues to rise, **THEN** **ISOLATE** feed flow to affected S/G.
9.	**CONTROL** S/G NR levels between 29% and 50%.	
10.	**INITIATE** BOP realignment: • **REFER TO** AOI-17, Turbine Trip.	

SQN BANK QUESTION

E-0-B.3.A 010

Given the following plant conditions:

- A reactor trip without SI has occurred.
- The operators are implementing E-0, Reactor Trip or Safety Injection.
- A check of rod position indications shows that one control rod has not fully inserted.
- The operators continue with the procedure without taking action concerning the stuck rod.

Given the conditions above, why is Emergency Boration **NOT** required at this time?

A. Emergency boration will be required, following the transition to ES-0.1, "Reactor Trip Response."

B. Maintaining adequate shutdown margin is ONLY a concern following a reactor trip with RCS temperature less than 547°F and decreasing.

C. Three or more control rods must be stuck to affect shutdown margin to the extent that emergency boration is required.

D. Verifying that the remaining rods are fully inserted ensures adequate Shutdown Margin is present.

The correct answer is D

Justification:

A: Incorrect: ES-0.1 will only emergency borate if 2 rods are > 12 steps out.

B. Incorrect: Shutdown margin is ALWAYS a concern not just temperature dependent.

C. Incorrect: 2 rods are > 12 steps out is the requirment for emergency boration

D. Correct. Shutdown margin is adequate when most reactive rod stuck out.

K/A: 007 EK1.02 3.4/3.8
007 EK3.01 4.0/4.6
005 AK3.01 4.0/4.3

Reference: EPM-3-E-0, page

Objective: OPI271c379, b.3.a

History:

Level: Low

Note:

I. **PROGRAM:**

Watts Bar Operator Training

II. **COURSE:**

A. License Training

B. License Requalification

III. **TITLE:**

E-0, Reactor Trip or Safety Injection

IV. **LENGTH OF LESSON:**

A. License training 4 Hours
B. License Operator Requal time will be determined after objectives are identified.

V. **TRAINING OBJECTIVES:**

AUO	RO	SRO	STA	
	X	X	X	1. Identify the 3 major actions of the E-0 procedure.
	X	X	X	2. Given a set of plant conditions determine if the RCP trip criteria and required conditions for pump shutdown are met.
	X	X	X	3. State when the Status Trees/SPDS should be monitored.
	X	X	X	4. State the action that must be taken if it is determined in Step 1 of E-0 that the reactor cannot be shutdown.
	X	X	X	5. [Explain the importance of quickly verifying a turbine trip following a reactor trip.]
	X	X	X	6. State the action that must be taken if it is determined that at least one train of shutdown boards cannot be energized.

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

AUO	RO	SRO	STA	
	X	X	X	7. State the action required if one or both trains of SI fail to actuate.
	X	X	X	8. Analyze a given set of plant conditions and determine required procedural transitions per E-0.
	X	X	X	9. Discuss the basis for monitoring RCS temp using T-cold when no RCPs are running as directed by ES-0.1.
	X	X	X	10. Given a set of plant conditions, determine if natural circulation is occurring in the RCS and identify actions required to establish natural circulation per ES-0.2.
	X	X	X	11. Given pzr level and RCS subcooling, determine if ES-0.2 SI actuation criteria are met and state the action which must be taken if SI is required.
	X	X	X	12. Given the results of RCS boron concentration samples taken while on natural circulation, explain why the sample taken from the RCS cold leg is higher than the pzr CB following RCS boration.
	X	X	X	13. Given plant conditions occurring as a result of depressurization during a natural circulation cooldown, determine whether or not RCS voiding is taking place.
	X	X	X	14. Using ES-0.2 determine the correct response for RCS parameters indicating RCS voiding.
	X	X	X	15. Explain the purpose for and basis of each step in E-0, ES-0.0, ES-0.1, ES-0.2, ES-0.3, and ES-0.4.

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

AUO	RO	SRO	STA	
	X	X	X	16. Explain the purpose of procedures ES-0.3 and ES-0.4 including when their use might be required.
	X	X	X	17. List the reasons why pzr level and RCS subcooling are increased prior to starting an RCP when using ES-0.3 or ES-0.4.
	X	X	X	18. Determine the RVLIS Static Range setpoint that is equal to the top of the RCS Hot Legs, and discuss the basis for maintaining level above this setpoint.
	X	X	X	19. Identify the parameter used to monitor RCS vessel head void growth in ES-0.4 (without RVLIS).
	X	X	X	20. Explain the basis for cycling through repressurization back to depressurization when pzr level is greater than 90% during the performance of ES-04.
	X	X	X	21. Describe the two limitations which apply to the implementation of ES-0.0, Rediagnosis.
	X	X	X	22. [Describe the reason(s) for terminating an uncontrolled cooldown after a reactor trip. (SOER 94-001, Rec. 4b)]
	X	X	X	23. Cite the immediate action steps including RNO for E-0.
	X	X	X	24. State the entry conditions for E-0.
	X	X	X	25. Explain the basis for ensuring AFW Pump Operation in Appendix A of E-0.
	X	X	X	26. Given a set of plant conditions, use E-0, ES-0.0, ES-0.1, ES-0.2, ES-0.3, ES0.4 and the Critical Safety Function Status Trees to correctly diagnose and implement: Action Steps, RNOs, Foldout Pages, Notes and Cautions.

2. 008 AG2.4.47 002
Given the following:

- Unit 1 is operating at 100% power with Pressurizer Heater Group C energized.
- Pressurizer PORV 68-340A begins leaking through.

Assuming **NO** operator action, which ONE of the following identifies...

(1) how 1-TI-68-318, PZR SURGE LINE TEMP, indication will respond

and

(2) the leakage rate when the associated Tech Spec Operation Leakage limit would be exceeded?

	(1)	(2)
A.	Rise	>1.0 gpm
B.	Rise	>10 gpm
C.	Lower	>1.0 gpm
D.	Lower	>10 gpm

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because another pressurizer parameter (the PORV tailpipe temperature indication) will rise as the valve begins leaking through and the RCS Operational Unidentified Leakage limit in accordance with Tech Specs is 1.0 gpm.*

B. *Incorrect, Plausible because another pressurizer parameter (the PORV tailpipe temperature indication) will rise as the valve begins leaking through and the RCS Operational Identified Leakage limit in accordance with Tech Specs being 10 gpm is correct.*

C. *Incorrect, Plausible because the Surge Line temperature dropping due to the PORV leaking is correct and the RCS Operational Unidentified Leakage limit in accordance with Tech Specs is 1.0 gpm.*

D. *Correct, with the PORV leaking, inventory will be lost in the pressurizer resulting in a flow from the RCS hot leg into the pressurizer. This replaces the outflow from the pressurizer. The Hot Leg temperature is lower than the temperature in the pressurizer which results in the Surge Line temperature dropping and the RCS Operational Identified Leakage limit in accordance with Tech Specs is 10 gpm.*

Question Number: 2

Tier: 1 **Group:** 1

K/A: 008 AG2.4.47
Vapor Space LOCA
Emergency Procedures / Plan
Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Importance Rating: 4.2 / 4.2

10 CFR Part 55: 41.10 / 43.5 / 45.12

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires the ability to understand how the Pressurizer Surge LIne Temperature indicator will trend during a vapor space leak and also know that this type of leakage is classified as Identified leakage; then be able to recognize the leakage rate when the Tech Spec Operational Leakage would be exceeded.

Technical Reference: SOI-68.03, Pressurizer Pressure and Spray Control System, Revision 0021

Proposed references to be provided: None

Learning Objective: 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	
Modified Bank	
Bank	X

Question History:

Comments:

3.0 PRECAUTIONS AND LIMITATIONS

A. If ΔT between Pzr-Spray is greater than 100°F, Spray use is restricted under the conditions in 1-SI-0-8 (To avoid cyclic stress, spray flow is **NOT** normally initiated if Pzr-Spray ΔT is greater than 100°F). Maximum Pzr-Spray ΔT is 320°F.

B. Aux Spray is to be initiated slowly by gradually raising charging flow to prevent thermal shock to line and nozzles.

C. To prevent Spray lines and Surge line from cooling below operating temp, and prevent spray line damage due to pressurized thermal shock, flow must be maintained in each Pzr spray line.

D. Maximum Pzr heatup rate is 100°F in any 1 hour. Maximum Pzr cooldown rate is 200°F in any 1 hour. Nil-Ductility may occur if limits are violated.

E. Pzr heaters are **NOT** energized unless Pzr level is greater than 17%. Pzr heaters are uncovered below 17%. Extensive damage will occur to heaters if energized while uncovered.

F. Work in a Radiological Control Area (RCA) requires the use of existing RWPs, and may require additional ALARA Preplans. Failure to follow posted Radiation Protection requirements can cause unnecessary radiation exposure. RADPRO should be notified of work having the potential to change radiological conditions.

G. Instrument Maintenance (IM) should be notified to ensure required instruments will be in service as necessary to support system operation.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE; Distractor

c. 10 gpm identified LEAKAGE; and Correct Answer

d. 150 gallons per day primary-to-secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary-to-secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary-to-secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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.
X	X	X	X	12. Identify the program setpoints, and describe any automatic actions relative to the pressurizer level program.

3. 015 AK3.03 003

Given the following:

- Unit 1 is operating at 18% power.
- RCP #2 motor winding temperatures have increased to 304°F.

Which ONE of the following identifies the action required and the reason for the sequence?

A. Trip the reactor and then stop RCP #2 because the unit will not be operated on 3 loop operation in Modes 1 or 2 due to conservative industry operating principles.

B. Trip the reactor and then stop RCP #2 because Tech Spec LCO 3.4.4 does not allow 3 loop operation with the Unit in Mode 1.

C. Stop RCP #2 and then trip the reactor because the challenge is to the pump and there is no challenge to DNB during 3-loop operation at the current reactor power level.

D. Stop RCP #2 and then trip the reactor because the challenge is to the pump and stopping the pump does not challenge the reactor trip system at the current reactor power level.

DISTRACTOR ANALYSIS:

A. Correct, the required sequence when an RCP is required to be stopped during Mode 1 or 2 operation is to trip the reactor first then stop the RCP. The reason as identified in AOI-24 is to conform to conservative industry operating principles.

B. Incorrect, Plausible because tripping the reactor prior to stopping the RCP is correct and there is a Tech Spec limitation on 3-loop operation and a violation would occur if the time limit was not met.

C. Incorrect, Plausible because 'Immediate Trip' criteria is met for RCP #2 and there is no DNB concern during 3-loop operation at the current low power level if the RCP was stopped before the reactor was tripped.

D. Incorrect, Plausible because 'Immediate Trip' criteria is met for RCP #2 and with the reactor level below P8 (48%) there would be no challenge to the reactor protection system if the RCP was stopped before the reactor was tripped.

Question Number: 3

Tier: 1 **Group:** 1

K/A: 015 AK3.03
Reactor Coolant Pump (RCP) Malfunctions
Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions:
Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction

Importance Rating: 3.7 / 4.0

10 CFR Part 55: 41.5, 41.10 / 45.6 / 45.13

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires knowledge of the reason for the sequence of tripping the reactor prior to stopping the RCP when the RCP is required to be stopped due to a malfunction.

Technical Reference: AOI-24, RCP Malfunctions During Pump Operation, Revision 0029

Proposed references to be provided: None

Learning Objective: 3-OT-AOI2400
10. Given a set of plant conditions, use AOI-24 to correctly:
    a. Recognize Entry Conditions.
    b. Identify Required Actions.
    c. Respond to Contingencies (RNO).
    d. Observe and Interpret Cautions and Notes.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: WBN bank question AOI0500.01020 modified

Comments:

Step	Action/Expected Response	Response Not Obtained

3.2 RCP Tripped Or Shutdown Required

NOTE 1 Malfunctions addressed by this procedure require RCP shutdown as soon as possible.

NOTE 2 Exceeding any of the limits listed on Attachment 2 of this procedure will require immediate shutdown of the affected RCP.

NOTE 3 Malfunctions resulting in high #1 seal leakoff will require closing #1 seal return FCV following RCP coastdown

1. **CHECK** RCP tripped

   MONITOR RCP immediate shutdown Criteria:

   - **REFER TO** ATTACHMENT 2, RCP Immediate Shutdown Criteria.

   1) **IF** RCP immediate shutdown required,
      THEN
      ** **GO TO** Step 2.

   2) **IF** RCP immediate shutdown NOT required,
      THEN
      ** **GO TO** Step 9

2. **CHECK** unit in Mode 1 or 2

   ** **GO TO** Step 4.

NOTE Control room staff should brief on Steps 3 through 6 prior to tripping the reactor. This ensures that the affected RCP is stopped and that appropriate actions are taken when unit is removed from service.

Step	Action/Expected Response	Response Not Obtained
3.	**TRIP** the reactor, and **GO TO** E-0, Reactor Trip or Safety Injection, WHILE continuing with this instruction.	
4.	**STOP** and **LOCK OUT** affected RCP(s).	
5.	**IF** in Mode 3, **THEN** **CHECK** any RCP Running	****GO TO** ES-0.2, Natural Circulation Cooldown, WHILE continuing with this instruction
	CAUTION If the RCP seal return flow control valve (FCV) is NOT closed within 5 minutes of stopping the RCP with excessive leakoff, seal damage may occur.	
6.	**MONITOR** RCP seal leakoff less than 6 gpm per pump: • 1-FR-62-24 [RCP 1 & 2] • 1-FR-62-50 [RCP 3 & 4] • ICS "RCP DATA" • ICS "RCP SEALS"	**WHEN** the RCP has coasted down (between 3 and 5 minutes), **THEN** **CLOSE** affected RCP seal return FCV: • 1-FCV-62-9 [RCP 1] • 1-FCV-62-22 [RCP 2] • 1-FCV-62-35 [RCP 3] • 1-FCV-62-48 [RCP 4]
7.	**CHECK** RCPs 1 and 2 running.	**CLOSE** affected loop's pressurizer spray valve.
8.	**GO TO** Step 15.	

4.0 DISCUSSION

Three general classes of conditions are addressed by AOI-24:

1. Conditions requiring immediate pump shutdown or pump trip.
2. Conditions requiring pump shutdown within 8 hours or following an orderly reactor shutdown as recommended by Plant Management.
3. Conditions requiring pump shutdown As Soon As Possible.

Condition 1 above, which includes high #1 seal leakoff flow that results in rising RCP lower bearing or #1 seal outlet temperature, requires an immediate reactor trip if in Mode 1 or 2. The RCP is stopped and the #1 seal return FCV is closed between 3 and 5 minutes of the RCP being stopped. Any other RCP trip setpoint being exceeded likewise requires an immediate reactor trip (Mode 1 or 2) and pump shutdown.

Condition 2 above are events requiring a RCP shutdown within 8 hours or following an orderly reactor shutdown as recommended by Plant Management include high or low #1 seal leakoff flow with stable RCP lower bearing and #1 seal leakoff temperatures. Gradual trends should be recognized and contingency actions developed by the plant staff prior to reaching any action level in this procedure. Recommendations for shutdown due to adverse RCP operating trends will be performed in the context of normal plant operating procedures even though the initial evaluation may have been initiated through the performance of this AOI.

Condition 3 above include failures that require the RCP to be removed from service as soon as possible in order to limit further pump degradation. These conditions include:

- #1 seal leakoff flow low with #2 seal leakoff high as evidenced by high standpipe level.
- #1 seal leakoff flow <6.0 gpm with RCP lower bearing or #1 seal leakoff temperature >180°F and rising.
- #2 seal leakoff flow determined to be >1.1 gpm.
- Standpipe level high with rising vibration (#2 seal failure).
- Standpipe level low with rising vibration (#3 seal failure).
- #3 seal leakoff high with limiting cntmt conditions (rad, temp, moisture).

Step	Action/Expected Response	Response Not Obtained

A damaged No. 1 seal may produce debris or leak at a high enough rate to exceed the injection rate and introduce unfiltered, high temperature reactor coolant into the seals. In the presence of debris, the ability of the No. 2 seal to survive and function as a backup for the No. 1 seal is more favorable under static conditions. Thus, it is preferred to have pump shutdown precede No. 1 seal leakoff isolation. RCP coastdown time is normally less than 2 minutes after tripping the pump. Actual coastdown time is dependent on the number of RCPs operating. Assuming 3 minutes for coastdown, the No. 1 seal leakoff is to be isolated between 3 and 5 minutes after tripping the RCP.

Failures involving running RCP seals are covered in this instruction. Seal problems before starting RCP are covered in SOI-68.02.

In Modes 1 and 2, four RCPs are required to be in operation to meet the safety analysis criteria for DNB. Three-loop operation is therefore an abnormal operating mode and is limited in accordance with Tech Specs. The LCO action allows 6 hours to place the plant in Mode 3 following the loss of one RCP at power. It should be emphasized, however, that the unit will NOT be operated on 3 loop operation in Modes 1 or 2 due to conservative industry operating principles. Plant management must authorize any RCP restart and return to power operation.

5.0 REFERENCES:

5.1 Performance

A. E-0, Reactor Trip or Safety Injection

B. ES-0.2, Natural Circulation Cooldown

C. GO-4, Normal Power Operation

D. GO-5, Unit Shutdown From 30% Reactor Power to Hot Standby

5.2 Developmental

A. WBN System Description Manual, N3-68-4001, Rx Coolant System

B. WBN-VTM-W120-660, Vendor Technical Manual for Rx Coolant Pumps

C. SOI-68.02, Reactor Coolant Pumps

Attachment 2
(Page 1 of 1)

RCP IMMEDIATE SHUTDOWN CRITERIA

NOTE Exceeding any of the following setpoints will require an immediate pump shutdown. Operating limits can be found in SOI 68.02. This list is immediate shutdown criteria only.

A. Shaft vibration greater than 20 mils or 15 mils with a rate of rise equal to 1 mil/hr (alarm at 15 mils). [Indicators located on 0-PNL-52-R139, Aux Inst Rm.]

B. Frame vibration greater than 5 mils or 3 mils with a rate of rise of 0.2 mil/hr. [Readings taken by Maint. at Aux Bldg L-Panels, el.737.]

C. Motor windings temp greater than 302°F.

D. Motor bearing temp greater than 195°F.

E. Pump bearing temp greater than 225°F.

F. Loss of CCS to oil coolers for greater than 10 minutes.

G. No. 1 seal outlet temp greater than 225°F.

H. No. 1 seal flow HIGH with rising pump bearing or #1 seal leakoff temperatures.

I. No. 1 seal ΔP less than or equal to 200 psid.

WBN BANK QUESTION

AOI0500.01 020

Given the following conditions:

- Unit 1 is at 60% power.
- Loop 1 RCP shaft vibration is at 25 mils.

Which of the following statements is correct per AOI-5, "Unscheduled Removal of One RCP", regarding the sequence of actions?

a. RCP should be tripped prior to the reactor trip, to minimize pump damage.

b. The reactor should be tripped prior to tripping the RCP, to prevent pressurizer level from dropping below 17%.

c. RCP should be tripped prior to the reactor trip, to prevent Reactor Coolant Bus voltage from dropping and tripping additional RCPs.

d. ✔ The reactor should be tripped prior to tripping the RCP, to prevent an automatic trip and an unnecessary challenge to a safety system.

I. PROGRAM

Watts Bar Operator Training

II. COURSE

A. License Training
B. Non-license Training

III. TITLE

AOI-24, RCP Seal Malfunctions during Pump Operation

IV. LENGTH OF LESSON

License Training 1 Hour
Non-License Training 1 Hour

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	01. Demonstrate knowledge of the Purpose/Goal of AOI-24.
X	X	X	X	02. Indicate how much leakage could result from total seal failure at full RCS pressure, describe action to mitigate the consequences of this leakage.
	X	X	X	03. Identify Alarms associated with RCP seal malfunctions.
X	X	X	X	04. List 4 Indications of RCP seal malfunctions per AOI-24.
				05. Deleted Objective
X	X	X	X	06. Assuming injection water to an RCP is lost and pump lower bearing temperature is above alarm setpoint, state what precautionary measure must be observed when restoring seal injection.
X	X	X	X	07. Given an RCP seal Standpipe Hi level alarm, describe how to calculate #2 seal leak-off rate from the Radwaste Panel.
	X	X	X	08. Explain why RCP Seal Standpipe Level Hi/Lo alarm comes in on #3 Seal leak-off Hi flow.

V. TRAINING OBJECTIVES (continued)

AUO	RO	SRO	STA	
	X	X	X	09. Identify the parameters listed in AOI-24 that require the RCP to be shutdown
	X	X	X	10. Given a set of plant conditions, use AOI-24 to correctly: a. Recognize Entry Conditions. b. Identify Required Actions. c. Respond to Contingencies (RNO). d. Observe and Interpret Cautions and Notes.
	X	X		11. Given a set of conditions, determine if RCP shutdown is required using AOI-24, Attachment 2.
	X	X		12. Describe basic Operator Actions to shut down an RCP.

VI. TRAINING AIDS

A. Marker Board & Markers
B. Multimedia/Overhead projector(s)

VII. MATERIALS

Attachment(s):
Attachment 1 - RCP Seal Flows
Attachment 2 - AOI-24, RCP Seal Malfunctions during Pump Operation (Latest Rev)
Attachment 3 - SOER 82-5, Reactor Coolant Pump Seal Failure

VIII. REFERENCES

ENGINEERING SYSTEM DESCRIPTION(S)		
Number	Title	Rev.
N3-62-4001	Chemical and Volume Control System	29
N3-68-4001	Reactor Coolant System	28
WBN FSAR		
Section	**Title**	**Amend.**
9.3.4	Chemical and Volume Control System	NA
Chapters 3, 5, 6, 7, 9, 15	Reactor Coolant System	NA
DRAWINGS		
Plant Drawings	**Title**	**Rev.**
	None	

4. 026 AA2.03 004
Given the following:

- Unit 1 is operating at 100% power.
- The C-S CCS Pump trips while supplying the B Train CCS.
- Operators have not taken any action.

Which ONE of the following identifies...

(1) how the 2B-B CCS pump will be started

and

(2) if the 1B-B CCS pump is used to supply B Train CCS, why its suction valve must be re-aligned?

A. (1) Must be manually started
(2) To protect train separation within the CCS surge tank

B. (1) Must be manually started
(2) To ensure a suction path to the 1B-B CCS pump

C. (1) Will automatically start on low B Train CCS pressure
(2) To protect train separation within the CCS surge tank

D. (1) Will automatically start on low B Train CCS pressure
(2) To ensure a suction path to the 1B-B CCS pump

DISTRACTOR ANALYSIS:

A. *Correct, The 2B-B CCS Pump does not automatically start on low header pressure and will need to be manually started, and the reason for re-aligning the 1B-B Pump's suction is to protect train separation (see SOI-70.01, "Component Cooling Water (CCS) System").*

B. *Incorrect, Plausible because the 2B-B CCS Pump does not automatically start on low header pressure and will need to be manually started. Also plausible, because there are pumps that must have their suction valves re-aligned to protect their suction source. In this case, the 1B-B pump has its suction lined up to the 1A side of the surge tank and would have a suction source, however it would connect the 1A and 1B CCS headers if it was not re-aligned.*

C. *Incorrect, Plausible because, although the 2B-B pump does not automatically start, the 1B-B CCS pump would auto start if the 1A-A pump failed to provide pressure. Also, the reason for re-aligning the 1B-B Pump's suction to protect train separation is correct (see SOI-70.01, "Component Cooling Water (CCS) System").*

D. *Incorrect, Plausible because, although the 2B-B pump does not automatically start, the 1B-B CCS pump would auto start if the 1A-A pump failed to provide pressure. Also plausible, because there are pumps that must have their suction valves re-aligned to protect their suction source. In this case, the 1B-B pump has its suction lined up to the 1A side of the surge tank and would have a suction source, however it would connect the 1A and 1B CCS headers if it was not re-aligned.*

Question Number: 4

Tier: 1 **Group:** 1

K/A: 026 AA2.03
Loss of Component Cooling Water (CCW)
Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water:
The valve lineups necessary to restart the CCWs while bypassing the portion of the system causing the abnormal condition.

Importance Rating: 2.6 / 2.9

10 CFR Part 55: 43.5 / 45.13

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires knowledge of the valve lineup for the 1B-B CCS Pump suction required for that pump to be used to bypass a failed C-S CCS pump.

Technical Reference: SOI-70.01, Component Cooling Water (CCS) System, Revision 0068
1-45W760-70-1 R25
2-45W760-70-1 R2 (as designed)

Proposed references to be provided: None

Learning Objective: 3-OT-SYS070A
03. Describe the CCS pumps, include power supply, pump type, capacity, lubrication, and logic.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: WBN Bank Question SYS070A.15 004 modified for use on the WBN 10/22011 NRC exam.

Comments:

Date_____ **INITIALS**

8.0 INFREQUENT OPERATION

8.1 Align Pump 1B-B to Supply Header 1B

CAUTION

Performance of this section with RCPs in service results in an increased vulnerability to single point failure. The loss of the 1A-A CCS Pump will result in an impact on the Unit 1 RCP oil coolers. Refer to AOI-24, RCP MALFUNCTIONS DURING PUMP OPERATION, for RCP Shutdown Criteria.

NOTE

Appendix R breakers and valves which are repositioned should be tracked per OR-14.10.

[1] **ENSURE** applicable items have been evaluated:

- PRA Risk _____
- Reactor Trip Risk _____
- Applicable Tech Spec LCOs _____

[2] **ENSURE** Pump 1B-B, **NOT** in service. _____

[3] **PLACE** 1-HS-70-38A, CCS PMP 1B-B, in the STOP/PULL-TO-LOCK position. _____

_____
CV

Date_____ INITIALS

8.1 Align Pump 1B-B to Supply Header 1B (continued)

[4] **PLACE** the following breakers to ON:

NOMENCLATURE	LOC	POSITION	UNID	PERF INITIAL	
480V Rx MOV Bd 1B2-B					
CCS PMP 1A/1B TO C-S DISCH XTIE (FCV-70-26)	C/14D	ON	1-BKR-70-26		CV
CCS PMP 1A/1B TO C-S DISCH XTIE (FCV-70-27)	C/17B	ON	1-BKR-70-27		CV
CCS PMP 1A/1B SUCT XTIE (1-FCV-70-34)	C/14E	ON	1-BKR-70-34		CV
CCS PMP 1A/1B TO C-S SUCT XTIE (1-FCV-70-64)	C/15D	ON	1-BKR-70-64		CV
CCS PMP 1A/1B TO C-S SUCT XTIE (1-FCV-70-74)	C/16A	ON	1-BKR-70-74		CV

CAUTION

Step 8.1[5] must be completed prior to continuing to prevent CCS headers 1-A and 1-B from being tied together.

NOTE

Independent Verification of Step 8.1[5] and 8.1[6] may be delayed until prior to Step 8.1[7].

[5] **CLOSE** the following valves:

NOMENCLATURE	LOC	POSITION	UNID	PERF INITIAL	VERIF INITIAL
CCS PMPS 1A & 1B SUCT XTIE	0-M-27B	CLOSED	1-HS-70-34A		IV
CCS PUMP 1A-A/1B-B DISCHARGE CROSSTIE	A3T/725	CLOSED	1-ISV-70-507		IV

Date_____ INITIALS

8.1 Align Pump 1B-B to Supply Header 1B (continued)

[6] **OPEN** the following valves:

NOMENCLATURE	LOC	POSITION	UNID	PERF INITIAL	VERIF INITIAL
CCS PMP 1B TO C-S DISCH XTIE	0-M-27B	OPEN	1-HS-70-26A		IV
CCS PMP 1B TO C-S DISCH XTIE	0-M-27B	OPEN	1-HS-70-27A		IV
CCS PMP 1B TO C-S SUCT XTIE	0-M-27B	OPEN	1-HS-70-64A		IV
CCS PMP 1B TO C-S SUCT XTIE	0-M-27B	OPEN	1-HS-70-74A		IV

[7] **PLACE** the following breakers to OFF:

NOMENCLATURE	LOC	POSITION	UNID	PERF INITIAL	VERIF INITIAL
480V Rx MOV Bd 1B2-B					
CCS PMP 1A/1B TO C-S DISCH XTIE (FCV-70-26)	C/14D	LOCKED-OFF	1-BKR-70-26		CV
CCS PMP 1A/1B TO C-S DISCH XTIE (FCV-70-27)	C/17B	LOCKED-OFF	1-BKR-70-27		CV
CCS PMP 1A/1B SUCT XTIE (1-FCV-70-34)	C/14E	LOCKED-OFF	1-BKR-70-34		CV
CCS PMP 1A/1B TO C-S SUCT XTIE (1-FCV-70-64)	C/15D	LOCKED-OFF	1-BKR-70-64		CV
CCS PMP 1A/1B TO C-S SUCT XTIE (1-FCV-70-74)	C/16A	LOCKED-OFF	1-BKR-70-74		CV

[8] **UNLOCK and THROTTLE** 1-ISV-70-505B, CCS PUMP 1B-B DISCHARGE ISOLATION to 25% OPEN. _____

CAUTION

CCS Pump damage may occur below 900 gpm per pump.

NOTE

Pump starting guidelines are in GOI-7.

[9] **ENSURE** a sufficient flow path to provide greater than the minimum flow allowed. _____

Date_____ **INITIALS**

8.1 Align Pump 1B-B to Supply Header 1B (continued)

[10] **START** CCS PMP 1B-B, with 1-HS-70-38A. _____

NOTE

1B Supply Header flow can be verified locally on 0-FI-70-201, [0-PNL-276-L643, A5-S/713].

[11] **OPEN SLOWLY** 1-ISV-70-505B, CCS PUMP 1B-B DISCHARGE ISOLATION, **THEN**

VERIFY flow between 900-6800 gpm. _____

[12] **LOCK OPEN** 1-ISV-70-505B, CCS PUMP 1B-B DISCHARGE ISOLATION. _____

_____
CV

[13] **ENSURE** started pump 480V ACB Closing Spring is CHARGED: [C.2]

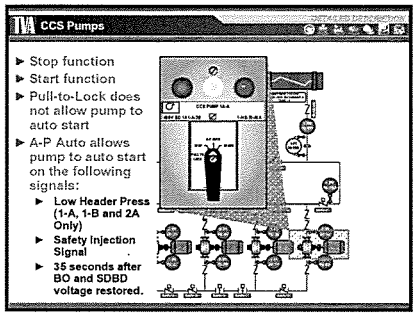
NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL
		480V SD Bd 1B1-B		
CCS PUMP 1B-B (1-PMP-70-38)	C/3C	CLOSING SPRING CHARGED	1-BKR-70-38	

[14] **REFER TO** Section 6.1 for temperature control of loop(s) placed in service. _____

End of Section

OBJECTIVE 4

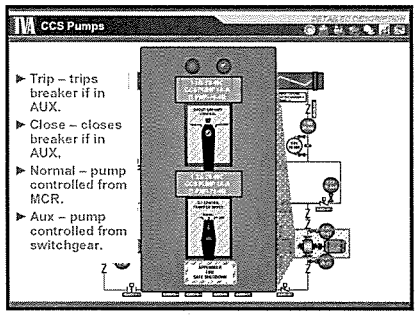
SLIDE 29



2. Discuss the MCR controls associated with the CCS Pumps.

- Stop function
- Start function
- Pull-to-Lock does not allow pump to auto start
- A-P Auto allows pump to auto start on the following signals:
  - Low Header Press (1-A, 1-B and 2A Only)
  - Safety Injection Signal
  - 35 seconds after BO and SDBD voltage restored.

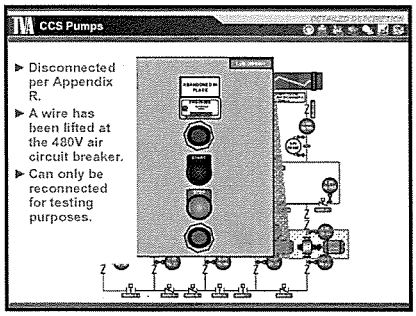
SLIDE 30



3. Discuss the switchgear controls associated with the CCS Pumps.

- Trip – trips breaker if in AUX.
- Close – closes breaker if in AUX,
- Normal – pump controlled from MCR.
- Aux – pump controlled from switchgear.

SLIDE 31



4. Discuss the local controls associated with the CCS Pumps.

- Disconnected per Appendix R.
- A wire has been lifted at the 480V air circuit breaker.
- Can only be reconnected for testing purposes.

WBN Bank Question

1. SYS070A.15 004
Given the following plant conditions:

- The Unit is at 99% power.
- CCS is in a NORMAL alignment with 1A-A, 2A-A, and C-S pump running.
- The Spent Fuel Pool Cooling System is aligned to Unit 2 CCS.
- C-S CCS pump suffers a catastrophic shaft failure.

Which of the following describes;

the effect of this failure,
and,
compensatory measures (if any) to support operations until the C-S CCS pump can be returned to service?

	Effect	Compensatory Measures
a.	1B-B CCS pump AUTO starts to supply Train B CCS cooling.	No compensatory measures are required
b.	2B-B CCS pump AUTO starts to supply Train B CCS cooling.	Spent Fuel Pit cooling must be swapped to Unit 1.
c.	2A-A CCS pump will be supplying the Train B CCS cooling	1B-B pump must be aligned to supply the Unit 1 Train B CCS
d. ✓	Cooling to Train B is lost until the 2B-B CCS pump is MANUALLY started	1B-B pump must be aligned to supply the Unit 1 Train B CCS

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training

B. NOTP

C. License Operator Requal

D AUO Requal

III. TITLE

Component Cooling System

IV. LENGTH OF LESSON

A. Licensed Training 1.5 hours

B. NOPT 3.0 hours

License Requalification and NAUO Requalification times will be determined after objectives are identified

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. State the design basis of the Component Cooling Water System (CCS) in accordance with FSAR section 9.2.2.
X	X	X	X	2. Sketch a basic drawing of the CCS, include all pumps, major heat exchangers, and blocks showing major uses of CCS.
X	X	X	X	3. Describe the CCS pumps, include power supply, pump type, capacity, lubrication, and logic.
X	X	X	X	4. Explain the logic associated with each valve/pump control in the CCS.
X	X	X	X	5. Explain the operation, purpose, and location of the C-S CCS Pump power supply transfer switch.
X	X	X	X	6. Describe the CCS heat exchangers, include cooling medium.

5. 027 AK3.02 005
Given the following:

- Unit 1 is operating at 100% power.
- 1-XS-68-340D, PZR PRESS CONTROL CHANNEL SELECT, is selected to PT-68-340 & 334.
- 1-XS-68-340B, PZR PRESS RECORDER CHANNEL SELECT, is selected to PT-68-334.
- 1-PI-68-334, PRESSURIZER PRESSURE, fails to the bottom of the scale due to transmitter failure.
- The other three pressurizer pressure instruments indicate:

  1-PI-68-340A - 2335 psig
  1-PI-68-323 - 2320 psig
  1-PI-68-322 - 2330 psig

- Operators are performing AOI-18, "Malfunction of Pressurizer pressure Control System," and have placed 1-XS-68-340D, PZR PRESS CONTROL CHANNEL SELECT, to PT-68-340 B322.

Which ONE of the following identifies how 1-XS-68-340B, PZR PRESS RECORDER CHANNEL SELECT, will be positioned and the reason why?

A. PT-68-323 because it is the channel with the lowest reading.

B. PT-68-322 because it is the channel selected to replace 1-PT-68-334.

C. PT-68-323 or PT-68-322 because neither is the channel selected for control.

D. Any position other than PT-68-334 because each is an operable channel.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because selecting the channel with the lowest reading would provide trend for the margin to DNB pressure limit.*

B. *Incorrect, Plausible because PT-68-322 is the pressure channel replacing the failed channel PT-68-334.*

C. *Incorrect, Plausible because PT-68-334 was not the controlling channel prior to its failure and these two positions are the only choices for selection of a non-controlling channel.*

D. *Correct, 1-XS-68-340B can be selected to any channel other than PT-334 because each of the other three are operable channels.*

Question Number: 5

Tier: 1 **Group:** 1

K/A: 027 AK3.02
Pressurizer Pressure Control System (PZR PCS) Malfunction
Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions:
Verification of alternate transmitter and/or plant computer prior to shifting flow chart transmitters

Importance Rating: 2.9* / 3.0

10 CFR Part 55: 41.5,41.10 / 45.6 / 45.13

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires knowledge of the reason for verifying the status of alternate transmitters prior to positioning the input to the chart recorder following the failure of the transmitter previously selected.

Technical Reference: AOI-18, Malfunction of Pressurizer Pressure Control System, Revision 0023

Proposed references to be provided: None

Learning Objective: 3-OT-AOI1800
7. Demonstrate ability/knowledge of AOI, to correctly:
   b. Respond to Action steps.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	X
Modified Bank	
Bank	

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

Comments:


CONTROL XFER
PZR PRESS CONTROL
CHANNEL SELECT
1-XS-68-340D
CONTROL
FROM
PT-68-340 & 334
PT-68-340
B322
PT-68-334
B323
PZR LEVEL CONTROL
CHANNEL SELECT
1-XS-68-339E
CONTROL
FROM
LI-68-339 & 335
LI-68-339
B 320
LI-68-335
B 320


RECORDER XFER
PZR PRESS RECORDER
CHANNEL SELECT
1-XS-68-340B
RECORD
PT-68-334
PT-68-340
PT-68-323
PT-68-322
PZR LEVEL RECORDER
CHANNEL SELECT
1-XS-68-339B
RECORD
LT-68-335
LT-68-339
LT-68-320

Step	Action/Expected Response	Response Not Obtained

3.1 Pressure Transient Due to Instrument or Controller Malfunction

NOTE Step 1 is an **IMMEDIATE ACTION** step.

1. **CHECK** PZR pressure STABLE or TRENDING to desired pressure:

   PLACE PZR master controller 1-PIC-68-340A in MANUAL and

   RESTORE press to normal.

   IF PZR pressure drop due failed PORV/Safety or Spray valve, **THEN**
   ****GO TO** Section 3.2.

2. **ENSURE** 1-XS-68-340D selected to OPERABLE channels for control and backup:

   - PT-68-340 & 334

     OR

   - PT-68-334 & 323

     OR

   - PT-68-340 & 322

Step	Action/Expected Response	Response Not Obtained

3.1 Pressure Transient Due to Instrument or Controller Malfunction (continued)

3.	**CHECK** PZR press STABLE or RISING.	**IF** RCS temp DROPPING **THEN** **REFER TO** AOI-38, Main Steam or Feedwater Line Leak. **IF** VCT and PZR level DROPPING, **THEN** **REFER TO** AOI-6, Small Reactor Coolant System Leak. **IF** PZR press can NOT be maintained above 1970 psig, **THEN** 1) **TRIP** Rx and **INITIATE** SI. 2) ****GO TO** E-0, Reactor Trip or Safety Injection.
4.	**DETERMINE** if PZR heaters and sprays are operating properly:	
	a. **CHECK** master control controlling sprays and heaters.	a. **CONTROL** individual heaters and sprays to maintain pressure on program.
	b. **CHECK** PZR press greater than 2250 psig.	b. **WHEN** PZR press is greater than 2250 psig, **THEN** **PERFORM** Step 4.c.

Step	Action/Expected Response	Response Not Obtained

3.1 Pressure Transient Due to Instrument or Controller Malfunction (continued)

	c. **ENSURE** all PZR heaters OFF	
	d. **ENSURE** PZR sprays maintain RCS press on program.	d. **IF** normal sprays unavailable, **THEN** **PLACE** Aux Spray in service USING ATTACHMENT 1, AUX SPRAY. • **IF** sprays open due to failed master control, **THEN** **CLOSE** sprays
5.	**ENSURE** operable channel selected for recording with 1-XS-68-340B.	
6.	**ENSURE** TR-68-2A placed to operable channel using 1-XS-68-2B, ΔT RCDR TR-68-2A LOOP SELECT [1-M-5].	
7.	**CHECK** PZR level at or trending to PROGRAM.	**REFER TO** AOI-20, Malfunction of Pressurizer Level Control Channel.
8.	**NOTIFY** Work Control to remove any failed channel from service and repair any failed controllers.	

Step	Action/Expected Response	Response Not Obtained

3.1 Pressure Transient Due to Instrument or Controller Malfunction (continued)

9. **WHEN** pressurizer pressure stable and equipment status supports returned to normal,

   THEN

   ENSURE the following in AUTO:

   - PZR Master controller
   - PZR spray controllers
   - All heater groups

10. **REFER TO** the following Tech Specs:

    - 3.3.1, RTS Instrumentation
    - 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation
    - 3.3.4, Remote Shutdown System
    - 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
    - 3.4.3, RCS Pressure and Temperature (P/T) Limits
    - 3.4.9, Pressurizer
    - 3.4.10, Pressurizer Safety Valves
    - 3.4.11, Pressurizer Power -Operated Relief Valves

11. **INITIATE** repairs to failed equipment.

12. **RETURN TO** Instruction in effect.

End of Section

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. LICENSE TRAINING
B. LICENSE REQUAL

III. TITLE

AOI-18, MALFUNCTION OF PZR PRESSURE CONTROL SYS

IV. LENGTH OF LESSON

LICENSE TRAINING 2.0 HOURS

License Requalification time will be determined after objectives are identified.

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	1. Demonstrate knowledge of the Purpose/goal of AOI-18.
	X	X	X	2. Identify Auto Actions associated with dropping RCS pressure.
	X	X	X	3. Identify Auto Actions associated with rising RCS pressure.
				4. Deleted
	X	X	X	5. Explain the Operator Actions for dropping RCS pressure.
	X	X	X	6. Explain the Operator Actions for rising RCS pressure.
	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 and Cautions.

6. 029 EK2.06 006

Given the following conditions:

- Unit 1 is at 100% power with SSPS testing on Train A.
- Reactor Trip and Bypass breaker indicating light positions:

	RED	GREEN
RTA -	DARK	LIT
RTB -	LIT	DARK
BYA -	LIT	DARK
BYB -	DARK	DARK

- A transient occurs that requires the initiation of a manual reactor trip.
- The OAC momentarily places the reactor trip handswitches in the TRIP position but Reactor Trip and Bypass Breaker indications remain the same.

Which ONE of the following identifies...

(1) how the Shunt Trip coils (52SH) on the Reactor Trip and Bypass Breakers that are closed should have responded when the OAC placed the Reactor Trip handswitches to TRIP

and

(2) the minimum local action that will result in a reactor trip?

A. (1) The coils should have energized.
(2) Open either RTB or BYA.

B. (1) The coils should have energized.
(2) Open both RTB and BYA.

C. (1) The coils should have de-energized.
(2) Open either RTB or BYA.

D. (1) The coils should have de-energized.
(2) Open both RTB and BYA.

DISTRACTOR ANALYSIS:

A. *Correct, The shunt trip coils energize to open a reactor trip or bypass breaker and opening either RTB or BYA will result in a reactor trip.*

B. *Incorrect, Plausible because the shunt trip coils do energize to open a reactor trip, and if the breakers closed had been in a parallel alignment both of the of the breakers would have to be opened to cause the reactor to trip.*

C. *Incorrect, Plausible because the reactor trip breaker undervoltage coils do de-energize to cause a reactor trip and because opening either one of the breakers causing a reactor trip is correct.*

D. *Incorrect, Plausible because the reactor trip breaker undervoltage coils do de-energize to cause a reactor trip and if the breakers closed had been in a parallel alignment both of the of the breakers would have to be opened to cause the reactor to trip.*

Question Number: 6

Tier: 1 **Group** 1

K/A: 029 EK2.06
Anticipated Transient Without Scram (ATWS)
Knowledge of the interrelations between the and the following an ATWS:
Breakers, relays, and disconnects

Importance Rating: 2.9* / 3.1*

10 CFR Part 55: 41.7 / 45.7

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to demonstrate the knowledge of how the reactor trip breakers and the relays that control the breakers operation are related to the reactor trip system if malfunctions occur and actions required to disconnect the power to control rods locally during an ATWS event.

Technical Reference: 1-47W611-99-1, R7
FR-S.1, Nuclear Power Generation/ATWS Rev. 0020
N3-99-4003, Reactor Protection System, Revision 0021

Proposed references to be provided: None

Learning Objective: 3-OT-FRS0001
03. List from memory and in order the two Immediate Operator Actions for procedure FR-S.1, Nuclear Power Generation / ATWS, and discuss the basis for each action.
3-OT-SYS099A
10. Describe two ways by which the SSPS opens the Reactor Trip Breakers.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: Combination of 029EK2.06 1 from Vogtle 2009 exam (28) and a SQN Bank question 029 EK2.06 008 used on the SQN 2/2010 exam.

Comments:

3.1.1 Reactor Trip System (continued)

6. Reactor Trip on a Turbine Trip (Reference 7.5.18)

   The reactor trip on a turbine trip is actuated by two out of three logic from low autostop oil pressure signals or by closed signals from the turbine steam stop valves. This trip is blocked below the P-9 interlock setpoint.

   Separate routing is maintained for the four turbine trip channel sets. Each of three sets include the signal from low autostop oil pressure and the signal from closing of the steam stop valve. The fourth set consists of the signal from closing of the steam stop valve. The separation of these four channel sets is maintained from the sensors to the reactor protection system logic input cabinets. These channel routings meet the redundance and separation requirements identical to those for Class 1E circuits. (Reference 7.2.4).

   One of the design bases considered in the protection system is the possibility of an earthquake. Equipment generating turbine trip signals are located in nonseismic structures. The trip signals are provided by contacts which change position when the turbine autostop oil pressure goes below its setpoint or when the turbine stop valves close. With respect to these contacts, their functioning is unrelated to a seismic event in that they are anticipatory to other diverse parameters which cause reactor trip. The contacts are closed during plant operation and open to cause reactor trip when the turbine is tripped. No power is provided to the protection system from the contacts; they merely serve to interrupt power to cause reactor trip. This design functions in a deenergize-to-trip fashion to cause a plant trip if power is interrupted in the trip circuitry. This ensures that the protection system will in no way be degraded by this anticipatory trip because seismic design considerations do not form part of the design bases for anticipatory trip sensors. (The reactor protection system cabinets which receive the inputs from the anticipatory trip sensors are seismically qualified). The anticipatory trips thus meet IEEE 279-1971, including redundancy, separation, single failure, etc. Seismic qualification of the contact's sensors is not required.

7. Safety Injection Signal Actuation Trip

   The RPS is designed such that a reactor trip occurs when the Safety Injection System is actuated. This trip protects the core against a loss of primary or secondary coolant.

8. Manual Trip

   The manual trip consists of two switches with two outputs on each switch (Reference 7.5.18). One output is used to actuate the train A trip breaker, the other output actuates the train B trip breaker. Operating a manual trip switch removes the voltage from the undervoltage trip coil and energizes the reactor trip breaker shunt trip coil.

   There are no interlocks which can block this trip.

B. Specific Control and Protection Interactions[1]

Vogtle BANK QUESTION

28.

Given the following conditions:

- Both Reactor Trip Breaker red lights are illuminated.
- Both Reactor Trip Breaker green lights are extinguished.
- The OATC momentarily places the reactor trip handswitch in the TRIP position and Reactor Trip Breaker indications **DO NOT CHANGE**.

Which of the following choices describes the actions that **should have occurred** when the OATC placed the handswitch to TRIP?

A. The Undervoltage coils energized.

The Shunt coils energized.

B. The Undervoltage coils de-energized.

The Shunt coils energized.

C. The Undervoltage coils energized.

The Shunt coils de-energized.

D. The Undervoltage coils de-energized.

The Shunt coils de-energized.

Step	Action/Expected Response	Response Not Obtained

CAUTION RCPs should not be tripped UNTIL reactor power is less than 5%.

NOTE Steps 1 and 2 are **IMMEDIATE ACTION** steps.

1. **ENSURE** Reactor Trip:
   - Reactor trip and bypass breakers OPEN.
   - RPIs at bottom of scale.
   - Neutron flux DROPPING.

   Manually **TRIP** reactor.

   IF reactor will **NOT** trip,
   THEN
   INSERT control rods.

2. **ENSURE** Turbine Trip:
   - All turbine stop valves CLOSED.

   Manually **TRIP** turbine.

   IF turbine will **NOT** trip,
   THEN
   Manually runback turbine.

   IF turbine can **NOT** be run back,
   THEN
   CLOSE MSIVs and bypasses.

3. **CHECK** AFW pumps operation:

   Manually **START** pumps and open valves as necessary.

   a. Both MD AFW pumps RUNNING.

   b. TD AFW pump RUNNING.

   c. LCVs in AUTO or controlled in MANUAL.

SQN BANK QUESTION

029 EK2.06 008

Given the following:

- Unit 1 is at 100% power with Reactor Trip Breaker "A" open due to SSPS testing on Train "A."
- A transient occurs that initiates a reactor trip signal, but the reactor fails to trip from the main control room.

Which ONE of the following identifies;

(1) the required immediate operator actions listed in FR-S.1, "Nuclear Power Generation/ATWS"

and

(2) the minimum local actions that will result in a reactor trip?

A. (1) Maintain rod insertion at maximum rate and trip the turbine only.
(2) Open either RTB or BYA

B. (1) Maintain rod insertion at maximum rate and trip the turbine only.
(2) Open both RTB and BYA

C. (1) Maintain rod insertion at maximum rate and trip the turbine and start AFW.
(2) Open either RTB or BYA

D. (1) Maintain rod insertion at maximum rate and trip the turbine and start AFW.
(2) Open both RTB and BYA

I. **PROGRAM:**

Watts Bar Operator Training

II. **COURSE:**

A. License Training

B. License Operator Requalification

III. **TITLE:**

Function Restoration Guidelines FR-S.1 & S.2

IV. **LENGTH OF LESSON:**

License Training 1.5 Hours

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

V. **TRAINING OBJECTIVES:**

AUO	RO	SRO	STA	
	X	X	X	1. Given a set of plant conditions, use the subcriticality Status Tree to identify which FR-S procedure, if any, should be implemented.
	X	X	X	2. Apply the rules of usage (TI-12.04) and analyze plant conditions to identify any required procedure transitions in FR-S.1 and FR-S.2.
	X	X	X	3. List from memory and in order the two Immediate Operator Actions for procedure FR-S.1, Nuclear Power Generation/ ATWS, and discuss the basis for each action.
	X	X	X	4. Explain the basis for checking RCS pressure less than 2335 psig during the performance of FR-S.1, Nuclear Power Generation/ ATWS.
	X	X	X	5. Given a set of plant conditions, determine if a SI signal should be manually actuated.

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

AUO	RO	SRO	STA	
	X	X	X	6. Discuss the reason for placing the main feedwater valve mode selector switch (HS-3-45) to long cycle recirculation during the performance of FR-S.1.
	X	X	X	7. Given a set of plant conditions, determine if FR-S.1 Appendix A, Boron Dilution Path Isolation, or 1-SI-62-1, Uncontrolled Dilution Paths, should be performed.
	X	X	X	8. Given a set of plant conditions, including time after shutdown and IR and SR readings, determine if the excore instruments are responding normally and identify the actions required to manually reinstate the SR instruments.
	X	X	X	9. Given a set of plant conditions, use FR-S.1, FR-S.2 and the Critical Safety Function Status Trees to correctly diagnose and implement: Action Steps, RNOs, Notes and Cautions.
	X	X	X	10. Explain the purpose for and basis of each step in FR-S.1 and FR-S.2
	X	X	X	11. List the four major action categories of FR-S.1, Nuclear Power Generation/ATWS
	X	X	X	12. List the three major action categories of FR-S.2, Loss of Core Shutdown

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training

B. Non-License Training

III. TITLE

Reactor Protection System (RPS)

IV. LENGTH OF LESSON

A. License Training 6 hours
1. Non-License Training 6 hours

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. Explain the purpose of the Reactor Protection System.
X	X	X	X	2. Sketch a basic drawing of the Solid State Protection System.
	X	X	X	3. Describe the actions that take place when a reactor trip is generated at 100% power.
X	X	X	X	4. Identify the functions which rely on ESFAS for initiation.
X	X	X	X	5 Explain how the fire pumps would be affected by an SI signal.
X	X	X	X	6 Briefly describe the inputs to the SSPS.
X	X	X	X	7 Deleted.
X	X	X	X	8 Briefly discuss the input relays, Logic Section and Output Section of the SSPS.
X	X	X	X	9 Explain how the two trains of SSPS are interconnected.
	X	X	X	10 Describe the two ways by which the SSPS opens the Reactor Trip breakers.
X	X	X	X	11 Describe the four basic outputs of the SSPS.
X	X	X	X	12 Explain the purpose of the reactor trip bypass breakers and how their use is made fail safe.

AUO	RO	SRO	STA	
	X	X	X	13 Describe the causes of "General Warning" on SSPS
	X	X	X	14 Identify where "General Warning" indications can be found.
	X	X	X	15 Identify the SSPS equipment which can be tested.
	X	X	X	16 Describe operator actions prior to allowing testing of SSPS train.
X	X	X	X	17 Identify the Reactor trips and give setpoints and list logic required for the Reactor trips.
	X	X	X	18 Given the condition/status of the Reactor Protection system/component and the appropriate sections of Tech Specs, determine if operability requirements are met and what actions, if any, are required.
X	X	X	X	19 Deleted
X	X	X	X	20 Deleted
	X	X	X	21 Deleted

7. 038 EA2.15 007

The following conditions exist on Unit 1:

- A S/G tube rupture is in progress.
- The crew is backfilling at 125 gpm per ES-3.1, "Post-SGTR Cooldown Using Backfill."
- A 75 gpm and the 45 gpm orifices are in service.
- The TSC has requested the crew to reduce the backfill rate from 125 gpm to 75 gpm.

Which ONE of the following actions would the crew take to reduce the backfill rate?

A. Lower RCS pressure

B. Raise RCS pressure

C. Lower ruptured SG pressure

D. Raise ruptured SG pressure

DISTRACTOR ANALYSIS:

A. Incorrect, Plausible because changing the RCS pressure would change the backfill flow rate but lowering the RCS pressure is the opposite of the direction needed to lower the backfill flow rate.

B. Correct, To reduce the backfill flow rate, the S/G to RCS ΔP must be reduced. This can be done either by reducing the ruptured S/G pressure or increasing RCS pressure. Lowering ruptured S/G pressure is not a viable option since it would require dumping steam through the S/G PORVs resulting in a release.

C. Incorrect, Plausible because lowering ruptured S/G pressure would reduce the backfill flow rate but would also require dumping steam through the S/G PORVs creating a release.

D. Incorrect, Plausible because changing the ruptured S/G pressure would change the backfill rate but raising the steam generator pressure is the opposite of the direction needed to lower the backfill flow rate.

Question Number: 7

Tier: 1 **Group:** 1

K/A: 038 EA2.15
Steam Generator Tube Rupture (SGTR)
Ability to determine or interpret the following as they apply to a SGTR:
Pressure at which to maintain RCS during S/G cooldown

Importance Rating: 4.2 / 4.4

10 CFR Part 55: 43.5 / 45.13

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires ability to determine the requirements for establishing and maintaining RCS pressure during a post-SGTR event when using the preferred cooldown method

Technical Reference: ES-3.1, Post-SGTR Cooldown Using Backfill, Revision 0014
WOG E-3 Background Documents, Rev 2

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	X
Modified Bank	
Bank	

Question History: SQN bank question ES-3.1-B.5.H 001 for use on the WBN 10/2011 NRC exam

Comments:

generators will also be cooled via reverse heat transfer to the cooler reactor coolant flowing through the tubes. However, the insulated steam space should be nearly unaffected by this reverse heat transfer. If all RCPs are stopped, flow through the ruptured loops may stagnate. The coolant temperature in those loops, as well as the liquid region in the affected steam generator secondary side, will cool more slowly than the active loops.

Depressurization of the primary side is more complicated than usual since the primary side is coupled to the affected steam generator via the failed tube. RCS pressure will tend to follow the pressure in the ruptured steam generator and vice versa. As the size of the tube failures increases, these two pressures will follow each other more closely. Whatever one does to affect RCS pressure, e.g. turn on heaters, initiate pressurizer spray, or open a pressurizer PORV, will also affect pressure in the ruptured steam generators in the same way by transferring coolant between the primary system and ruptured steam generator.

This coupling has two major effects on RCS pressure control. First, since one is controlling pressure in both the primary system and ruptured steam generator and, therefore, a larger volume, normal RCS pressure control is less effective. More steam must be condensed or discharged from the pressurizer to decrease pressure. However, this only effects the rate at which pressure can be reduced. The capability to depressurize the RCS is preserved.

The most significant peculiarity of the primary and secondary coupling is the transfer of coolant with pressure changes. As RCS pressure is reduced below that of the ruptured steam generator, secondary-to-primary leakage increases the inventory in the primary system and decreases inventory in the ruptured steam generator. Pressurizer level increases and ruptured steam generator level decreases reflecting this transfer of coolant. Conversely, if RCS pressure is increased, or ruptured steam generator pressure decreases, primary-to-secondary leakage decreases RCS inventory and increases the inventory in the ruptured steam generator. Hence, unlike normal operations, pressurizer pressure control can also control pressurizer and ruptured steam generator inventories.

In the backfill recovery method, pressure in the RCS and ruptured steam generators is controlled by a combination of pressurizer pressure control and AFW flow. As RCS pressure is decreased below that of the ruptured steam generator using pressurizer spray or a PORV, liquid from the affected steam generator flows in to the primary, i.e., backflow occurs. This backflow decreases the energy of the steam space in the affected steam generator by expanding the steam bubble. Consequently, pressure in that steam generator also decreases. On the primary side, the secondary-to-primary leakage increases reactor coolant inventory which leads to an increase in pressurizer level. This increase in inventory is accommodated by coolant shrinkage and letdown flow so that pressurizer level can be managed and pressurizer pressure control can be maintained.

Level in the ruptured steam generator(s) will continue to decrease as the primary system is depressurized. If the water level is permitted to decrease too low, the layer of warm water which insulates the steam space from the cold water in the tube region could become too thin. In that case, energy transfer from the steam space to the cold water may depressurize the affected steam generator uncontrollably. This would cause primary-to-secondary leakage and may require SI reinitiation. To avoid this condition, it may be necessary to refill the affected steam generator using AFW to maintain level in the narrow range.

Introduction of cold AFW flow into an isolated steam generator can have many diverse effects. In one scenario, the cold water would lift the insulating layer of warm water and compress the steam bubble in the top of the steam generator. Pressure would increase and could eventually lift a relief valve. On the other hand, the cold AFW water could mix with the warmer water and condense steam as the water level increases. For this scenario, pressure would decrease. A combination of these scenarios is the most likely. The extent of AFW mixing depends on the location of the AFW line and water level in the steam generator. However, as long as the water level is maintained in the narrow range, the steam space is expected to be sufficiently insulated to prevent an uncontrolled depressurization. In either case, AFW flow will aid

STEP DESCRIPTION TABLE FOR ES-3.1 Step 9

STEP: Depressurize RCS To Backfill From Ruptured SG(s)

PURPOSE: To depressurize the RCS and ruptured SG to RHR system operating conditions

BASIS:

The ruptured steam generator will act like a large pressurizer to the RCS and inhibit RCS depressurization. In order to reduce primary pressure to maintain normal pressure/temperature limits and to establish RHR cooling, the RCS and ruptured SG must be depressurized concurrently. In the backfill method this is accomplished simply by decreasing RCS pressure using normal spray supported by auxiliary spray and pressurizer PORVs as backup. As RCS pressure is reduced below the pressure in the ruptured SG(s), secondary-to-primary leakage occurs which expands the steam bubble in the ruptured SG thereby reducing its pressure. This also promotes contact between the steam and cooler metal which further depressurizes the ruptured SG by condensing steam, as discussed in Step 7. Pressurizer level may increase as backflow in excess of coolant shrinkage accumulates in the pressurizer. If pressurizer level is near the upper limit of the control band, it may be difficult to stop backfill before exceeding the upper limit. Even if RCS depressurization is stopped, a primary-to-secondary differential pressure will exist and backfill will continue. Consequently, RCS makeup flow, letdown, and cooldown rate must be adjusted to maintain pressurizer level on span. Depressurization of the RCS should be controlled to maintain a stable pressurizer level, within the limits established by the previous step, as letdown flow and primary coolant shrinkage compensate for the backflow. This action will avoid a water solid condition, to maintain pressurizer pressure control and prevent water relief through the PORVs if sprays are unavailable. Pressure should also be maintained greater than saturation to avoid steam formation in the RCS and subsequent SI reinitiation.

Step	Action/Expected Response	Response Not Obtained

8. (continued)

   d. ** **REFER TO** GO-6, Unit Shutdown From Hot Standby to Cold Shutdown as necessary.

9. **MONITOR** Ruptured S/G NR level greater than 29% [39% ADV].

   REFILL Ruptured S/G NR level to 77% [59% ADV] with AFW.

   IF ANY of the following:

   - Ruptured S/G pressure drops in an uncontrolled manner,

     OR

   - Ruptured S/G pressure rises to 1100 psig,

     OR

   - Ruptured S/G pressure drops to 400 psig **AND**

     Ruptured S/G NR level greater than 29% [39% ADV], **THEN**

     STOP AFW flow to the Ruptured S/G.

10. **CONTROL** pzr level between 29% and 63% [47% and 58% ADV].

    - **ADJUST** charging and letdown as necessary.

Step	Action/Expected Response	Response Not Obtained

CAUTION Cycling of the pzr PORV should be minimized to improve PORV reliability.

NOTE

- If RCPs are **NOT** running, the upper head region may void during RCS depressurization. This will result in a rapidly rising pzr level.
- RCS depressurization rate should be controlled to maintain pzr level stable.

11. **INITIATE** backfilling Ruptured S/G into RCS:

a. **DEPRESSURIZE** RCS below Ruptured S/G pressure with normal pzr sprays.

a. **IF** letdown is in service, **THEN**

ALIGN aux spray USING Appendix B (ES-3.1), ALIGN AUX SPRAY.

IF letdown is **NOT** in service, **THEN**

USE one pzr PORV, **AND**

MONITOR the following:

- Vessel head void formation.
- Pzr level rise.
- PRT rupture.

b. **ENSURE** backfill flow:

- Pzr level stable or controlled.
- Ruptured S/G level dropping.

Step	Action/Expected Response	Response Not Obtained

12. **MAINTAIN** RCS depressurization criteria:

a. Pzr level less than 63% [58% ADV].

a. **PERFORM** the following:

- **EQUALIZE** RCS pressure and Ruptured S/G pressure.
- **LOWER** pzr level to less than 63% [58% ADV] before continuing depressurization.

b. Ruptured S/G NR level greater than 29% [39% ADV].

b. **PERFORM** the following:

1) **EQUALIZE** RCS pressure and Ruptured S/G pressure.
2) **CONTROL** AFW to raise Ruptured S/G NR level to greater than 77% [59% ADV] before continuing RCS depressurization.

Step continued on the next page

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 target incore temp prior 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.

8. 040 AK1.05 008
Given the following:

- Unit 1 is operating at 75% power steady-state conditions.
- A steam line leak occurs on a steam header outside the north valve room.

Which ONE of the following identifies...

(1) the time in core life that results in the largest addition of positive reactivity

and

(2) a resulting condition requiring a manual reactor trip in accordance with AOI-38, "Main Steam or Feedwater Leak?"

	Time in Core Life	Condition requiring Reactor Trip
A.	BOL	Final reactor power of 79%.
B.	BOL	Tavg/Tref cannot be maintained within 3°F.
C.	EOL	Final reactor power of 79%
D.	EOL	Tavg/Tref cannot be maintained within 3°F

DISTRACTOR ANALYSIS:

A. *Incorrect, the largest reactivity change would occur at EOL (not at BOL) due to the moderator temperature coefficient (MTC) change but unable to maintain Tavg/Tref mismatch within 5°F would require a reactor trip. Plausible because the reactivity effect does change over core life and steam leak greater than 3% requiring a reactor trip is correct.*

B. *Incorrect, the largest reactivity change would occur at EOL (not at BOL) due to the moderator temperature coefficient (MTC) change but steam leak greater than 3% power would require a reactor trip. Plausible because the reactivity effect does change over core life and a higher Tavg/Tref mismatch would require a reactor trip.*

C. *Correct, the largest reactivity change would occur at EOL due to the moderator temperature coefficient (MTC) being much larger than at BOL, and the condition requiring a reactor trip is a steam leak greater than 3% power.*

D. *Incorrect, the largest reactivity change would occur at EOL due to the moderator temperature coefficient (MTC) change but unable to maintain the Tavg/Tref mismatch within 5°F would require a reactor trip. Plausible because the reactivity effect does change over core life with EOL being correct and a higher Tavg/Tref mismatch would require a reactor trip.*

Question Number: 8

Tier: 1 **Group** 1

K/A: 040 AK1.05
Steam Line Rupture
Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture:
Reactivity effects of cooldown

Importance Rating: 4.1 / 4.4

10 CFR Part 55: 41.8 / 41.10 / 45.3

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires knowledge of how time in core life affects the reactivity change due to a steam leak and the operational limits on the change in unit conditions due to the steam leak and corresponding reactivity change.

Technical Reference: AOI-38, Main Steam or Feedwater Line Break, Revision 0010

Proposed references to be provided: None

Learning Objective: 3-OT-AOI3800
4. Given a set of plant conditions, use the AOI to correctly:
   a. Recognize Entry Conditions.
   b. Identify Required Actions.
   c. Respond to Contingencies (RNO).
   d. Observe and Interpret Cautions and Notes.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	
Bank	X

Question History: WBN bank question AOI-3800 006 with the choices repositioned to relocate the correct answer and minor wording changes is stem and values in choices, but not sufficiently to call it a modified question.

Comments:

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS

1. **IF** leak threatens personnel safety, **THEN**:
   a. **(ρ) TRIP** Rx.
   b. **CLOSE** the following:
      - MSIVs.
      - MSIV bypass valves.
   c. ** **GO TO** E-0, Reactor Trip or Safety Injection

2. **CHECK** S/G PORVs CLOSED.

   IF S/G press less than 1130 psig, **THEN**
   CLOSE S/G PORV or local isolation valve.

3. **CHECK** steam dump valves CLOSED.

   IF T_{avg} less than T_{ref}, **THEN**

   CLOSE steam dumps by turning 1-HS-1-103A and 1-HS-1-103B to OFF,

   OR

   CLOSE local isolation valves.

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS (continued)

4.	**MAINTAIN** reactor power less than or equal to 100% • Loop ΔT. • NIS power range monitors .	**(ρ) REDUCE** turbine load by 10% using the valve position limiter: a. **VERIFY** load reduction as indicated by either: • 1-PI-1-72 • 1-PI-1-73 • MWe meter b. **IF** Rx power returns to 100%, **THEN** 1) **(ρ) TRIP** Rx. 2) **CLOSE** all MSIVs and bypass valves. 3) ** **GO TO** E-0, Reactor Trip or Safety Injection.
5.	**MONITOR** T_{avg} and T_{ref} • T_{avg} trending to $T_{ref.}$ • Mismatch less than 5°F. D/B Reactor	**(ρ) CONTROL** T_{avg} with Control Rods in manual. **IF** T_{avg} and T_{ref} mismatch can NOT be maintained less than 5°F, **THEN** a. **(ρ) TRIP** Rx. b. **CLOSE** all MSIVs and bypass valves. c. ** **GO TO** E-0, Reactor Trip or Safety Injection.

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS (continued)

NOTE

It is a normal condition for Turbine load and Rx power to exhibit a nominal mismatch for loads less than 50%.

6.	**MONITOR** leak less than 3% of required steam or FW flow: **IF** power greater that 50%, **THEN COMPARE** turbine load to reactor power and ΔT.	**PERFORM** the following: a. **(ρ) TRIP** Rx. b. **CLOSE** all MSIVs and bypass valves. c. ** **GO TO** E-0, Reactor Trip or Safety Injection.
7.	**ENSURE** SG levels on program.	**IF** SG levels can **NOT** be maintained, **THEN** a. **(ρ) TRIP** Rx. b. ** **GO TO** E-0, Reactor Trip or Safety Injection.

CAUTION

FW or condensate leaks upstream of the FW isol valves should be promptly dealt with to limit hazards to Turbine Bldg personnel.

8.	**DISPATCH** personnel to perform secondary plant inspection for main steam and feedwater leaks.	

WBN BANK QUESTION

AOI3800 006

Given the following plant conditions:

- The Unit is stable at 60% power.
- Rod control is in Manual.
- A S/G Safety valve begins to leak through.

Which of the following identifies both....

the time in core life that will result in the largest addition of positive reactivity,
and,
a condition requiring a manual reactor trip per AOI-38, Steam or Feedwater Leak?

time in core life	condition requiring reactor trip
a. ✔ EOL	final Reactor power of 65%.
b. EOL	Tavg/Tref cannot be maintained within 3°F.
c. BOL	final Reactor power of 65%.
d. BOL	Tavg/Tref cannot be maintained within 3°F.

I. PROGRAM

Watts Bar Operator Training

II. COURSE

License Training

III. TITLE

AOI-38, Main Steam Or Feedwater Line Leak

IV. LENGTH OF LESSON

License Training 1 Hour

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	1. Demonstrate knowledge of the Purpose/Goal of AOI.
	X	X	X	2. Identify 3 Alarms that indicate this event is occurring.
	X	X	X	3. Give Operator Actions if leak is Hazardous to personnel
	X	X		4. Given a set of plant conditions, use the AOI to correctly: a. Recognize Entry Conditions. b. Identify Required Actions. c. Respond to Contingencies (RNO). d. Observe and Interpret Cautions and Notes.
	X	X	X	5. Identify the required action if reactor power is verified to be greater than 100% due to steam leak.

VI. TRAINING AIDS:

A. Marker Board and markers
B. Multimedia/Overhead projector(s)

VII. MATERIALS:

Attachment(s)

Attachment 1 - AOI-38, Main Steam or Feedwater Line Leak (Latest Rev.)

Attachment 2 - OE19418 and OE5464

Attachment 3 – Watts Bar Nuclear Plant MSR C2 Steam Relief Inadvertent Opening at Power

9. 054 AA1.02 009

Given the following:

- Unit 1 is operating at 100% power when an ATWS occurs.

Which ONE of the following identifies...

(1) the maximum time that operators have to manually start the AFW pumps

and

(2) the limiting ATWS accident which determines the maximum AFW start time?

A. (1) 30 seconds
(2) Failure of the turbine to trip

B. (1) 30 seconds
(2) Loss of main feedwater

C. (1) 60 seconds
(2) Failure of the turbine to trip

D. (1) 60 seconds
(2) Loss of main feedwater

DISTRACTOR ANALYSIS:

A. Incorrect, Plausible because 30 seconds is the amount of time that the operators have to trip the turbine during an ATWS and failure of the turbine to trip is an analyzed ATWS initiation event.

B. Incorrect, Plausible because 30 seconds is the amount of time that the operators have to trip the turbine during an ATWS and Loss of Main Feedwater is the correct event for the limiting time of 60 seconds.

C. Incorrect, Plausible because 60 seconds is the correct amount of time that the operators have to start AFW pumps during an ATWS and failure of the turbine to trip is an analyzed ATWS initiation event.

D. Correct, 60 seconds is the correct amount of time that the operators have to start AFW pumps during an ATWS and Loss of Main Feedwater is the correct event for the limiting time of 60 seconds.

Question Number: 9

Tier: 1 **Group** 1

K/A: 054 AA1.02
Loss of Main Feedwater (MFW)
Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW):
Manual startup of electric and steam-driven AFW pumps.

Importance Rating: 4.4 / 4.4

10 CFR Part 55: 41.7 / 45.5 / 45.6

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires the knowledge that the AFW pumps must be started when Main Feedwater is lost during an ATWS.

Technical Reference: WOG Emergency Procedure FR-S.1 Background Document, Revision 2
Watts Bar Nuclear Plant Updated Final Safety Analysis Report, Amendment 8

Proposed references to be provided: None

Learning Objective: 3-OT-FRS0001
10. Explain the purpose for and basis of each step in FR-S.1 and FR-S.2.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	X
Modified Bank	
Bank	

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

Comments:

STEP: Check AFW Pumps Running

PURPOSE: To ensure AFW pumps are running

BASIS:

The MD AFW pumps start automatically on an SI signal and SG low level to provide feed to the SGs for decay heat removal. If SG levels drop below the appropriate setpoint, the turbine-driven AFW pump will also automatically start to supplement the MD pumps. The ATWS analyses have shown that actuation of AFW within 60 seconds after the failure to scram provides acceptable results.

ACTIONS:

- Determine if MD AFW pumps are running
- Determine if the turbine-driven AFW pump is running if necessary
- Start MD AFW pumps
- Open steam supply valves to turbine-driven AFW pump

INSTRUMENTATION:

- MD AFW pumps status indication
- Turbine-driven AFW pump status indication
- Turbine-driven AFW pump steam supply valve position indication

CONTROL/EQUIPMENT:

Switches for:
- MD AFW pumps
- Turbine-driven AFW pump steam supply valves

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

Whenever a flood above plant grade is anticipated, an orderly shutdown to hot shutdown and a cooldown to cold shutdown will be initiated immediately. In a little more than 5 hours after reactor shutdown, the secondary system pressure will be reduced to approximately 100 psig. Within 27 hours after reactor shutdown, the fire-protection system piping will be connected to the auxiliary feedwater discharge piping by means of special spool pieces not normally installed, and the secondary system pressure will be maintained ≤125 psia. Operator action is taken, during a flood transient at 17 hours, to reduce SG operating pressure to ≤80 psig, to facilitate the supply of cooling water from the HPFP system in the event of the failure of 0-PCV-26-18 to close. Valve 0-PCV-26-18 is mounted in the HPFP pump recirculation line to the river and is non-safety grade. This pressure is sufficient for decay heat removal. When the flood exceeds plant grade, the auxiliary feedwater pumps will be inoperable, and the fire protection pumps, which are located above the maximum possible flood elevation, will supply feedwater.

Appropriate portions of the HPFP system are designed to function under normal conditions as well as for the maximum possible flood with the coincident or subsequent loss of the upstream and/or downstream dams. The HPFP pumps are located in the intake station above the flood line and are arranged to supply water directly to the steam generators in the event the auxiliary feedwater pumps are flooded.

The portion of the HPFP system which supplies auxiliary feedwater to the steam generators is ASME Section III, Class 3, Seismic Category I with the exception of the fire pumps discharge relief valve. These valves are replaced by ASME Section III, Class 3 blind flanges during flood mode preparations to ensure the integrity of the ASME Section III, Class 3 auxiliary feedwater supply piping during flood mode operation.

The AFW system is required to be available in the event of an ATWS event. The most severe ATWS scenarios have been determined to be those in which there is a complete loss of normal feedwater (Reference WCAP-10858). The design basis events for the AMSAC are Loss of Normal Feedwater/ATWS and Loss of Load/ATWS. Since there is a complete loss of normal feedwater during both of these transients, the accident analysis of both transients (Chapter 15) assumed AFW reaches full flow within 60 seconds after the initiating event for long term reactor protection. Also, the Loss of Normal Feedwater transient assumed a turbine trip within 30 seconds after the initiating event to maintain short term pressures below ASME Service Level C pressure limits. Normally these features will be actuated by the RPS. However, if a common mode failure to the RPS incapacitates AFW initiation and/or turbine trip in addition to prohibiting a scram, then an alternate method of providing AFW flow and a turbine trip is required to maintain RCS pressure below ASME Service Level C pressure limits. These two functions, turbine trip and AFW flow actuation, are provided via the AMSAC.

The AFW piping system layout has been optimized to prevent water hammer occurrences induced by the piping system.

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

AUO	RO	SRO	STA	
	X	X	X	6. Discuss the reason for placing the main feedwater valve mode selector switch (HS-3-45) to long cycle recirculation during the performance of FR-S.1.
	X	X	X	7. Given a set of plant conditions, determine if FR-S.1 Appendix A, Boron Dilution Path Isolation, or 1-SI-62-1, Uncontrolled Dilution Paths, should be performed.
	X	X	X	8. Given a set of plant conditions, including time after shutdown and IR and SR readings, determine if the excore instruments are responding normally and identify the actions required to manually reinstate the SR instruments.
	X	X	X	9. Given a set of plant conditions, use FR-S.1, FR-S.2 and the Critical Safety Function Status Trees to correctly diagnose and implement: Action Steps, RNOs, Notes and Cautions.
	X	X	X	10. Explain the purpose for and basis of each step in FR-S.1 and FR-S.2
	X	X	X	11. List the four major action categories of FR-S.1, Nuclear Power Generation/ATWS
	X	X	X	12. List the three major action categories of FR-S.2, Loss of Core Shutdown

10. 055 EA1.01 010
Given the following:

Unit 1 is in Mode 3 following a loss of offsite power 20 minutes ago.

Both of the Unit 1 diesel generators failed to start.

Which ONE of the following identifies...

(1) how the MCR crew will monitor Core Exit Thermocouples

and

(2) the minimum number of thermocouples required to to be operable in accordance with Tech Spec LCO 3.3.3, Post Accident Monitoring?

A. (1) Plasma displays on the control board or Integrated Computer System (ICS)
(2) A minimum of 2 quadrants with at least 4 thermocouples per train

B. ✔ (1) Plasma displays on the control board or Integrated Computer System (ICS)
(2) Each of the 4 quadrants must have at least 2 operable thermocouples per train

C. (1) Integrated Computer System (ICS) ONLY
(2) A minimum of 2 quadrants with at least 4 thermocouples per train

D. (1) Integrated Computer System (ICS) ONLY
(2) Each of the 4 quadrants must have at least 2 operable thermocouples per train

DISTRACTOR ANALYSIS:

A. *Incorrect, The plasma displays and the ICS being available during the blackout is correct, but the Tech Spec requirement is not a minimum of 2 quadrants with at least 4 thermocouples per train. Plausible because the first part is correct and the requirement is to have a minimum of 8 thermocouples in each train but it is '4 quadrants with 2', not '2 quadrants with 4'. The statement from TS basis (see below) that states "There are two isolated systems, with each system monitoring at least four thermocouples per quadrant" could be incorrectly interpreted by the applicant.*

B. *Correct, The plasma displays and the ICS computer both receive power from inverters which are powered from batteries during a loss of all AC power. The batteries are design to supply the loads for longer than the 20 minutes provided in the question stem. Tech Spec LOC 3.3.3, Post Accident Monitoring, requires each of the 4 quadrants to have two trains with a minimum of 2 thermocouples each.*

C. *Incorrect, The exclusive use of the ICS would not be required because the plasma displays would also be available during the blackout and the Tech Spec requirement is not 'a minimum of 2 quadrants with at least 4 thermocouples per train'. Plausible because there are indications that would be lost during a blackout and the statement from TS basis (see below) that states "There are two isolated systems, with each system monitoring at least four thermocouples per quadrant" could be incorrectly interpreted by the applicant.*

D. *Incorrect, The exclusive use of the ICS would not be required because the plasma displays would also be available during the blackout. Plausible because there are indications that would be lost during a blackout and the requirement to have 2 trains in each quadrant with each train having a minimum of 2 thermocouples is stated correctly in the second part of the choice.*

Wording from 3.3.3 Bases
17, 18, 19, 20. Core Exit Temperature
Core Exit Temperature is provided for verification and long term surveillance of core cooling. Core exit thermocouples, in conjunction with RCS wide range temperatures, are sufficient to provide indication of radial distribution of the coolant enthalpy rise across representative sections of the core.
Core Exit Temperature is used to support determination of whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.
The Inadequate Core Cooling Monitor (ICCM) is used to monitor the core exit thermocouples. There are two isolated systems, with each system monitoring at least four thermocouples per quadrant. The plasma display gives the average quadrant value, the high quadrant value, and the low quadrant value for each quadrant.
Two OPERABLE channels are required in each quadrant to provide adequate indication of coolant temperature rise in representative regions of the core. Two isolated channels of two thermocouples each ensure a single failure will not disable the ability to identify significant temperature gradients.

Question Number: 10

Tier: 1 **Group** 2

K/A: 055 EA1.01
Loss of Offsite and Onsite Power (Station Blackout)
Ability to operate and monitor the following as they apply to a Station Blackout:
In-core thermocouple temperatures.

Importance Rating: 3.7 / 3.9

10 CFR Part 55: 41.7 / 45.5 / 45.6

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires knowledge of the operability and availability of the In-core thermocouple temperature outputs during a station blackout.

Technical Reference: 1-45W700-1 R31
1-45W706-5 R18
Tech Spec 3.3.3, PAM instrumentation, Amendment 72
Control Board Photo

Proposed references to be provided: None

Learning Objective: 3-OT-T/S0303
03. Given plant parameters/conditions, correctly determine applicable Limiting Conditions for Operation or Technical Requirement limits for the various instrumentation systems covered by T/S or T/R.
04. Given plant parameters/conditions, correctly determine the OPERABILITY of the various instrumentation systems covered by T/S or T/R.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	
Modified Bank	
Bank	X

Question History: WBN bank question 055 EA1.01 010 used on 5/2008 exam

Comments:

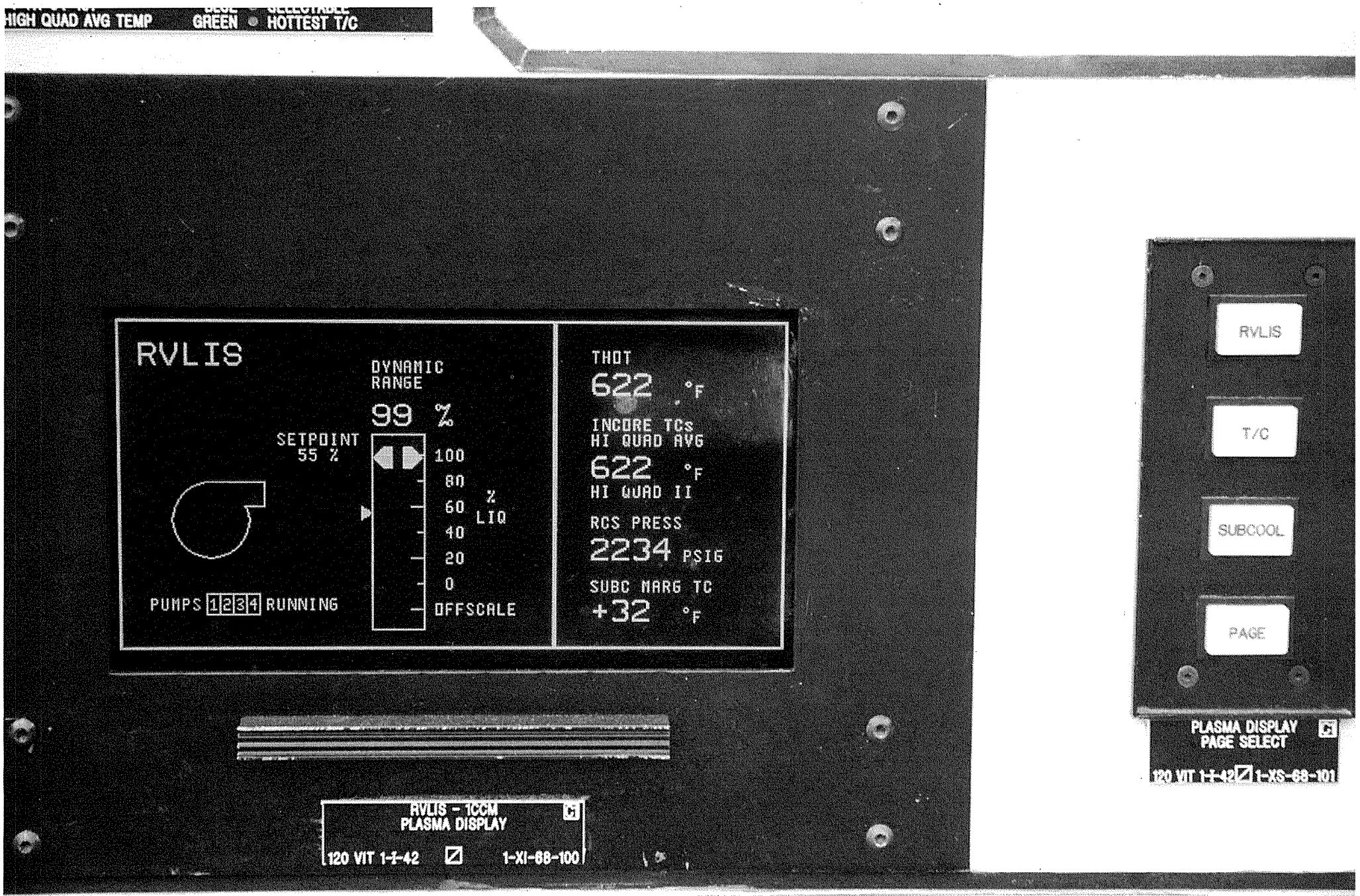

HIGH QUAD AVG TEMP
GREEN
HOTTEST T/C
RVLIS
DYNAMIC RANGE
99 %
SETPOINT 55 %
100
80
60
40
20
0
%
LIQ
OFFSCALE
PUMPS 1 2 3 4 RUNNING
THOT
622 °F
INCORE TCs HI QUAD AVG
622 °F
HI QUAD II
RCS PRESS
2234 PSIG
SUBC MARG TC
+32 °F
RVLIS
T/C
SUBCOOL
PAGE
PLASMA DISPLAY PAGE SELECT
120 VIT 1-1-42
1-XS-68-101
RVLIS - 1CCM PLASMA DISPLAY
120 VIT 1-1-42
1-XI-68-100

Table 3.3.3-1 (page 1 of 2)
Post Accident Monitoring Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS/TRAINS	CONDITION REFERENCED FROM REQUIRED ACTION D.1
1.	Intermediate Range Neutron Flux(g)	1[(a)], 2[(b)], 3	2	E
2.	Source Range Neutron Flux	2[(c)], 3	2	E
3.	Reactor Coolant System (RCS) Hot Leg Temperature (T-Hot)	1,2,3	1 per loop	E
4.	RCS Cold Leg Temperature (T-Cold)	1,2,3	1 per loop	E
5.	RCS Pressure (Wide Range)	1,2,3	3	E
6.	Reactor Vessel Water Level [(f)(g)]	1,2,3	2	F
7.	Containment Sump Water Level (Wide Range)	1,2,3	2	E
8.	Containment Lower Comp. Atm. Temperature	1,2,3	2	E
9.	Containment Pressure (Wide Range) (g)	1,2,3	2	E
10.	Containment Pressure (Narrow Range)	1,2,3	4	E
11.	Containment Isolation Valve Position (g)	1,2,3	2 per penetration flow path [(d)(i)]	E
12.	Containment Radiation (High Range)	1,2,3	2 upper containment 2 lower containment	F
13.	RCS Pressurizer Level	1,2,3	3	E
14.	Steam Generator (SG) Water Level (Wide Range)[(g)]	1,2,3	1/SG	E
15.	Steam Generator Water Level (Narrow Range)	1,2,3	3/SG	E
16.	AFW Valve Status [(j)]	1,2,3	1 per valve	E
17.	Core Exit Temperature-Quadrant 1[(f)]	1,2,3	2 [(e)]	E

(continued)

Table 3.3.3-1 (page 2 of 2)
Post Accident Monitoring Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS/TRAINS	CONDITION REFERENCED FROM REQUIRED ACTION D.1
18.	Core Exit Temperature-Quadrant 2[f]	1,2,3	2[e]	E
19.	Core Exit Temperature-Quadrant 3[f]	1,2,3	2[e]	E
20.	Core Exit Temperature-Quadrant 4[f]	1,2,3	2[e]	E
21.	Auxiliary Feedwater Flow	1,2,3	2/SG	E
22.	Reactor Coolant System Subcooling Margin Monitor [h]	1,2,3	2	E
23.	Refueling Water Storage Tank Water Level	1,2,3	2	E
24.	Steam Generator Pressure	1,2,3	2/SG	E
25.	Auxiliary Building Passive Sump Level [j]	1,2,3	2	E

(a) Below the P-10 (Power Range Neutron Flux) interlocks.
(b) Above the P-6 (Intermediate Range Neutron Flux) interlocks.
(c) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(d) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, pressure relief valve, or check valve with flow through the valve secured.
(e) A channel consists of two core exit thermocouples (CETs).
(f) The ICCM provides these functions on a plasma display.
(g) Regulatory Guide 1.97, non-Type A, Category 1 Variables.
(h) This function is displayed on the ICCM plasma display and digital panel meters.
(i) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.
(j) Watts Bar specific (not required by Regulatory Guide 1.97) non-Type A Category 1 variable.

BASES

LCO (continued)

16. AFW Valve Status

The status of each AFW swap over to Essential Raw Cooling Water (ERCW) valve is monitored with non-Type A Category 1 indication in the control room. Indication on each valve for fully open or fully closed position is provided. AFW valve status is monitored to give verification to the operator that automatic transfer to ERCW has taken place.

17, 18, 19, 20. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

Core exit thermocouples, in conjunction with RCS wide range temperatures, are sufficient to provide indication of radial distribution of the coolant enthalpy rise across representative sections of the core. Core Exit Temperature is used to support determination of whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.

The Inadequate Core Cooling Monitor (ICCM) is used to monitor the core exit thermocouples. There are two isolated systems, with each system monitoring at least four thermocouples per quadrant. The plasma display gives the average quadrant value, the high quadrant value, and the low quadrant value for each quadrant.

Two OPERABLE channels are required in each quadrant to provide adequate indication of coolant temperature rise in representative regions of the core. Two isolated channels of two thermocouples each ensure a single failure will not disable the ability to identify significant temperature gradients.

The incore thermocouple monitoring system described in Reference 4 supports the plant operating procedures.

(continued)

Question Number: 10

K/A: 000055 EA1.01
Ability to operate/monitor the following as they apply to a Station Blackout: In-core thermocouple temperatures.

Tier:	**1**	**RO Imp: 3.7**	**RO Exam:**	**Yes**	**Cognitive Level:**	**Low**
Group:	**1**	**SRO Imp: 3.9**	**SRO Exam:**	**Yes**	**Source:**	**NEW**

Applicable 10CFR55 Section: 41.7/45.5/45.6

Learning Objective: 3-OT-ECA0000, Objective 8: Given a set of plant conditions, use ECA-0.0, ECA-0.1, and ECA-0.2 to correctly diagnose and implement: Action Steps, RNOs, Notes and Cautions.

References: ECA-0.0, Rev. 19.

Question:

Unit 1 is in Mode 3 following a loss of offsite power and the failure of both the Unit 1 diesel generators to start.

Which ONE of the following identifies how the MCR crew will monitor Core Exit Thermocouples and the effect on the post accident monitoring (PAM) instrumentation Tech Spec LCO for Core Exit Temperature?

A. Plasma displays on the control board.
PAM Tech Spec LCO entry is required.

B. Plasma displays on the control board.
PAM Tech Spec LCO entry is NOT required.

C. Integrated Computer System (ICS) since the plasma displays on the control board will be unavailable.
PAM Tech Spec LCO entry is required.

D. Integrated Computer System (ICS) since the plasma displays on the control board will be unavailable.
PAM Tech Spec LCO entry is NOT required.

DISTRACTOR ANALYSIS

a. Incorrect. The plasma displays would be available during the blackout, but no PAM instrumentation LCO entry would be required. Plausible because the candidate could conclude that the LCO entry would be required due to the loss of power.
b. CORRECT. The plasma displays would be available during the blackout. The plasma displays are PAM instruments and no PAM instrumentation LCO entry would be required.
c. Incorrect. The use of the ICS would not be required because the plasma displays would be available during the blackout, therefore no PAM instrumentation LCO entry would be required. Plausible because the candidate could conclude that the LCO entry would be required with the plasma display unavailable.
d. Incorrect. The use of the ICS would not be required because the plasma displays would be available during the blackout. Additionally, the second part of the distractor is correct in that no PAM instrumentation LCO entry would be required. Plausible because the candidate could conclude that the LCO entry would not be required but conclude that the plasma display would be unavailable.

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. LICENSE TRAINING
B. LICENSED REQUAL

III. TITLE

T/S AND T/R 3.3, INSTRUMENTATION AND BASES

IV. LENGTH OF LESSON

A. LICENSE TRAINING 1 Hrs

LICENSED OPERATOR REQUAL TIME WILL BE DETERMINED AFTER OBJECTIVES ARE IDENTIFIED.

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	1. Demonstrate the ability to extract specific information from the Technical Specification, and Technical Requirements, as they pertain to Instrumentation Systems.
	X	X	X	2. Determine the bases for OPERABILITY of the Reactor Trip System or other instrumentation system.
	X	X	X	3 Given plant parameters/conditions, correctly determine applicable Limiting Conditions for Operation or Technical Requirement limits for the various instrumentation systems covered by T/S or T/R.
	X	X	X	4. Given plant parameters/conditions, correctly determine the OPERABILITY of the various instrumentation systems covered by T/S or T/R.
	X	X	X	5. Given plant parameters/conditions, correctly determine applicable Action Conditions, Required Actions, and Completion Times for the various instrumentation systems covered by T/S or T/R.

11. 056 AK3.01 011

Which ONE of the following identifies the pump that will be sequenced on first following a blackout signal on a 6900v Shutdown board and a reason?

	First Pump	Reason
A.	CCP	Largest blackout load on the board.
B.	CCP	To restore Reactor Coolant Pump seal cooling.
C.	ERCW pump	Largest blackout load on the board.
D.	ERCW pump	To restore Diesel Generator engine cooling.

DISTRACTOR ANALYSIS:

A. Incorrect, The CCP does sequence on first (5 seconds after SD board voltage is restored), however the ERCW pump is the largest load.

B. Correct, The CCP does sequence on first (5 seconds after SD board voltage is restored), and does supply RCP seal cooling and inventory makeup to the RCS.

C. Incorrect, ERCW pump is not be first pump to sequence on (20 seconds after SD board voltage is restored) but plausible because the ERCW pump is the largest load on the board and is used to cool the DG supplying power to the board.

D. Incorrect, ERCW pump is not be first pump to sequence on (20 seconds after SD board voltage is restored) but plausible because the ERCW pump is the largest load on the board and is used to cool the DG supplying power to the board.

Question Number: 11

Tier: 1 **Group** 1

K/A: 056 AK3.01
Loss of Offsite Power
Knowledge of the reasons for the following as they apply to the loss of Offsite Power:
Order and time to initiation of power for the load sequencer.

Importance Rating: 3.5 / 3.9

10 CFR Part 55: 41.5/41.10 / 45.6 / 45.13

10CFR55.43.b: Not applicable

K/A Match: This question matches K/A by having candidate determine the Status of CCPs and ERCW pumps following a loss of offsite power and a subsequent SI and the expected timed sequence for restart of the equipment.

Technical Reference: 1-45W760-62-1, R16
1-45W760-67-1, R25
WBN Final Safety Analysis Report, Amendment 8

Proposed references to be provided: None

Learning Objective: 3-OT-SYS201B
16. Identify the equipment and associated (sequence timer) ST setpoint that will start with the following:
a. A blackout with a return of voltage.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	
Modified Bank	
Bank	X

Question History: SQN Bank question 056 AK3.01 012 used on the SQN 2/2010 exam.

Comments:

The pump shaft, seal housing, thermal barrier, bolting ring and motor stand can be removed from the casing without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

The performance characteristic, shown in Figure 5.5-2, is common to all of the fixed speed mixed flow pumps, and the "knee" at about 45% design flow introduces no operational restrictions, since the pumps operate at full speed.

5.5.1.3 Design Evaluation

5.5.1.3.1 Pump Performance

The RCPs are sized to deliver flow at rates which equal or exceed the required flow rates. Initial reactor coolant system tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation.

The reactor trip system ensures that pump operation is within the assumptions used for loss of coolant flow analyses, which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a RCP is lost during operation.

An extensive test program was conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long term tests were conducted on less than full scale prototype seals as well as on full size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled leakage shaft seal pump design.

The support of the stationary member of the number 1 seal ("seal ring") is such as to allow large deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearing, the shaft could not deflect far enough to cause opening of the controlled leakage gap. The "spring constant" of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Testing of pumps with the number 1 seal entirely removed (full reactor pressure on the number 2 seal) has shown that relatively small leakage rates would be maintained for long periods of time (approximately 100 hours) even if the number 1 seal fails entirely. The plant operator is warned of this condition by the increase in number 1 seal leakoff and has time to close the number 1 seal leakoff line, and to conduct a safe plant shutdown without significant leakage of reactor coolant to the containment. Thus, it may be concluded that gross leakage from the pump does not occur, even if seals were to suffer physical damage.

The effect of loss of offsite power on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the cooling water for seal and bearing cooling. The emergency diesel generators are started automatically due to loss of offsite power so that component cooling flow is automatically restored. Seal water injection flow is subsequently restored by restarting a charging pump on diesel power.

1. 056 AK3.01 012

SQN BANK Question

Which ONE of the following identifies the pump that will be sequenced on first following a blackout signal on a 6900v Shutdown board and a reason?

	First Pump	Reason
A.	CCP	Largest blackout load on the board.
B.	CCP	Reactor Cooling Pump seal cooling restoration.
C.	ERCW pump	Largest blackout load on the board.
D.	ERCW pump	Diesel Generator cooling restoration.

DISTRACTOR ANALYSIS:

A. *Incorrect, The CPP does sequence on first (2 seconds after SD board voltage is restored), however the ERCW pump is the largest load.*

B. *Correct, The CPP does sequence on first (2 seconds after SD board voltage is restored), and does supply RCP seal cooling and inventory makeup to the RCS.*

C. *Incorrect, ERCW pump is not be first pump to sequence on (15 seconds after SD board voltage is restored) but plausible because the ERCW pump is the largest load on the board and is used to cool the DG supplying power to the board.*

D. *Incorrect, ERCW pump is not be first pump to sequence on (15 seconds after SD board voltage is restored) but plausible because the ERCW pump is the largest load on the board and is used to cool the DG supplying power to the board.*

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

AUO	RO	SRO	STA	
X	X	X	X	16. Identify the equipment and associated (sequence timer) ST setpoint that will start with the following: a. A blackout with a return of voltage. b. A blackout with a return of voltage together with a safety injection signal.
X	X	X	X	17. Identify the equipment that has a half-second (0.5 sec) timer associated with it start-ST-timer, and explain the purpose of this half-second timer.
X	X	X	X	18. Identify how DG control is regained when recovering from a blackout.
X	X	X	X	19. Identify the 480V boards that tripped during a blackout from the shutdown boards.
X	X	X	X	20. Identify when the blackout relays should be reset when recovering from a blackout.

12. 058 G2.1.30 012
Given the following:

- Unit 1 is operating at 100% power.
- A loss of 125V DC Vital Battery Board III occurs.
- Operators are performing AOI-21.03, "125V DC Vital Battery Board III," Appendix A, "Transfer of 125v DC Buses."

Which ONE of the following identifies...

(1) why the operator is directed to 'DEPRESS and HOLD' the 6.9kV SD Bd 2A-A BO-RESET switch in logic Panel 2A-A when control power is being restored to Shutdown Board 2A-A

and

(2) where the switch to transfer the 6.9kV SD Bd 2A-A 125V DC NORMAL BUS power supply is located?

A. (1) To prevent inadvertent equipment tripping.
(2) Inside the 125V DC Vital Battery Board III room.

B. (1) To prevent inadvertent equipment tripping.
(2) Inside a compartment on 6.9kV Shutdown Board 2A-A.

C. (1) To prevent unwanted starts of equipment.
(2) Inside the 125V DC Vital Battery Board III room.

D. (1) To prevent unwanted starts of equipment.
(2) Inside a compartment on 6.9kV Shutdown Board 2A-A.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because it is held to prevent unwanted operation of equipment but the operation is equipment starting not tripping that is being prevented and there are switches associated with 125v DC vital power inside the 125V DC Vital Battery Board III room.*

B. *Incorrect, Plausible because it is held to prevent unwanted operation of equipment but the operation is equipment starting not tripping that is being prevented and the transfer switch being located inside a compartment of the 6.9kV Shutdown Board is correct.*

C. *Incorrect, Plausible because preventing unwanted start of equipment is correct and there are switches associated with 125v DC vital power inside the 125V DC Vital Battery Board III room.*

D. *Correct, As identified in AOI-21.03, the reason for depressing and holding the BO-RESET switch is to prevent unwanted start of Train A equipment (See below) and the transfer switch is located inside a compartment on the 6.9Kv shutdown Board.*

Appendix A
(Page 1 of 1)
Transfer of 125V DC Buses
NOTE
Holding the BO-RESET Switch prevents unwanted start of Train A equipment, such as CCPs, AFW pumps, ERCW pumps, CCS pumps, and PZR Htrs when BO relays energize, and requires two operators to perform.

Question Number: 12

Tier: 1 **Group:** 1

K/A: 058 G2.1.30
Loss of DC Power
Conduct of Operations
Ability to locate and operate components, including local controls.

Importance Rating: 4.4 / 4.0

10 CFR Part 55: 41.7 / 45.7

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires the ability to locate a local control associated with a loss of DC power and the ability to understand why a separate local control pushbutton is being 'held' in a pushed in position.

Technical Reference: AOI-21.03, Loss of 125V DC Vital Battery Bd III, Revision 0020
1-45W724-3 R24

Proposed references to be provided: None

Learning Objective: 3-OT-AOI2100
9. Discuss transfer of 125V DC buses using AOI-21.01 or AOI-21.02, Appendix B for 125V DC Battery Board I or II respectively.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	X
Modified Bank	
Bank	

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

Comments:

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS

1.	**DISPATCH** personnel to restore power USING SOI-236.03, 125V DC Vital Battery Board III.	
2.	**PLACE** 1-XSW-46-DC, T-D AFW PUMP DC MANUAL TRANSFER SWITCH to EMERGENCY [A2T/692].	
3.	**ENSURE** ERCW flow to all running D/G's.	**EMERGENCY STOP** any D/G without cooling flow.
4.	**TRANSFER** Shutdown Board 125V DC control power to alternate supply USING Appendix A.	
5.	**PERFORM** Appendix C to place ventilation systems in service.	
6.	**PLACE** Handswitch 43TL, in 6.9KV Logic Panel 2A-A, in the TEST position.	

Appendix A
(Page 1 of 1)

Transfer of 125V DC Buses

NOTE

Holding the BO-RESET Switch prevents unwanted start of Train A equipment, such as CCPs, AFW pumps, ERCW pumps, CCS pumps, and PZR Htrs when BO relays energize, and requires two operators to perform.

A. **DEPRESS** and **HOLD** the 6.9kV SD Bd 2A-A BO-RESET switch [Logic Panel 2A-A].

B. **WHEN** DC bus is energized (125V DC CONTROL BUS ENERGIZED red light ON, panel 2), **THEN**

RELEASE the 6.9kV SD Bd 2A-A BO-RESET switch [Logic Panel 2A-A].

C. **PLACE** 6.9kV SD Bd 2A-A 125V DC SUPPLY TRANSFER SWITCH NORMAL BUS in the alternate position [panel 2].

D. **PLACE** 480V SD Bd 2A1-A 125V DC control power normal bus transfer switch in the alternate position [compt 5A].

E. **PLACE** 480V SD Bd 2A2-A 125V DC control power normal bus transfer switch in the alternate position [compt 5A].

F. **DEPRESS** and **HOLD** the 6.9kV SD Bd 1A-A BO-RESET switch [Logic Panel 1A-A].

G. **WHEN** DC bus is energized (125V DC CONTROL BUS ENERGIZED red light ON, panel 17), **THEN**

RELEASE the 6.9kV SD Bd 1A-A BO-RESET switch [Logic Panel 1A-A].

H. **PLACE** 6.9kV SD Bd 1A-A 125V DC SUPPLY TRANSFER SWITCH BACKUP BUS in the alternate position [panel 17].

I. **PLACE** 480V SD Bd 1A1-A 125V DC control power backup bus transfer switch in the alternate position [compt 6A].

J. **PLACE** 480V SD Bd 1A2-A 125V DC control power backup bus transfer switch in the alternate position [compt 6A].

K. **IF** CSST C switchgear NORMAL control power feed is available, **THEN**

ENSURE CSST C switchgear on NORMAL control power feed.

Appendix B
(Page 1 of 1)
Restoration of 125V DC Buses

NOTE

Holding the BO-RESET Switch prevents unwanted start of Train A equipment, such as CCPs, AFW pumps, ERCW pumps, CCS pumps, and PZR Htrs when BO relays energize, and requires two operators to perform.

A. **DEPRESS** and **HOLD** the 6.9kV SD Bd 2A-A BO-RESET switch [Logic Panel 2A-A].

B. **WHEN** DC bus is energized (125V DC CONTROL BUS ENERGIZED red light ON, panel 2), **THEN**

RELEASE the 6.9kV SD Bd 2A-A BO-RESET switch [Logic Panel 2A-A].

C. **PLACE** 6.9kV SD Bd 2A-A 125V DC SUPPLY TRANSFER SWITCH NORMAL BUS in the normal position [panel 2].

D. **PLACE** 480V SD Bd 2A1-A 125V DC control power normal bus transfer switch in the normal position [compt 5A].

E. **PLACE** 480V SD Bd 2A2-A 125V DC control power normal bus transfer switch in the normal position [compt 5A].

F. **DEPRESS** and **HOLD** the 6.9kV SD Bd 1A-A BO-RESET switch [Logic Panel 1A-A].

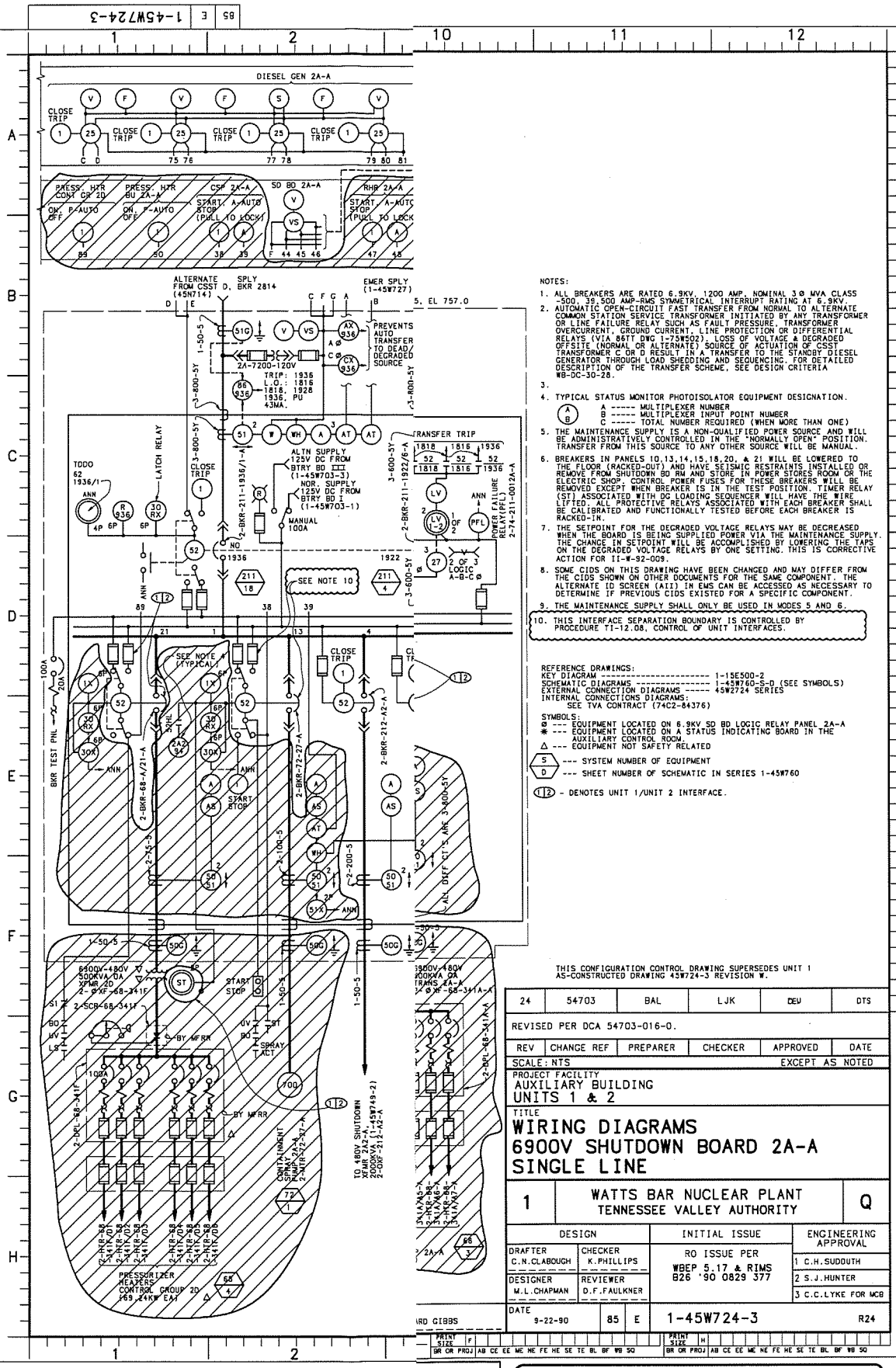
G. **WHEN** DC bus is energized (125V DC CONTROL BUS ENERGIZED red light ON, panel 17), **THEN**

RELEASE the 6.9kV SD Bd 1A-A BO-RESET switch [Logic Panel 1A-A].

H. **PLACE** 6.9kV SD Bd 1A-A 125V DC SUPPLY TRANSFER SWITCH BACKUP BUS in the normal position [panel 17].

I. **PLACE** 480V SD Bd 1A1-A 125V DC control power backup bus transfer switch in the normal position [compt 6A].

J. **PLACE** 480V SD Bd 1A2-A 125V DC control power backup bus transfer switch in the normal position [compt 6A].



NOTES:

1. ALL BREAKERS ARE RATED 6.9KV, 1200 AMP, NOMINAL 3 Ø MVA CLASS -500, 39,500 AMP-RMS SYMMETRICAL INTERRUPT RATING AT 6.9KV.
2. AUTOMATIC OPEN-CIRCUIT FAST TRANSFER FROM NORMAL TO ALTERNATE COMMON STATION SERVICE TRANSFORMER INITIATED BY ANY TRANSFORMER OR LINE FAILURE RELAY SUCH AS FAULT PRESSURE, TRANSFORMER OVERCURRENT, GROUND CURRENT, LINE PROTECTION OR DIFFERENTIAL RELAYS (VIA B&T DWG 1-75W502). LOSS OF VOLTAGE & DEGRADED OFFSITE (NORMAL OR ALTERNATE) SOURCE OF ACTUATION OF CSST TRANSFORMER C OR D RESULT IN A TRANSFER TO THE STANDBY DIESEL GENERATOR THROUGH LOAD SHEDDING AND SEQUENCING. FOR DETAILED DESCRIPTION OF THE TRANSFER SCHEME, SEE DESIGN CRITERIA WB-DC-30-28.
3.
4. TYPICAL STATUS MONITOR PHOTOISOLATOR EQUIPMENT DESIGNATION.
   (A/B) A ----- MULTIPLEXER NUMBER
   B ----- MULTIPLEXER INPUT POINT NUMBER
   C ----- TOTAL NUMBER REQUIRED (WHEN MORE THAN ONE)
5. THE MAINTENANCE SUPPLY IS A NON-QUALIFIED POWER SOURCE AND WILL BE ADMINISTRATIVELY CONTROLLED IN THE "NORMALLY OPEN" POSITION. TRANSFER FROM THIS SOURCE TO ANY OTHER SOURCE WILL BE MANUAL.
6. BREAKERS IN PANELS 10,13,14,15,18,20, & 21 WILL BE LOWERED TO THE FLOOR (RACKED-OUT) AND HAVE SEISMIC RESTRAINTS INSTALLED OR REMOVE FROM SHUTDOWN BD RM AND STORE IN POWER STORES ROOM OR THE ELECTRIC SHOP. CONTROL POWER FUSES FOR THESE BREAKERS WILL BE REMOVED EXCEPT WHEN BREAKER IS IN THE TEST POSITION. TIMER RELAY (ST) ASSOCIATED WITH DG LOADING SEQUENCER WILL HAVE THE WIRE LIFTED. ALL PROTECTIVE RELAYS ASSOCIATED WITH EACH BREAKER SHALL BE CALIBRATED AND FUNCTIONALLY TESTED BEFORE EACH BREAKER IS RACKED-IN.
7. THE SETPOINT FOR THE DEGRADED VOLTAGE RELAYS MAY BE DECREASED WHEN THE BOARD IS BEING SUPPLIED POWER VIA THE MAINTENANCE SUPPLY. THE CHANGE IN SETPOINT WILL BE ACCOMPLISHED BY LOWERING THE TAPS ON THE DEGRADED VOLTAGE RELAYS BY ONE SETTING. THIS IS CORRECTIVE ACTION FOR II-W-92-009.
8. SOME CIDS ON THIS DRAWING HAVE BEEN CHANGED AND MAY DIFFER FROM THE CIDS SHOWN ON OTHER DOCUMENTS FOR THE SAME COMPONENT. THE ALTERNATE ID SCREEN (AII) IN EMS CAN BE ACCESSED AS NECESSARY TO DETERMINE IF PREVIOUS CIDS EXISTED FOR A SPECIFIC COMPONENT.
9. THE MAINTENANCE SUPPLY SHALL ONLY BE USED IN MODES 5 AND 6.
10. THIS INTERFACE SEPARATION BOUNDARY IS CONTROLLED BY PROCEDURE TI-12.08, CONTROL OF UNIT INTERFACES.

REFERENCE DRAWINGS:
KEY DIAGRAM ---------- 1-15E500-2
SCHEMATIC DIAGRAMS ---------- 1-45W760-5-D (SEE SYMBOLS)
EXTERNAL CONNECTION DIAGRAMS ------ 45W2724 SERIES
INTERNAL CONNECTIONS DIAGRAMS: SEE TVA CONTRACT (74C2-84376)

SYMBOLS:
Ø --- EQUIPMENT LOCATED ON 6.9KV SD BD LOGIC RELAY PANEL 2A-A
✱ --- EQUIPMENT LOCATED ON A STATUS INDICATING BOARD IN THE AUXILIARY CONTROL ROOM.
△ --- EQUIPMENT NOT SAFETY RELATED
(5/D) --- SYSTEM NUMBER OF EQUIPMENT / --- SHEET NUMBER OF SCHEMATIC IN SERIES 1-45W760
(1/2) - DENOTES UNIT 1/UNIT 2 INTERFACE.

THIS CONFIGURATION CONTROL DRAWING SUPERSEDES UNIT 1 AS-CONSTRUCTED DRAWING 45W724-3 REVISION W.

24	54703	BAL	LJK	DEV	DTS
REVISED PER DCA 54703-016-0.					
REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE

SCALE: NTS — EXCEPT AS NOTED

PROJECT FACILITY
AUXILIARY BUILDING
UNITS 1 & 2

TITLE
WIRING DIAGRAMS 6900V SHUTDOWN BOARD 2A-A SINGLE LINE

1	WATTS BAR NUCLEAR PLANT TENNESSEE VALLEY AUTHORITY	Q

DESIGN		INITIAL ISSUE	ENGINEERING APPROVAL
DRAFTER C.N.CLABOUGH	CHECKER K.PHILLIPS	RO ISSUE PER WBEP 5.17 & RIMS B26 '90 0829 377	1 C.H.SUDDUTH
DESIGNER M.L.CHAPMAN	REVIEWER D.F.FAULKNER		2 S.J.HUNTER
			3 C.C.LYKE FOR MCB

DATE 9-22-90 | 85 | E | 1-45W724-3 | R24

PRINT SIZE F — BR OR PROJ | AB CE EE ME NE FE HE SE TE BL BF WB SQ
PRINT SIZE H — BR OR PROJ | AB CE EE ME NE FE HE SE TE BL BF WB SQ

RD GIBBS

DRAWING

CONFIGURATION CONTROL DRAWING

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. LICENSE TRAINING
B. NON-LICENSE TRAINING

III. TITLE

AOI-21, LOSS OF 125V DC VITAL BATTERY BDS

IV. LENGTH OF LESSON

1.5 Hours, All Courses

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	1. Identify Indications of a Loss of 125V Vital Battery Bd I.
	X	X	X	2. Describe Auto Actions for Loss of 125V Vital Battery Bd I.
	X	X	X	3. Identify 125V Vital Battery Bds whose loss results in Rx Trip.
	X	X	X	4. Explain why Aux Bldg Gen Supply & Exhaust fans are stopped on loss of Vital Battery Bd I or II.
	X	X	X	5. Describe basic procedure for shutting down unneeded DGs after loss of a Vital Battery Bd.
X	X	X	X	6. Determine local action required on loss of Vital Battery Bd II that causes C & SS Air Compressors to unload.
	X	X	X	7. Describe significance of loss of dc to Protection and Control systems (SOER 83-5, Rec 9).
	X	X	X	8. Describe significance of a partial loss of dc (SOER 81-15, Rec 2C).
X	X	X	X	9. Discuss transfer of 125V DC buses using AOI-21.01 or AOI-21.02, Appendix B for 125V DC Battery Board I or II respectively.
X	X	X	X	10. Discuss transfer of C&SS Air Compressors 125V DC from Normal to Alternate.
X	X	X	X	11. Discuss transfer of Turbine Driven Auxiliary Feedwater Pump 1A-S DC power from Normal to Emergency per AOI-21.03, Sec. 3.0.

13. 062 AA1.06 013

Given the following:

- Unit 1 is operating at 100% power.
- All available Train A ERCW pumps are in service.
- ERCW pump B-A trips.

Which ONE of the following identifies...

(1) the CCS heat exchanger whose outlet flow control bypass valve will be adjusted, if needed, to maintain CCS temperature on the Unit 1 Miscellaneous Equipment Header

and

(2) how the ERCW flow rate on CCS Heat Exchanger B will respond when the bypass valve is adjusted?

	CCS Heat Exchanger	ERCW flow to CCS Heat Exch B
A.	C	Drops
B.	A	Drops
C.	C	Remains Constant
D.	A	Remains Constant

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because the CCS heat exchanger C outlet bypass valve being the valve to be throttled more open to reduce the temperature could have been correct if a Train B pump had tripped and because the ERCW flow rate on CCS heat exchanger B is reduced is correct.*

B. *Correct, the B-A pump trip will lower the 2A ERCW header pressure which will drop flow through Component Cooling Water System (CCS) heat exchangers A and B. This will cause CCS temperature on the U-1 Miscellaneous Equipment Header to increase requiring 1-FCV-67-143, CCS HX A OUTLET ERCW FLOW CNTL BYP, to be manually throttled more open to restore the temperature. Placing more flow on CCS heat exchanger A cause a further reduction on the ERCW header pressure, thus lowering ERCW flow on CCS heat exchanger B.*

C. *Incorrect, Plausible because the CCS heat exchanger C outlet bypass valve being the valve to be throttled more open to reduce the temperature could have been correct if a Train B pump had tripped and because the ERCW flow rate on the CCS heat exchanger C does remain constant, but the ERCW flow rate on CCS heat exchanger B is reduced.*

D. *Incorrect, Plausible because the CCS heat exchanger A outlet bypass valve is the valve that will be throttled more open to reduce the temperature and because while the ERCW flow rate on the CCS heat exchanger C does remain constant, the ERCW flow rate on CCS heat exchanger B is reduced.*

Question Number: 13

Tier: 2 **Group:** 1

K/A: 062 AA1.06
Loss of Nuclear Service Water
Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS):
Control of flow rates to components cooled by the SWS

Importance Rating: 2.9 / 2.9

10 CFR Part 55: 41.7 / 45.5 / 45.6

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires service water system (ERCW) flow rate changes that are made following a loss of a service water system pump in order to maintain temperature on a system (CCS) cooled by the service water system.

Technical Reference: SOI-70.01, Component Cooling Water System, Revision 0068
1-47W845-1 R57
1-47W845-2 R80
1-47W859-1 R49

Proposed references to be provided: None

Learning Objective: 3-OT-SYS067A
3. Describe the ERCW System flow path from the river to the cooling tower basin and discharge holding pond including:
b. Major components
3-OT-SYS070A
19. Given a set of plant conditions, determine the correct response of the CCS system.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	X
Modified Bank	
Bank	

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

Comments:

Table of Contents

Date_____ **INITIALS**

6.1.1 CCS Heat Exchanger A Temperature Control

NOTES

1) Adjustment of ERCW flow may impact concurrent functions or evolutions of the ERCW system

2) CCS temperature indication [0-M-27B] is provided by:
1-TI-70-161, CCS HX A OUTLET TEMP, **OR**
1-P/TR-70-161, CCS HX A TEMP & PMP PRESS.

3) Adequate time should be allowed for CCS temperature to stabilize following ERCW flow adjustment.

4) Steps 6.1.1[2] AND 6.1.1[3] may be repeated as required to achieve the desired CCS temperature.

5) 1-FCV-67-143 does **NOT** have a full travel seal-in feature. It should be adjusted by "bumping" 1-HS-67-143A, CCS HX A DISCH TO HDR B, [0-M-27A] in the OPEN or CLOSE direction. Local electrical controls are disconnected, and handwheel may be used for local MANUAL adjustment.

6) Local valve manipulation should be performed 1) SLOWLY to prevent abrupt system perturbations, and 2) in continuous contact with UO monitoring system parameters.

7) Power may be restored to 1-FCV-67-146-A, CCS HX A OUTLET CONTROL, and the valve may be opened as needed in Modes 4, 5 and 6, or defueled.

8) When CCS HX outlet valve 1-FCV-67-146 is aligned and open to intermediate positions, the bypass valve 1-FCV-67-143 should be closed to avoid excessive flow thru the CCS HX A and to ensure adequate flow to other equipment.

[1] **REFER TO** SOI-67.01, Essential Raw Cooling Water System for system parameter monitoring and Pump starting or stopping instructions. _____

[2] **IF** CCS temperature is HIGHER than desired, **OR**

IF ERCW pressure is HIGHER than desired,

THEN THROTTLE OPEN 1-FCV-67-143, CCS HX A OUTLET ERCW FLOW CNTL BYP [A10T/737]. _____

Date_____ **INITIALS**

6.1.1 CCS Heat Exchanger A Temperature Control (continued)

[3] **IF** CCS temperature is LOWER than desired, **OR**

IF ERCW pressure is LOWER than desired, **THEN**

THROTTLE CLOSED 1-FCV-67-143, CCS HX A OUTLET ERCW FLOW CNTL BYP [A10T/737]. _____

[4] **IF** in Mode 4, 5, 6 or Defueled and additional ERCW flow through CCS HX A is needed, **THEN**

[4.1] **ENTER** Operating Requirement (OR) for Appendix R valve opening. _____

[4.2] **UNLOCK** and **CLOSE** bkr on 480V RX MOV BD 1A2-A, C/11A, 1-FCV-67-146-A. _____

[4.3] **OPEN** 1-FCV-67-146-A, CCS HX A OUTLET CONTROL, as required. _____

[4.4] **CHECK** for cavitation at valve, AND

OPEN further, as necessary to STOP any cavitation. _____

[4.5] **CLOSE** 1-FCV-67-143, CCS HX A OUTLET ERCW FLOW CNTL BYP. _____

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training

B. Non-License Training

III. TITLE

Essential Raw Cooling Water System

IV. LENGTH OF LESSON

A. Licensed Training 2 hour

B. Non-Licensed Training 3.5 hours

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X		1. State the function of the Essential Raw Cooling Water System in accordance with the system description.
	X	X	X	2. State the design basis of the Essential Raw Cooling Water System in accordance with FSAR Section 9.2.1.
X	X	X	X	3. Describe the ERCW System flow path from the river to the cooling tower basin and discharge holding pond including: a. Interfaces b. Major components c. Paths to and from the Auxiliary Building
X	X	X	X	4. Deleted.
X	X	X	X	5. Deleted.
X	X	X	X	6. Explain the purpose of Biocide injection.
X	X	X	X	7. Discuss the precaution associated with contact of Bromide with the body.
X	X	X	X	8. State the ERCW System normal discharge path and given a failure of the path, discuss the alternate discharge paths.
X	X	X	X	9. Describe the purpose of the hydraulic gradient.
X	X	X	X	10. Describe the purpose of the discharge overflow structure.
X	X	X	X	11. Deleted.

AUO	RO	SRO	STA	
X	X	X	X	12. Deleted.
	X	X	X	13. Given a loss of power, determine the correct response of the ERCW System including: a. "C" CCS Heat Exchanger outlet valves. b. ERCW Pumps.
X	X	X	X	14. Explain how and when the ERCW System would be used to feed the Steam Generators.
X	X	X	X	15. List the places where the ERCW System can be interconnected to another system with a spool piece.
X	X	X	X	16. Given a failure in the ERCW System, determine how radiation would be detected.
	X	X	X	17. Given a loss of instrument air/control power, determine the effect on temperature control valves.
X	X	X	X	18. [Identify 2 indications of biofouling in heat exchangers as stated in SOER 84-1. (REC. #4)]
X	X	X	X	19. Describe the ERCW pumps including capacity, logic, power supplies, type, lubrication, heat removal means and alarm setpoints.
X	X	X	X	20. Describe the normal and backwash flow paths of the ERCW strainers.
X	X	X	X	21. Describe the purpose of the ERCW prelube, what components it supplies and from where it is supplied.
X	X	X	X	22. Describe the operation of the traveling screen system, normal and backwash, and how ΔP is determined.
X	X	X	X	23. Briefly describe how to place the ERCW System in service per SOI-67.01.
	X	X	X	24. Regarding Technical Specifications and Technical Requirements for this system: a. Identify the conditions and required actions with completion time of one hour or less. b. Explain the Limiting Conditions for Operation, Applicability, and Bases. c. Given a status/set of plant conditions, apply the appropriate Technical Specifications and Technical Requirements.
X	X	X	X	25. List the ERCW parameters governed by Tech Specs.

AUO	RO	SRO	STA	
X	X	X	X	26. Correctly locate control room controls and indications associated with the Essential Raw Cooling Water System, including: a. Pump and Selector Switch Controls b. Header Isolation Valves and Flow Indications c. Supply and Discharge Headers d. CCS Heat Exchanger Alignments
X	X	X	X	27. [Identify the action(s) to be taken by the operator if significant heat exchanger degradation due to fouling is detected. (SOER 84-1, Rec. 4)]

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training

B. NOTP

C. License Operator Requal

D AUO Requal

III. TITLE

Component Cooling System

IV. LENGTH OF LESSON

A. Licensed Training 1.5 hours

B. NOPT 3.0 hours

License Requalification and NAUO Requalification times will be determined after objectives are identified

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. State the design basis of the Component Cooling Water System (CCS) in accordance with FSAR section 9.2.2.
X	X	X	X	2. Sketch a basic drawing of the CCS, include all pumps, major heat exchangers, and blocks showing major uses of CCS.
X	X	X	X	3. Describe the CCS pumps, include power supply, pump type, capacity, lubrication, and logic.
X	X	X	X	4. Explain the logic associated with each valve/pump control in the CCS.
X	X	X	X	5. Explain the operation, purpose, and location of the C-S CCS Pump power supply transfer switch.
X	X	X	X	6. Describe the CCS heat exchangers, include cooling medium.

AUO	RO	SRO	STA	
X	X	X	X	7. Given a tube rupture in a CCS heat exchanger, describe the resulting flow path.
X	X	X	X	8. Describe the thermal barrier system; include purpose, pump capacity, and logic.
X	X	X	X	9. Describe the CCS Surge Tanks; include purpose, capacity, and method of makeup to them.
X	X	X	X	10. Explain how the CCS pumps are sealed and how the seal leakage return unit operates.
X	X	X	X	11. Identify ten (10) uses of CCS during normal and post accident conditions.
X	X	X	X	12. Identify the automatic actions that occur upon detection of CCS high radiation.
	X	X	X	13. Describe the actions which must be taken if CCS is lost to the RCP motors.
X	X	X	X	14. [Identify two indications of biofouling in a heat exchanger. (SOER 84-1, Rec. 4)]
	X	X	X	15. Describe the effect of a loss of CCS to the major equipment supply headers: a. Miscellaneous equip. & Reactor Bldg. Headers b. ESF Equipment Header A c. ESF Equipment Header B d. Spent Fuel Pit Supply Header
	X	X	X	16. Regarding Technical Specifications and Technical Requirements for this system: a. Identify the conditions and required actions with completion time of one hour or less. b. Explain the Limiting Conditions for Operation, Applicability, and Bases. c. Given a status/set of plant conditions, apply the appropriate Technical Specifications and Technical Requirements.

AUO	RO	SRO	STA	
X	X	X	X	17. Identify the in-plant location of each of the following: a. Component Cooling Water Pumps. b. Component Cooling Water Heat Exchangers. c. Thermal Barrier Booster Pumps. d. Component Cooling System Surge Tanks. e. Seal Leakage Return Tank and Pumps. f. C-S CCS Pump Power Supply Throw-over Switch. g. CCS Flood Mode Spool Pieces. h. RHR Heat Exchangers.
	X	X	X	18. Correctly locate all control room controls and indications associated with the Component Cooling System.
	X	X	X	19. Given a set of plant conditions, determine the correct response of the CCS system.
	X	X	X	20. Given a CCS instrument and failure mode, identify how the instrument will respond and what interlock(s) or control function(s) will be affected, including effects on system/component operation.
	X	X	X	21. Given a loss of instrument air/control power, determine the effect on the following valves: a. Surge tank make up valve. b. Surge tank vent valve. c. Letdown HEAT EXCHANGER. temp. control valve.
	X	X	X	22. Explain how the failure of CCS or its support systems could lead to core damage.
	X	X	X	23. [Identify the action(s) to be taken by the Operator if significant heat exchanger degradation due to fouling is detected. (SOER 84-1, Rec. 4)]

14. 065 G2.2.44 014
Given the following:

- Unit 1 is currently in Mode 3, with the reactor trip breakers open.
- A control air leak develops inside containment.
- 1-FCV-62-86, Alt Charging to Loop 4, is in service and 1-FCV-62-85, Norm Charging to Loop 1, begins drifting open.
- AOI-10, "Loss of Control Air," is entered.
- The operating crew stops RCPs #1, #3, and #4.

Which ONE of the following identifies...

(1) the control air header that has developed the leak

and

(2) how stopping the RCPs affect the plant in accordance with AOI-10?

A. (1) Auxiliary Air Train B
(2) Reduces the level in the pressurizer.

B. (1) Auxiliary Air Train B
(2) Restores control of pressurizer spray.

C. (1) Non-essential Control Air
(2) Reduces the level in the pressurizer.

D. (1) Non-essential Control Air
(2) Restores control of pressurizer spray.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because 1-FCV-62-85 is a B Train valve and it would be reasonable that it be supplied with B Train Auxiliary Air. Also, all but one of the RCPs being evaluated and stopped during a loss of control air in Mode 3 in order to minimize heat input to the RCS to limit pressurizer level increase is correct.*

B. *Incorrect, Plausible because 1-FCV-62-85 is a B Train valve and it would be reasonable that it be supplied with B Train Auxiliary Air. Also, stopping the all of the RCPS would reduce the heat input to the RCS which would limit the rise in pressurizer level but it would cause an unwanted natural circulation condition and is not in accordance with the AOI.*

C. *Correct, While 1-FCV-62-85 is a Train B valves for isolation purposes, the valve receives non-essential control air. All but one RCP being stopped, after an evaluation during a loss of control air in Mode 3, would be in order to minimize heat input to the RCS to limit the pressurizer level increase.*

D. *Incorrect, Plausible because 1-FCV-62-85 being supplied by non-essential control air is correct. Also, stopping the all of the RCPS would reduce the heat input to the RCS which would limit the rise in pressurizer level but it would cause an unwanted natural circulation condition and is not in accordance with the AOI.*

Question Number: 14

Tier: 1 **Group:** 1

K/A: 065 G2.2.44
Loss of Instrument Air
Equipment Control
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Importance Rating: 4.2 / 4.4

10 CFR Part 55: 41.5 / 43.5 / 45.12

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires the interpretation of control room indications (valve positions) to determine the failure mode of the instrument air system and it also requires the operator to recall why the RCP operation would affect the plant (pressurizer level).

Technical Reference: AOI-10, Loss of Control Air, Rev. 0040

Proposed references to be provided: None

Learning Objective: 3-OT-AOI1000
05. Explain the actions to control PZR level if Non-Essential Air is lost in Mode 3.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	X
Modified Bank	
Bank	

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

Comments:

Step	Action/Expected Response	Response Not Obtained

3.3 Loss of Control Air in Mode 1, 2, 3 or 4 (continued)

CAUTION **Charging FCVs 62-85, 62-86, 62-89, and 62-93 fail OPEN.**

NOTE With letdown isolation failed CLOSED, pzr level will continue to rise. Reducing seal flow will slow rate of rise.

13. **CHECK** normal letdown in service.

CLOSE charging isol, 1-FCV-62-90 and 1-FCV-62-91.

Locally **ADJUST** in-service seal water filter outlet valve 1-ISV-62-549 OR 1-ISV-62-550 [A5T/713] to greater than 6 gpm/pump.

14. **CHECK** VCT level between 20% and 41%.

IF VCT level less than 7%,
THEN
ENSURE CCP suction from RWST.

IF VCT level greater than 13%,
THEN
ENSURE CCP suction from VCT.

15. **CHECK** pzr level less than 70%.

IF pzr level rising uncontrolled,
THEN:

a. **REDUCE** power to lower pzr level:

- **REFER TO** AOI-39, Rapid Load Reduction.

b. **MINIMIZE** RCP seal flow at greater than 6 gpm/pump.

Step	Action/Expected Response	Response Not Obtained

3.3 Loss of Control Air in Mode 1, 2, 3 or 4 (continued)

NOTE In Mode 3, consider shutting down all but one RCP (Rx trip bkrs open) or two RCPs (Rx trip bkrs closed) to reduce RCS heat input. Loop 2 RCP is preferred to run.

16. **IF** reactor is tripped,
THEN
EVALUATE availability of pzr sprays, and **STOP** RCPs, as desired.

17. **START** available ABGTS:

- **REFER TO** Appendix A for affected dampers, and
- **REFER TO** SOI-30.06, Auxiliary Building Gas Treatment System.

18. **START** available EGTS:

- **REFER TO** Appendix A for affected dampers, and
- **REFER TO** SOI-65.02, Emergency Gas Treatment System.

19. **ENSURE** the following STOPPED [1-M-9]:

- Cntmt Purge Sup and Exh fans.
- Inst Rm Purge Sup and Exh fans.
- A B Gen Supply and Exhaust fans, Unit 1 and 2.
- A and B F H Area Exh fans.
- CRDM coolers.

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training
B. Non-License Training

III. TITLE

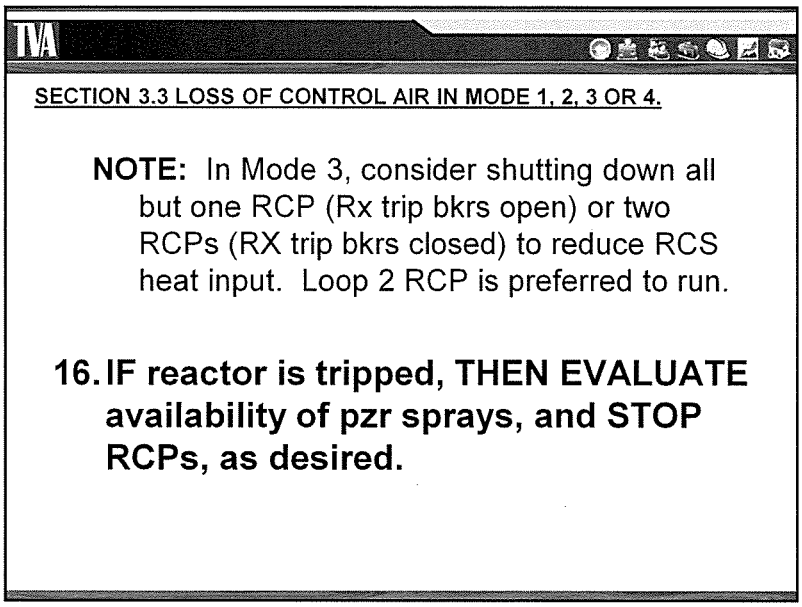
AOI-10, Loss of Control Air

IV. LENGTH OF LESSON

A. License Training 2 Hour
B. Non-License Training 2 Hour

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	1. Demonstrate knowledge of the Purpose/Goal of AOI-10.
	X	X	X	2. Describe Auto Actions for Loss of Control Air per AOI-10 (SOER 88-1, Rec 2).
	X	X	X	3. Identify Operator Action if Loss of Control Air is due to loss of 6.9kV SD Bds (SOER 88-1, Rec 2).
	X	X	X	4. Describe local Manual Actions to control RCS temperature if Train A or B Aux Air header depressurizes and Non-Essential Air is not available.
X	X	X	X	5. Explain the actions to control PZR level if Non-Essential Air is lost in Mode 3.
	X	X	X	6. Summarize the method(s) for RCS temperature control on RHR, upon loss of Control Air (SOER 88-1, Rec 2).



NOTE: In Mode 3, consider shutting down all but one RCP (Rx trip bkrs open) or two RCPs (RX trip bkrs closed) to reduce RCS heat input. Loop 2 RCP is preferred to run.

Basis: Informs the operator about options to stop RCPs, with the unit in Mode 3, to reduce RCS heat input causing cooldown and pzr level drop. Loop 2 is preferred to be left in service due to availability of pzr spray.

Basis 16: Aux Air supplies air pressure to the spray valves; therefore, they should be available for pressure control. All but loop 2 RCP should be stopped at this time to reduce heat input to the RCS.

Additional Information:

3.4.5 RCS Loops – MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and either:

a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

Bases:

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

15. 077 AA2.10 015

Given the following:

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

  Megawatts 1210 MWe
  Megavars +20 Mvars
  Gen Voltage 23.6 Kv

- A disturbance occurs on the Transmission Grid resulting in the following:

  Megawatts 1210 MWe
  Megavars -180 Mvars
  Gen Voltage 23.6 Kv

- Annunciator 1-C, STATOR TEMP HI, alarms.

- The Stator Coil Outlet temperature is determined to be 177°F and rising at 2°F/minute.

Which ONE of the following identifies...

(1) how the turbine trip circuit is affected by the conditions

and

(2) an action associated with generator exciter current that could reduce stator temperature?

A. (1) A timer to trip the turbine has been initiated.
(2) Lower generator excitation current.

B. (1) A timer to trip the turbine has been initiated.
(2) Raise generator excitation current.

C. (1) The timer to trip the turbine has **NOT** been initiated, but will be if the temperature continues to rise.
(2) Lower generator excitation current.

D. (1) The timer to trip the turbine has **NOT** been initiated, but will be if the temperature continues to rise.
(2) Raise generator excitation current.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because the temperature to start the turbine trip timer also brings in an alarm which could be mistaken for the alarm that is LIT and lowering the generator excitation current would be correct if the MVAR load had been a positive value.*

B. *Incorrect, Plausible because the temperature to start the turbine trip timer also brings in an alarm which could be mistaken for the alarm that is LIT and raising the generator excitation current is correct with the MVAR loading being a negative value.*

C. *Incorrect, Plausible because the turbine trip timer being not started is correct and lowering the generator excitation current would be correct if the MVAR load had been a positive value.*

D. *Correct, the alarm setpoint is 176°F for the lit annunciator but the turbine trip 45 second timer does not start timing until the temperature reaches 194°F, so the timer has not started and raising exciter current will reduce the MVAR load on the generator allowing the stator to cool.*

Question Number: 15

Tier: 1 **Group** 1

K/A: 077 AA2.10
Generator Voltage and Electric Grid Disturbances
Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances:
Generator overheating and the required actions.

Importance Rating: 3.6 / 3.8

10 CFR Part 55: 41.5 and 43.5 / 45.5, 45.7, and 45.8

10CFR55.43.b: Not applicable

K/A Match: The K/A is matched because the question requires knowledge of the correct operator actions for a condition where a grid disturbance causes generator overheating due to a change in reactive loading on the generator.

Technical Reference: ARI 1-7, Electrical Control, Revision 0030

Proposed references to be provided: None

Learning Objective: 3-OT-SYS035C
5. Describe the Generator Trips associated with SCW system.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	X
Modified Bank	
Bank	

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

Comments:

1-C

Source	Setpoint
1-TE-35-214	176°F Stator Coil Water Discharge Temperature
1-PDS-35-120A	Less than or equal to 21 psid across stator coil (low stator coil water flow)

STATOR TEMP HI

R

(Page 1 of 1)

NOTE

Turbine Generator trips at 194°F (90°C) stator outlet temperature or stator bar ΔP less than or equal to 17 psid when unit load is greater than 15% and condition exists for 45 seconds.

Probable Cause:

A. Stator Coolant System failure
   1. Low flow through stator water heat exchangers
   2. Malfunction of 1-TCV-24-52, STATOR COIL HX 1A/1B RCW OUT HDR TEMP CNTROL
   3. SCW pumps tripped

B. Under excitation

Corrective Action:

[1] **INITIATE** stator temperature monitoring per SOI-35.01, Generator Hydrogen Cooling System.
[2] **ENSURE** generator operating within capability curve limits.
[3] **CONSIDER** load reduction per GO-4, Normal Power Operation or AOI-39, Rapid Load Reduction, to keep temperature within limits.
[4] **DISPATCH** Operator to investigate and check proper operation of 1-TCV-24-52.
[5] **IF** 1-TCV-24-52 NOT controlling temperature properly, **THEN THROTTLE OPEN** 1-BYV-24-926, STATOR COIL HS 1A/1B TEMP CNTL VLV BYPASS.
[6] **MONITOR** stator coil discharge temperature on Plant Computer point, T3098A.
[7] **IF** alarm is due to loss or partial loss of Raw Cooling Water, **THEN REFER TO** AOI-46, Loss of Raw Cooling Water.

References:

1-75W1507,
GO-4, SOI-35.01, SOI-35.03

1-B

Source	Setpoint
1-TS-35-104A 1-TS-35-104B 1-TS-35-104D	Stator outlet H2O temp greater than or equal to 194°F (90°C) (2 out of 3 TS's)
1-PDS-35-120B 1-PDS-35-120D 1-PDS-35-120E	Stator Bar ΔP less than or equal to 17 psid (2 out of 3 PDS's)

STATOR COOLING FAILURE

(Page 1 of 1)

Probable Cause:

A. Loss of RCW to stator cooling system
B. Stator coolant system pipe break
C. Stator coolant pump tripped

NOTE

The unit will trip after a 45 second time delay, if unit load is greater than 15%.

Corrective Action:

[1] **DISPATCH** Operator to Panel 1-L-39 determine cause of alarm and initiate corrective action.
[2] **REDUCE** unit load to less than 15% of rated power.
[3] **IF** turbine trips above 50% power, **THEN** **ENSURE** reactor trip, and **GO TO** E-0, Reactor Trip Or Safety Injection.
[4] **IF** turbine trips below 50% power, **THEN** **GO TO** AOI-17, Turbine Trip.
[5] **IF** alarm is due to loss or partial loss of Raw Cooling Water, **THEN** **REFER TO** AOI-46, Loss of Raw Cooling Water.
[6] **NOTIFY** Work Control to initiate corrective action, if necessary.

References:

1-45W600-47-2
AOI-17
E-0

I. PROGRAM

Watts Bar Operator Training

II. COURSES

License Training

NOTP

License Requalification

NAUO Requalification

III. TITLE

Stator Cooling Water System

IV. LENGTH OF LESSON

A. License Training 1.5 Hours
B. NOTP 1.5 Hours
License Requalification and NAUO Requalification times will be determined after objectives are identified.

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. Describe the purpose of the Stator Cooling Water (SCW) System.
X	X	X	X	2. Identify the systems that interface with the SCW System.
X	X	X	X	3. Describe auto start of the Standby SCW Pump.
	X	X	X	4. Identify maximum load that can be carried with one HX out.
	X	X	X	5. Describe the Generator Trips associated with the SCW System.
X	X	X	X	6. Explain how the SCW System demineralizers are used.
	X	X	X	7. Give maximum allowable SCW System water conductivity.

16. W/E04 EK1.1 016
Given the following:

- Unit 1 was operating at 100% power when a small break LOCA occurred.
- The crew has implemented ECA-1.2, "LOCA Outside Containment," due to alarm "AUX BLDG HIGH ENERGY LINE BREAK" being LIT.
- When the crew closes RHR Train A cold leg injection valve, 1-FCV-63-93, the RCS pressure begins to rise.
- RHR pump 1A-A is stopped and its suction valve closed.
- RCS pressure is now 1535 psig and slowly rising.

Which ONE of the following pumps is currently contributing the most to the decrease in RWST level?

A. 1B-B Safety Injection Pump

B. 1A-A Containment Spray Pump

C. 1A-A Centrifugal Charging Pump

D. 1B-B Residual Heat Removal Pump

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible since the SI pumps are started and would be delivering flow to the RCS, however they have a shut-off head of ~1560 psig. With the RCS pressure near the shutoff head , each of the SI pumps would only be injecting ~50 gpm and would be contributing less than the CCP to lowering the level in the RWST.*

B. *Incorrect, Plausible since the Containment Spray pumps are of greatest capacity (4500 gpm) during a LOCA, and if candidate does not realize that the LOCA is outside CNMT and there would be no need for this pump to be operating.*

C. *Correct, Since the CCPs re-align through the CCPIT during a LOCA event, and with RCS pressure at 1535 psig the CCPs are designed to deliver ~340 gpm (170 gpm/each) to the RCS. The CCP would be the largest contributor to RWST level lowering at this time during the accident.*

D. *Incorrect, Plausible since this pump is running during the LOCA event. However the RCS pressure is above the shut-off head of the RHR pump and the pump is not injecting.*

Simulator - RCS pressure 1520 psig

CCP flow 340 gpm/total (170 gpm each)
SIP flow A-50 gpm B 40 gpm

Question Number: 16

Tier: 1 **Group** 1

K/A: W/E04 EK1.1
LOCA Outside Containment
Knowledge of the operational implications of the following concepts as they apply to the (LOCA Outside Containment):
Components, capacity, and function of emergency systems.

Importance Rating: 3.5 / 3.9

10 CFR Part 55: 41.8 / 41.10

10CFR55.43.b: Not applicable

K/A Match: This question matches the K/A by having the candidate recall the capacity of several ECCS pumps and relating that information to the ECCS system response during a LOCA.

Technical Reference: N3-63-4001, Safety Injection System, Rev. 0027

Proposed references to be provided: None

Learning Objective: 3-OT-SYS063A
27. List each subsystem of the ECCS with its design injection pressure and flow.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	
Bank	X

Question History: SQN bank question W/E04 EK1.01 written for SQN SEPT 2010 NRC exam.

Comments:

8.1 Figure 1 - CCP Composite Curve

Curve	Maximum Composite Curve	FSAR Curve**	Test Acceptance Curve*
Flow (gpm)	Head (ft)	Head (ft)	Head (ft)
0	6180.0	5590.0	5822
50	6125.0	5575.0	5807
100	6100.0	5560.0	5792
150	6083.0	5550.0	5782
200	5800.0	5200.0	5432
250	5490.0	4800.0	5032
300	5100.0	4325.0	4557
350	4690.0	3800.0	4032
400	4210.0	3200.0	3432
450	3610.0	2575.0	2807
500	2800.0	1860.0	2092
550	1750.0	1140/1110.0	1342

* Note : Based On a 4% head (5800 ft * 4%) margin between the Minimum Acceptance Curve and the FSAR Curve

** Note : The FSAR curve is based on the "Generic" Pump Curve data. The Generic pump curve data was used in the original ECCS Analysis. The only exception is at 550 gpm, 1110 ft was used instead of 1140 ft. since 1110 ft was used in the previous analysis.

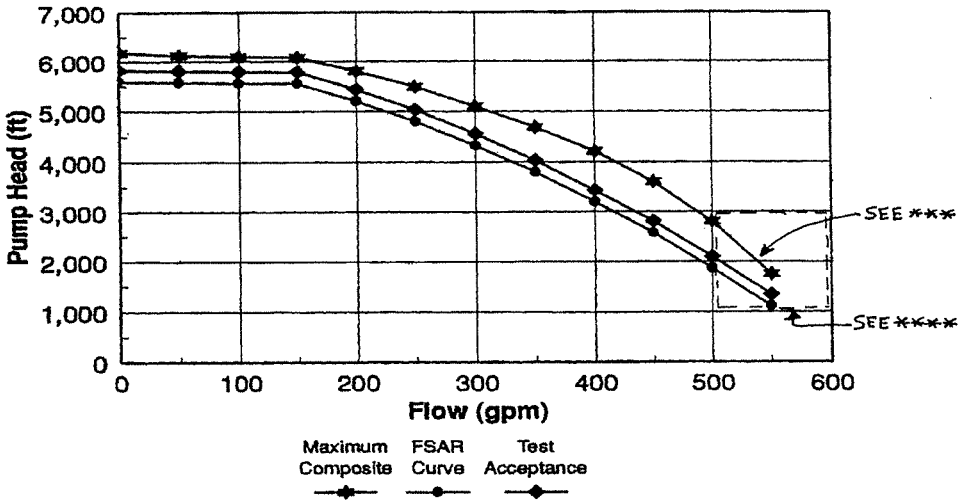


FIGURE 1 CCP COMPOSITE CURVE
(REF 7.5.84)

*** As-found test data reflected slightly higher head/flow values than the maximum curve. This bounding test point (bounding for both 1A-A & 1B-B test data) has been evaluated by Westinghouse in WAT-D-10872 and found to be acceptable.

**** FUTURE PROCUREMENTS: On pages 8 and 9 of Reference 7.5.117, note the Runout Evaluation History section and the Charging Runout Limits table. The information provided is the current basis for "upper band" operating limits on the CCPs and should be utilized for any future procurements of rotating elements after the Reference 7.5.117 data. See Section 3.2.1A and Table 5 of this system description. Due to cold leg injection flow balance requirements, the performance curve(s) of the replacement rotating element(s) shall be analyzed by Westinghouse for suitability prior to change out (Ref. 7.5.84, 7.5.119, 7.5.120).

8.2 Figure 2 - SIP Composite Curve

	Maximum Composite Curve	FSAR	Test Acceptance Curve
Flow (gpm)	Head (ft)	Head (ft)	Head (ft)
0	3560	3345	3428
50	3505	3300	3383
100	3495	3290	3373
150	3460	3240	3323
200	3415	3200	3283
250	3355	3150	3233
300	3250	3050	3133
350	3105	2905	2988
400	2990	2750	2833
450	2795	2550	2633
500	2600	2380	2463
550	2395	2175	2258
600	2190	1980	2063
650	1990	1725	1808

Note: Based on a 3 % head (2750 ft X 3 %) margin between the Test Acceptance Curve and the FSAR Curve.

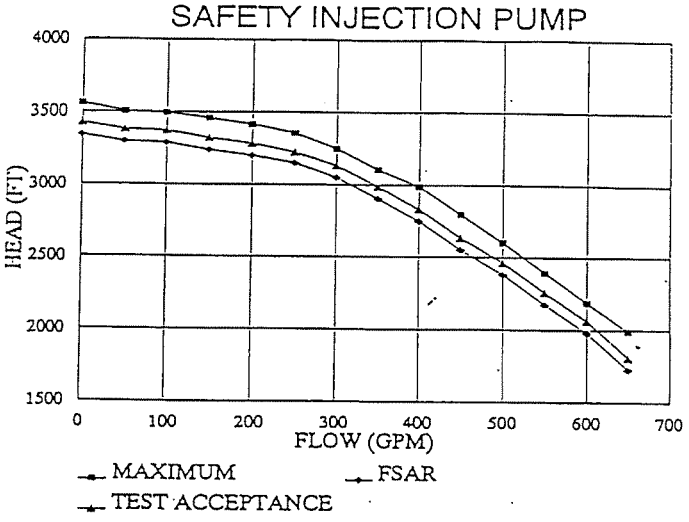


FIGURE 2 SIP COMPOSITE CURVE
(REF 7.5.84)

8.3 Figure 3 - RHRP Composite Curve

Flow (gpm)	MAXIMUM COMPOSITE CURVE Head(ft)	TEST ACCEPTANCE CURVE Head(ft)	FSAR* Head(ft)
0	487	450	432.5
680	474	411**	400.5***
2000	436	392**	381.5***
2500	423	392	381.5***
3000	409	385	367.5
3500	395	375	357.5
4000	374	355	337.5
4500	351	325	307.5
5000	322	295	277.5
5500	290	260	247.5
6000	254	210	192.5

* NOTE: Based on a 5% head (350ft * 5% = 17.5 ft) margin between the Test Acceptance Curve and the FSAR Curve.
** NOTE: Determined from test data.
***NOTE: Data from 680 to 2500 gpm is degraded by 3% (or 10.5 ft) from the test data.

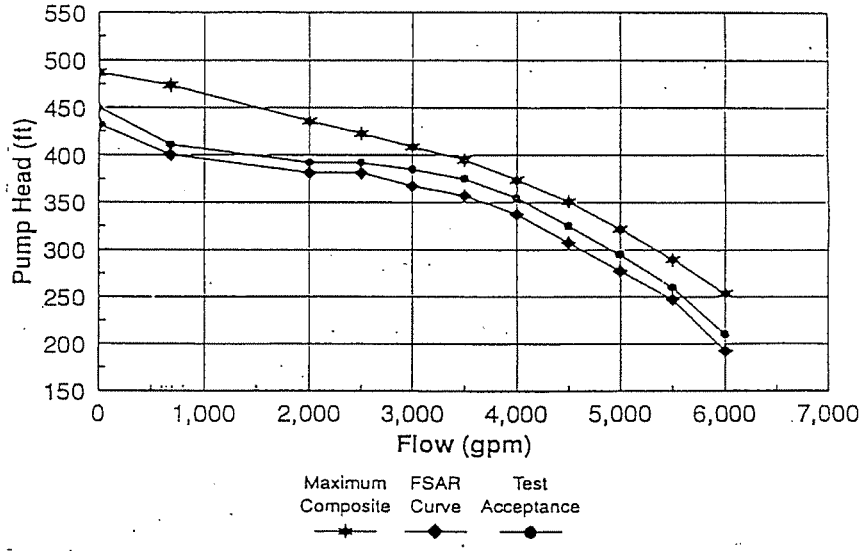
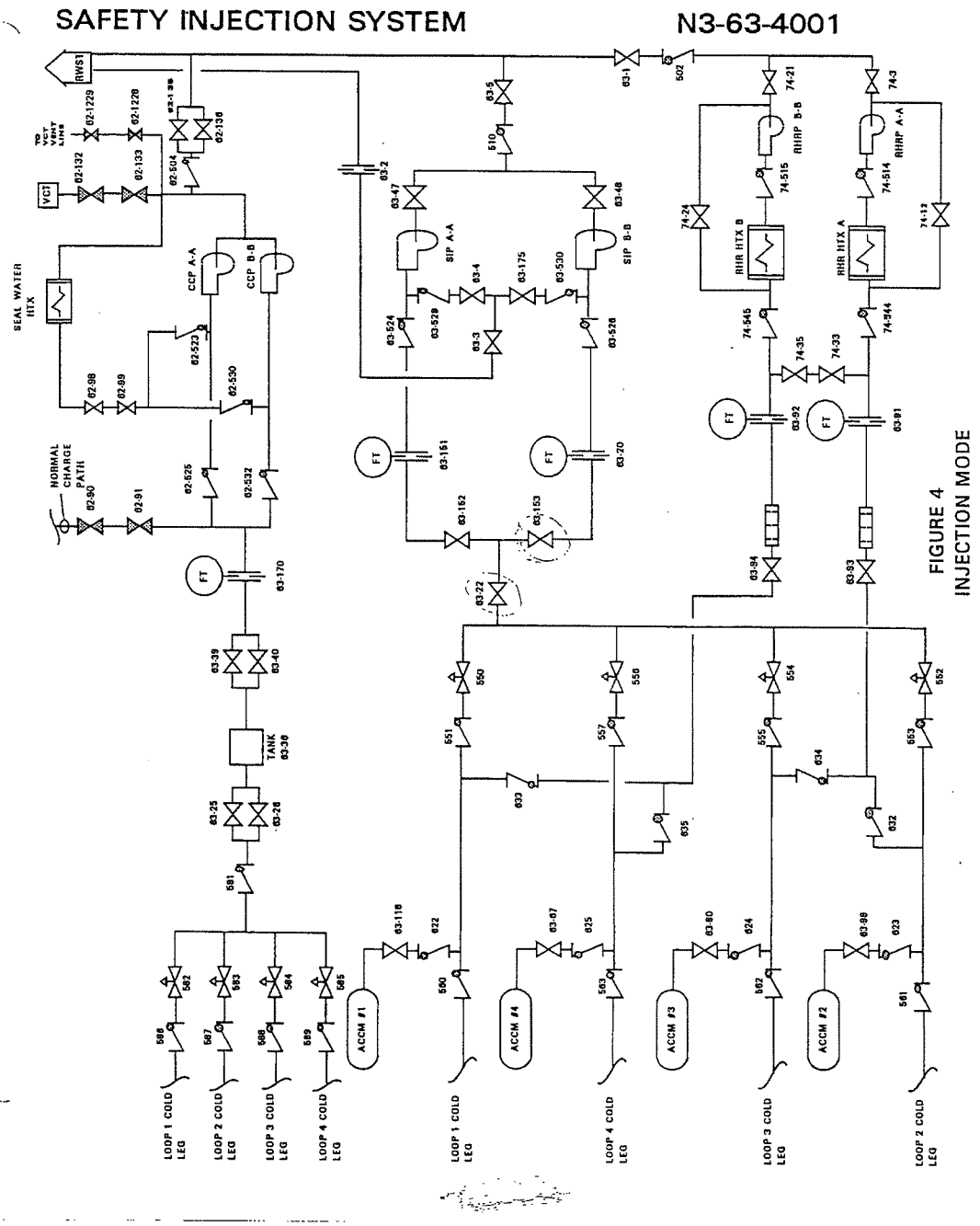


FIGURE 3 RHRP COMPOSITE CURVE
(REF 7.5.84)

8.4 Figure 4 - Injection Mode Simplified Flow Diagram



9.5 Table 5 - SIS-CCP Design and Operating Parameters

TABLE 5
SIS-CCP DESIGN AND OPERATING PARAMETERS
(Ref 7.4.3, 7.4.7, 7.5.34, 7.5.35, 7.5.51, 7.5.52, 7.5.57, 7.5.81, 7.5.84)

Component	Characteristic	Value
CCPs	Type	Centrifugal
	Number of Pumps per unit	2
	Design Press	2800 psig
	Design Temp	300°F
	Design Flow Rate(original design point)	150 gpm
	Design Head (original design point)	5800'
	Max Flow Rate (inj. mode/recirc mode)	550 gpm/560 gpm
	Head required @ Max Flow Rate (inj. mode)	1342'****
	Disch Shutoff Head (Design)	5900'
	Disch Shutoff Head (ECCS Analysis)	5822'****
	NPSH Required @ Max Flow (inj. mode)	25'*****
	Pump Water Volume	1.9 ft^3
	Motor Capacity (Original/Upgraded)	600 HP/720 HP
	Amps, Full Load (Original)	45.3
	Amps, Locked Rotor (Original)	273
	CCS Min, Flow Required	28 gpm
	Gear Oil CLR CCS Flow, gpm	8
	Lube Oil CLR CCS Flow, gpm	20
	Max Starting Time, Sec	5 ***
Piping Injection		
Beyond Stop Valves	Design Pressure	2485 psig
FCV-63-25/26	Design Temperature	650°F
Piping, Suction From	Design Pressure	150 psig
RWST to LCV-62-135, 136	Design Temperature	150°F
Piping, Pump Suction	Design Pressure	220 psig
	Design Temperature	250°F
Relief Valve 63-577	Design Pressure	150 psig
Discharge Piping	Design Temperature	200°F
All Other Piping	Design Pressure	2735 psig
	Design Temperature	200°F
BIT	Number per unit	1
	Total Volume	900 gal
	Usable Volume @ Operating Conditions	900 gal
	Boron Conc (as BA) Minimum	0* ppm
	Design Pressure	2,735 psig
	Design Temperature, °F	300°F
	Operating Pressure	2500 psig ambient
	Operating Temperature	6**
	Number of Tank Heaters	

* Minimum concentration is 0 ppm since this was the value used in the W BIT elimination reanalysis (ref 7.5.48). Actual concentration may be as high as the RWST.

** Power to these heaters disconnected. Heaters are not needed due to the reduction in BIT boron concentration.

*** Time for pump to reach full speed, head and flow (on miniflow) after receipt of a start signal and power available.

**** The required head has been re-established for the corresponding flow rate based on the test acceptance curve contained in reference 7.5.84. However, when purchasing spare or replacement internal elements, the characteristic curve should be 1400 ft at 550 gpm to allow margin for OM Code degradation. Also, see Section 3.2.1A and Figure 1 for additional future procurement limitations that must be considered.

***** The as-tested NPSH at less than maximum flow has been evaluated by DCN S-31896-A and determined to be acceptable.

9.6 Table 6 - SIS-SIP Design and Operating Parameters

Table 6 SIS-SIP Design and Operating Parameters (Ref 7.4.7 and 7.5.34, 7.5.35, 7.5.47, 7.5.52, 7.5.56, 7.5.84)		
Component	Characteristic	Value
SIPs	Type Number per unit Design Pressure Design Temperature Design Flow Rate original design point) Design Head (original design point) Max Flow Rate (inj. mode/recirc mode) Head required @ Max Flow Rate (inj. mode) Disch Shutoff Head (Design) Disch Shutoff Head (ECCS Analysis) NPSH Required @ Max Flow (inj. mode) Pump Water Volume Bearing Oil Cooler, Flow Bearing Oil Cooler, Heat Load Motor Capacity, hp Amps, Full Load Amps, Locked Rotor CCS Min Flow Required Max Starting Time	Centrifugal 2 1750 psig 300°F 400 gpm 2700' 650 gpm/675 gpm 1808'** 3500' 3483'** 30'(Pump 1A)*** 28'(Pump 1B)*** 1.6 ft^3 15 0.046x10^6Btu 400 31.5 amp 218 amp 30 gpm 5 sec *
SI Injection Piping Beyond Stop Valves FCV-63-22, -23, -156, -157	 Design Pressure Design Temperature	 2485 psig 650°F
SI Suction Piping From The RWST To FCV-63-5	Design Pressure Design Temperature	150 psig 150°F
SIP Suction Piping	Design Pressure Design Temperature	220 psig 200°F
SIP Miniflow Line Beyond FCV-63-3	Design Pressure Design Temperature	150 psig 200°F
Discharge Piping For Relief Valves 63-534, -535, -536	Design Pressure Design Temperature	100 psig 400°F
All Other SI Piping	Design Pressure Design Temperature	1750 psig 200°

* Time for pump to reach full speed, head and flow (on miniflow) after receipt of a start signal and power available.

** The required head has been re-established for the corresponding flow rate based on the test acceptance curve contained in reference 7.5.84.

*** The as-tested NPSH at less than maximum flow has been evaluated by DCN S-31896-A, and S-34674-A, and determined to be acceptable.

9.7 Table 7 - SIS-CLA Design and Operating Parameters

Table 7
SIS CLA Design and Operating Parameters
Ref. 7.4.7, 7.5.52, 7.5.58, 7.5.104, 7.5.111, 7.5.115, 7.5.122, 7.5.125. 7.5,126, 7.5.127

Component	Characteristic	Value
CL Accumulator	Number Per Unit	4
	Design Pressure	700 psig
	Design Temperature	300°F
	Operating Temperatures	60 - 150°F
	Min Press Safety Analysis Limit	585 psig
	Lo Press Alarm	Ref. 7.1.12
	High Press Alarm	Ref. 7.1.12
	Maximum Press (Safety Analysis)	690 psig
	Minimum Inventory (Safety Analysis	1005 ft^3
	Lo Level Alarm	Ref. 7.1.12
	High Level Alarm	Ref. 7.1.12
	Maximum Inventory (Safety Analysis)	1095 ft^3
	Volume to Upper Level Tap	1080.6 ft^3
	Nominal Vol to Upper Head Tempest	1116.84 ft^3
	Nominal Vol of Tank	1355.59 ft^3
	Minimum Total Verifiable Vol with Level Just At Upper Tap	1059.6 ft^3
	BA Concentration	3,150 (nom) ppm 3,000 (min)
	Relief Valve Setpoint	700 psig
	External Pressure Rise**	
	Pressure	12 psi
	Time	10 sec
	Vol N_2 Gas - Nominal	305.59 ft^3
	N_2 Supply Isolates	632 psig
	Hi-Level Alarm	1165 ft^3
Accumulator Outlet To Valves FCV-63-67, -80, -98, -118	Design Pressure	700 psig
	Design Temperature	300°F
Accumulator Lines From Valves FCV-63-67, -80, -98, -118 To RCS	Design Pressure	2485 psig
	Design Temperature	650°F
All Other Accumulator Piping	Design Pressure	700 psig
	Design Temperature	650°F

** In Cntmt

SQN BANK QUESTION

1. W/E04 EK1.1 015

Given the following:

- Unit 1 was operating at 100% power when a small break LOCA occurred.
- The crew has implemented ECA-1.2, "LOCA Outside Containment," due to alarm "AUX BLDG HIGH ENERGY LINE BREAK" being LIT.
- The crew closes RHR Train A cold leg injection valve, FCV-63-93, in accordance with ECA-1.2.
- RCS pressure is now 1495 psig and slowly rising.

Which ONE of the following pumps is contributing the most to the decrease in RWST level?

A. 1B-B Safety Injection pump

B. 1A-A Containment Spray pump

C. 1B-B RHR pump

D. 1A-A CCP injecting through CCPIT

AUO	RO	SRO	STA	
X	X	X	X	11. Describe the purpose for the hot leg recirculation mode of ECCS operation and what it is designed to prevent.
X	X	X	X	12. Describe the components of the ECCS subsystems.
X	X	X	X	13. Describe the cold leg accumulators, include capacity, boron concentration, type of charging agent, and pressure at which they inject.
X	X	X	X	14. Briefly explain how to make up nitrogen and water to the CLA's.
X	X	X	X	15. Describe each of the following ECCS pumps, include lubrication, cooling and type: a. RHRP. b. SIP. c. CCP.
X	X	X	X	16. Describe the RWST; include capacity, the means of maintaining boron in solution, overflow detection and where temperature/level can be read.
X	X	X	X	17. List all the systems that are supplied directly by the RWST.
X	X	X	X	18. Sketch all the connections to and from the RWST.
X	X	X	X	19. Describe the RWST solution, include how it is made up to and maintained.
X	X	X	X	20. Describe the containment sump, include location and setpoints.
X	X	X		21. Identify and explain the initiation signals to the ECCS, include setpoints.
	X	X	X	22. Identify the accident(s) for which each safety injection signal affords protection.
X	X	X	X	23. Identify each system that is automatically activated from a safety injection signal.
	X	X	X	24. Given a set of plant conditions, determine the correct response of the Emergency Core Cooling System.
	X	X		25. Describe how to reset the safety injection signal, include P-4 interlock, also how and when to block the SI signal.
X	X	X	X	26. Describe how a safety injection can be reset after activation.
X	X	X	X	27. List each subsystem of the ECCS with its design injection pressure and flow.
	X	X		28. Identify the technical specifications and bases associated with each of the following components or conditions: a. Cold Leg Injection Accumulators b. ECCS Subsystems - Tavg $\geq$ 350 $^{\circ}$ F c. ECCS Subsystems - Tavg < 350 $^{\circ}$F d. RWST
X	X	X	X	29. State the ECCS parameters that are governed by Tech Specs.
	X	X	X	30. Given the condition/status of the Emergency Core Cooling system/component and the appropriate sections of Tech Specs, determine if operability requirements are met and what actions, if any, are required.
X	X	X	X	31. Deleted.
X	X	X	X	32. Explain the ECCS precautions and limitations.
X	X	X	X	33. Deleted.

17. W/E05 EK2.2 017

Given the following conditions:

- The Crew has entered FR-H.1, "Loss of Secondary Heat Sink," due to a loss of inventory in the S/Gs and failure of the AFW pumps to start.

	SG Wide Range Levels (%)				*AFW*	*Cnmt*
Time	*SG#1*	*SG#2*	*SG#3*	*SG#4*	*Flow*	*Press*
0800	41	39	38	37	0 gpm	1.70 psid
0810	37	32	34	33	0 gpm	2.40 psid
0820	30	28	27	26	0 gpm	2.70 psid
0830	27	25	22	24	0 gpm	2.50 psid

Which ONE of the following is the earliest time, if any, that the initiation of Bleed and Feed is required?

A. Bleed and Feed **NOT** required.

B. 0810

C. 0820

D. 0830

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible if any the criteria are not known – setpoint, number of levels below the setpoint, or when adverse containment numbers are to be applied (and would have been correct for the previous level setpoints.) 0800 would be a correct answer if containment pressure had been above 2.8 psid because 3 of the levels are less than 39%.*

B. *Incorrect, Plausible if any the criteria are not known – setpoint, number of levels below the setpoint, or adverse containment numbers. 0810 would be a correct answer if the containment pressure had been above 2.8 psid because all of the levels are less than 39%.*

C. *Incorrect, Plausible if any the criteria are not known – setpoint, number of levels below the setpoint, or adverse containment numbers. 0820 is plausible because three of the four levels are below the minimum heat sink required level of 29% which could be misapplied instead of the WR level of 26%. Also, one of the levels is below the required setpoint and the time would be correct if 2 additional levels, which are near the minimum setpoint, were below the minimum required 26%.*

D. *Correct, Bleed and Feed criteria is met when 3 of the 4 S/G wide range levels are at or below 26% when Adverse Containment values are not required. Adverse containment values are not used until containment pressure is above 2.8 psid.*

Question Number: 17

Tier: 1 **Group** 1

K/A: W/E05 EK2.2
Loss of Secondary Heat Sink
Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following:
Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Importance Rating: 3.9 / 4.2

10 CFR Part 55: 41.7 / 45.7

10CFR55.43.b: Not applicable

K/A Match: KA is matched because the applicant must apply the plant conditions to the required actions in the applicable procedure being used in response to a loss of all feed water to identify when the emergency

K/A Match: KA is matched because the applicant must apply the plant conditions to the required actions in the applicable procedure being used in response to a loss of all feed water to identify when the emergency coolant system would be initiated to provide core cooling.

Technical Reference: FR-H.1, Loss of Secondary Heat Sink, Rev 18

Proposed references to be provided: None

Learning Objective: 3-OT-FRH0001
07. Given a set of plant conditions, evaluate and determine when RCS bleed and feed should be initiated.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	
Bank	X

Question History: WBN bank question W/E05 EK2.2 016 used an AUDIT exam in 2010 with one level value changed in the 0830 time line.

Comments:

Step	Action/Expected Response	Response Not Obtained

CAUTION RCS bleed and feed criteria must be monitored for immediate response if the criteria is exceeded.

3. **DETERMINE** if RCS bleed and feed required:

   a. **CHECK** RCS bleed and feed required:

   - Any THREE S/G WR levels less than or equal to 26% [36% ADV].

   OR

   - RCS pressure greater than or equal to 2335 psig.

   a. **MONITOR** RCS bleed and feed criteria:

   WHEN criteria are met, **THEN**

   PERFORM Substep 3b.

   b. **STOP** all RCPs, **AND**

   ** **GO TO** Cautions prior to Step 18 to initiate RCS bleed and feed.

   ** **GO TO** Step 4.

WBN BANK QUESTION

1. W/E05 EK2.2 016

Given the following conditions:

- The Crew has entered FR-H.1, "Loss of Secondary Heat Sink," due to a loss of inventory in the S/Gs and failure of the AFW pumps to start.

	SG Wide Range Levels (%)				*AFW*	*Cnmt*
Time	*SG#1*	*SG#2*	*SG#3*	*SG#4*	*Flow*	*Press*
0800	41	39	38	37	0 gpm	1.70 psid
0810	37	32	34	33	0 gpm	2.40 psid
0820	30	28	27	26	0 gpm	2.70 psid
0830	27	26	22	24	0 gpm	2.50 psid

Which ONE of the following is the earliest time, if any, that the initiation of Bleed and Feed is required?

A. Bleed and Feed **NOT** required.

B. 0810

C. 0820

D. 0830

V. TRAINING OBJECTIVES (continued)

AUO	RO	SRO	STA	
	X	X	X	4. Explain the basis for tripping the RCPs during the performance of FR-H.1.
	X	X	X	5. Identify the action that is taken if, during the performance of FR-H.1, it is determined that no CCPs are available for operation. Explain the basis for initiating the above referenced action.
	X	X	X	6. Given a set of plant conditions, use procedures FR-H.1, .2, .3, .4, & .5 to correctly identify any required procedure transition.
	X	X	X	7. Given a set of plant conditions, evaluate and determine when RCS bleed and feed should be initiated.
	X	X	X	8. Explain why RCS Bleed and Feed must be established quickly once the Bleed and Feed criteria in FR-H.1 has been met.
	X	X	X	9. Identify the minimum RCS feed path (ECCS Pumps) that must be established before a PZR PORV is opened to establish Bleed and Feed per FR-H.1.
	X	X	X	10. Deleted
	X	X	X	11. Deleted.
	X	X	X	12. [Discuss the adverse effects of S/G overfill (SOER 83-2, Rec. 13 e & f)].
	X	X	X	13. Justify the basis for checking RCS T-hot less than 545°F when determining and dealing with the cause of a S/G overpressure condition per FR-H.2.

18. W/E11 EK2.2 018
Given the following conditions:

0900 - A reactor trip occurs.

0910 - A small break LOCA occurs.

0930 - The crew transitions to ECA-1.1, "Loss of RHR Sump Recirculation," due to the failure of both RHR pumps.

0940 - Crew reduces ECCS flow to 1 SIP and 1 CCP per ECA-1.1.

0950 - Crew determines SI flow cannot be terminated due to lack of subcooling.

1000 - The crew is performing ECA-1.1 Step 19 RNO to establish the minimum required ECCS flow to remove decay heat.

Which ONE of the following identifies the minimum flow rate that meets the intent of ECA-1.1, Step 19 RNO, and requirements for using ECCS pumps in meeting this flow rate?

REFERENCE PROVIDED

A. Establish 460 gpm ECCS flow.
ECCS pumps may be started and stopped as necessary to accomplish the desired flow rate.

B. Establish 460 gpm ECCS flow.
ECCS pumps are **NOT** permitted to be started and stopped as necessary to accomplish the desired flow rate.

C. Establish 370 gpm ECCS flow.
ECCS pumps may be started and stopped as necessary to accomplish the desired flow rate.

D. Establish 370 gpm ECCS flow.
ECCS pumps are **NOT** permitted to be started and stopped as necessary to accomplish the desired flow rate.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because 460 gpm meets the curve requirement however it does not meet the intent of the step which is to meet the minimum flow requirements while still meeting the curve requirements. Second part correct in ECCS pumps may be started and stopped as necessary to accomplish the desired flow rate.*

B. *Incorrect, Plausible because 460 gpm meets the curve requirement however it does not meet the intent of the step which is to meet the minimum flow requirements while still meeting the curve requirements. Plausible if student does not know the pumps can be started or stopped pumps as necessary. The applicant may think that due to a LOCA pumps must not be stopped.*

C. *Correct, 0900 - 1000 (60 Min) Using ECA-1.1, Figure 1, the minimum value is slightly below 370 gpm. The Basis states "the operator is then instructed to establish the minimum ECCS flow needed to match decay heat in order to further decrease ECCS pump Flow and delay RWST depletion. This value of 370 gpm is in the acceptable region using the graph from time of trip and meets the requirement of Minimum Flow to delay RWST depletion. The ECCS pumps may be started and stopped as necessary to accomplish the desired flow rate.*

D. *Incorrect, Correct flow rate wrong action. Plausible if student does not know that the pumps can be started or stopped pumps as necessary. The applicant may think that due to a LOCA pumps must not be stopped.*

Question Number: 17

Tier: 1 **Group** 1

K/A: W/E11 EK2.2
Loss of Emergency Coolant Recirculation
Knowledge of the interrelations between the Loss of Emergency Coolant Recirculation and the following:
Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Importance Rating: 3.9 / 4.3

10 CFR Part 55: 41.7 / 45.7

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the questions requires knowledge of minimum ECCS flow during a loss of recirculation event. The candidate must understand proper operation (minimum flow for the

K/A Match: K/A is matched because the questions requires knowledge of minimum ECCS flow during a loss of recirculation event. The candidate must understand proper operation (minimum flow for the time since trip) for the ECCS system.

Technical Reference: ECA-1-1, Loss of RHR Sump Recirculation, Rev 12

Proposed references to be provided: ECA-1-1, Loss of RHR Sump Recirculation, Rev 12, (2 pages- page 12 and Figure 1 page 29 (with Acceptable Region and Unacceptable region deleted on the figure))

Learning Objective: 3-OT-ECA0101
02. Given the time of reactor trip, be able to use ECA-1.1 figure 1 to identify the minimum required SI flow.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: WBN bank question W/E11 EK2.2 017 used on AUDIT 08/2010

Comments:

Step	Action/Expected Response	Response Not Obtained

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

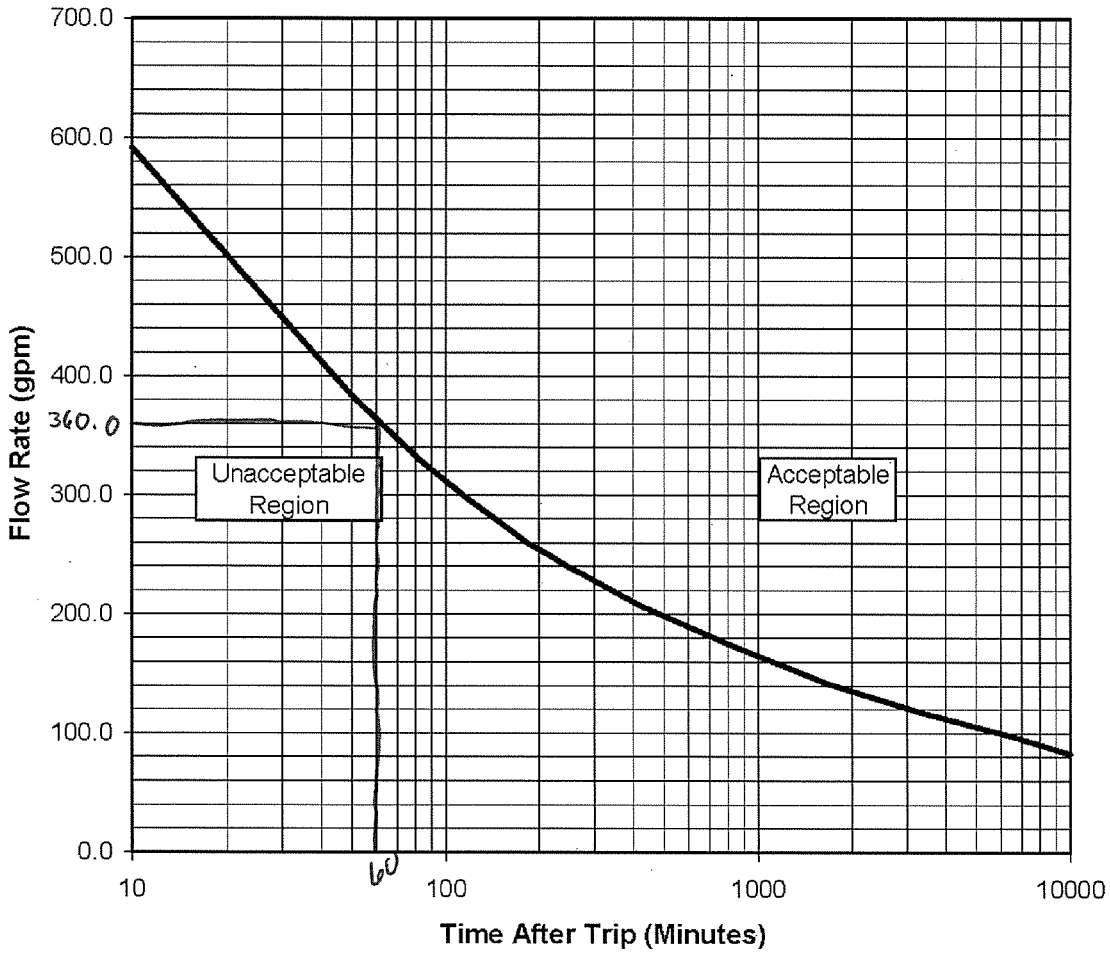
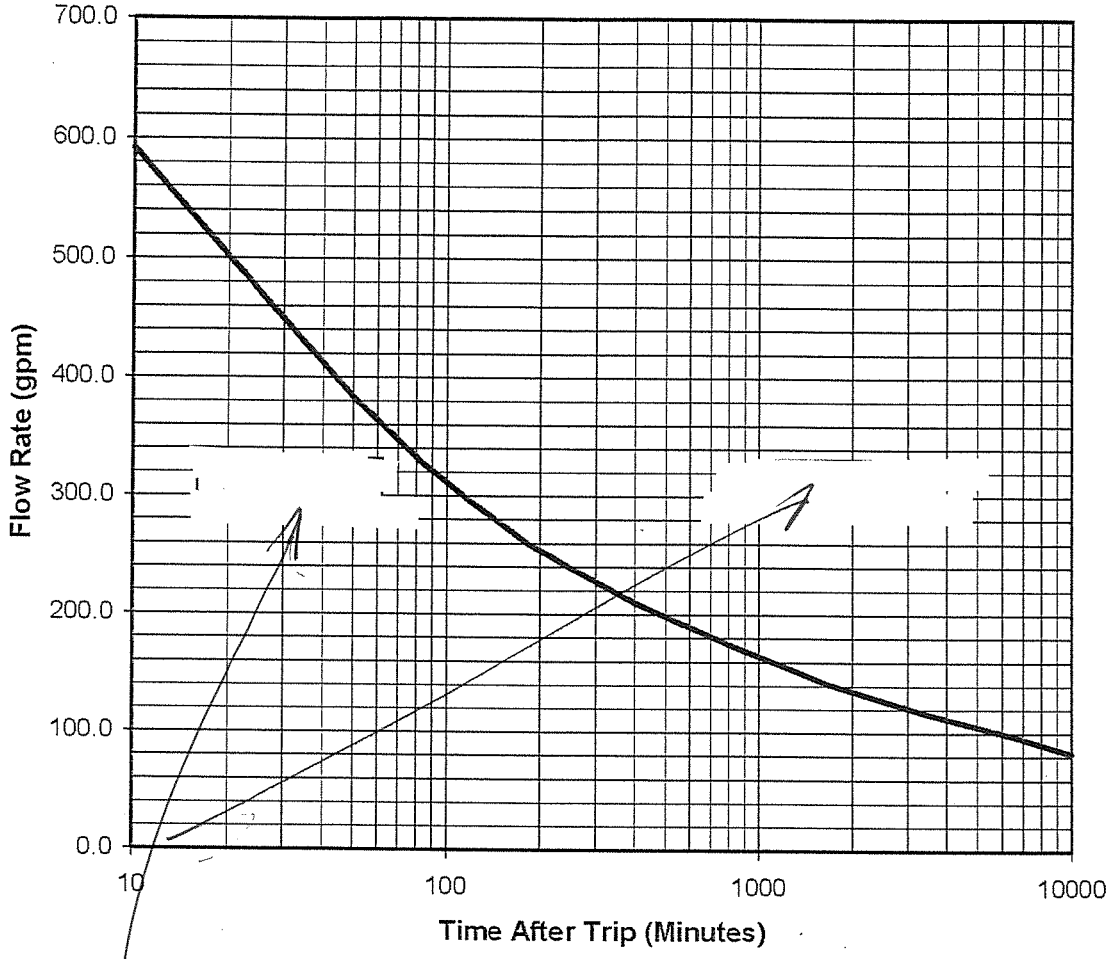


Figure 1
(Page 1 of 1)

Minimum SI Flow for Decay Heat vs. Time After Trip



PROPOSED REFERENCE PAGE 12 & this page with ACCEPTABLE & UNACCEPTABLE REGIO. Deleted

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.

WBN BANK QUESTION

1. W/E11 EK2.2 017

Given the following conditions:

0900 - A reactor trip occurs.
0910 - A small break LOCA occurs.
0930 - The crew transitions to ECA-1.1, "Loss of RHR Sump Recirculation," due to the failure of both RHR pumps.
0940 - Crew reduces ECCS flow to 1 SIP and 1 CCP per ECA-1.1.
0950 - Crew determines SI flow cannot be terminated due to lack of subcooling.
1000 - The crew is performing ECA-1.1 Step 19 RNO to establish the minimum required ECCS flow to remove decay heat.

Which ONE of the following identifies the minimum flow rate that meets the intent of ECA-1.1, Step 19 RNO, and requirements for using ECCS pumps in meeting this flow rate?

REFERENCE PROVIDED

A. Establish 460 gpm ECCS flow.
ECCS pumps may be started and stopped as necessary to accomplish the desired flow rate.

B. Establish 460 gpm ECCS flow.
ECCS pumps are **NOT** permitted to be started and stopped as necessary to accomplish the desired flow rate.

C. Establish 370 gpm ECCS flow.
ECCS pumps may be started and stopped as necessary to accomplish the desired flow rate.

D. Establish 370 gpm ECCS flow.
ECCS pumps are **NOT** permitted to be started and stopped as necessary to accomplish the desired flow rate.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible due to 460 gpm meets the curve requirment however it does not meet the intent of the step which is to meet the minimum flow requirements while still meeting the curve requirments. Second part correct in ECCS pumps may be started and stopped as necessary to accomplish the desired flow rate.*

B. *Incorrect, Plausible due to 460 gpm meets the curve requirment however it does not meet the intent of the step which is to meet the minimum flow requirements while still meeting the curve requirments. Plausible if student does not know the pumps can be started or stopped pumps as neccessary. The applicant may think that due to a LOCA pumps must not be stopped.*

C. *Correct, 0900 - 1000 (60 Min) Using ECA-1.1, Fiqure 1, the minimum value is slightly below 370 gpm. The Basis states "the operator is then instructed to establish the minimum ECCS flow needed to match decay heat in order to further decrease ECCS pump Flow and delay RWST depletion. This value of 370 gpm is in the acceptable region using the graph from time of trip and meets the requirement of Minimum Flow to delay RWST depletion. The ECCS pumps may be started and stopped as necessary to accomplish the desired flow rate.*

D. *Incorrect, Correct flow rate wrong action. Plausible if student does not know that the pumps can be started or stopped pumps as neccessary . The applicant may think that due to a LOCA pumps must not be stopped.*

Question Number: 17

Tier: 1 **Group** 1

K/A: W/E11 EK2.2
Loss of Emergency Coolant Recirculation
Knowledge of the interrelations between the Loss of Emergency Coolant Recirculation and the following:
Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Importance Rating: 3.9 / 4.3

10 CFR Part 55: 41.7 / 45.7

10CFR55.43.b: Not applicable

K/A Match:

Technical Reference: ECA-1-1, Loss of RHR Sump Recirculation, Rev 11

Proposed references to be provided: ECA-1-1, Loss of RHR Sump Recirculation, Rev 11, Figure 1 page 29

Learning Objective:

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: Modified the SQN NRC EXAM 1/2008 question, question orignally on WBN NRC EXAM 2006 as WE11G2.1.13.

Comments:

19. 001 AA1.07 019

Given the following:

- Unit 1 is operating at 60% power when the Control Rods begin an uncontrolled withdrawal at 72 steps/minute.
- The OAC places 1-RBSS, ROD BANK SELECT to MAN stopping the rod movement.
- The operating crew determines a failure of an RCS Loop 3 temperature input to the Rod Control System resulted in the uncontrolled rod movement.
- The Loop input to the system has been blocked in accordance with AOI-2, "Malfunction of Rod Control System."
- Current conditions are:
  - Tavg is 0.9°F above Tref.
  - ROD SPEED indicates 48 Steps/min.
  - PASSIVE SUMMER ROD DEMAND indicates slightly over +1.0.
  - Control Bank D rods are at 176 steps withdrawn.

If 1-RBSS, ROD BANK SELECT is returned to AUTO, which ONE of the following identifies how the control rods will respond?

Control Rod position will...

A. remain at 176 steps withdrawn.

B. decrease because of the Tavg-Tref error.

C. increase because the Rod Speed has not decayed to '0'.

D. decrease because of the PASSIVE SUMMER ROD DEMAND.

DISTRACTOR ANALYSIS:

A. *Correct, The rod position indications will not change when the rod control is restored to automatic with the conditions stated in the question.*

B. *Incorrect, Plausible for the rod position to decrease because the rods would insert if the delta between Tavg and Tref had been higher.*

C. *Incorrect, Plausible for the rod position to increase because the rods would withdraw if the demand signal had not decayed after the loop was defeated. However, the rod speed indication is not a function of the actual system demand signal.*

D. *Incorrect, Plausible for the rod position to decrease because the rods would insert if the Passive Summer Demand had been higher. However, it would have had to be a minimum of 1.5 to cause motion.*

Question Number: 19

Tier: 1 **Group:** 2

K/A: 001 AA1.07
Continuous Rod Withdrawal
Ability to operate and / or monitor the following as they apply to the Continuous Rod Withdrawal
RPI

Importance Rating: 3.3 / 3.1

10 CFR Part 55: 41.7 / 45.5 / 45.6

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires ability to determine how the control rod position indications will change when restoring to normal conditions following a Continuous Rod Withdrawal event in accordance with the abnormal operating procedure.

Technical Reference: AOI-2, Malfunction of Reactor Control System, Revision 0038

Proposed references to be provided: None

Learning Objective: 3-OT-AOI0200
04. Explain the Subsequent Actions for Continuous Rod Insertion/Withdrawal.

Cognitive Level:

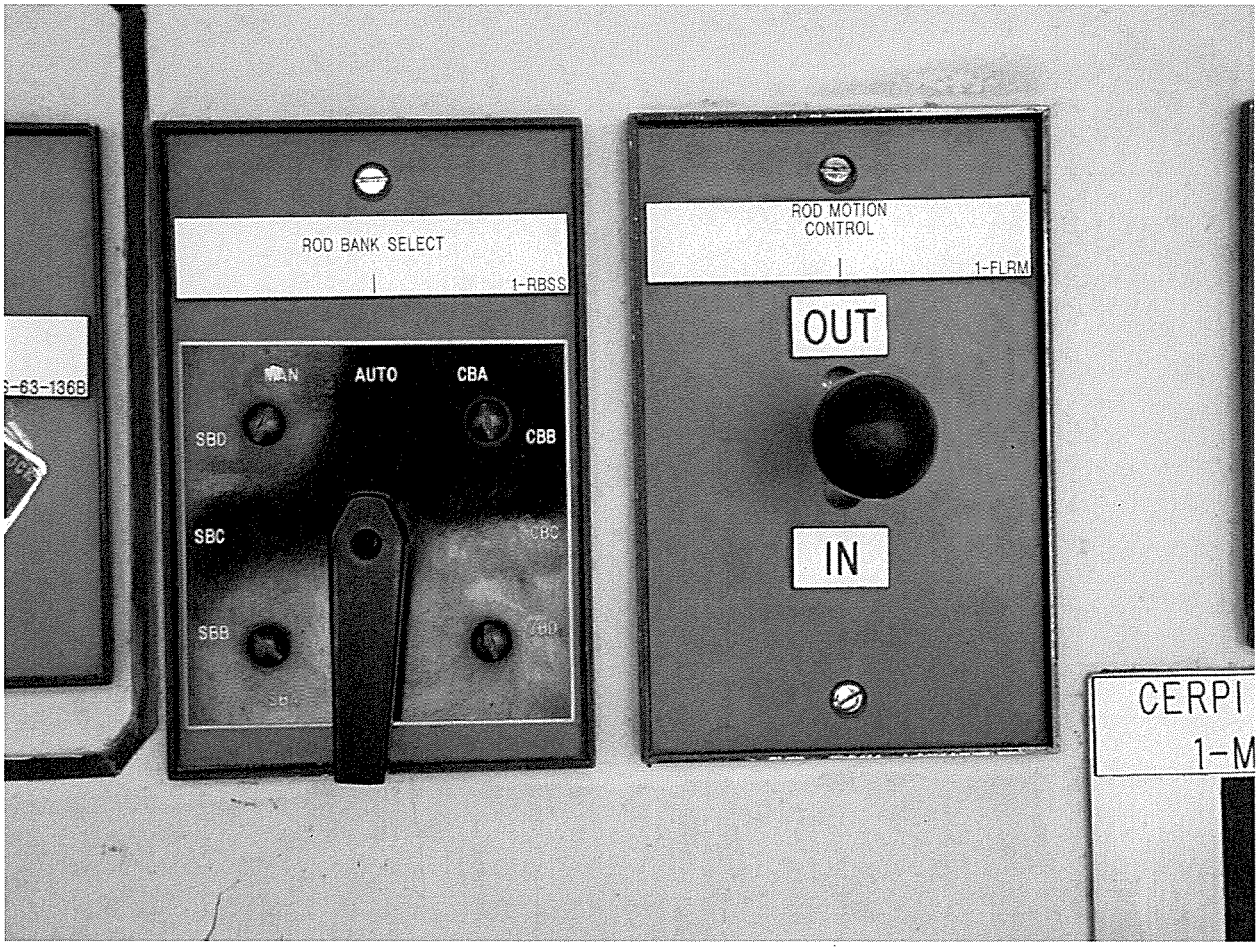
Higher	X
Lower	

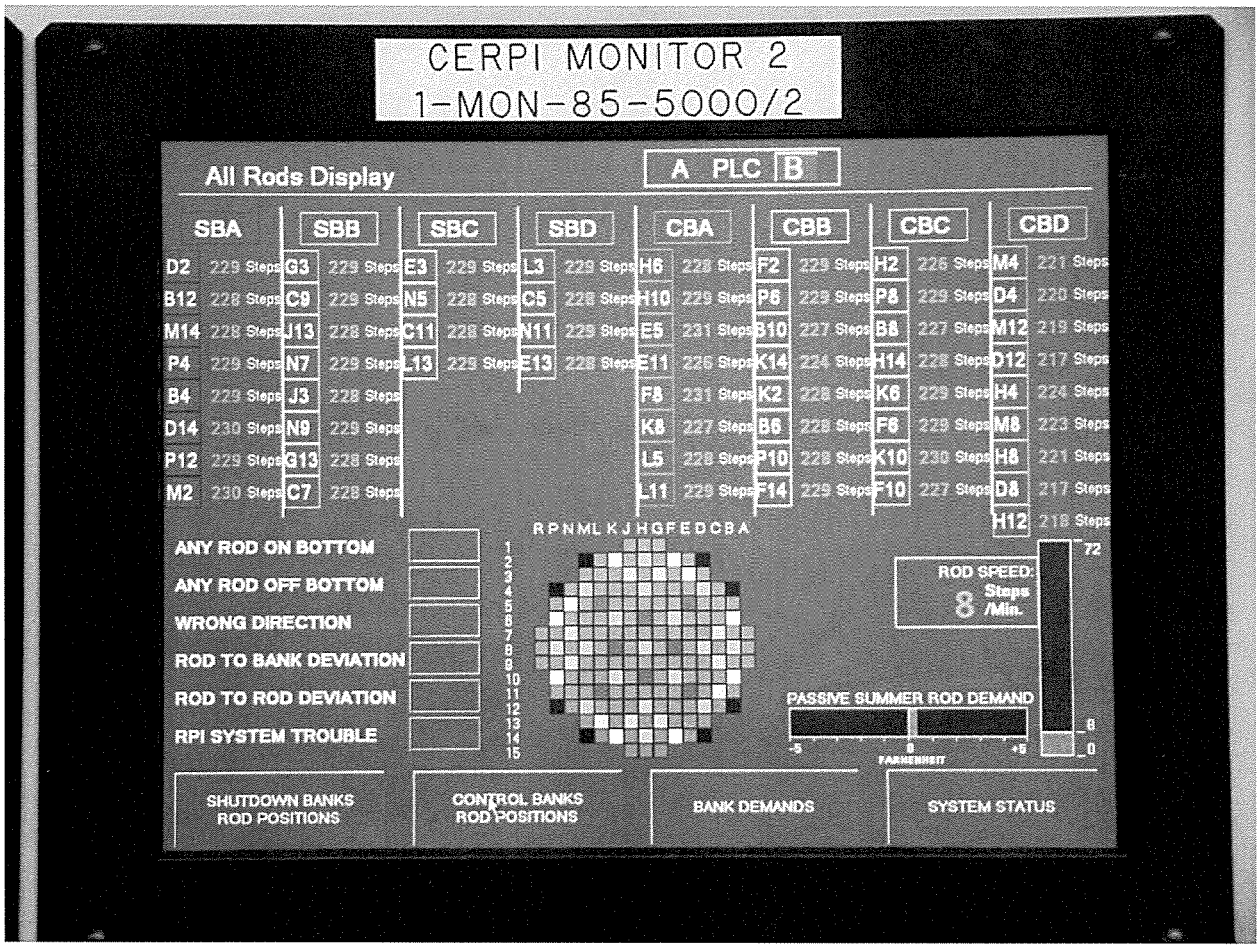
Question Source:

New	X
Modified Bank	
Bank	

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

Comments:


ROD BANK SELECT
1-RBSS
MAN
AUTO
CBA
SBD
CBB
SBC
CBC
SBB
CBD
ROD MOTION
CONTROL
1-FLRM
OUT
IN
CERPI
1-M


CERPI MONITOR 2
1-MON-85-5000/2
All Rods Display
A PLC B
SBA
SBB
SBC
SBD
CBA
CBB
CBC
CBD
D2 229 Steps
B12 228 Steps
M14 228 Steps
P4 229 Steps
B4 229 Steps
D14 230 Steps
P12 229 Steps
M2 230 Steps
G3 229 Steps
C9 229 Steps
J13 228 Steps
N7 229 Steps
J3 228 Steps
N9 229 Steps
G13 228 Steps
C7 228 Steps
E3 229 Steps
N5 228 Steps
C11 228 Steps
L13 228 Steps
L3 229 Steps
C5 228 Steps
N11 229 Steps
E13 228 Steps
H6 228 Steps
H10 229 Steps
E5 231 Steps
E11 226 Steps
F8 231 Steps
K8 227 Steps
L5 228 Steps
L11 229 Steps
F2 229 Steps
P6 229 Steps
B10 227 Steps
K14 224 Steps
K2 228 Steps
B6 228 Steps
P10 228 Steps
F14 229 Steps
H2 226 Steps
P8 229 Steps
B8 227 Steps
H14 228 Steps
K6 229 Steps
F6 229 Steps
K10 230 Steps
F10 227 Steps
M4 221 Steps
D4 220 Steps
M12 219 Steps
D12 217 Steps
H4 224 Steps
M8 223 Steps
H8 221 Steps
D8 217 Steps
H12 218 Steps
ANY ROD ON BOTTOM
ANY ROD OFF BOTTOM
WRONG DIRECTION
ROD TO BANK DEVIATION
ROD TO ROD DEVIATION
RPI SYSTEM TROUBLE
R P N M L K J H G F E D C B A
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15
ROD SPEED:
8 Steps /Min.
72
8
0
PASSIVE SUMMER ROD DEMAND
-5
0
+5
FAHRENHEIT
SHUTDOWN BANKS ROD POSITIONS
CONTROL BANKS ROD POSITIONS
BANK DEMANDS
SYSTEM STATUS

3.0 OPERATOR ACTIONS

3.1 Diagnostics

IF	GO TO SECTION	PAGE
Uncontrolled rod movement (Rod movement NOT due to actual T-ave to T-ref mismatch or change in reactor-turbine power)	3.2	6
Instrument failure (e.g. T-ave, NIS, PT-1-73) with Rod Control in MAN	3.2	6
Dropped RCCA (actual change in core power distribution parameters)	3.3	12
RCCA Misalignment in Modes 1 and 2	3.4	21
Rod Position Indicator (RPI) Malfunction (actual core power distribution parameters normal)	3.5	33
Failure of Control Rods to Move on Demand	3.6	36
RCCA Misalignment in Modes 3, 4, or 5	3.7	42

Step	Action/Expected Response	Response Not Obtained
	3.2 Uncontrolled Rod Bank Movement	
	NOTE Step 1 is an **IMMEDIATE ACTION** step	
1.	**STOP** uncontrolled rod motion:	
	a. **PLACE** control rods in MAN.	
	b. **CHECK** control rod movement STOPPED.	b. **(ρ) TRIP** reactor. ****GO TO** E-0, Reactor Trip or Safety Injection.
2.	**MAINTAIN** T-ave on PROGRAM. (Reference Attachment 1) • (ρ) **USE** control rods. OR • (ρ) **ADJUST** turbine load.	

Step	Action/Expected Response	Response Not Obtained

3.2 Uncontrolled Rod Bank Movement (continued)

3.	**CHECK** loop T-ave channels NORMAL.	**DEFEAT** failed channels: a. **PLACE** 1-XS-68-2D, ΔT CHANNEL DEFEAT, to failed channel position, **AND PULL.** b. **PLACE** 1-XS-68-2M, TAVG CHANNEL DEFEAT, in failed channel position **AND PULL.** **ENSURE** TR-68-2A placed to a Loop with operable ΔT/OTΔT/OPΔT channels using 1-XS-68-2B. **NOTIFY** Maintenance to implement IMI-160 for failed channel. **WHEN** at least 5 minutes have elapsed since failed T-ave channel is defeated **AND** 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.
4.	**CHECK** Auct T-avg NORMAL on 1-TR-68-2B.	**CONTROL** PZR level in MAN with 1-FCV-62-93 and 1-FCV-62-89. (Reference Attachment 1)
5.	**CHECK** NIS power range channels NORMAL.	****GO TO** AOI-4, Nuclear Instrumentation Malfunction.

I. PROGRAM

Watts Bar Operator Training

II. COURSE

License Training

III. TITLE

AOI-2, Malfunction of Reactor Control System

IV. LENGTH OF LESSON

License Training 2 hours

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	01. Describe the Purpose/goal of AOI.
	X	X	X	02. Identify Alarms for a Continuous Rod Withdrawal/Insertion.
	X	X	X	03. Describe Initial Actions for Continuous Rod Withdrawal.
	X	X	X	04. Explain the Subsequent Actions for Continuous Rod Insertion/Withdrawal.
X	X	X	X	05. Discuss the Symptoms of a Dropped RCCA.
X	X	X	X	06. [Describe adverse effects of a Misaligned rod at power. (SOER 84-02, Rec 7b)]
X	X	X	X	07. Identify Indications of RCCA Misalignment.
	X	X	X	08. Explain Operator Actions for an RCCA Misalignment.
	X	X	X	09. [Describe adverse effects of withdrawing affected bank after realigning it with a low rod. (SOER 84-02, Rec 7a & 7b)]
X	X	X	X	10. Identify Operator Actions for failure of Rods to Move on Demand.

AUO	RO	SRO	STA	
	X	X	X	11. Describe the Symptoms of an RPI malfunction.
	X	X	X	12. Given a set of plant conditions, use the AOI to correctly: a. Recognize Entry Conditions. b. Identify Required Actions. c. Respond to Contingencies (RNO). d. Observe and Interpret Cautions and Notes.
	X	X	X	13. Describe plant conditions which require Manual Reactor trip due to malfunctions of Reactor Control System.

20. 036 AK3.03 020

Given the following:

- During receipt of new fuel, a non-irradiated fuel assembly is dropped on the refueling floor.
- A plant announcement is made per AOI-29, "Dropped or Damaged Fuel or Refueling Cavity Seal Failure," to evacuate the area.

Which ONE of the following identifies why AOI-29 requires the area to be evacuated?

A. Potential high radiation rates

B. Potential exposure to Alpha particles

C. Potential transporting of hot particles

D. Potential exposure to Beta particles

DISTRACTOR ANALYSIS:

A. Incorrect, Plausible because increasing radiation rates are normally associated with dropped fuel assemblies, and would be correct if an irradiated assembly had been damaged,

B. Correct, AOI-29 section 4.0, "Discussion," states that, "Radiation levels, in general, should not increase, but Alpha (α) radiation may be produced from release of tramp uranium and/or fuel."

C. Incorrect, Plausible because 'hot particles' is a concern around other components in the plant.

D. Incorrect, Plausible because Beta is a type of radiation and is a particle like Alpha.

Question Number: 20

Tier: 2 **Group** 1

K/A: 036 AK3.03
Fuel Handling Incidents
Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents:
Guidance contained in EOP for fuel handling incident.

Importance Rating: 3.7 / 4.1

10 CFR Part 55: 41.8 / 41.10 / 45.6 / 45.13

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires knowledge of why AOI-29 requires the evacuation of the area during a fuel handling incident.

Technical Reference: AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure, Revision 0021

Proposed references to be provided: None

Learning Objective:
6. Demonstrate ability/knowledge of AOI-29, to correctly:
a. Recognize Entry conditions.
b. Respond to Action steps.
c. Respond to Contingencies (RNO column).
d. Respond to Notes and Cautions.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	X
Modified Bank	
Bank	

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

Comments:

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS

3.1 Diagnostics

NOTE

- Visual confirmation of a dropped or damaged fuel assembly or Refueling Cavity Seal failure will come from the Refueling SRO or his designee.
- If fission products are released into the refuel water the RHR and Spent Fuel Pool Cooling Systems may exhibit high radiation levels.

IF	GO TO Section	Page
Refueling Cavity Seal has failed	3.2	5
IRRADIATED fuel assembly has been dropped or damaged INSIDE containment	3.3	10
IRRADIATED fuel assembly has been dropped or damaged OUTSIDE containment	3.4	13
NEW fuel assembly has been dropped or damaged	3.5	16
Spent Fuel Pit Gate leakage	3.6	18

Step	Action/Expected Response	Response Not Obtained

3.5 Dropped or Damaged New Fuel Assembly

CAUTION If irradiated fuel is involved entry into area shall be coordinated with Radiation Protection.

1. **EVACUATE** the affected area:
   - Fuel Handling SRO **NOTIFY** personnel in the general area of radiation concern.
   - **ANNOUNCE** for all personnel to evacuate the affected area.

2. **STOP** all non-fuel handling related work on the Refuel Floor UNTIL damaged assembly is inspected and stored.

3. **REFER TO** EPIP-10, Medical Emergency Response, to treat any injured personnel.

4. **NOTIFY** Shift Manager of dropped or damaged fuel assembly.

5. **NOTIFY** Radiation Protection of dropped or damaged fuel assembly, **AND**

   REQUEST survey of El 757, Refuel Floor.

6. **REFER TO** EPIP-1, Emergency Plan Classification Flowchart.

Step	Action/Expected Response	Response Not Obtained

3.5 Dropped or Damaged New Fuel Assembly (continued)

7. **WHEN** Radiation Protection grants approval, **THEN**

   INSPECT fuel assembly visually and determine extent of damage.

8. **REFER TO** FHI-1, Receiving, Returning, Inspecting, and Storing New Fuel and Insert, for handling dropped or damaged fuel assembly.

9. **GO TO** Section 3.7, Recovery.

End of Section

4.0 DISCUSSION

A. Section 3.2, Refueling Cavity Seal Failure, provides guidance to:

- Set down, into an approved storage location, any fuel assembly the Refueling Machine may be carrying.
- Pick up any fuel assembly in the RCCA Change Fixture and place it into any approved location in the Rx Vessel.
- Make up to the SFP and/or Refueling Cavity.
- Limit the loss of water from the SFP.

A maximum leak rate of 3176 gpm has been calculated assuming a deflated seal and a resulting 1/16" gap between the seal and vessel. With the Fuel Transfer Tube closed, this calculation provides 70 minutes for the refueling cavity to drain to the Rx Vessel flange elevation.

The following elevations and dimensions are provided to aid decision making in handling the seal failure:

Refuel Floor	El 757
Rx Vessel Flange	El 725' 1.5"
Top of Upender, vertical	El 725' 1.75"
Top of Spent Fuel Pit racks	El 723' 3.75"
Flange to Upper Internals storage area	11' 3"
Flange to Transfer Canal	15' 10.75"
Height of Rx Vessel Head Guide Studs	26' 9.6"

B. Section 3.3, Dropped or Damaged Irradiated Fuel Assembly in Containment, provides guidance to reduce the release of radioactive material if an irradiated fuel assembly is dropped or damaged during handling in the refueling cavity.

C. Section 3.4, Dropped or Damaged Irradiated Fuel Assembly in the Spent Fuel Pit Area, provides guidelines to reduce the release of radioactive material if an irradiated fuel assembly is dropped or damaged during handling in the Spent Fuel Pit or Transfer Canal areas.

D. Section 3.5, Dropped or Damaged New Fuel Assembly, provides guidelines to handle a non-irradiated fuel assembly. Radiation levels, in general, should **NOT** increase, but Alpha (α) radiation may be produced from release of tramp uranium and/or fuel.

I. **PROGRAM:**

Watts Bar Operator Training

II. **COURSES:**

A. License Training
B. NOTP
C. Licensed Operator Requal
D. AUO Requal

III. **TITLE:**

AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure

IV. **LENGTH OF LESSON:**

A. License Training	1 Hour
B. NOTP	4 Hours
C. Initial License Training	2 Hours

Licensed Operator and AUO REQUAL times will be determined after objectives are identified.

V. **TRAINING OBJECTIVES:**

AUO	RO	SRO	STA	
X	X	X	X	1. Demonstrate knowledge of purpose/goal of AOI-29.
	X	X	X	2. Identify 2 Alarms that indicate Refueling Cavity Seal Failure.
	X	X	X	3. Identify 5 Alarms that indicate Dropped Fuel Assembly (inside or outside Cntmt).
	X	X	X	4. Identify possible Auto Actions from: a. Dropped fuel assembly in SFP area. b. Dropped fuel assembly in Cntmt.
				5. Deleted (10/04/93)
	X	X	X	6. Demonstrate ability/knowledge of AOI-29, to correctly: a. Recognize Entry conditions. b. Respond to Action steps. c. Respond to Contingencies (RNO column). d. Respond to Notes and Cautions.

21. 037 AK1.01 021

Given the following:

- Unit 1 was operating at 100% when a steam generator tube leak occurred.
- The unit has been placed in Mode 3.
- The leaking steam generator pressure is 950 psig.
- The operating crew is currently determining the target incore temperature for RCS cooldown, per the following table in AOI-33, "Steam Generator Tube Leak."

LEAKING SG PRESSURE (PSIG)	TARGET INCORE TEMP (°F)
1100	491°F
1000	479°F
900	466°F
800	451°F
700	434°F
690	433°F

Which ONE of the following identifies the steam pressure setpoint on the steam dump system that would maintain the target incore temperature?

A. 480 psig

B. 495 psig

C. 545 psig

D. 560 psig

DISTRACTOR ANALYSIS:

A. *Correct, The given S/G pressure of 950 psig requires the operator to use the target of 466°F as the step to determine the target says that if the ruptured S/G pressure is between the listed values, then use the lower value. From the steam tables, 466°F corresponds to 495 psia, which is approximately 480 psig.*

B. *Incorrect, Plausible because the ruptured S/G pressure of 950 psig corresponds to a target of 466°F, which would give a saturation pressure of 495 psia. Failure to convert psia to psig will result in 495 psig appearing correct.*

C. *Incorrect, Plausible if the applicant chooses the higher value (1000 psig) from the table rather than the correct, lower value (950 psig). 1000 psig gives a target temperature of 479°F which corresponds to a pressure of 560 psia, which is approximately 545 psig.*

D. *Incorrect, Plausible if the applicant chooses the higher value (1000 psig) from the table rather than the correct, lower value (950 psig). 1000 psig gives a target temperature of 479°F which corresponds to a pressure of 560 psia. Failure to convert psia to psig will result in 545 psig appearing correct.*

Question Number: 21

Tier: 1 **Group** 2

K/A: 037 AK1.01
Steam Generator (S/G) Tube Leak
Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak:
Use of steam tables.

Importance Rating: 2.9* / 3.3

10 CFR Part 55: 41.8 / 41.10 / 45.3

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires knowledge of how to use the steam tables to determine setpoint (steam dumps) used in the plant for a steam generator tube leak.

Technical Reference: AOI-33, Steam Generator Tube Leak, Revision 0034

Proposed references to be provided: Steam Tables

Learning Objective: 3-OT-AOI3300
06. Given the leaking SG's pressure, determine Target Incore Temperature for RCS depressurization.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: WBN bank question EOP0300 127 modified for the NRC 10/2011 exam.

Comments:

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS (continued)

25. **DETERMINE** Target Incore Temp For RCS Cooldown:

- **IF** leaking SG pressure is between listed values,
  THEN
  USE lower value.

LEAKING SG PRESSURE (PSIG)	TARGET INCORE TEMP (°F)
1100	491°F
1000	479°F
900	466°F
800	451°F
700	434°F
690	433°F

26. **CHECK** Shutdown Margin ADEQUATE for initial cooldown.

DO NOT CONTINUE until Shutdown Margin is adequate for initial cooldown.

WBN BANK QUESTION

EOP0300 127

Given the following plant conditions:

- S Steam Generator Tube Rupture has occurred.
- E-3, Steam Generator Tube Rupture is in progress.
- The operators have completed the RCS cooldown to the target incore temperature of 480°F.

Which of the following identifies the Steam Dumps pressure setpoint to control RCS temperature at 480°F?

a. 580 - 585 psig.

b. 550 - 555 psig.

c. 580 - 585 psia.

d. 550 - 555 psia.

I. PROGRAM

Watts Bar Operator Training

II. COURSES

License Training

III. TITLE

AOI-33, Steam Generator Tube Leak

IV. LENGTH OF LESSON:

License Training 1.5 Hours

V. TRAINING OBJECTIVES:

AUO	RO	SRO	STA	
	X	X	X	1. Identify the purpose/goal of AOI-33.
	X	X	X	2. Identify Alarms that indicate a Steam Generator (SG) Tube Leak.
	X	X	X	3. Identify Auto Action that occurs on a SG Tube Leak.
	X	X		4. Objective deleted.
	X	X	X	5. Describe the means of identifying a S/G Tube Leak.
	X	X	X	6. Given the leaking SG's pressure, determine Target Incore Temperature for RCS depressurization.
	X	X	X	7. Identify the 2 conditions given in AOI-33 that require the operator to stop RCS depressurization.
	X	X	X	8. Given a set of plant conditions, use AOI-33 to correctly: a. Recognize Entry Conditions. b. Identify Required Actions. c. Respond to Contingencies (RNO). d. Observe and Interpret Cautions and Notes.
	X	X	X	9. Identify the methods available to quantify a S/G Tube Leak as given in AOI-33.
	X	X	X	10. Identify when AOI-33, Appendix A, SG Tube Leak Monitoring and Data Sheet 1 is terminated once a S/G tube leak has been identified.

22. 051 AA2.02 022

Given the following plant conditions:

- Main Turbine load is 660 MWe.
- Condenser vacuum is decreasing at 0.1 in-HgA/min due to air in-leakage.
- The crew has entered AOI-11, "Loss of Condenser Vacuum."
- Currently Condenser pressures are:

Zone 'A'	Zone 'B'	Zone 'C'
3.75 in-HgA	4.93 in-HgA	6.06 in-HgA

Which ONE of the following identifies...

(1) the status of the turbine automatic 'low vacuum trip' function

and

(2) the condenser pressure that is currently above the setpoint requiring a reactor trip in accordance with AOI-11?

REFERENCE PROVIDED

A. (1) The turbine 'Low Vacuum' trip has failed.
(2) Zone 'B'

B. (1) The turbine 'Low Vacuum' trip has **NOT** failed.
(2) Zone 'B'

C. (1) The turbine 'Low Vacuum' trip has failed.
(2) Zone 'C'

D. (1) The turbine 'Low Vacuum' trip has **NOT** failed.
(2) Zone 'C'

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because the turbine 'Low Vacuum' trip automatic function is set above the values requiring a manual trip (which is not normally the way systems are set up) and the pressure in Zone 'B' being greater than the setpoint allowed is correct.*

B. *Correct, The turbine automatic trip function has not failed. It is set to trip the turbine as the pressure rises somewhere between 6 and 12 in HgA in accordance with AOI-17, "Turbine Trip." the pressure in Zone 'B' is greater than the allowed pressure and with the load being 660 MWe the reactor power is greater than 50%, thus a reactor trip is required.*

C. *Incorrect, Plausible because the turbine 'Low Vacuum' trip automatic function is set above the values requiring a manual trip (which is not normally the way systems are set up) and with the information removed from the provided reference identifying what the lines are representing is not recalled correctly, the wrong zone could be chosen.*

D. *Incorrect, Plausible because the turbine 'Low Vacuum' trip automatic function not failing is correct (See A above.) and with the information removed from the provided reference identifying what the lines are representing is not recalled correctly, the wrong zone could be chosen.*

Question Number: 22

Tier: 1 **Group** 2

K/A: 051 AA2.02
Loss of Condenser Vacuum
Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum:
Conditions requiring reactor and/or turbine trip.

Importance Rating: 43.5 / 45.13

10 CFR Part 55: 3.9 / 4.1

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires the knowledge of the conditions where an automatic turbine trip will occur and the conditions that require a manual trip of the turbine/reactor to be initiated.

Technical Reference: AOI-17, Turbine Trip, Rev. 0048
AOI-11, Loss of Condenser Vacuum, Revision 0029

Proposed references to be provided: AOI-11, Appendix A, Condenser Vacuum ICS Graph (1 page with the Condenser Zone identifiers deleted)

Learning Objective: 3-OT-AOI1100
2. Identify Conditions that must be satisfied for Continuous Low-Load operation of the Turbo-Generator.
4. Describe the automatic Low-Vacuum Turbine Trip.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	X
Modified Bank	
Bank	

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

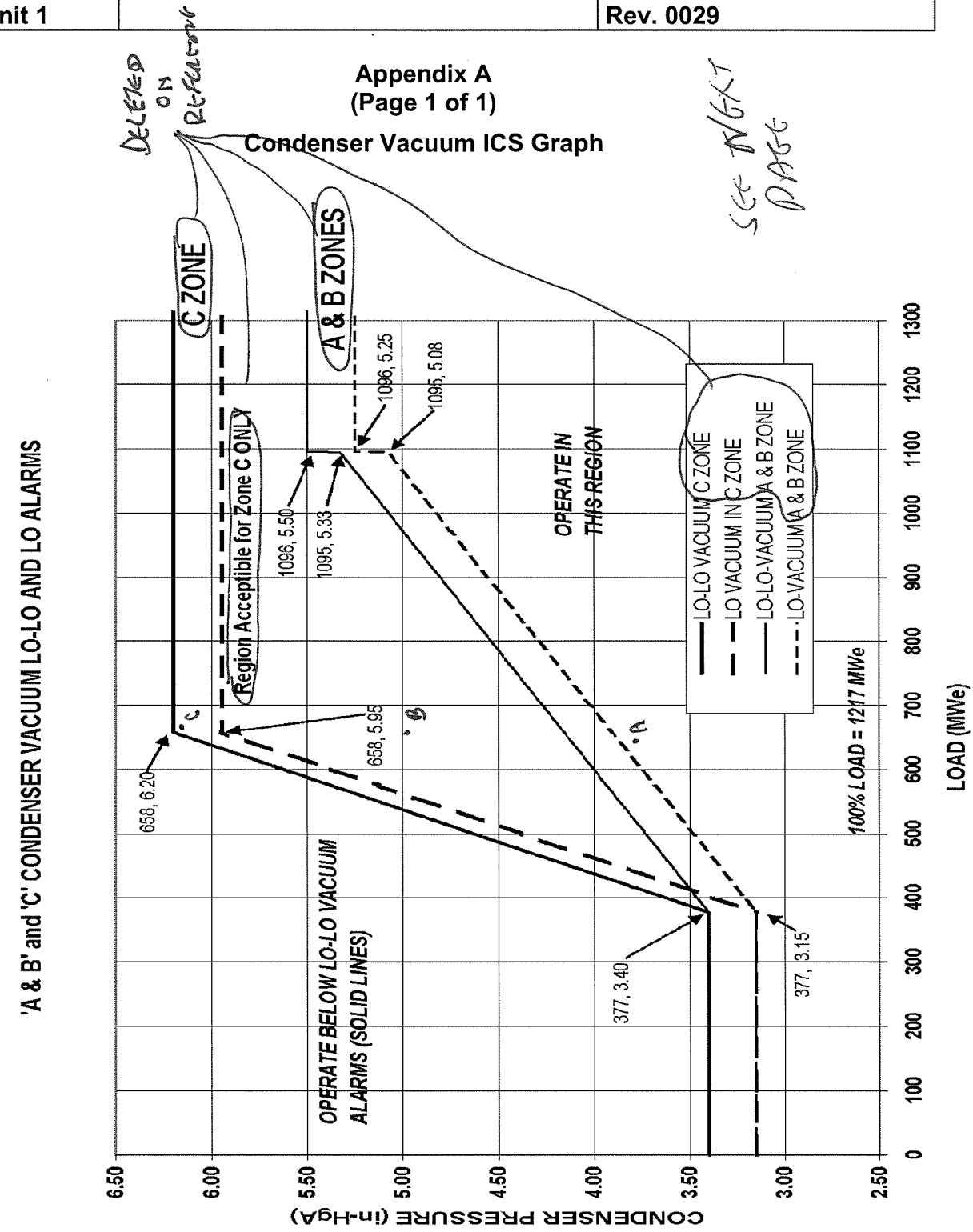
Comments:

4.0 DISCUSSION (continued)

Automatic Turbine Trip occurs from any of the following:

1.	AUTO STOP OIL press	(less than 45 psig)
2.	BEARING OIL PRESS LO	(less than 7 psig)
3.	EXCESS THRUST BEARING WEAR	(greater than 60 psig oil press)
4.	CONDENSER VACUUM LO	(greater than 6 to 12 in Hga)
5.	MFPT 1A AND 1B TRIPPED	
6.	GENERATOR COOLING FAILURE	(If greater than 15% load; 45 sec TD)
7.	OVERSPEED	(110% mechanical, 111% electrical)
8.	EHC POWER FAILURE	(loss of ±15V, or 48V dc supply)
9.	SSPS TRAIN A	(Rx trip-P4 contacts)
10.	SSPS TRAIN B	(Rx trip-P4 contacts)
11.	SAFETY INJECTION	K621 (FWIS) relay
12.	S/G LEVEL HI-HI	K621 (FWIS) relay (≥ 82.4% NR)
13.	ELECTRICAL TROUBLE	
14.	AMSAC	(40% turbine load and S/G [3/4] NR level less than 12%; 25 sec TD.)
15.	GENERATOR BREAKER OPEN	

Appendix A
(Page 1 of 1)
Condenser Vacuum ICS Graph

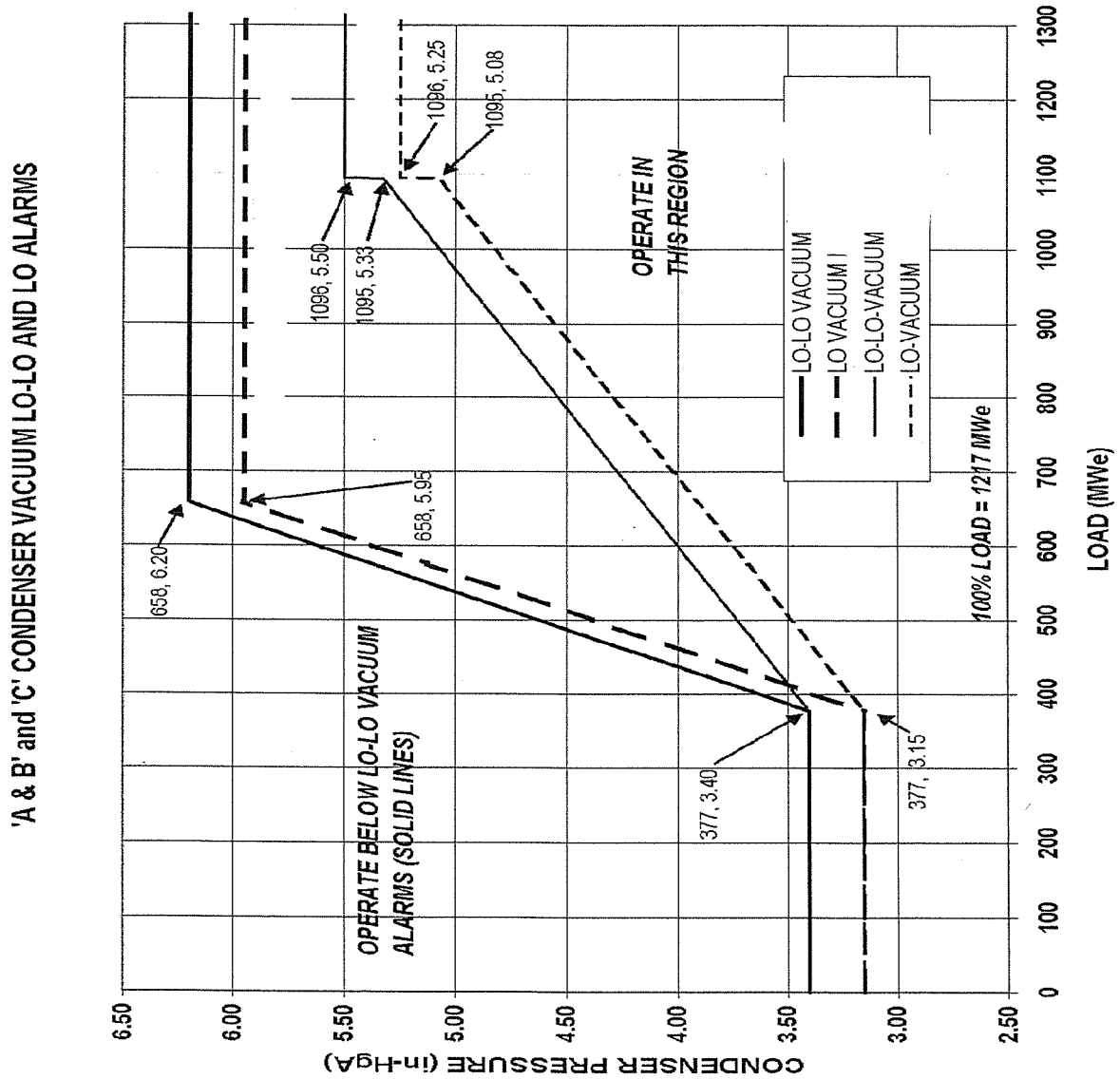


NOTE: This Appendix is an ICS Controlled Graph and should **NOT** be modified without Corporate Computer Engineering acknowledgement.

Appendix A
(Page 1 of 1)

Condenser Vacuum ICS Graph

Reference Proposed To Be Supplied



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

I.PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSES

LICENSE TRAINING

NON-LICENSE TRAINING

III. TITLE

AOI- 11, Loss of Condenser Vacuum

IV. LENGTH OF LESSON

LICENSE TRAINING	1 hour
NON-LICENSE TRAINING	1 hour

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	1. Describe the Purpose/Goal of AOI-11.
	X	X	X	2. Identify Conditions that must be satisfied for Continuous Low-Load operation of the Turbo-Generator.
X	X	X	X	3. List 4 Checks an NAUO makes to determine/correct the Cause of Low Vacuum.
	X	X	X	4. Describe the automatic Low-Vacuum Turbine Trip.
	X	X	X	5. Identify Alarms and Indications of a partial or complete Loss of Condenser Vacuum.

23. 068 AA2.03 023

Given the following:

- Unit 1 was operating at 100% power when a Main Control Room evacuation was performed in accordance with AOI-30.2, "Fire Safe Shutdown," Series procedure C.69, "Fire Safe Shutdown Control Building."

Which ONE of the choices below completes the following statement relative to the available indications for Tcold and Thot in the Aux. Control Room?

Tcold is determined using _____(1)_____ installed instrumentation.

Thot is determined using _____(2)_____ installed instrumentation.

	(1)	(2)
A.	RCS temperature	Incore Thermocouple
B.	RCS temperature	RCS temperature
C.	SG pressure	Incore Thermocouple
D.	SG pressure	RCS temperature

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because while Thot instruments are available, RCS Tcold instruments are not available in the ACR and while using an alternate parameter is sometimes used (as in Tcold), Thot is not determined using Incore Thermocouple instruments. Incore thermocouple temperatures are not available in the ACR.*

B. *Incorrect, Plausible because while Thot instruments are available, RCS Tcold instruments are not available in the ACR and using the installed RCS indicators to determine Thot is correct.*

C. *Incorrect, Plausible because using associated loop SG pressure instrument to determine Tcold is correct but while using an alternate parameter is sometimes used (as in Tcold), Thot is not determined using Incore Thermocouple instruments. Incore thermocouple temperatures are not available in the ACR.*

D. *Correct, As identified in Table 1 of the procedure, the Tcolds are determined using the Tsat scale on the associated SG pressure instrument and the Hot leg temperatures are determined using installed temperature indicators on each of the RCS loops.*

Question Number: 23

Tier: 1 **Group:** 2

K/A: 068 AA2.03
Control Room Evacuation
Ability to determine and interpret the following as they apply to the Control Room Evacuation:
T-hot, T-cold, and in-core temperatures

Importance Rating: 4.0 / 4.2

10 CFR Part 55: 43.5 / 45.13

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires the ability to identify the instruments used following a control room evacuation to determine the status of Tcold and Thot.

Technical Reference: AOI-30.2, Fire Safe Shutdown, Revision 0031
AOI-30.2 C.69, Fire Safe Shutdown Control Building, Revision 0005
Auxiliary Control Room pictures

Proposed references to be provided: None

Learning Objective: 3-OT-AOI3000
12. Demonstrate Ability/knowledge of AOI-30.1 and 30.2 by:
    a. Recognizing entry conditions
    b. Responding to required actions of the AOI
    c. Responding to contingencies (RNO)
    d. Responding to Notes/Cautions

Cognitive Level:

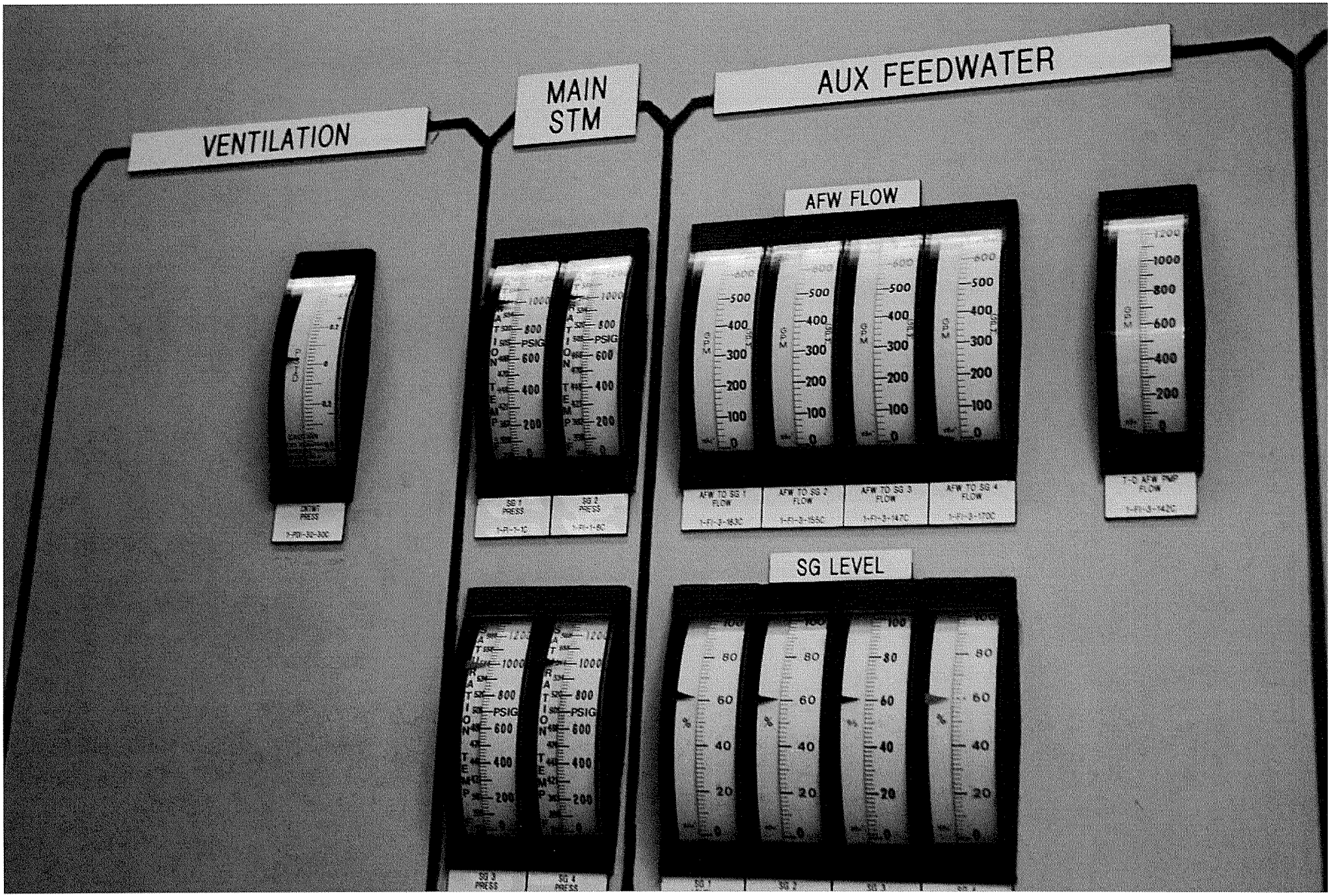
Higher	
Lower	X

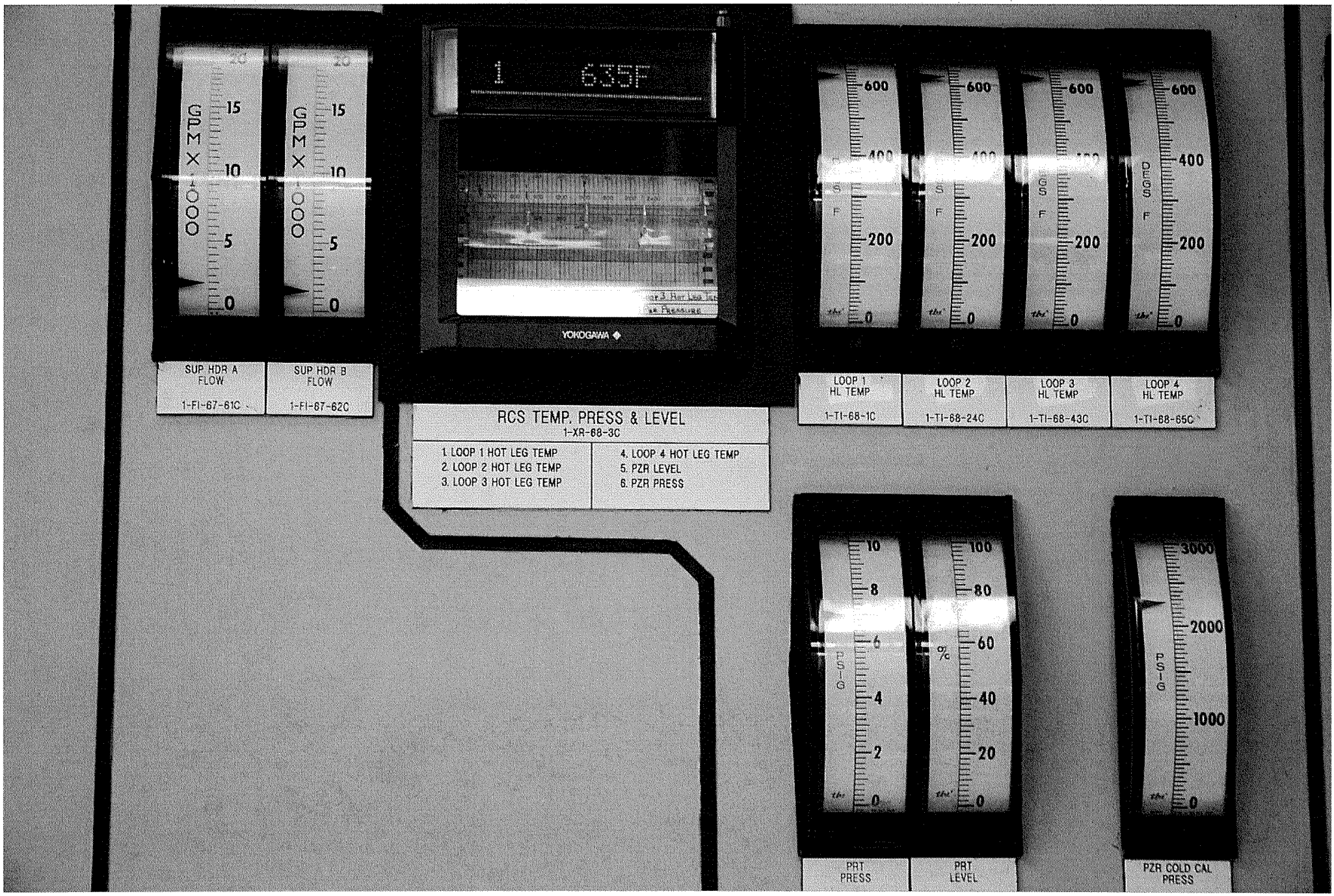
Question Source:

New	X
Modified Bank	
Bank	

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

Comments:


VENTILATION
MAIN STM
AUX FEEDWATER
AFW FLOW
SG LEVEL
PSIG
AFW TO SG 1 FLOW
AFW TO SG 2 FLOW
AFW TO SG 3 FLOW
AFW TO SG 4 FLOW
SG 1 PRESS
SG 2 PRESS
SG 3 PRESS
SG 4 PRESS


SUP HDR A FLOW
1-FI-67-61C
SUP HDR B FLOW
1-FI-67-62C
YOKOGAWA
RCS TEMP, PRESS & LEVEL
1-XR-68-3C
1. LOOP 1 HOT LEG TEMP
2. LOOP 2 HOT LEG TEMP
3. LOOP 3 HOT LEG TEMP
4. LOOP 4 HOT LEG TEMP
5. PZR LEVEL
6. PZR PRESS
LOOP 1 HL TEMP
1-TI-68-1C
LOOP 2 HL TEMP
1-TI-68-24C
LOOP 3 HL TEMP
1-TI-68-43C
LOOP 4 HL TEMP
1-TI-68-65C
PRT PRESS
PRT LEVEL
PZR COLD CAL PRESS

4.0 OPERATOR ACTIONS

NOTE

- The decision to trip the unit and declare an Appendix R fire is left to the judgment of the Unit SRO/SM and must be based on the magnitude of the fire and its potential effect on the equipment/components necessary to achieve and maintain cold shutdown
- For an Appendix R fire, this procedure takes precedence over the Emergency Operating Procedures
- AUO local operator actions should be assigned as early as possible by an SRO or UO NOT involved with immediate actions of this procedure.

ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1) **DETERMINE** the fire location has the potential to affect equipment needed for safe shutdown.	**RETURN** to AOI 30.1
2) **CHECK** either 1A or 1B CCS Pump RUNNING.	**START** either 1A CCS Pump or 1B CCS Pump, from either the MCR or locally.
3) **CHECK** 1 FCV 67 143 A, CCS HX A ERCW OUTLET FLOW CNTRL OPEN.	**OPEN** 1 FCV 67 143 A, CCS HX A ERCW OUTLET FLOW CNTL and **THROTTLE** as required.
4) **REFER** to AOI-30.2 APP B, Elevation Diagrams, to determine applicable AOI-30.2 C-Series procedure.	
5) **ANNOUNCE** Appendix R fire over the PA system.	
6) **IF** fire requires MCR evacuation, **THEN** **PERFORM** AOI-30.2 C.69 WHILE continuing in this procedure.	**PERFORM** the applicable AOI-30.2 C-Series procedure for the identified fire location, **WHILE** continuing in this procedure.

1.0 MAIN/AUXILIARY CONTROL ROOM OPERATOR ACTIONS

CAUTION

Handswitches are positioned to ensure equipment is in its required safe shutdown configuration, by ensuring the proper signal is maintained on each device. The equipment may appear to be in the condition called for in the procedure step, but the indication may be false due to faulted control circuits or instrumentation.

NOTES

1) Based on the location and severity of the fire, Operations personnel must determine which Train of equipment is reliable for plant cooldown. Some action steps (such as VCT isolation) require this information be passed to AUOs for manual actions.

2) Alternate means (flow, pressure, amps, etc.) may be needed for checking component or system status.

3) Table 1 is a listing of instrumentation which should remain available during a fire in this location.

4) Table 2 is a summary of AUO assignments.

5) Table 3 provides a log for RCS / Pzr Cooldown.

[1] **ENSURE** Reactor trip:

- Reactor trip and bypass breakers OPEN.
- RPIs at bottom of scale.
- Neutron flux dropping.

CAUTION

A Spurious SI may occur due a fire in this area.

NOTE

A fire in the vicinity of 1-M-5 could affect both the RCP MCR trip circuits and the Pressurizer Spray Valve control circuits.

[2] **STOP** RCPs [1-M-5].

1.0 MAIN/AUXILIARY CONTROL ROOM OPERATOR ACTIONS (continued)

NOTE

Steps 1.0[4] thru 1.0[8] are to be performed only if time is available prior to abandoning the MCR.

[3] **OPEN** 1-TCV-70-192, LETDOWN HX OUTLET TEMP CNTL by placing 1-HIC-62-78A, LETDOWN HX OUTLET TEMP TCV-70-192 CNTL in MANUAL and reducing controller output to ZERO.

[4] **ENSURE** turbine trip and steam valves closed:

[4.1] **ENSURE** Turbine stop valves CLOSED.

[4.2] **CLOSE** MSIVs as follows:

[4.2.1] **LOWER** S/G PORV setpoints until PORVs open enough to allow steam dump closure.

[4.2.2] **PLACE** HS-1-103A and HS-1-103B, STEAM DUMP CONTROL switches to OFF.

[4.2.3] **CLOSE** all MSIVs.

CAUTION

Loss of Offsite Power may occur due to a fire in this room, but the D/Gs are credited.

[5] **ENSURE** 1A or 1B 6.9KV shutdown board energized:

- From CSST (offsite),

OR

- From D/G (blackout).

[6] **DEENERGIZE** the RCP Start Buses at the Electrical Control Board as follows:

[6.1] **PLACE** transfer switches RCP ST BUS A and RCP ST BUS B in MAN.

1.0 MAIN/AUXILIARY CONTROL ROOM OPERATOR ACTIONS (continued)

[6.2] **TRIP** the following normal and alternate supply breakers:

- ACB-2514
- ACB-2612
- ACB-2512
- ACB-2614

[7] **CLOSE** PZR PORVs:

- **PLACE** 1-HS-68-334A, PZR PORV 334, to CLOSE.
- **PLACE** 1-HS-68-340AA, PZR PORV 340A, to CLOSE.

[8] **ANNOUNCE** "Abandoning Control Room" over PA system.

NOTES

1) All procedures addressed in this instruction are maintained in the ACR locker.
2) SM's Clerk may be contacted for radio and charger retrieval.

[9] **WHEN** evacuation of the MCR is required, **THEN**

PERFORM the following:

[9.1] **TAKE** the following:

- This procedure (AOI-30.2) and all Instructions in progress.
- All available radios.
- IC (Incident Commander) key ring.

[9.2] **PROCEED TO** the ACR.

[10] **PERFORM** ACR Checklist 30.2-6 immediately upon arrival.

[11] **MONITOR** plant parameters per TABLE 1, AVAILABLE INSTRUMENTATION.

Table 1
(Page 1 of 1)
AVAILABLE INSTRUMENTATION

1.0 AVAILABLE INSTRUMENTATION

[1] **IF** instrumentation in this Table is OR becomes DEENERGIZED, **THEN**

REFER TO AOI-30.2 APP E.

[2] **MONITOR** the following instrumentation for this fire area:

[2.1] Loop Instrumentation:

PARAMETER	LOOP 1	LOOP 2	LOOP 3	LOOP 4
HL TEMP	1-TI-68-1C	1-TI-68-24C	1-TI-68-43C	1-TI-68-65C
CL TEMP	T_{sat} scale on associated SG pressure instrument should be used.			
S/G PRESS	1-PI-1-1C	1-PI-1-8C	1-PI-1-19C	1-PI-1-26C
S/G LEVEL	1-LI-3-164C	1-LI-3-156C	1-LI-3-148C	1-LI-3-171C
AFW FLOW	1-FI-3-163C	1-FI-3-155C	1-FI-3-147C	1-FI-3-170C

[2.2] PZR/RHR Instrumentation:

PZR	RHR/TI
1-PI-68-342C, PZR-COLD CAL PRESS	1-TI-68-43C (Loop 2 HL Temp for inlet temp)
1-LI-68-325C, PZR LEVEL	1-TI-74-38C (HX-A Outlet)
1-LI-68-326C, PZR LEVEL	1-TI-74-40C (HX-B Outlet)

[2.3] Other Instrumentation:

PARAMETER	TRAIN A/CHANNEL	TRAIN B/CHANNEL
AFP SUCTION PRESS (local)	1-PI-3-117	1-PI-3-127
AFP DISCHARGE PRESS	1-PDIC-3-122C	1-PDIC-3-132C
ERCW SUP HDR FLOW	1-FI-67-61C	1-FI-67-62C
ERCW SUP HDR FLOW	2-FI-67-61C	2-FI-67-62C
NEUTRON MON	1-NI-92-138	
VCT LEVEL	1-LI-62-129C	
CHG	1-FI-62-93C	

I. PROGRAM: WATTS BAR OPERATOR TRAINING

II. COURSE: LICENSE TRAINING

III. TITLE: AOI-30.1, 30.2 PLANT FIRES

IV. LENGTH OF LESSON: 3 HOURS

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. Describe the Purpose/goal of AOI-30.1 &30.2.
X	X	X	X	2. When a VALID fire is reported to the Main Control Room (MCR), describe the information obtained from the person reporting the fire.
	X	X	X	3. Describe the three elements of the Fire Protection program designed to provide "defense in depth" to fire protection of areas important to safety as described in 10CFR50 App R.
	X	X	X	4. Define the 10CFR50 Appendix R requirements with respect to: a. Water supplies b. Manual fire suppression c. Automatic fire detection d. Separation of cables and equipment and associated non-safety circuits e. Fire Brigade f. Emergency lighting
	X	X	X	5. State the major actions of AOI-30.1 PLANT FIRES.
X	X	X	X	6. State the criteria for determining if a transition to AOI-30.2 is required during performance of AOI-30.1 (Define Appendix R fire).

V. TRAINING OBJECTIVES (continued)

AUO	RO	SRO	STA	
X	X	X	X	7. Identify the section of AOI-30.2 giving procedural guidance relative to each of the following: a. Location of component(s) within Auxiliary, Control, or Reactor buildings or Intake pumping station b. Control Air c. Ventilation Systems with failed fire dampers
	X	X	X	8. Identify parameters/conditions which the unit SRO/SOS must evaluate when judging whether AOI-30.2 must be initiated.
	X	X	X	9. List the assumptions (3) made for analysis as described in AOI-30.2 with respect to an Appendix R fire.
X	X	X	X	10. State the two primary limiting safety conditions which must be maintained following a postulated Appendix R fire as specified in AOI-30.2.
X	X	X	X	11. State the assumption(s) made relative to fires on or affecting electrical boards as discussed in AOI-30.2
	X	X	X	12. Demonstrate Ability/knowledge of AOI-30.1 and 30.2 by: a. Recognizing entry conditions b. Responding to required actions of the AOI c. Responding to contingencies (RNO) d. Responding to Notes/Cautions

V. TRAINING OBJECTIVES (continued)

AUO	RO	SRO	STA	
X	X	X	X	13. For the following systems, describe the required system configuration and basis for the alignment as described in AOI-30.2 a. Main/Reheat Steam b. Main/Auxiliary Feedwater c. Control Air System d. Fuel Oil System e. Ventilation (HVAC) f. CVCS g. Safety Injection System h. Essential Raw Cooling Water System i. Reactor Coolant System j. Component Cooling System k. Containment Spray System l. Residual Heat Removal System m. Primary Makeup Water System n. Reactivity Control System o. Nuclear Instrumentation System
X	X	X	X	14. Describe AUO Responsibilities and Actions for a Plant Fire.
X	X	X	X	15. Describe AUO Responsibilities and Actions for when a Appendix R fire has been declared.
X	X	X	X	16. From memory, describe the requirements and responsibilities of the fire brigade organization in accordance with FPDP-4.

VI. TRAINING AIDS

A. Marker Board and Markers

B. Students

24. 074 EA1.06 024
Given the following:

- Following a LOCA the operating crew is performing FR-C.1, "Inadequate Core Cooling."
- Containment pressure peaked at 5.2 psid.
- Depressurizing the S/Gs to atmospheric pressure is ineffective in reducing incore T/C temperatures.
- All RCP support systems are available for RCPs 2 and 4.
- RCPs 1 and 3 have no ERCW supply for the motor coolers.
- S/G Narrow Range levels are:

#1	#2	#3	#4
40%	41%	32%	43%

- Core exit thermocouples have exceeded 1200°F.
- RCPs are to be started one at a time in an attempt to reduce core exit thermocouples below 1200°F.

Which ONE of the following identifies the RCPs that would be started in an attempt to reduce core exit thermocouple temperatures?

A. ANY three of the RCPs

B. ALL four of the RCPs

C. RCPs 2 and 4 ONLY

D. RCPs 1, 2, and 4 ONLY

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because there is a condition where one of the RCPs would not be started but would be saved for future use.*

B. *Incorrect, Plausible because if the #3 steam generator levels had been greater than 39%, all four of the RCPs would be started, one at a time, until core exit temperatures were less than 1200°F even if the support systems are not available.*

C. *Incorrect, Plausible because normally the RCPs are required to have their support systems available to allow the pump to be started.*

D. *Correct, For loops with steam generator levels greater than 39%, FR-C.1 directs each of the RCPs to be started one at a time until core exit temperatures are less than 1200°F even if the support systems are not available. RCPs 1, 2, and 4 meet this criteria.*

Question Number: 24

Tier: 1 **Group** 2

K/A: 074 EA1.06
Inadequate Core Cooling
Ability to operate and monitor the following as they apply to a Inadequate Core Cooling:
RCPs

Importance Rating: 3.6 / 3.9

10 CFR Part 55: 41.7 / 45.5 / 45.6

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires the ability to analyze conditions to determine which RCPs would be started during the performance of the procedure for inadequate core cooling.

Technical Reference: FR-C.1, Inadequate Core Cooling, Revision 0016

Proposed references to be provided: None

Learning Objective: 3-OTFRC0001
3. Given a set of plant conditions, use FR-C.1, C.2, and C.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 FRC0001 018 modified.

Comments:

Step	Action/Expected Response	Response Not Obtained

CAUTION
- When the reactor vessel head vent block valve is opened, the throttle valve will cycle open and closed.
- Slowly opening (stroke time 5 seconds) the head vent valve will prevent water hammer and pipe damage.

NOTE Normal conditions are desired but **NOT** required for starting the RCPs.

Distraction #2 & #4 only ones having support systems-

27. **CHECK** if RCPs should be started.

a. **CHECK** if an idle RCS cooling loop is available.
- NR S/G level greater than 29% [39% ADV]. (eliminates #3)
- RCP in associated loop - AVAILABLE and **NOT** OPERATING.

a. **PERFORM** the following:

1) **OPEN** all pzr PORVs and block valves.

2) **IF** Incore T/Cs remain greater than 1200 °F, **THEN**:

RESTORE power to head vents:
- **PLACE** 1-SW-68-394-A disconnect switch to ON [125V Vital Batt Bd Rm I].
- **PLACE** 1-SW-68-395-B disconnect switch to ON [125V Vital Batt Bd Rm II] **AND**

OPEN all reactor vessel head vent and block valves.

3) ** **GO TO** Step 28.

b. **START** RCP oil lift pump two minutes prior to starting RCP.

Step continued on next page

Step	Action/Expected Response	Response Not Obtained
22.	**DEPRESSURIZE** All Intact S/Gs to Atmospheric Pressure: a. **DUMP** steam to condenser at maximum rate.	a. **IF** condenser **NOT** available, **THEN** **USE** Intact S/G PORVs.

23.	**ENSURE** ECCS Flow.
	• FLOW through BIT.
	
	OR
	• SI pump flow.
	
	OR
	• RHR pump flow.
		CONTINUE efforts to establish ECCS injection flow from any source available.
		
		IF incore T/Cs less than 1200°F, **THEN**
		
		** **GO TO** Step 22.
		
		IF incore T/Cs greater than or equal to 1200°F, **THEN**
		
		** **GO TO** Step 26.

24.	**CHECK** Core Cooling: a. Incore T/Cs less than 1200°F. b. At least two RCS hot leg temperatures less than 350°F. c. RVLIS greater than 60%.	a. ** **GO TO** Step 26. b. ** **GO TO** Step 22. c. ** **GO TO** Step 22.
25.	** **GO TO** E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 17.	
26.	**CHECK** Incore T/Cs greater than 1200 °F.	** **GO TO** Step 28.

Step	Action/Expected Response	Response Not Obtained
27.	(continued) c. **START** RCP in one idle RCS cooling loop. d. **STOP** RCP oil lift pump one minute after RCP start. e. ** **GO TO** Step 26.	
28.	Manually or Locally **DEPRESSURIZE** All Intact S/Gs to atmospheric pressure: • USE condenser steam dump **OR** • USE S/G PORV **OR** • USE TD AFWP steam supply.	**IF** no intact S/G available, **THEN** **DEPRESSURIZE** Faulted S/G, **OR** **DEPRESSURIZE** Ruptured S/G.
29.	**CHECK** Incore T/Cs less than 1200°F.	**IF** incore T/Cs dropping, **THEN** ** **GO TO** Step 26. **IF** incore T/Cs stable or rising, **AND** all available RCPs running, **THEN** ** **GO TO** SACRG-1, Severe Accident Control Room Guideline Initial Response.

WBN BANK QUESTION

FRC0001 018

Given the following plant conditions:

- A LOCA has occurred.
- FR-C.1, Inadequate Core Cooling, is in progress.
- Core Exit thermocouple temperature is 1205°F and rising.
- Containment pressure is 3.6 psig.
- All S/Gs are Intact with levels of...

#1	#2	#3	#4
32% NR	17% NR	26% NR	35% NR

- Step [22], "Depressurize All Intact S/Gs to Atmospheric Pressure" is in progress.

Which of the following identifies the number of RCPs that will be started per FR-C.1, for the above conditions?

a. 1

b. 2

c. 3

d. 4

I. **PROGRAM:**

Watts Bar Operator Training

II. **COURSE:**

A. License Training

B. Onsite

C. License Operator Requal

III. **TITLE:**

Function Restoration Guidelines FR-C.1, FR-C.2, & FR-C.3

IV. **LENGTH OF LESSON:**

A. License training 4 Hours

B. Onsite 4 Hours

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

V. **TRAINING OBJECTIVES:**

AUO	RO	SRO	STA	
	X	X	X	1. Given a set of plant conditions, use the FR-0 Core Cooling Status Tree to identify and implement the correct procedure.
	X	X	X	2. Identify the 3 major actions for restoring core cooling as prescribed by procedure FR-C.1, and explain the basis for each of these actions.
	X	X	X	3. Given a set of plant conditions, use FR-C.1, C.2, and C.3 to correctly diagnose and implement: Action Steps, RNOs, Foldout Pages, Notes and Cautions.
	X	X	X	4. Explain why the RCPs are stopped prior to depressurizing all intact S/Gs to atmospheric pressure during the performance of FR-C.1 and C.2.

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

AUO	RO	SRO	STA	
	X	X	X	5. Identify the two major actions for restoring core cooling as prescribed by procedure FR-C.2, and explain the basis for each of these actions.
	X	X	X	6. Explain the basis for stopping one RCP, if all four RCPs are in service as directed by procedure FR-C.2, Degraded Core Cooling.
	X	X	X	7. Describe when and why it might be necessary to vent Nitrogen gas from a cold leg accumulator as directed in procedure FR-C.2, Degraded Core Cooling.
	X	X	X	8. Explain the purpose for and basis of each step in FR-C.1, FR-C.2, and FR-C.3.

25. W/E03 EK2.2 025

Given the following:

- A small break LOCA occurred on Unit 1.
- ES-1.2, "Post LOCA Cooldown and Depressurization," is in progress.
- RCS pressure is 1420 psig and one charging pump has been stopped.
- The crew is ready to stop the first SI pump.

Which ONE of the following identifies how subcooling will be affected when the SI pump is stopped and why?

Subcooling will...

A. remain the same due to the running SI pump flow increasing to maintain ECCS flow to equal the break flow.

B. remain the same due to reduced ECCS injection flow causing RCS temperature and pressure to increase.

C. drop and stabilize at a lower value due to an increase in RCS temperature with lower ECCS injection flow.

D. drop due to reduced ECCS injection flow and stabilize at a lower value when break flow equals ECCS injection flow.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because the flow on the other SI pump will increase when the first SI pump is stopped but the overall SI pump discharge pressure will be less and the break flow will exceed injection flow causing pressure to drop until a new lower pressure equalization is established.*

B. *Incorrect, Plausible because with lower ECCS flow, there is a reduction in the amount of ECCS water being injected but cooling is still being supplied and the SGs are also providing cooling. The reduction does not cause an RCS pressure and temperature increase.*

C. *Incorrect, Plausible because the subcooling value dropping is correct but it is not due to a temperature increase due to lower ECCS flow.*

D. *Correct, The subcooling value will first drop due RCS pressure dropping because of a reduction in the ECCS injection flow when the SI pump is stopped. Then as the pressure in the RCS drops the break flow will drop. Eventually the RCS break flow and the ECCS injection flow will reach equilibrium at a lower pressure.*

Question Number: 25

Tier: 1 **Group:** 2

K/A: W/E03 EK2.2
LOCA Cooldown and Depressurization
Knowledge of the interrelations between the (LOCA Cooldown and Depressurization) and the following:
Facility*s heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Importance Rating: 3.7 / 4.0

10 CFR Part 55: 41.7 / 45.7

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires the applicant to determine how RCS conditions including subcooling are affected due to stopping a pump supplying water used for heat removal during a post LOCA cooldown and depressurization event.

Technical Reference: ES-1.2, Post LOCA Cooldown and Depressurization, Revision 0015

Proposed references to be provided: None

Learning Objective: 3-OT-EOP0100
18. Analyze and explain the process that leads to a new RCS equilibrium pressure following the shutdown of an ECCS pump during the ES-1.2 reduction sequence

Cognitive Level:

Higher	
Lower	X

Question Source:

New	
Modified Bank	
Bank	X

Question History: SQN bank question ES-1.2-B.2 009 with some wording changes in the stem and choices but not enough to be a modified question.

Comments:

Step	Action/Expected Response	Response Not Obtained

CAUTION During the SI reduction sequence, injection flow and break flow must reach equilibrium after each pump is stopped. RCS pressure may be reduced to maintain pressurizer level.

17. **CONTROL** RCS press to maintain pzr level:

- **MAINTAIN** subcooling greater than 65°F [85°F ADV].
- **MAINTAIN** pzr level between 29% and 63% [47% and 58% ADV].

NOTE
- After a charging pump is stopped, RCS press should be allowed to stabilize or rise before another ECCS pump is stopped.
- The charging pumps and SI pumps should be stopped on alternate trains when possible.

18. **DETERMINE** if charging pump shutdown criteria met:

a. **CHECK** both charging pumps RUNNING.

a. **IF** one or both charging pumps stopped, **THEN**

** **GO TO** Notes prior to Step 20.

b. **CHECK** at least one SI pump RUNNING.

b. **IF** both SI pumps stopped, **THEN**

** **GO TO** Step 19.

Step continued on next page

Step	Action/Expected Response	Response Not Obtained
18.	(continued)	

	c. **CHECK** RCS subcooling greater than 81°F [102°F ADV].	c. **IF** T-hot greater than 360°F, **THEN**
		
		** **GO TO** Step 32.
		
		IF T-hot less than 360°F **AND** both RHR pumps stopped, **THEN**
		
		START one RHR pump.
		
		IF neither of the RHR pumps can be started, **THEN**
		
		** **GO TO** Step 32.

	d. **CHECK** pzr level greater than 29% [47% ADV].	d. ** **GO TO** Caution prior to Step 15.
	e. **STOP** one charging pump.	
	f. ** **GO TO** Notes prior to Step 20.	

Step	Action/Expected Response	Response Not Obtained
19.	**DETERMINE** if one charging pump should be stopped:	
	a. **CHECK** both charging pumps RUNNING.	a. ** **GO TO** Notes prior to Step 20.
	b. **CHECK** both SI pumps STOPPED.	b. ** **GO TO** Notes prior to Step 18.

	c. **CHECK** RCS subcooling greater than 118°F [141°F ADV].	c. **IF** T-hot greater than 360°F, **THEN** ** **GO TO** Step 32.
		
		IF T-hot less than 360°F **AND** both RHR pumps stopped, **THEN**
		
		START one RHR pump.
		
		IF neither of the RHR pumps can be started, **THEN**
		
		** **GO TO** Step 32.

	d. **CHECK** pzr level greater than 29% [47% ADV].	d. ** **GO TO** Caution prior to Step 15.
	e. **STOP** one charging pump.	

Step	Action/Expected Response	Response Not Obtained

NOTE

- If RCS subcooling greater than 85°F [106°F ADV], then RCS depressurization may be continued to maintain pzr level.
- RCS pressure should be allowed to stabilize or rise before SI pump is stopped.

20. **DETERMINE** if SI pump shutdown criteria met:

a. **CHECK** both SI pumps RUNNING.

a. **IF** both SI pumps stopped, **THEN**

** **GO TO** Step 22.

IF one SI pump stopped, **THEN**

** **GO TO** Notes prior to Step 21.

b. **CHECK** RCS subcooling greater than 85°F [106°F ADV].

b. **IF** T-hot greater than 360°F, **THEN**

** **GO TO** Step 32.

IF T-hot less than 360°F **AND** both RHR pumps stopped, **THEN**

START one RHR pump.

IF neither of the RHR pumps can be started, **THEN**

** **GO TO** Step 32.

c. **CHECK** pzr level greater than 29% [47% ADV].

c. ** **GO TO** Caution prior to Step 15.

d. **STOP** one SI pump.

Step	Action/Expected Response	Response Not Obtained

NOTE

- If RCS subcooling greater than 188°F [213°F ADV], then RCS depressurization may be continued to maintain pzr level.
- RCS pressure should be allowed to stabilize or rise before SI pump is stopped.

21. **DETERMINE** if remaining SI pump should be stopped:

a. **CHECK** one SI pump RUNNING.

a. ** **GO TO** Step 22.

b. **CHECK** one charging pump RUNNING.

b. ** **GO TO** Step 32.

c. **CHECK** RCS subcooling greater than 188°F [213°F ADV].

c. **IF** T-hot greater than 360°F, **THEN**

** **GO TO** Step 32.

IF T-hot less than 360°F **AND** both RHR pumps stopped, **THEN**

START one RHR pump.

IF neither of the RHR pumps can be started, **THEN**

** **GO TO** Step 32.

d. **CHECK** pzr level greater than 29% [47% ADV].

d. ** **GO TO** Caution prior to Step 15.

e. **STOP** remaining SI pump.

Step	Action/Expected Response	Response Not Obtained
22.	**CHECK** if normal charging should be established:	
	a. **CHECK** both of the following criteria met: • Both SI pumps STOPPED. • ONLY one charging pump RUNNING.	a. **DO NOT** align charging. ** **GO TO** Step 32.

	b. **CHECK** RCS subcooling greater than 57°F [77°F ADV].	b. **IF** T-hot greater than 360°F, **THEN**
		** **GO TO** Step 32.
		IF T-hot less than 360°F **AND** both RHR pumps stopped, **THEN**
		START one RHR pump.
		IF neither of the RHR pumps can be started, **THEN**
		** **GO TO** Step 32.

	c. **CHECK** pzr level greater than 29% [47% ADV].	c. ** **GO TO** Caution prior to Step 15.

ES-1.2-B.2 009

SQN BANK QUESTION

Given the following plant conditions:

- A small break LOCA has occurred.
- The crew is implementing ES-1.2, Post LOCA Cooldown and Depressurization.
- The crew has determined that one Safety Injection Pump can be stopped.

Which ONE of the following explains what will happen to the subcooling value when the SIP is stopped? Subcooling _____

A. remains the same due to running SIP flow increasing to equal break flow.

B. lowers due to reduced ECCS injection flow and stablizes at a lower value as RCS temperature rises.

C. remains the same due to reduced ECCS injection flow causing RCS temperature and pressure to rise.

D. lowers due to reduced ECCS injection flow and stablizes at a lower value when break flow and ECCS flow equal.

Justification:

A. Incorrect. While the running SIP will increse its flow; total SIP flow will decrease. This would result in lower RCS pressure; therefore, lower subcooling margin.

B. Incorrect. RCS temperature should not rise since majority of cooling provied by dumping steam from steam generator.

C. Incorrect. RCS temperature should not rise since majority of cooling provied by dumping steam from steam generator.

D. Correct. When SIP is stopped, RCS pressure drops. Thus T_{sat} for RCS lowers reducing the amount of subcooling. This is the bases for the subcooling margins for step 20 of ES-1.2.

Notes:

K/A {CFR}:	E03EK2.2	(3.7/4.0)	[41.7]
	E03EK3.1	(3.3/3.7)	[41.5, 41.10]
	E03EA1.2	(3.7/3.9)	[41.7]

References: ES-1.2, Post LOCA Cooldown and Depressurization

LP/Objectives: OPL271ES12, obj 2.b.
OPL271ES12, obj 3.a.
OPL271ES12, obj 4.b
OPL273C502 (series), Obj 7

History: 2/04 - Indian Point 2 exam Mar 2003. Modified to fit SQN.
1/05 - reordered objectives from short to long.

Level: Comprehension

Est Time: 4 min

Comments:

I. **PROGRAM:**

Watts Bar Operator Training

II. **COURSE:**

A. License Training

B. License Operator Requal

III. **TITLE:**

E-1, Loss of Reactor or Secondary Coolant

IV. **LENGTH OF LESSON:**

A.	License training	3 Hours
B.	License Operator Requal	License operator REQUAL time will be determined after objectives are identified.

V. **TRAINING OBJECTIVES:**

AUO	RO	SRO	STA	
	X	X	X	1. Describe the purpose of procedure E-1 as listed in Section 1.0 of the procedure.
	X	X	X	2. Explain the basis for tripping the RCPs in an accident situation given the following conditions: a. RCS press less than 1500 psig b. Phase B isolation signal initiated.
	X	X	X	3. List the condition that must be checked and satisfied before removing a RCP from service in accident conditions due to low RCS pressure (< 1500 psig).
	X	X	X	4. Explain the basis for the continuous action step to monitor containment pressure and stop the CS pumps when containment pressure is verified less than 2.0 psig.
	X	X	X	5. For a given H_2 concentration in containment determine if the H_2 igniter should be energized and explain why or why not.

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

AUO	RO	SRO	STA	
	X	X	X	6. Explain the basis for isolating the CLAs when RCS press decreases to less than 250 psig.
	X	X	X	7. Explain the reason for transfer to Hot Leg recirc following a LOCA including the location of the worst case break for this concern.
	X	X	X	8. Given a set of plant conditions, use E-1, ES-1.1, ES-1.2, ES-1.3, and ES-1.4 to correctly diagnose and implement: Action Steps, RNOs, Foldout Pages, Notes, and Cautions.
	X	X	X	9. List the four parameters (not setpoints) that must be verified prior to SI termination.
	X	X	X	10. Determine the correct procedure transition if during the SI termination steps of ES-1.1 it is determined that PZR level cannot be maintained using the normal charging flowpath.
	X	X	X	11. Explain the basis for waiting for a faulted S/G to complete depressurization before checking RCS press stable or increasing following the establishment of normal charging and prior to stopping any running SI pumps.
	X	X	X	12. Discuss the purpose of ES-1.2 Post LOCA Cooldown and Depressurization.
	X	X	X	13. Justify the procedure step to shutdown the RHR pumps if RCS pressure is greater than 150 psig.
	X	X	X	14. Identify the procedural transition required if any S/G level continues to increase with feedflow isolated.

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

AUO	RO	SRO	STA	
	X	X	X	15. Explain the basis for limiting the RCS cooldown rate to 100°F in one hour.
	X	X	X	16. Discuss the requirement to check RCS subcooling greater than 65°F prior to RCS depressurization.
	X	X	X	17. Describe how depressurization of the RCS might result in the capability to maintain PZR level when PZR level could not be maintained prior to depressurization.
	X	X	X	18. Analyze and explain the process that leads to a new RCS equilibrium pressure following the shutdown of an ECCS pump during the ES-1.2 reduction sequence.
	X	X	X	19. Explain why subcooling is minimized following the alignment of normal charging in procedure ES-1.2.
	X	X	X	20. Discuss and justify the priority of usage given to procedure ES-1.3, Transfer to RHR Containment Sump.
	X	X	X	21. Justify the ES-1.3 procedural requirement to shutdown the SI pumps if RCS press increase to greater than 1350 psig while aligned for sump recirc.
	X	X	X	22. Identify and explain the basis of the interlock on the RHR pump discharge to the SI and CCP suction (FCV-63-8 and 11).
	X	X	X	23. State from memory the action required if offsite power is lost following transfer to RHR containment sump cold leg recirc. Explain the basis for the required action.

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

AUO	RO	SRO	STA	
	X	X	X	24. Discuss the basis for ensuring the CCP suction from the RWST (LCV-62-135 and 136) handswitches are left in the A-Auto position following transfer to cold leg recirc in procedure ES-1.3.
	X	X	X	25. Explain why procedure ES-1.3 directs the operator to leave the containment spray pumps aligned to the RWST until RWST level is less than 8%.
	X	X	X	26. Identify the action required if RWST level decreases to 8% during swapover to CL sump recirc.
	X	X	X	27. Explain the basis for limiting temperature above current conditions after transition to SI termination. (SOER 94-001, Rec. 4b.)
X	X	X	X	28. Describe the actions in ES-1.1, SI Termination, required in the event that SI does not reset.

26. W/E08 EK3.3 026

Following a Small Break LOCA, the crew is performing the actions contained in FR-P.1, "Pressurized Thermal Shock."

Which ONE of the following describes the difference in SI termination Criteria for FR-P.1 as opposed to the criteria in ES-1.1, "Safety Injection Termination" and the reason for the difference?

The criteria in FR-P.1 is...

A. less restrictive to allow for a faster reduction in RCS pressure.

B. more restrictive to allow for a more controlled reduction in RCS pressure.

C. less restrictive because subsequent RCP restart is likely to cause propagation of any existing flaw in the reactor vessel walls.

D. more restrictive because subsequent RCP restart is likely to cause propagation of any existing flaw in the reactor vessel walls.

DISTRACTOR ANALYSIS:

A. Correct, The criteria are less restrictive so that SI reduction can aid the RCS pressure reduction. RVLIS level rather than PRZR level is used as a measure of inventory.

B. Incorrect, Plausible if the applicant believes the opposite of actual reason to be true.

C. Incorrect, RCP restart will not cause a crack to propagate under any of the conditions analyzed for this procedure. However, it is plausible because propagation of flaws is a major concern in a PTS event.

D. Incorrect, RCP restart will not cause a crack to propagate under any of the conditions analyzed for this procedure. However, it is plausible because propagation of flaws is a major concern in a PTS event.

Question Number: 26

Tier: 1 **Group** 2

K/A: W/E08 EK3.3
Pressurized Thermal Shock

K/A: W/E08 EK3.3
Pressurized Thermal Shock
Knowledge of the reasons for the following responses as they apply to the (Pressurized Thermal Shock)
Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

Importance Rating: 3.7 / 3.8

10 CFR Part 55: 41.5 / 41.10, 45.6, 45.13

10CFR55.43.b: Not applicable

K/A Match: KA is matched because the question requires knowledge of the reasons controls may be manipulated to terminate ECCS flow under different conditions than normally required in the EOP network as allowed by the procedure when responding to a pressurized thermal shock.

Technical Reference: FR-P.1, Pressurized Thermal Shock, Revision 0015
WOG Background Document , FR-P.1 Background, Revision 2

Proposed references to be provided: None

Learning Objective: 3-OT-FRP0001
4. Justify the basis for using a less restrictive SI termination criteria when performing FR-P.1.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	
Bank	X

Question History: Beaver Valley bank question (2005 exam)

Comments:

Step	Action/Expected Response	Response Not Obtained

NOTE Either Loop 1 or 2 pzr spray valve is effective for Loop 2 RCP in service or for Loops 1, 3, & 4 RCPs in service.

8. **CHECK** SI termination criteria:

   - RVLIS greater than 60% with NO RCP running

     OR

     RVLIS greater than 63% with ANY RCP running.

   - RCS subcooling greater than 115°F [135°F ADV].

   PERFORM the following:

   a. **IF** RCS subcooling greater than 65°F [85°F ADV] **AND** NO RCP running, **THEN**

      REFER TO Attachment 1, RCP Emergency Restart Criteria.

   b. **START** RCP(s) oil lift pump two minutes prior to starting RCP.

   c. **WHEN** start conditions established, **THEN**:

      1) **START** one RCP, loop 2 preferred.

      2) **IF** Loop 2 RCP can **NOT** be started, **THEN**

         START ALL other RCPs.

      3) **STOP** RCP(s) oil lift pump one minute after RCP start.

      4) **CONTINUE** ECCS Flow.

      5) ** **GO TO** Note prior to Step 28.

Step	Action/Expected Response	Response Not Obtained

CAUTION If offsite power is lost after SI reset, then manual action will be required to restart the SI pumps and RHR pumps.

9. **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.

10. **RESET** Phase A and Phase B, **AND**

    INITIATE Appendix A, (FR-P.1), CLA Breaker Operation.

11. **ENSURE** cntmt air in service:

    a. Aux air press 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 DESCRIPTION TABLE FOR FR-P.1 Step 6

STEP: Check If SI Can Be Terminated

PURPOSE: To determine if conditions have been established which indicate that full SI flow is no longer required

BASIS:

Following SI actuation, RCS conditions may be restored to within acceptable limits for SI termination to be allowed. The combination of a minimum subcooling and sufficient liquid level in the vessel to cover the core represents less restrictive SI termination criteria in this guideline than those present in the ORGs since, for an imminent PTS condition, SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure.

The subcooling criterion will ensure subcooled conditions and the RVLIS indication ensures the existence of an adequate vessel inventory such that core cooling is ensured. Refer to document SI TERMINATION/REINITIATION in the Generic Issues section of the Executive Volume.

If either of the termination criteria are not satisfied, then SI is required to ensure core cooling and should not be terminated. Most likely the cold leg/downcomer low temperature condition is due to SI water mixing effects and an RCP restart is attempted.

Of the transients considered in PTS, the SBLOCA transient may result in a condition whereby Safety Injection (SI) flow cannot be terminated. In Westinghouse Owners Group (WOG) reports OG-110 and OG-117 titled "Evaluation of Alternate RCP Trip Criteria" and "Justification of Manual RCP Trip for Small Break LOCA Events" respectively, a range of SBLOCAs were identified where continued RCP operation or conversely untimely RCP restart could result in increased RCS inventory loss. The loss of additional inventory could ultimately result in deeper core uncovery transients which could in turn result in fuel cladding temperatures in excess of the plant's design basis FSAR analysis result. Therefore, from a SBLOCA standpoint, RCP restart at an inopportune time could result in a degraded core cooling scenario.

In WCAP-10319 titled "A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, from Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Power Plants", numerous transient analyses including those of SBLOCA have been analyzed without RCP restart. The results

BASIS:

of the stagnant loop evaluation demonstrate that the total expected frequency of significant flaw extension in a typical W PWR reactor vessel due to PTS, including the contributions from stagnant loop SBLOCA transients, does not exceed the NRC required RT_{PTS} screening value of 270°F for axial flaws. Therefore, based on analyses results, RCP restart is not required to meet the NRC PTS risk goal for a typical W plant.

Therefore, an additional support condition, RCS subcooling, in addition to plant specific minimum support conditions is recommended to assure that no potential RCS inventory aggravation will occur due to RCP restart.

An analysis of the effect of an RCP restart has been made to ensure the safety of this action relative to vessel integrity. For conservatism in the analysis the assumption was made that a small preexisting flaw had grown and arrested at 75 percent of wall thickness before RCP start. Starting an RCP was shown not to result in any further flaw propagation and loss of vessel integrity. For a case where a flaw has not grown prior to RCP start, the subsequent heat-up of the downcomer region will decrease the possibility of flaw initiation.

Therefore, in order to mix the cold incoming SI water and the warm reactor coolant water and thereby decrease the likelihood of a PTS condition, an RCP restart is attempted. Whether an RCP is started or not, the next step performed (Step 24), if SI is still required, provides guidance on subsequent cooldown restrictions.

ACTIONS:

- Determine if RCS subcooling (based on core exit TCs) is greater than (R.12)°F [(R.13)°F for adverse containment]
- Determine if RVLIS full range indication indicates greater than (K.02) if no RCP running
- Determine if RVLIS dynamic head range indicates greater than (L.08) if one RCP running
- Determine if RVLIS dynamic head range indicates greater than (L.07) if two RCPs running
- Determine if RVLIS dynamic head range indicates greater than (L.06) if three RCPs running
- Determine if RVLIS dynamic head range indicates greater than (L.05) if four RCPs running
- Determine if RCS subcooling (based on core exit TCs) is greater than (R.01)°F [(R.02)°F for adverse containment]
- Determine if no RCP is running
- Attempt to start one RCP