DISTRACTOR ANALYSIS:

A. *Correct, Tech Spec 3.6.6 identifies that while containment spray is required in Mode 4, the RHR spray is not needed to supplement the containment spray and is not required. As the heat up continues to Mode 3, the RHR spray is required and if the valve is not repaired the provisions of LCO 3.0.4.b can be implemented to allow entry into Mode 3.*

B. *Incorrect, Plausible because no Tech Spec LCO being required while in Mode 4 is correct and since the valve status will prevent meeting the LCO after the mode change. The required actions of LCO 3.6.6 do not permit continued operation in the Mode, thus 3.0.4.a will prevent the Mode change unless the requirements of 3.0.4.b are met and there are LCOs that will not allow the use of 3.0.4.b.*

C. *Incorrect, Plausible because Tech Spec LCO 3.6.6 entry would have been required if the RCS temperature had been at least 350°F (Mode 3) and implementing the provisions of LCO 3.0.4.b to allow the Mode change is correct.*

D. *Incorrect, Plausible because Tech Spec LCO 3.6.6 entry would have been required if the RCS temperature had been at least 350°F (Mode 3) and since the valve status will prevent meeting the LCO after the mode change. The required actions of LCO 3.6.6 do not permit continued operation in the Mode, thus 3.0.4.a will prevent the Mode change unless the requirements of 3.0.4.b are met and there are LCOs that will not allow the use of 3.0.4.b.*

Question Number: 90

Tier: 2 **Group** 1

K/A: 026 G2.2.37
026 Containment Spray System
2.2 Equipment Control
2.2.37 Ability to determine operability and/or availability of safety related equipment.

Importance Rating: 3.6 / 4.6

10 CFR Part 55: 41.7 / 43.5 / 45.12

10CFR55.43.b: 2

K/A Match: K/A is matched because the questions requires knowledge of the containment spray system Tech Spec and is SRO because the questions requires knowledge of the provision of LCO 3.0.4

Technical Reference: Tech Spec LCO 3.0, Amendment 55
Tech Spec LCO 3.6.6, Containment Spray System

Proposed references to be provided: None

Learning Objective: 3-OT-T/S0306
4. Given plant conditions and parameters correctly, determine the applicable LIMITING CONDITION FOR OPERATION, OR TECHNICAL REQUIREMENTS for the various components of the Containment System.

Cognitive Level:
Higher X
Lower

Question Source:
New X
Modified Bank
Bank

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

Comments:

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.
LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours; b. MODE 4 within 13 hours; and c. MODE 5 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.
LCO 3.0.4	When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.7.2.18, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray System

LCO 3.6.6 Two containment spray trains and two residual heat removal (RHR) spray trains shall be OPERABLE.

---------------------------NOTE---------------------------
The RHR spray train is not required in MODE 4.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. One RHR spray train inoperable.	B.1 Restore RHR spray train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.3	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.6.4	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.6.5	Verify each spray nozzle is unobstructed.	At first refueling <u>AND</u> 10 years
SR 3.6.6.6	Perform SR 3.5.2.2 and SR 3.5.2.4 for the RHR spray system.	In accordance with Applicable SRs

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. LICENSE TRAINING
B. LICENSED REQUAL

III. TITLE

T/S AND T/R 3.6, CONTAINMENT SYSTEMS AND BASES

IV. LENGTH OF LESSON

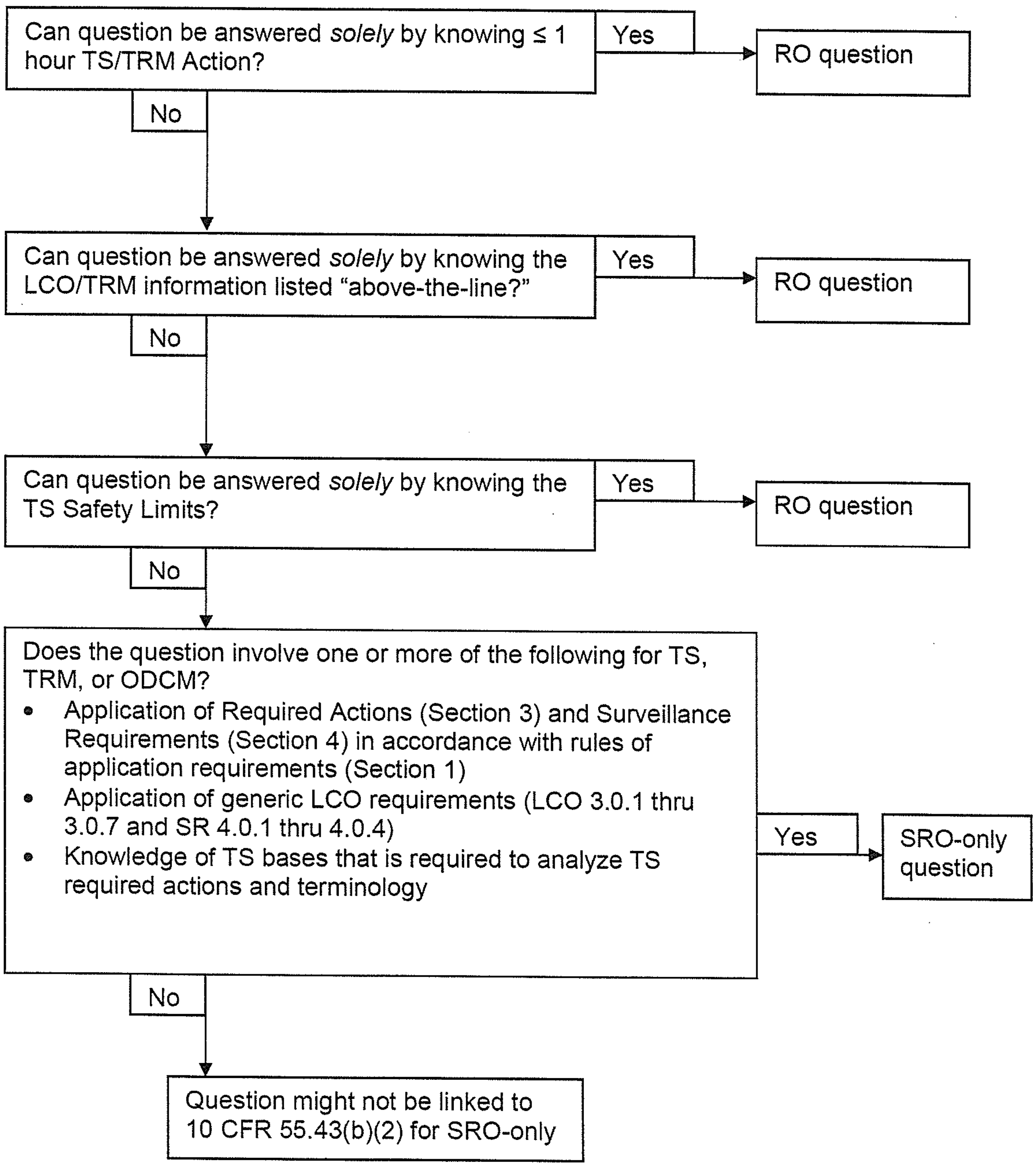
A. LICENSE TRAINING 1 Hrs

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 the Containment System.
	X	X	X	2. Determine the bases for each specification, as applicable, to the Containment System.
	X	X	X	3. Given plant conditions and parameters, correctly determine the OPERABILITY of components associated with the Containment System.
	X	X	X	4. Given plant conditions and parameters correctly, determine the applicable LIMITING CONDITION FOR OPERATION, OR TECHNICAL REQUIREMENTS for the various components of the Containment System.
	X	X	X	5. Given plant conditions and parameters, determine applicable Action Conditions, Required Actions, and Completion Times associated with the Containment System.

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



91. 034 G2.4.30 091

Given the following:

- Unit 1 is defueled with fuel shuffles in progress in the spent fuel pool.
- Personnel report that level is dropping in the spent fuel pool.
- An irradiated fuel assembly is dropped causing damage and the release of fuel pellets.
- 0-RM-90-102 and 0-RM-90-103 both increase to the alarm setpoint.
- Makeup is established to the spent fuel pool stabilizing level below Technical Specification minimum, but above the fuel.

Which ONE of the following identifies the event declaration to be made and the external notification having the shortest allowed time?

REFERENCE PROVIDED

	Declaration	Notification
A.	NOUE	NRC
B.	NOUE	TEMA
C.	ALERT	NRC
D.✓	ALERT	TEMA

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because the conditions for NOUE declaration do exist for EAL 7.4, "Fuel Handling." Also plausible because an NRC notification is required for the event, but it is a one hour notification.*

B. *Incorrect, Plausible because the conditions for NOUE declaration do exist for EAL 7.4, "Fuel Handling." Also plausible because there is a 10 minute notification to TEMA which is correct for an event declaration.*

C. *Incorrect. Plausible because based on EAL 7.4, "Fuel Handling," the conditions for an Alert exist: alarms on 0-RM-90-102 and 103 AND reports of damage to irradiated fuel resulting in rupture of the fuel rods. Also plausible because an NRC notification is required for the event, but it is a one hour notification.*

D. *Correct, Based on EAL 7.4, "Fuel Handling," the conditions for an Alert exist: alarms on 0-RM-90-102 and 103 AND reports of damage to irradiated fuel resulting in rupture of the fuel rods. Also, a 10 minute notification to TEMA is correct for the event declaration.*

Question Number: 91

Tier: 2 **Group** 2

K/A: 034 G2.4.30
Fuel Handling
Emergency Procedures / Plan
Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator

Importance Rating: 2.7 / 4.1

10 CFR Part 55: 41.10 / 43.5 / 45.11

10CFR55.43.b: 7

K/A Match: K/A is matched because the question requires knowledge of the reporting requirements for an event that is declared for a dropped fuel assembly. SRO because the question requires knowledge of reporting requirements and Emergency Classifications associated with fuel handling facilities and procedures.

Technical Reference: EPIP-1, "Emergency Plan Classification Logic," Revision 0035
EPIP-3, Alert, Revision 0033

Proposed references to be provided: EPIP-1, "Emergency Plan Classification Logic." Rev 0035, Page 49 of 51

Learning Objective: 3-OT-PCD-048C
01. Classify emergency events.
26. Understand the critical times associated with:
- Event Declaration
- Offsite Notification
- Facility Staffing
- Printed EPS Report

Cognitive Level:

Higher	___
Lower	X

Question Source:

New	X
Modified Bank	___
Bank	___

Question History: Question written for 10/2011 Watts Bar NRC exam

Comments:

Attachment 7
(Page 6 of 7)

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

Date ________

3.0 INSTRUCTIONS (continued)

D. **IF** the ODS CANNOT be contacted within 10 minutes, **THEN** ☐

NLO 3

- **FAX** Appendix A to TEMA at 9-1-615-242-9635. ☐
- **NOTIFY** the Tennessee Emergency Management Agency (TEMA) of the Radiological Emergency Plan activation by calling 9-1-800-262-3300 or 9-1-615-741-0001 or 9-1-800-262-3400. ☐

E. **ANNOUNCE** to the crew: "An ALERT is being declared based on ________________. I will be the Site Emergency Director." ☐

F. **TRACK** dispatched personnel by name, and **PERFORM** one of the following:

- **IF** OSC is not staffed, **THEN** **INFORM** Rotating Maintenance Organization Supervisor of names for team tracking. ☐
- **WHEN** OSC is staffed, **THEN** **INFORM** OSC manager of names for team tracking. ☐

G. **GO TO** Step 3.0[5]. ☐

[3] **IF** the TSC has been activated and the CECC has not been activated **THEN**

A. **IF** the CECC has been activated and assumed responsibility for event notifications, **THEN** **GO TO** Step 3.0[4]. ☐

B. **INITIATE** Appendix A, Initial Notification Form for ALERT. ☐

V. TRAINING OBJECTIVES

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

V. TRAINING OBJECTIVES (continued)

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

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

Some examples of SRO exam items for this topic include:

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

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

Some examples of SRO exam items for this topic include:

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

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

92. 072 A2.02 092

Given the following:

- Unit 1 operating at 100% power.

- An internal electrical failure causes the output of 0-RM-90-103, "Spent Fuel Pit Area Radiation Monitor," to fail above the HI RAD setpoint.

- The operating crew is performing SOI-30.05, "Auxiliary Bldg HVAC Systems," Section 8.8, "Recovery from Aux Bldg Isolation (ABI)," to restore the system.

Which ONE of the following identifies...

(1) an additional SOI that has a section required to be performed to restore the system to normal

and

(2) whether the ABI actuation is required to be reported to the NRC as an Immediate Notification in accordance with NPG-SPP-03.5, "NRC Reporting Requirements?"

A. (1) SOI-30.06, "Auxiliary Building Gas Treatment System"
(2) NRC report is required.

B. (1) SOI-31.02, "Post Accident Sampling Facility Ventilation System"
(2) NRC report is required.

C✓ (1) SOI-30.06, "Auxiliary Building Gas Treatment System"
(2) NRC report is **NOT** required.

D. (1) SOI-31.02, "Post Accident Sampling Facility Ventilation System"
(2) NRC report is **NOT** required.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because SOI-30.06 being required is correct and ESF actuations are normally reportable.*

B. *Incorrect, Plausible because performance of a section of SOI-31.02 would be required if the Auxiliary Building isolation had been caused by any of the signals (other than SFP rad monitor) and ESF actuations are normally reportable.*

C. *Correct, SOI-30.06, "Auxiliary Building Gas Treatment System" would be required to restore the ABGTS fan to a standby alignment and the actuation would be an invalid actuation, thus no 8-hour notification would be required.*

D. *Incorrect, Plausible because performance of a section of SOI-31.02 would be required if the Auxiliary Building isolation had been caused by any of the signals (other than SFP rad monitor) and ESF actuation not being reportable is correct because the actuation was invalid.*

Question Number: 92

Tier: 2 **Group** 2

K/A: 072 A2.02
Area Radiation Monitoring System
Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Detector failure

Importance Rating: 2.8 / 2.9

10 CFR Part 55: 41.5 / 43.5 / 43.3 / 45.13

10CFR55.43.b: 5, 7

K/A Match: K/A is matched and the question is SRO because the question requires the ability to predict the impact (ventilation system changes & NRC reportability) and use of procedures to mitigate the consequences of the failure.

Technical Reference: SOI-30.05, Auxiliary Bldg HVAC Systems, Rev. 0051
NPG-SPP-03.5, Regulatory Reporting Requirements, Revision 0003

Proposed references None

to be provided:

Learning Objective: 3-OT-SYS030A

6. List the ABI initiation signals.
7. Explain what events take place on an ABI and why.

3-OT-SSP0305

10. Given a set of normal or abnormal plant conditions, determine whether the event requires reporting to the NRC, the FAA, or TEMA.

Cognitive Level:

Higher	X
Lower	

Question Source:

New	
Modified Bank	X
Bank	

Question History: Sequoyah bank question 072 A2.02 modified for use on the WBN 10/2011 NRC exam. SQN question used on the SQN 01/09 NRC exam.

Comments:

Date______ Initials

8.8 Recovery from Aux Bldg Isolation (ABI)

NOTES

1) If ABI occurred due to an exhaust vent Rad monitor, an air sample by Radiological Protection may be necessary.

2) 0-RE-90-102 AND -103, FUEL POOL RADIATION MONITORS, signals do **NOT** require ABI reset; only detectors must be cleared.

[1] **PLACE** the following AB Gen HVAC fans in STOP, PULL-TO-LOCK:

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL
AB GEN EXHAUST FAN 1A & DISCH FCO-30-159	1-M-9	STOP, PULL-TO-LOCK	1-HS-30-159A	
AB GEN EXHAUST FAN 1B & DISCH FCO-30-162	1-M-9	STOP, PULL-TO-LOCK	1-HS-30-162A	
AB GEN SUPPLY FAN 1A & DISCH FCO-30-103	1-M-9	STOP, PULL-TO-LOCK	1-HS-30-103A	
AB GEN SUPPLY FAN 1B & DISCH FCO-30-102	1-M-9	STOP, PULL-TO-LOCK	1-HS-30-102A	
AB GEN EXHAUST FAN 2A & DISCH FCO-30-274	1-M-9	STOP, PULL-TO-LOCK	2-HS-30-274A	
AB GEN EXHAUST FAN 2B & DISCH FCO-30-278	1-M-9	STOP, PULL-TO-LOCK	2-HS-30-278A	
AB GEN SUPPLY FAN 2A & DISCH FCO-30-104	1-M-9	STOP, PULL-TO-LOCK	2-HS-30-104A	
AB GEN SUPPLY FAN 2B & DISCH FCO-30-105	1-M-9	STOP, PULL-TO-LOCK	2-HS-30-105A	
FH AREA EXH FAN A & DISCH FCO-30-136	1-M-9	STOP, PULL-TO-LOCK	0-HS-30-136A	
FH AREA EXH FAN B & DISCH FCO-30-139	1-M-9	STOP, PULL-TO-LOCK	0-HS-30-139A	

Date______ **Initials**

8.8 Recovery from Aux Bldg Isolation (ABI) (continued)

[2] **WHEN** initiating signal is **CLEAR, THEN**

RESET ABI signal with 1-HS-30-101A, AUX BLDG ISOL TR-A, and 1-HS-30-101B, AUX BLDG ISOL TR-B [1-M-6]. ________

[3] **OPEN** the following dampers with HSs listed below: ________

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL
AB GEN EXH FAN 1A SUCT	1-M-9	OPEN	1-HS-30-160	
AB GEN EXH FAN 1A SUCT	1-M-9	OPEN	1-HS-30-161	
AB GEN EXH FAN 1B SUCT	1-M-9	OPEN	1-HS-30-166	
AB GEN EXH FAN 1B SUCT	1-M-9	OPEN	1-HS-30-167	
AB GEN EXH FAN 2A SUCT	1-M-9	OPEN	2-HS-30-271	
AB GEN EXH FAN 2A SUCT	1-M-9	OPEN	2-HS-30-272	
AB GEN EXH FAN 2B SUCT	1-M-9	OPEN	2-HS-30-275	
AB GEN EXH FAN 2B SUCT	1-M-9	OPEN	2-HS-30-276	
F H AREA FAN A EXH DISCH	1-M-9	OPEN	0-HS-30-137	
F H AREA FAN A EXH DISCH	1-M-9	OPEN	0-HS-30-138	
F H AREA FAN B EXH DISCH	1-M-9	OPEN	0-HS-30-140	
F H AREA FAN B EXH DISCH	1-M-9	OPEN	0-HS-30-141	
U2 AB GEN SUP OUTLET	1-M-9	OPEN	2-HS-30-108	
U2 AB GEN SUP OUTLET	1-M-9	OPEN	2-HS-30-109	
U2 AB GEN SPACES SUPPLY	1-M-9	OPEN	2-HS-30-22	
U1 AB GEN SPACES & FH AREA SUP	1-M-9	OPEN	1-HS-30-106	
U1 AB GEN SPACES & FH AREA SUP	1-M-9	OPEN	1-HS-30-107	
U1 AB GEN SPACES SUPPLY	1-M-9	OPEN	1-HS-30-86	
U1 AB GEN SPACES SUPPLY	1-M-9	OPEN	1-HS-30-87	
CASK LOAD AREA SUPPLY	1-M-9	OPEN	0-HS-30-129	
CASK LOAD AREA SUPPLY	1-M-9	OPEN	0-HS-30-130	
CASK LOAD AREA EXHAUST	1-M-9	OPEN	0-HS-30-122	
CASK LOAD AREA EXHAUST	1-M-9	OPEN	0-HS-30-123	
U2 AB GEN SPACES SUPPLY	1-M-9	OPEN	2-HS-30-21	

Date______ Initials

8.8 Recovery from Aux Bldg Isolation (ABI) (continued)

[4] **RESET** the following for Exhaust Fans that indicate tripped on 1-M-9, Fuel Handling Area and AB General Exh Fans Shunt Trip Bkr Status (**N/A** Bkrs **NOT** tripped).

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL
AB GEN EXH FAN 1A ISOLATION SWITCH	480V SD Bd Rm 1A	RESET/ON	1-SW-30-159	
AB GEN EXH FAN 1B ISOLATION SWITCH	480V SD Bd Rm 1B	RESET/ON	1-SW-30-162	
AB GEN EXH FAN 2A ISOLATION SWITCH	480V SD Bd Rm 2A	RESET/ON	2-SW-30-274	
AB GEN EXH FAN 2B ISOLATION SWITCH	480V SD Bd Rm 2B	RESET/ON	2-SW-30-278	
FUEL HANDLING EXH FAN A ISOLATION SWITCH	480V SD Bd Rm 2A	RESET/ON	0-SW-30-136	
FUEL HANDLING EXH FAN B ISOLATION SWITCH	480V SD Bd Rm 2B	RESET/ON	0-SW-30-139	

[5] **START** AB General Supply, Exhaust Fans and Fuel Handling Exhaust Fan per Section 8.16. ________

CORRECT

[6] **ENSURE** ABGTS is SHUTDOWN and aligned to STANDBY per SOI-30.06, UNLESS needed for cleanup. ________

[7] **RETURN** ESF to Area Coolers to Auto per Section 5.7. ________

NOTE

N/A the following step if ABI signal was **NOT** initiated. (0-RE-90-102 AND -103, FUEL POOL RADIATION MONITORS, do **NOT** initiate an ABI signal; therefore, the PASF is **NOT** affected.

DISTRACTOR

[8] **ENSURE** PASF recovered from ABI per SOI-31.02, Section 8.3, Recovery from Aux Bldg Isolation (ABI). ________

[9] **ENSURE** AB Hydrogen System in desired configuration per SOI-77.09. ________

End of Section

Appendix A
(Page 1 of 13)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

1.0 PURPOSE

This Appendix identifies reporting requirements; and instructions for determining reportability, preparation, and transmittal of LERs; and notification to NRC for events occurring at TVA's licensed nuclear plants.

2.0 SCOPE

TVA is required by §50.72 and §50.73 to promptly report various types of conditions or events and provide written follow-up reports, as appropriate. This appendix provides reporting guidance applicable to licensed power reactors.

NOTES

1) Appendix B provides additional reporting criteria found in §Part 20, 30, 40, and 70 that may be applicable to events involving byproduct, source or special nuclear material possessed by the licensed nuclear plant. Site Licensing and Site RadCon are responsible for making the reportability determinations for §Part 20, 30, 40, or 70 events associated with their site. Corporate Licensing and Corporate RadChem are responsible for making the reportability determinations for §Part 20, 30, 40, or 70 events associated with all other TVA licensed activities. Licensing is responsible for developing (with input from affected organizations) and submitting the immediate notification and written reports to NRC in accordance with §Part 20, 30, 40, or 70 requirements. Reporting requirements for personnel exposure required by §Part 20 are contained in RCTP-105, Personnel Inprocessing and Dosimetry Administrative Processes.

2) Appendix C contains the criteria for reporting if events or conditions affecting ISFSI. TVA, as the general licensee of the ISFSI, is required by §72.216 to make initial and written reports in accordance with §72.74 and §72.75. Operations is responsible for making the reportability determinations for §72.74 and §72.75 reports. For any event, condition, or issue having the potential for being reportable, contact Site Licensing for consultation and concurrence on the reportability determination. In no event shall the lack of licensing concurrence result in a failure to meet specified reporting timeframes. Operations is responsible for making the immediate notification to NRC in accordance with §72.74. Operations is responsible for making the immediate, 4-hour, and 24-hour notifications to NRC in accordance with §72.75. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by §72.75.

3) Reporting requirements for events or conditions affecting the physical protection of the licensed nuclear plant specified in §73.71 are contained in NSDP-1, Safeguards Event Reporting Guidelines. Responsibilities for reportability determinations and immediate notification requirements are assigned to Site Nuclear Security and Corporate Nuclear Security. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by §73.71.

Appendix A
(Page 2 of 13)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.0 REQUIREMENTS

NOTES

1) Internal management notification requirements for plant events are found in Appendix D. The Operations Shift Manager is responsible for notifying Site Operations Management and the Duty Plant Manager. The Duty Plant Manager is responsible for making the remaining internal management notifications.

2) NRC NUREG-1022, Supplements and subsequent revisions should be used as guidance for determining reportability of plant events pursuant to §50.72 and §50.73. A text searchable copy of NUREG-1022 is maintained on the TVA NPG Nuclear Licensing Webpage at address http://tvanweb.cha.tva.gov/licensing/Pages/NRC-Industry_Guidance_Documents.htm.

3.1 Immediate Notification - NRC

TVA is required by §50.72 to notify NRC immediately if certain types of events occur. This appendix contains the types of events and the allotted time in which NRC must be notified. (Refer to Form NPG-SPP-03.5-1 or NRC Form 361). Operations is responsible for making the reportability determinations for §50.72 and §50.73 reports. For any event, condition, or issue having the potential for being reportable, contact Site Licensing for consultation and concurrence on the reportability determination. In no event shall the lack of licensing concurrence result in a failure to meet specified reporting timeframes. Operations is responsible for making the immediate notification to NRC in accordance with §50.72.

Notification is via the Emergency Notification System. If the Emergency Notification System is not operative, use a telephone, telegraph, mailgram, or facsimile.

NOTE

The NRC Event Notification Worksheet may be used in preparing for notifying the NRC. This Worksheet may be obtained directly from the NRC website (www.nrc.gov) under Form 361, or TVA NPG Form NPG-SPP-03.5-1 may be used.

A. The Immediate Notification Criteria of §50.72 is divided into 1-hour, 4-hour, and 8- hour phone calls. Notify the NRC Operations Center within the applicable time limit for any item which is identified in the Immediate Notification Criteria.

B. The following criteria require 1-hour notification:

1. (Technical Specifications) - Safety Limits as defined by the Technical Specifications which have been violated.

2. §50.72 (a)(1)(i) - The declaration of any of the Emergency classes specified in the licensee's approved Emergency Plan.

Appendix A
(Page 3 of 13)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

> **NOTE**
>
> If it is discovered that a condition existed which met the Emergency Plan criteria but no emergency was declared and the basis for the emergency class no longer exists at the time of discovery, an ENS notification (and notification of the Operations Duty Specialist), within one hour of discovery of the undeclared (or misclassified) event, shall be made. However, actual declaration of the emergency class is not necessary in these circumstances.

3. §50.72(b).(1)) - Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x).

C. The following criteria require 4-hour notification:

1. §50.72(b)(2)(i) - The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

2. §50.72(b)(2)(iv)(A) - Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

3. §50.72(b)(2)(iv)(B) - Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

> **NOTE**
>
> NPG-SPP-05.14 provides additional instructions regarding addressing and informally communicating events to outside agencies involving radiological spills and leaks.

4. §50.72(b)(2)(xi) - Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactive contaminated materials.

D. The following criteria require 8-hour notification:

Appendix A

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

> **NOTE**
>
> The non-emergency events specified below are only reportable if they occurred within three years of the date of discovery.

1. §50.72(b)(3)(ii)(A) - Any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

2. §50.72(b)(3)(ii)(B) - Any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.

3. §50.72(b)(3)(iv)(A) - Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) [see list below], except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

   a. Reactor protection system (RPS) including: Reactor scram and reactor trip.

   > **NOTE**
   >
   > Actuation of the RPS when the reactor is critical is also reportable under §50.72(b)(2)(iv)(B) above.

   (1) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

   (2) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: High-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

   (3) ECCS for boiling water reactors (BWRs) including: core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

   (4) BWR reactor core isolation cooling system.

   (5) PWR auxiliary or emergency feedwater system.

   (6) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

Appendix A
(Page 5 of 13)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

(7) Emergency ac electrical power systems, including: Emergency diesel generators (EDGs).

4. §50.72(b)(3)(v) - Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:

   (A) Shut down the reactor and maintain it in a safe shutdown condition;

   (B) Remove residual heat;

   (C) Control the release of radioactive material; or

   (D) Mitigate the consequences of an accident.

NOTE

According to §50.72 (b)(3)(vi) events covered by §50.72(b)(3)(v) may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.

5. §50.72(b)(3)(xii) - Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

6. §50.72(b)(3)(xiii) - Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of control room indication, emergency notification system, or offsite notification system).

E. Follow-up Notification (§50.72(c))

With respect to the telephone notifications made under paragraphs (a) and (b) [§50.72 (a) and §50.72 (b), respectively] of this section [§50.72], in addition to making the required initial notification, during the course of the event:

1. Immediately report:

Appendix A
(Page 6 of 13)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

(i) Any further degradation in the level of safety of the plant or other worsening plant conditions including those that require the declaration of the Emergency Classes, if such a declaration has not been previously made; or

(ii) Any change from one Emergency Class to another, or

(iii) A termination of the Emergency Class.

(1) Immediately report:

(i) The results of ensuing evaluations or assessments of plant conditions,

(ii) The effectiveness of response or protective measures taken, and

(iii) Information related to plant behavior that is not understood.

(2) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.

3.2 Twenty-Four Hour Notification - NRC

Any violation of the requirement contained in specific operating license conditions, shall be reported to NRC in accordance with the license condition.

3.3 Two-Day Notification - NRC

§50.9(b) - The NRC shall be notified of incomplete or inaccurate information which contains significant implications for the public health and safety or common defense and security. Notification shall be provided to the administrator of the appropriate regional office within two working days of identifying the information. Licensing is responsible for determining reportability (with input from affected organizations) and notifying NRC in accordance with §50.9.

3.4 Sixty-Day Verbal Report

§50.73(a)(2)(iv)(A) requires that any event or condition that resulted in manual or automatic actuation of the specified systems be reported as a Licensee Event Report (LER [Refer to Appendix A, Section 3.5]). This CFR section also allows that in the case of an invalid actuation, other than actuation of the reactor protection system when the reactor is critical, an optional telephone notification may be placed to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER.

3.4 Sixty-Day Verbal Report (continued)

A. Verbal Report Required Content:

If the verbal notification option is selected (NUREG 1022, Revision 2, Section 3.2.6., System Actuation), instead of an LER, the verbal report:

1. Is not considered an LER.

2. Should identify that the report is being made under §50.73(a)(2)(iv)(A).

3. Should provide the following information:

   a. The specific train(s) and system(s) that were actuated.

   b. Whether each train actuation was complete or partial.

   c. Whether or not the system started and functioned successfully.

NOTE

Licensing will ensure that the information that is provided to NRC during the Sixty-Day Verbal Report is verified in accordance with BP-213.

B. Verbal Report Development and Review

Licensing will:

1. Develop (with input from responsible organization) the response (i.e., report summary) to address the required input.

2. Ensure that the reporting details are approved by site vice president or his designee prior to making the verbal report.

C. Telephone Report Timeliness

Operations will make the 60-day telephone report promptly after the response is approved by the site vice president or his designee.

3.5 Written Report - NRC

A. A report on a Safety Limit Violation shall be submitted to the NRC, the NSRB, and the Site Vice President if required by Technical Specifications.

B. Any violation of the requirements contained in the Operating license conditions in lieu of other reporting requirements requires a written follow-up report if specified in the license.

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

C. Reporting Radiation Injuries

1. §140.6(a) requires, as promptly as possible, submittal of a written notice [e.g., report] in the event of:

    a. Bodily injury or property damage arising out of or in connection with the possession or use of the radioactive material at the licensee's facility [location]; or

    b. In the course of transportation; or

    c. In the event any radiation exposure claim is made. (Refer to RCDP-9, Radiological and Chemistry Control Radiological Exposure Inquiries)

2. The written notice shall contain particulars sufficient to identify the licensee and reasonably obtainable information with respect to time, place, and circumstances thereof, or the nature of the claim.

D. Licensee Event Reports

A written report shall be prepared in accordance with §50.73(a)(i) for items in the 60-day report criteria or Technical Specifications. The report shall be complete and accurate in accordance with the methods outlined in this procedure. The completed forms shall be submitted to the USNRC, Document Control Desk, Washington, DC 20555. NUREG 1022, Revision 2, contains the instructions for completion of the LER form. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports (or optional telephone reports [refer to Appendix A, Section 3.4]) required by §50.73.

> **NOTE**
>
> Unless otherwise specified in the reporting criteria below, an event shall be reported if it occurred within three years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

E. Report Criteria

1. §50.73(a)(2)(i)(A) - The completion of any nuclear plant shutdown required by the plant's Technical Specifications.

2. §50.73(a)(2)(i)(B) - Any operation or condition which was prohibited by the plant's Technical Specifications, except when:

    a. The Technical Specification is administrative in nature;

072 A2.02 093

Given the following:

SQN BANK QUESTION

- Unit 1 operating at 100% power.
- Unit 2 in MODE 6 with the core off-load in progress.
- An internal electrical failure causes the output of 0-RM-90-103, "Spent Fuel Pit Area Radiation Monitor" to fail above the HI RAD setpoint.

Which ONE of the following identifies the recovery required using 0-SO-30-10 "Auxiliary Building Ventilation Systems" as a result of the Auxiliary Building Isolation (ABI) and whether the actuation is required to be reported to the NRC in accordance with SPP-3.5, "NRC Reporting Requirements"?

A. Only Train B recovery is required;
8-hour notification required.

B.✓ Only Train B recovery is required;
8-hour notification NOT required.

C. Only Train A recovery is required;
8-hour notification required.

D. Only Train A recovery is required;
8-hour notification NOT required.

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training

B. Non-License Training

C. License Requal

D. Non-License Requal

III. TITLE

AUXILIARY BUILDING VENTILATION SYSTEM

IV. LENGTH OF LESSON

A. Licensed Training	2.0 hours
B. Non-Licensed Training	2.0 hours
C. NOTP	4.0 hours

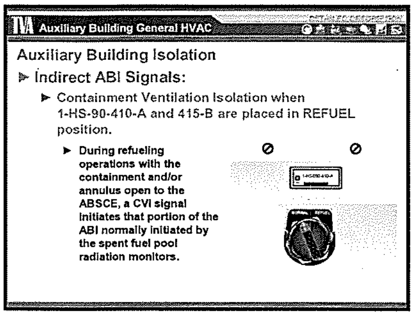
V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. State the design basis of the Auxiliary Building Heating/Ventilation/Air Conditioning system in accordance with FSAR section 9.4.3.
	X	X	X	2. 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	3. Explain when Auxiliary Building area temperature monitoring is applicable.

AUO	RO	SRO	STA	
X	X	X	X	4. Describe all Auxiliary Building Heating/Ventilation/Air-Conditioning System major components as to power supplies, start and stop logic and operations.
X	X	X	X	5. Explain why building pressure control is important in the Auxiliary, Control and Annulus Buildings.
X	X	X	X	6. List the ABI initiation signals.
X	X	X	X	7. Explain what events take place on an ABI and why.
X	X	X	X	8. Discuss how the Shutdown Board rooms are ventilated and cooled.
X	X	X	X	9. Describe the purposes of the 480v Electrical Board Rooms Pressurizing Fans.
X	X	X	X	10. Describe the purpose for ventilation in the 125v Vital Battery rooms.
X	X	X	X	11. State the temperature at which the exhaust fans for the north and south valve vault rooms start.
X	X	X	X	12. Explain the primary concern with a temperature of < 50°F in the north and south valve vaults.
X	X	X	X	13. Describe the electrical logic, including trips, for the Fuel Handling Exhaust Fans.
X	X	X	X	14. Describe how the ECCS rooms and ESF areas are cooled.
X	X	X	X	15. Describe the emergency ventilation systems provided for the Turbine Driven Auxiliary FW Pump Rooms.
X	X	X	X	16. Explain how the Post Accident Sampling Facility is ventilated.
X	X	X	X	17. [Explain the need for checking the position of backdraft dampers on ventilation equipment when standby fans are placed in service and/or on shift routines.] WBPER920218
X	X	X	X	18. Explain the purpose of the Auxiliary Building Heating System.
X	X	X	X	19. State the purpose and describe the basic operation of the major components of the Auxiliary Building Heating System.

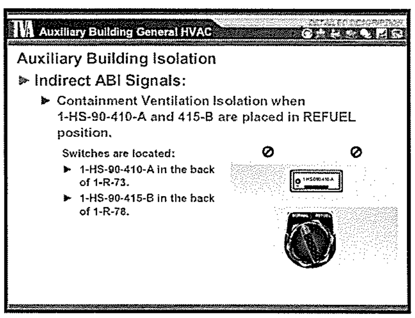
A U O	R O	S R O	S T A	
X	X	X	X	20. Explain how the hot water flow to the Auxiliary Building preheat coils is regulated.
X	X	X	X	21. State how to place the Auxiliary Building air preheating system in service.
X	X	X	X	22. List the precautions concerning the Auxiliary Building heat system.
X	X	X	X	23. Describe what must be done if the air temperature entering the auxiliary building heating and cooling coils is < 35°F.
X	X	X	X	24. Explain the purpose of the Auxiliary Building Cooling System.
X	X	X	X	25. Explain the basic operating principle of the Auxiliary Building General Vent Chiller Package.
X	X	X	X	26. Describe the chilled water flowpath through the Auxiliary Building Cooling System.
	X	X	X	27. Correctly locate control room controls and indications associated with the Auxiliary Building Heating/Ventilation/Air Conditioning system, including: a. Auxiliary Building General Supply Fans b. Auxiliary Building General Exhaust Fans c. Tornado Dampers d. Battery Room Exhaust Fans e. Shutdown Board Room Pressurizing Fans
X	X	X	X	28. Describe the in-plant location of the following: a. Auxiliary Building General Supply and Exhaust Fans b. Fuel Handling Area Exhaust Fans c. Auxiliary Building General Vent Chillers d. Auxiliary Building General Vent Chillers Chilled Water System Components

SLIDE 46



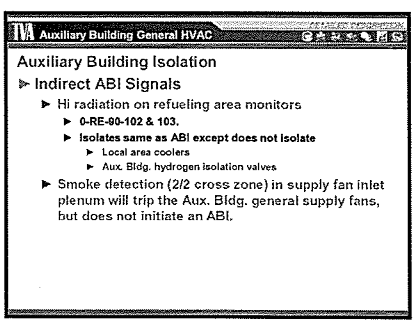
- Indirect ABI Signals:
  - Containment Ventilation Isolation when 1-HS-90-410-A and 415-B are placed in REFUEL position
  - During refueling operations with the containment and/or annulus open to the ABSCE, a CVI signal initiates that portion of the ABI normally initiated by the spent fuel pool radiation monitors.

SLIDE 47



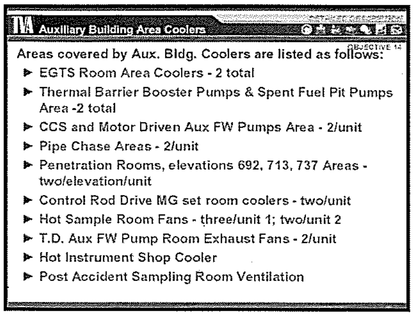
- Switches are located:
  - 1-HS-90-410-A in the back of 1-R-73.
  - 1-HS-90-415-B in the back of 1-R-78.

SLIDE 48



- Hi radiation on refueling area monitors
  - 0-RE-90-102 & 103.
  - Isolates same as ABI except doesn't isolate
    - Local area coolers
    - Aux. Bldg. hydrogen isolation valves
- Smoke detection (2/2 cross zone) in supply fan inlet plenum will trip the Aux. Bldg. general supply fans, but does not initiate an ABI.

SLIDE 111

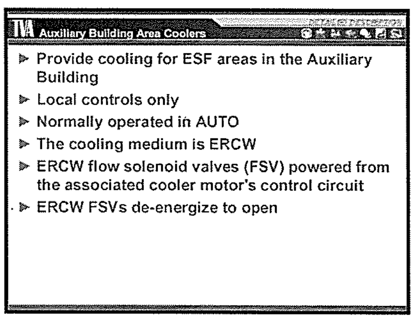


Auxiliary Building Area Coolers

1. Discuss the areas covered by the Auxiliary Building Area Coolers.

- EGTS Room Area Coolers - 2 total
- Thermal Barrier Booster Pumps & Spent Fuel Pit Pumps Area -2 total
- CCS and Motor Driven Aux FW Pumps Area - 2/unit
- Pipe Chase Areas - 2/unit
- Penetration Rooms, elevations 692, 713, 737 Areas - two/elevation/unit
- Control Rod Drive MG set room coolers - two/unit
- Hot Sample Room Fans - three/unit 1; two/unit 2
- T.D. Aux FW Pump Room Exhaust Fans - 2/unit
- Hot Instrument Shop Cooler
- Post Accident Sampling Room Ventilation.

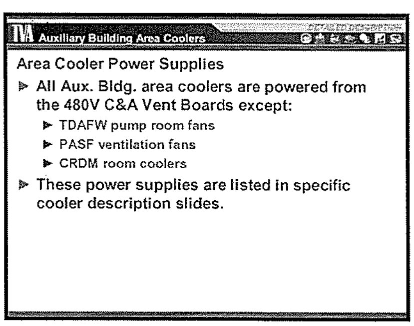
SLIDE 112



2. Discuss the function and operation of the Auxiliary Building Area Coolers.

- Area coolers provide cooling for ESF areas in the Auxiliary Building.
- The coolers have local controls only.
- All area coolers are normally operated in AUTO.
- The cooling medium for the Aux. Building area coolers is ERCW.
- The ERCW flow solenoid valves (FSV) are powered from the associated cooler motor's control circuit.
- ERCW FSVs de-energize to open.

SLIDE 113



3. Discuss the Auxiliary Building Area Cooler power supplies.

- All Aux. Bldg. area cooler are powered from the 480V C&A Vent Boards except:
  - TDAFW pump room fans
  - PASF ventilation fans
  - CRDM room cooler
- These power supplies are listed in specific cooler description slides.

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. Initial License Training
B. Licensed Operator Requal
C. Shift Technical Advisor

III. TITLE

NPG-SPP-03.5, REGULATORY REPORTING REQUIREMENTS

IV. LENGTH OF LESSON

A. Initial License Training 2.0 Hours
B. Shift Technical Advisor 2.0 Hours
C. Licensed Operator Requalification times will be determined when objectives are identified.

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	1. Identify the Plant events requiring immediate notification of the NRC per 10CFR50.72 (1 HOUR), as specified in NPG-SPP-03.5 Appendix A.
	X	X	X	2. Identify the Plant events requiring immediate notification of the NRC per 10CFR50.72 (4 HOUR), as specified in NPG-SPP-03.5 Appendix A.
	X	X	X	3. Identify the Plant events requiring immediate notification of the NRC per 10CFR50.72 (8 HOUR), as specified in NPG-SPP-03.5 Appendix A.
	X	X	X	4. Identify the Plant events requiring immediate notification of the NRC per 10CFR50.72 (24 HOUR), as specified in NPG-SPP-03.5 Appendix A.
	X	X	X	5. Identify the criteria requiring submission of a License Event Report (LER) specified in 10CFR50.73 and outlined in NPG-SPP-03.5 Appendix A

	X	X	X	6. Identify the criteria requiring immediate notification, 1-hour, 4-hour and 24-hour notification as specified in NPG-SPP-03.5 Appendix B.
	X	X	X	7. Identify the criteria requiring immediate notification, 1-hour, 4-hour and 24-hour notification to the NRC in accordance with 10CFR72.74 and 72.75 as specified in NPG-SPP-03.5 Appendix C
	X	X	X	8. Identify the criteria requiring notification per Appendix D and Appendix E of NPG-SPP-03.5.
	X	X	X	9. Identify criteria covered in Appendix F through K of NPG-SPP-03.5.
	X	X	X	10. Given a set of normal or abnormal plant conditions, determine whether the event requires reporting to the NRC, the FAA, or TEMA.

VI. **TRAINING AIDS**

A. Marker Boards & Markers
B. 10CFR Parts 1 to 50 and 10CFR Parts 51 to 199
C. Instructor preference-NRC web page with 10CFR documents

VII. **MATERIALS**

Appendix A
Copy of Reporting Requirements Worksheet and Key

Attachment 1
Copy of NPG-SPP-03.5 Regulatory Reporting Requirements, Current Revision
Attachment 2
PowerPoint Presentation, current revision

VIII. **REFERENCES**

A. [Commitments]

1. NRC None

2. [Text in this lesson plan which is enclosed in brackets [] shall not be altered or deleted without approval by the Operations Training Manager.]

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

Some examples of SRO exam items for this topic include:

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

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

Some examples of SRO exam items for this topic include:

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

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

93. 086 G2.2.22 093
Which ONE of the following identifies...

- Unit 1 is at 4% power after being restarted from a refueling outage.
- An Appendix R valve is determined to be out of its required Fire Safe Shutdown (FSSD) condition.

Which ONE of the following identifies..

(1) the maximum time allowed by OR 14.10 to restore the valve to the proper FSSD condition before additional action is required

and

(2) how the planned entry into Mode 1 will be affected?

	Max. Time	Mode 1 entry
A.	14 days	Is not restricted by the action statement.
B.	14 days	Prohibited until the action statement is cleared.
C✓	30 days	Is not restricted by the action statement.
D.	30 days	Prohibited until the action statement is cleared.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because 14 days is the maximum time allowed for other 14.10 actions and being in an action not restricting Mode changes is correct. Restricting the entry if a required valve is not is the proper position (similiar to Tech Spec) is plausible because there are restrictions associated with OR 14.10 and there are mode reduction requirements similar to Tech Spec 3.0.3.*

B. *Incorrect, Plausible because 14 days is the maximum time allowed for other 14.10 actions and because there are Tech Spec Conditions and Required Actions that do restrict mode changes. Restricting the entry if a required valve is not is the proper position (similar to Tech Spec) is plausible because there are restrictions associated with OR 14.10 and there are mode reduction requirements similar to Tech Spec 3.0.3.*

C. *Correct, Section 14.10.2 states "With one or more of the breakers and/or valves specified in design output documents not in the noted position or condition, return the breakers and/or valve to the required position within 30 days." Section 14.0 has a statement identifying 'The Fire Protection Report does not have a requirement similar to Technical Specifications 3.0.4 preventing mode changes while in an action statement.'*

D. *Incorrect, Plausible because 30 days being the maximum time allowed is correct and because there are Tech Spec Conditions and Required Actions that do restrict mode changes. Restricting the entry if a required valve is not is the proper position (similar to Tech Spec) is plausible because there are restrictions associated with OR 14.10 and there are mode reduction requirements similar to Tech Spec 3.0.3.*

Question Number: 93

Tier: 2 **Group** 2

K/A: 086 G2.2.22
Fire Protection System
Equipment Control
Knowledge of limiting conditions for operations and safety limits.

Importance Rating: 4.0 / 4.7

10 CFR Part 55: 41.5 / 43.2 / 45.2

10CFR55.43.b: 2

K/A Match: K/A is matched and is SRO because the question requires knowledge of the limiting conditions for operations for components identified in OR 14.10 because it requires knowledge of how a

K/A Match: K/A is matched and is SRO because the question requires knowledge of the limiting conditions for operations for components identified in OR 14.10 because it requires knowledge of how a limiting conditions for operations will affect a planned Mode change.

Technical Reference: WBN Fire Protection Report, Revision 46
Part II - Fire Protection Report,
14.0 Fire Protection Systems and Features Operating Requirements (OR) page revision 19
14.2 Water Supply, page revisions 34 and 28

Proposed references to be provided: None

Learning Objective: 3-OT-SYS26A
26. Given the condition/status of the HPFP system/ component and the appropriate sections of Fire Protection Plan, determine if operability requirements are met and what actions, if any, are required.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	X
Modified Bank	
Bank	

Question History: WBN bank question MSC0047 001modified for the WBN 10/2011 exam

Comments:

SECTION 14.0 TABLE OF CONTENTS

14.0 FIRE PROTECTION SYSTEMS AND FEATURES OPERATING REQUIREMENTS (OR)

The OR established in this section have been developed to ensure adequate fire protection capability is available and maintained, to detect, control, and extinguish fires occurring in any portion of the plant where safety-related or FSSD equipment are located.

Fire protection systems and features at WBN are not assumed to be operable to mitigate the consequences of a Design Basis Accident (DBA) or plant transient. The bases for this assumption are contained in Section I of Appendix R which states that the need to limit fire damage to systems required to achieve and maintain FSSD conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of DBAs. As a result, Section I identifies that fire protection features must be capable of limiting fire damage so that:

1. One train of systems necessary to achieve and maintain hot shutdown conditions from either the control room, auxiliary control room, or emergency control stations is free of fire damage; and

2. Systems necessary to achieve and maintain cold shutdown from either the control room, auxiliary control room, or emergency control stations can be repaired within 72 hours.

3. Alternate shutdown capability is provided at WBN, when needed, to achieve and maintain cold shutdown within 72 hours.

Operability of the fire protection systems and features are required whenever safety-related equipment and fire safe shutdown systems protected by the fire protection systems and features are required to be Operable.

The Fire protection Report provides applicable action statements and thus does not have a requirement similar to Technical Specification 3.0.3 except for equipment listed in Section 14.10. When a piece of equipment in section 14.10 is out of service, there are mode reduction requirements similar to Technical Specification 3.0.3. However, equivalent methods (documented in an engineering evaluation in accordance with site procedures) that ensure fire safe shutdown can be achieved per the requirements of 10CFR50, Appendix R may be used to delay or remove the mode reduction requirements. These equivalent methods once documented by engineering evaluation provide alternatives to the applicable actions statements when equipment listed in Part II, Section 14.10 must be declared inoperable.

The Fire Protection Report does not have a requirement similar to Technical Specifications 3.0.4 preventing mode changes while in an action statement.

The Testing and Inspection Requirements (TIRs) for the WBN fire protection systems and features have been developed taking into consideration industry practice (e.g., similar methods approved for use by other licensed nuclear power facilities), NFPA consensus standards, and insurance carrier loss prevention recommendations.

14.9 Emergency Battery Lighting Units

Emergency battery lighting units provided for FSSD shall be Operable whenever the illuminated associated fire safe shutdown equipment is required.

14.9.1 With any of the emergency battery lighting units provided for FSSD inoperable, restore the inoperable units to Operable status within 24 hours -OR- ensure alternate lighting is available.

14.9.2 Restore the inoperable emergency battery lighting unit to Operable status within 14 days. If not restored within 14 days, continue the compensatory actions AND perform 10CFR50.72 and/or 10CFR50.73 reviews per site administrative procedures.

14.10 Fire Safe Shutdown Equipment

The equipment listed on Table 14.10 is required for Fire Safe Shutdown(FSSD) and shall be Operable (or in its FSSD condition) when the unit is in modes 1, 2, and 3. The non-System 26 valves noted on the plants mechanical flow diagrams as being administratively locked in the open, closed, or throttled position (with breaker open) for Appendix R shall be maintained in that condition when the unit is in Modes 1, 2 and 3.

14.10.1 With one or more required equipment in Table 14.10 inoperable (or not in its FSSD condition), restore to operable status (or its FSSD condition) within 30 days.

14.10.2 With one or more of the breakers and/or valves specified in design output documents not in the noted position or condition, return the breakers and/or valve to the required position within 30 days.

14.10.3 If required action and associated completion time cannot be met,

a. place the equipment in the condition required for FSSD, -OR-

b. provide a back-up means of instrumentation monitoring for the equipment in Table 14.10, -OR-

c. perform an evaluation to justify using alternate means to provide FSSD, -OR-

d. be in Mode 3 within 6-hours and Mode 4 within the following 12-hours.

BASES - OPERATING REQUIREMENTS (OR)
FIRE SAFE SHUTDOWN EQUIPMENT

B.14.10 A minimum set of plant systems and components has been identified at WBN to ensure that the plant can achieve and maintain safe shutdown in the event of plant fires (see Part III, Safe Shutdown Capabilities). In the majority of cases the identified plant systems and components are addressed by WBN Technical Specifications and Technical Requirements Manual which list surveillance requirements for verifying the Operability of the systems and components. This OR lists the systems and components which are not included as part of a Technical Specification or Technical Requirement.

Thermal overloads that are by-passed during accident conditions must remain operable during normal plant operation. This will ensure that valves that are required for a Control Building fire are not damaged due to a hot short that could by-pass the torque switch. In addition, the thermal overloads are required for limiting current flow in the event of fire induced multiple high impedance faults and documented in the Multiple High Impedance Fault Analysis. The Technical Requirements Manual, Table 3.8.3-1, "Motor-Operated Valves Thermal Overload Devices Which Are Bypassed Under Accident Conditions" provides the list of thermal overloads this statement addresses.

This OR is provided to ensure that systems and components which are required for safe shutdown are maintained operable and tested to ensure operability. The intent of this OR is to ensure the equipment listed in Table 14.10 is capable of performing its FSSD function. To ensure this, equipment listed in Table 14.10 shall satisfy the FSSD Condition listed by being Operable, capable of achieving its FSSD Condition, or in its FSSD Condition. The equipment listed in Table 14.10 is considered inoperable when it is not in or can not achieve its listed FSSD Condition. The actions are based on Technical Specifications 3.3.4, Remote Shutdown System.

B.14.10.1 With a safe shutdown component shown in Table 14.10 inoperable, the inoperable component must be restored within 30 days when the unit is in modes 1, 2, or 3.

Table 14.10 defines the Fire Safe Shutdown (FSSD) condition as “OPERABLE” for the Temperature Control Valves (TCVs) supplying for the Lower Compartment Coolers (LCCs) and Control Rod Drive Mechanism (CRDM) coolers. In Modes 1, 2, or 3, these valves are required to modulate to control temperature to their respective cooler. “OPERABLE” for these TCVs is a position to ensure that their respective cooler has sufficient cooling flow to maintain the Reactor Building Lower Compartment temperature. Functional Evaluations (FEs) have been performed on a single TCV, and determined that with the TCV in the open position it meets the requirement for FSSD condition.

B.14.10.2 With a breaker and/or valve specified in design output documents as being administratively controlled for Appendix R out of it's required position (as noted on the drawing), the breaker and/or valve must be returned to the required position within 30 days when the unit is in Modes 1, 2, or 3. These breakers and/or valves are administratively controlled to prevent inadvertent operation during an Appendix R fire event. There is no TIR associated with the OR since the valves and/or breaker positions are controlled by the applicable System Operating Instruction and the plant's configuration control program.

B.14.10.3 If the required action and associated completion time are not met, the plant must be placed in a condition where the OR does not apply. If possible, the inoperable or misconfigured component can be placed in the condition required for safe shutdown (i.e., close a valve, shutdown a pump, lock open a breaker), or a backup instrument can be provided for monitoring temperature, flow, or pressure. If this cannot be accomplished, an evaluation can be performed to justify using an alternate means to achieve compliance with Appendix R FSSD requirements. The plant's Temporary Control and Alteration process (TACF) along with a 10CFR50.59 review can also be used to provide the alternate means of FSSD compliance. If none of the above actions can be accomplished, the unit must be brought to at least Mode 3 within 6-hours and to Mode 4 within the following 12-hours.

BASES - TESTING AND INSPECTION REQUIREMENTS (TIR)
FIRE SAFE SHUTDOWN EQUIPMENT

B.14.10.a TIR 14.10.a is performance of a terminal voltage check and on alignment check of the plant's 250 VDC Batteries 1 and 2. This provides assurance that the batteries are operable and aligned to the appropriate DC bus. This check will be performed at least once every 31 days when the plant is in modes 1, 2, or 3.

B.14.10.b TIR 14.10.b is performance of a breaker alignment check for the 250 VDC Battery Boards 1 and 2 and Distribution Panels 1 and 2. This check provides assurance that breakers which supply control power to steam load trip circuits and RCP breaker trip circuits are aligned properly. This check will be performed at least once every 31 days when the plant is in modes 1, 2, or 3.

B.14.10.c TIR 14.10.c verifies every 18 months that main steam system valves are capable of being closed via Main Control Room switch or locally by manual operation of the valve. This verifies that each valve operates properly to ensure the isolation of main steam loads should main steam isolation valves become inoperable in the event of a fire damage. The valves are tested every 18 months when the unit is shutdown since operation of the valve via the hand switch during operation can cause a reactor trip.

B.14.10.d TIR 14.10.d is performance of a channel calibration on instruments required for safe shutdown. Many of these instruments are required for local operation of plant systems and components during a fire event. The performance of the calibration ensures the accuracy of these instruments should they be required for use. This calibration is performed once per 18 months.

B.14.10.e TIR 14.10.e is performance of in-service testing for CCS pump 2B-B under the augmented in-service testing program. This pump is needed to support Unit 1 fire safe shutdown requirements. The augmented in-service testing program requires a flow verification at least once per 92 days to ensure that the pump is operable.

B.14.10.f TIR 14.10.f verifies every 92 days that RCS Pressurizer Spray Valves are capable of being closed from the Main Control Room controller. The valves are tested every 92 days (quarterly) in accordance with the augmented in-service testing program.

B.14.10.g TIR 14.10.g verifies every 18 months that the Control Rod Drive Cooler Motors and associated dampers operate properly from MCR controls. The CRDM Coolers and dampers are tested every 18 months when the unit is shutdown since these coolers are normally in operation during unit operation. Also, cycling these coolers on and off during plant operation could have an adverse effect on the Rod Position Indication System.

B.14.10.h TIR 14.10.h verifies every 18 months that the Generator Control System Solenoid can be operated from its associated hand switch in the MCR. This test is performed every 18 months when the unit is shutdown since operation of this solenoid will cause a unit trip. The solenoid is tested every 18 months as part of the Technical Requirements Surveillance Program.

B.14.10.i TIR 14.10.i verifies every 18 months that the Lower Compartment Cooler System Temperature Control Valves (TCVs) operate properly from MCR controls. The TCVs are tested every 18 months when the unit is shutdown since these coolers are required for Containment cooling during unit operation.

B.14.10.j TIR 14.10.j.a verifies every 31 days that the nitrogen tanks have the quantity and pressure of nitrogen required for operation of the valves. This check will be performed at least once every 31 days when the plant is in modes 1, 2 or 3.

TIR 14.10.j.b verifies every 18 months that the SG PORVs and AFW LCVs can be operated properly from backup control stations using the compressed nitrogen. The PORVs and LCVs are tested every 18 months when the unit is shutdown since these valves are required to be operable per plant Technical Specifications when the plant is in operating modes 1 through 4 and testing these valves utilizing the nitrogen system would make the valves inoperable.

B.14.10.k TIR 14.10.k verifies every 92 days that the Auxiliary Control Air Compressors are capable of starting automatically if the air receiver pressure drops below a pre-established setpoint. Re-establishing and maintaining system pressure ensure adequate capacity to meet the needs of the small set of components credited for remote pneumatic operation during Fire Safe Shutdown.

B.14.10.l TIR-14.10.l is for tracking only. The thermal overload bypass devices are tested by the Technical Requirements Manual and no further testing is needed. The concern for the FPR is for the overloads to be bypassed and thus defeating their protection features as addressed in the bases to OR-14.10. This provides a method for the surveillance program to ensure OR-14.10 is entered should the associated tests not be performed and the overloads are bypassed.

B.14.10.m TIR 14.10.m verifies every 18 months that the CREATCS Appendix R transfer switches (0-XS-31-12-A and 0-XS-31-11-B) function as intended by the performance of a continuity check. This will ensure that CREATCS is available for local control during an Appendix R fire that takes out the normal control circuit. The continuity test is consistent with the surveillance requirements for other safety-related transfer switches (reference Technical Specification Bases SR3.3.4.2) .

MSC0047 001 WBN BANK QUESTION

Which of the following identifies the allowed out of service time and the MODE applicability for a component listed in Part II - Fire Protection Plan Section 14.10, Fire Safe Shutdown Equipment determined to be inoperable?

	Out of Service time	MODE Applicability
a.	14 days	MODES 1 - 3
b.	14 days	MODES 1 - 4
c.✔	30 days	MODES 1 - 3
d.	30 days	MODES 1 - 4

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. License Training

B. Non-License Training

III. TITLE

High Pressure Fire Protection

IV. LENGTH OF LESSON

A. Licensed Training 2 hours

B. Non-Licensed Training 4 hours

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. State the design basis of the HPFP system in accordance with System Description, N3-26-4002.
X	X	X	X	2. State the function of the HPFP system in accordance with the system description.
X	X	X	X	3. Sketch a basic drawing of the HPFP equipment located at the intake structure; include pumps, strainers, pressure control valve, intake pit and mud valve.
X	X	X	X	4. Describe the HPFP pumps; include capacity, power supplies, type and logic.
	X	X	X	5. Correctly locate and state the normal alignment of control room controls and indications associated with the HPFP system, including: a. Pump handswitches b. Valve handswitches
X	X	X	X	6. State the places the fire pumps can be started and stopped.
	X	X	X	7. Upon HPFP actuation, state the sequence in which the fire pumps start.
X	X	X	X	8. Describe the local checks to be made on the HPFP when in standby and when running.
X	X	X	X	9. State when the valves 0-FCV 26-3 and 0-FCV 26-8, mud valves would be opened.

AUO	RO	SRO	STA	
X	X	X	X	10. Discuss the purpose of 0-PCV 26-18, Back pressure Control Valve.
X	X	X	X	11. State how the HPFP system remains charged / pressurized while the HPFP pumps are off.
				12. Objective Deleted.
X	X	X	X	13. List all the systems that use the Maxitrol Sentry valve.
X	X	X	X	14. Describe the basic construction and operation of a Maxitrol Sentry deluge valve.
X	X	X	X	15. List the different type sprinkler heads used in the HPFP system.
X	X	X	X	16. Describe the basic construction and operation of a Viking deluge valve.
X	X	X	X	17. Explain the difference between a wet header and dry header system.
X	X	X	X	18. Explain how some dry headers are monitored to ensure the heads are intact.
X	X	X	X	19. Given a loss of instrument air/control power and fire header pressure, determine the effect on the following valves. a. Maxitrol Sentry b. Viking
X	X	X	X	20. Explain how the HPFP foam units operate.
X	X	X	X	21. Describe how the 5th DG and the Backup Security DG buildings are protected during fire.
X	X	X	X	22. Briefly explain how to reset the HPFP system after actuation, include resetting of the deluge valves and the foam tanks.
X	X	X	X	23. State the different ways to initiate all the HPFP systems automatically and manually.
X	X	X	X	24. List all uses of HPFP in the Reactor Building.
X	X	X	X	25. Briefly describe the purpose of the periodic HPFP flushes and biocide injection.
	X	X	X	26. Given the condition/status of the HPFP system/component and the appropriate sections of Fire Protection Plan, determine if operability requirements are met and what actions, if any, are required.
X	X	X	X	27. State the reason for not initiating deluge systems on energized equipment except in a case of emergency. (This objective is a commitment and shall not be altered or deleted without the approval of the OTM.)
X	X	X	X	28. Describe the installation and construction of the Diesel Driven Fire Pump.

AUO	RO	SRO	STA	
X	X	X	X	29. Describe the control functions for the Diesel Driven Fire Pump; specifically how the pump is started and stopped (manually and automatically) include interlocks.

VI. TRAINING AIDS

A. Whiteboard and markers.
B. Multimedia projector.
C. Overhead projector

VII. MATERIALS

One copy of each of the following for each participant as required:

Appendices

None

B. Attachments

None

C. WBN Prints (Latest Revision)

1) 47W845-1

2) 47W845-2

3) 47W845-3

4) 47W845-4

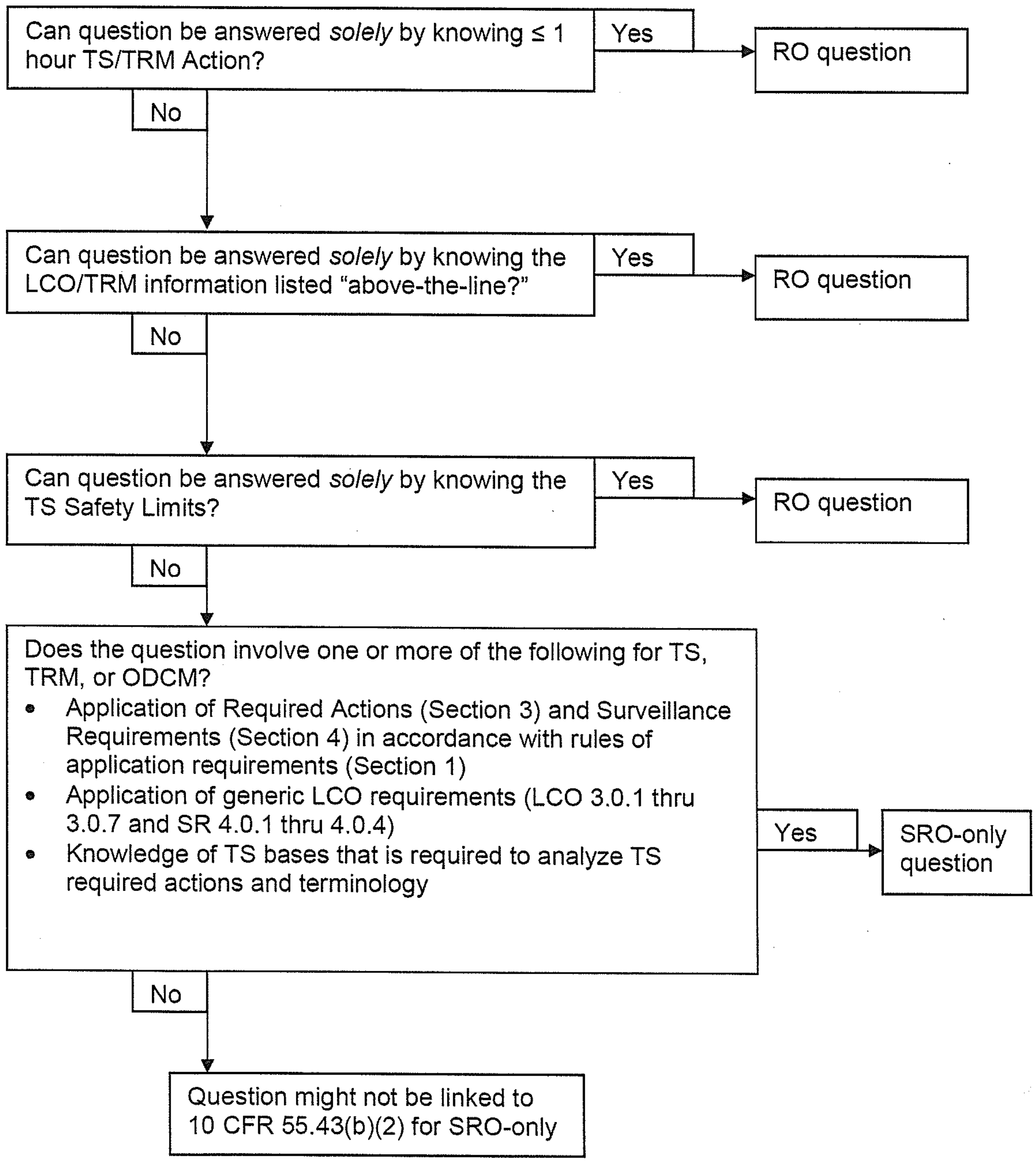
5) 47W845-5

VIII. REFERENCES

A.

COMMITMENTS	
NUMBER	TITLE
1	BFN RCA 89-101 (BFN LER 89-028-01) Fire Deluge System Contributes to Electrical Failure
2	(Internal) Add statement which cautions against initiating fire suppression water deluge systems on energized equipment.
	Text in this lesson plan which is annotated as commitments shall not be deleted without approval by the Operations Training Manager.

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



94. G 2.1.18 094
Given the following plant conditions:

- Unit 1 is Mode 3 following a reactor trip on the previous shift.
- Auxiliary Air Compressor B is tagged for maintenance.
- A failed temperature switch results in an Auxiliary Building Isolation (ABI).
- ABGTS fan A-A trips due to motor failure when it attempts to starts.

In accordance with OPDP-8, "Limiting Conditions for Operation Tracking," which ONE of the following identifies ...

(1) how the ABGTS Train A entry will be designated in the OPDP-8, "LCO Tracking Log"

and

(2) if a 'Loss of Safety Function' has occurred?

	(1)	(2)
A.✓	'Active' LCO	Has occurred
B.	'Information Only' LCO	Has occurred
C.	'Active' LCO	Has **NOT** occurred
D.	'Information Only' LCO	Has **NOT** occurred

DISTRACTOR ANALYSIS:

A. Correct, The ABGTS is required in Mode 3, thus the fan failure resulting in Train A being inoperable would be entered in the LCO Tracking Log as an Active LCO and with the Train B auxiliary air compressor already out of service a Loss of Safety Function would exist because the qualified air supply to ABGTS Train B is not available. The auxiliary air is a support system for several systems, including ABGTS.

B. Incorrect, Plausible because Mode applicability changes as the plant enters different operating modes which result in Information Only entries into the Tech Spec Tracking Log and because a Loss of Safety Function occurring as a result of the ABGTS fan failure is correct since the Train B does not have its essential air supply auxiliary air compressor available.

C. Incorrect. Plausible because the ABGTS Train A failure being entered into the LCO Tracking Log as an Active LCO is correct and because there is control air available for the other Train of ABGTS.

D. Incorrect, Plausible because Mode applicability changes as the plant enters different operating modes which result in Information Only entries into the Tech Spec Tracking Log and because there is control air available for the Train B of ABGTS from the plant control air system.

Question Number: 94

Tier: 3 **Group** N/A

K/A: G 2.1.18
Conduct of Operations
Ability to make accurate, clear, and concise logs, records, status boards, and reports.

Importance Rating: 3.6 / 3.8

10 CFR Part 55: 41.10 / 45.12 / 45.13

10CFR55.43.b: 2

K/A Match: K/A is matched because the question requires knowledge of the LCO Tracking Log entry that is required to be made for the given mode and condition. The question is SRO because the SRO is responsible for completing the Tech Spec Tracking log and also for performing the Safety Function Determinations when SSC are made inoperable.

Technical Reference: Tech Spec LCO 3.7.12, "Auxiliary Building Gas

Technical Reference: Tech Spec LCO 3.7.12, "Auxiliary Building Gas Treatment System (ABGTS)"
Tech Spec 5.7.2.18, Safety Function Determination Program
OPDP-8 Limiting Conditions for Operation Tracking, Revision 0005

Proposed references to be provided: None

Learning Objective: 3-OT-OPDP-8
03. Identify the responsibilities of the Unit Supervisor described in OPDP-8, Limiting Condition for Operation Tracking.

Cognitive Level:
- **Higher** X
- **Lower** ____

Question Source:
- **New** ____
- **Modified Bank** X
- **Bank** ____

Question History: Watts Bar question T/S0331 096 modified for use on the 10/2011 NRC exam.

Comments:

5.7 Procedures, Programs, and Manuals

5.7.2.16 Diesel Fuel Oil Testing Program (continued)

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to the 7 day storage tanks; and

c. Total particulate concentration of the fuel oil in each of the four interconnected tanks which constitute a 7 day storage tank is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.7.2.17 (removed from Technical Specifications)

5.7.2.18 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.7.2.19 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The Fall 2007 end date for conducting the 10 year interval containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than Fall 2012.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 15.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

(continued)

3.7 PLANT SYSTEMS

3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

LCO 3.7.12 Two ABGTS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies in the fuel handling area.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ABGTS train inoperable.	A.1 Restore ABGTS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4. <u>OR</u> Two ABGTS trains inoperable in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the fuel handling area.	C.1 Place OPERABLE ABGTS train in operation. <u>OR</u> C.2 Suspend movement of irradiated fuel assemblies in the fuel handling area.	Immediately Immediately

(continued)

Appendix B
(Page 2 of 6)

Safety Function Determination Program (SFDP)
WBN Only

3.0 LOSS OF SAFETY FUNCTION DETERMINATION

In accordance with TS 5.7.2.18, an LOSF exists when, assuming no concurrent single failure, a safety function assumed (explicitly or implicitly) in the accident analysis cannot be performed. An LOSF may exist when a support system is inoperable, and:

- A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
- A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- A required system redundant to the support system(s) for the supported systems above is also inoperable.

3.1 Initiating an LOSF Evaluation

A. Upon determining that an SSC is in a degraded or nonconforming condition and has been declared inoperable, an LOSF evaluation shall be initiated. A flowchart of the process is depicted in FIGURE 1.

B. Determine if the inoperable SSC is directly addressed by a TS LCO. If the SSC is addressed in TS, then go to Step 3.1D, otherwise continue with Step C.

C. If the inoperable SSC is not addressed in TS, then determine if it renders another SSC directly addressed in TS inoperable. If the initially inoperable SSC does not render a SSC addressed in TS inoperable, then no further action is required, and no further LOSF evaluation is required.

D. Enter the applicable TS conditions and required actions for the inoperable TS SSCs determined in Steps 3.1B and/or 3.1C, above.

E. Determine if other inoperable SSCs exist for a TS LCO. If only one TS LCO action has been entered, then follow the required actions for that LCO, and no further LOSF evaluation is required. If more than one LCO is entered, then go to Step 3.1F.

F. Determine if the LCO whose required actions were entered is a support system. If the inoperable SSC is a support system SSC, then continue to Step 3.1G. LCOs which are not support systems will generally not result in an LOSF when taken in combination with additional inoperabilities. Therefore, no further LOSF evaluation is required and the required actions of the applicable LCOs shall be met in accordance with LCO 3.0.2.

G. If more than one LCO's required actions were entered, then determine if all the LCOs have been entered for the same train or channel. If all LCOs were entered for the same train or channel, then no further LOSF evaluation is required. If more than one train or channel LCOs were entered, then continue to Step 3.1H.

Safety Function Determination Program (SFDP)
WBN Only

3.1 Initiating an LOSF Evaluation (continued)

NOTES

1) The inoperable SSC must be evaluated against each other inoperable SSC and appropriate combinations of inoperable SSCs, to determine if an LOSF exists.

2) For support system inoperabilities, LCO 3.0.6 requires declaring the support system LCO not met, but suspends performing of the supported systems ACTIONS.

H. Determine if inoperabilities within redundant trains or channels result in the loss of capability to fulfill a safety function. The LOSF verification can be performed through evaluation of the impact of all inoperabilities being tracked. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to the support system(s) for the supported systems 1 and 2 above is also inoperable.

I. If no LOSF is identified, then the required actions for the LCOs address the condition (some actions for supported systems are addressed in the support system LCOs, e.g., LCOs 3.7.7 and 3.7.8).

J. If an LOSF is identified, ensure that the most appropriate action is taken considering the current MODE and plant conditions:

1. The appropriate conditions and required actions of the LCO in which the LOSF exist are entered, or

2. If no appropriate LCO condition and required actions exist for the LOSF, then LCO 3.0.3 shall be evaluated for entry and other appropriate limitations and remedial or compensatory actions are to be taken commensurate with the LOSF, or

3. If no LCO exist for the LOSF and the plant is in a MODE where LCO 3.0.3 is not applicable, then appropriate limitations and remedial or compensatory actions are to be taken commensurate with the LOSF.

K. An Unit Log entry is to be made documenting the LCO or the LOSF entry, and the actions taken.

Safety Function Determination Program (SFDP)
WBN Only

4.0 EXTENDING SUPPORTED SSC COMPLETION TIMES

Refer to TS 3.0.6 and 5.7.2.18 for LCO entry relative to support systems and supported systems. TS 1.3 addresses extending LCO Action completion times.

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Safety Function Determination Program (SFDP)
WBN Only

1.0 PURPOSE

> **NOTE**
>
> OPERABILITY determinations are conducted separate and apart from the SFDP. A loss of safety function (LOSF) determination should be performed after OPERABILITY is determined.

The SFDP ensures that an LOSF is detected and appropriate actions taken. The SFDP contains:

A. Provisions for cross-train checks to ensure a loss of capability to perform the safety function assumed in the accident analysis does not go undetected.

B. Provisions for ensuring the plant is in a safe condition if an LOSF condition exists.

C. Provisions for ensuring that an inoperable supported system's completion time is not inappropriately extended as a result of multiple support system inoperabilities; and

D. Other appropriate limitations and remedial or compensatory action.

> **NOTE**
>
> In the context of the SFDP, references to the TS also include the Technical Requirements Manual (TRM), Fire Protection Report (FPR) and Offsite Dose Calculation Manual (ODCM).

2.0 SCOPE

> **NOTE**
>
> An LOSF may occur through a combination of structures, systems and components (SSCs) inoperabilities addressed in the TS, as well as SSCs that are not addressed in TS.

The SFDP includes those SSCs required to be OPERABLE. If an SSC is declared inoperable, then an LOSF evaluation shall be performed. An LOSF evaluation should also be performed before MODE changes (e.g., MODE 4 to 3, MODE 3 to 4) if multiple LCOs or inoperabilities are being tracked.

Appendix B
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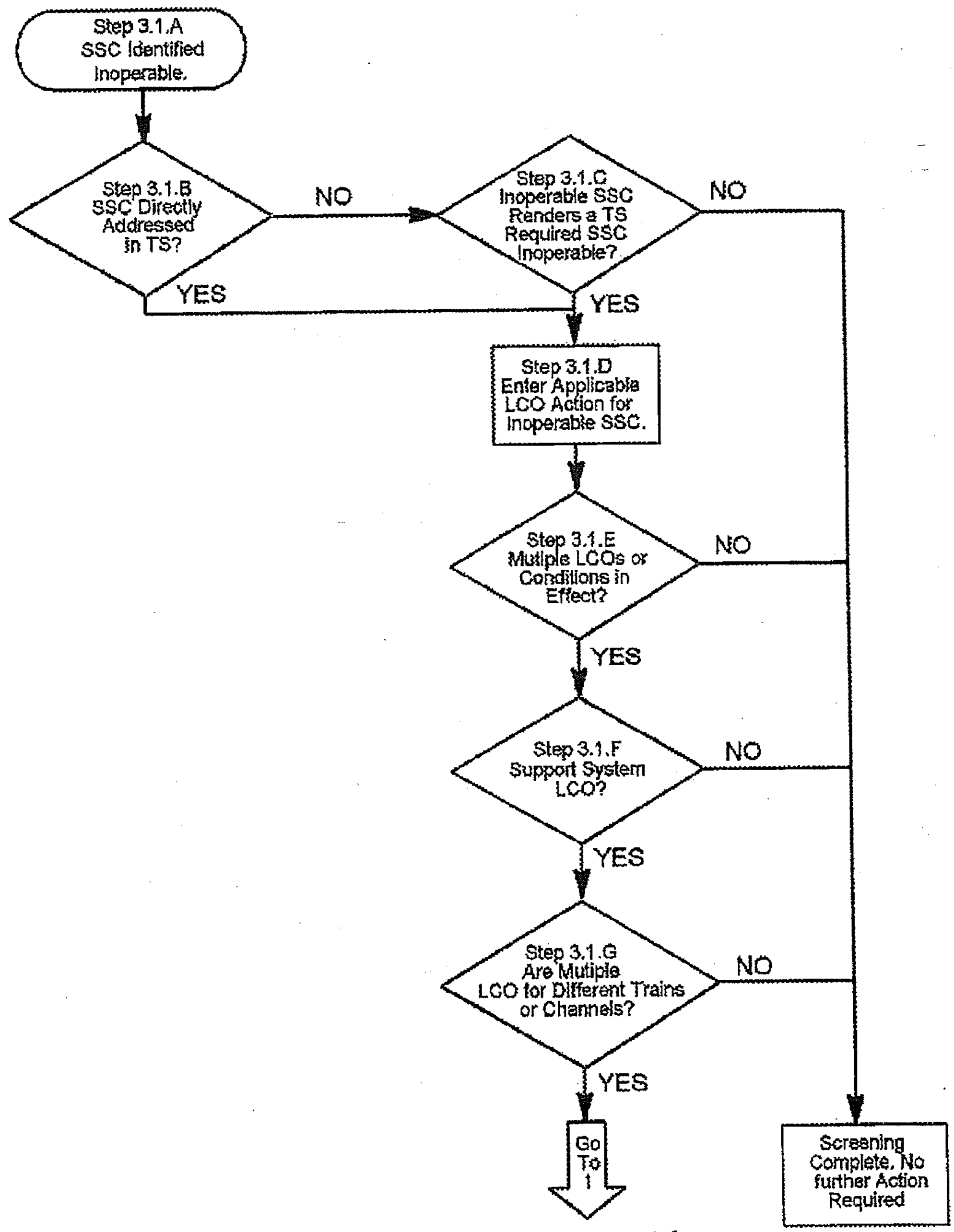
Safety Function Determination Program (SFDP)
WBN Only

5.0 SAFETY FUNCTION DETERMINATION FLOW CHART FIGURE 1

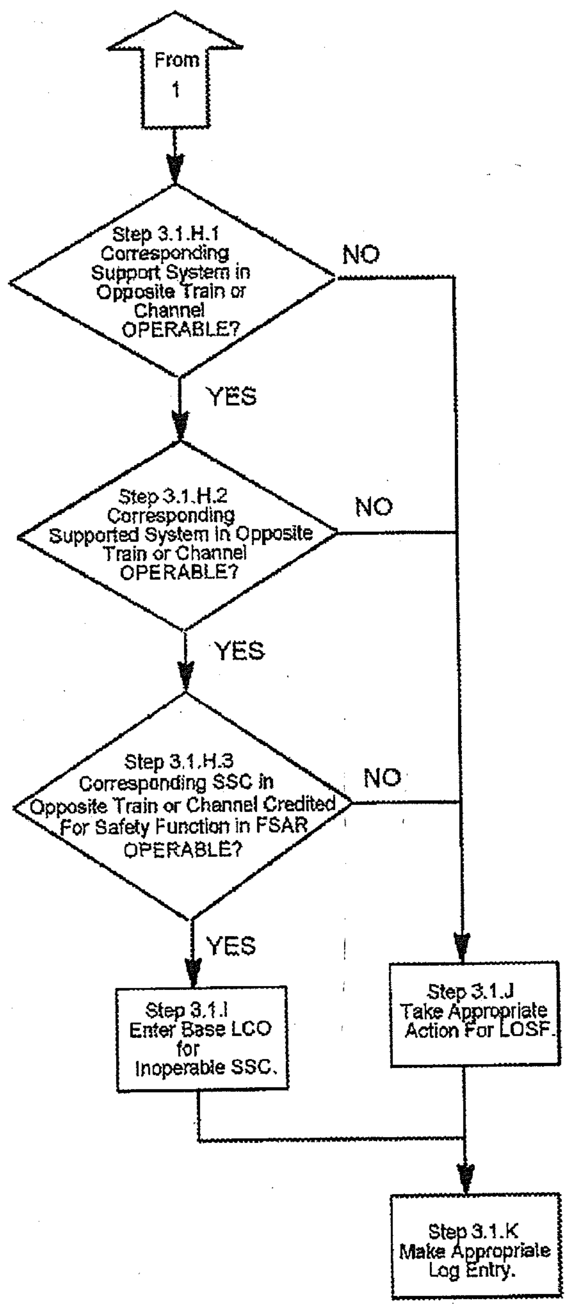


NEXT PAGE

Appendix B
(Page 6 of 6)

Safety Function Determination Program (SFDP)
WBN Only

5.0 SAFETY FUNCTION DETERMINATION FLOW CHART FIGURE 1 (continued)



WBN BANK QUESTION

T/S0331 096

Given the following plant conditions:

- The Unit is at 100% power.
- Train 'A' SSPS is out of service for surveillance testing.
- During the test it is determined that a modification installed during the last refueling outage has resulted in the Train 'A' Reactor Trip System being unable to automatically trip the reactor.
- Manual trip capability is available.
- Subsequently, it is determined the Train 'B' Reactor Trip System was modified in the same manner and is also inoperable.

Which of the following identifies ...

the maximum time allowed to place the unit in Mode 3 per Technical Specifications, and,
after the unit is in Mode 3 with the reactor trip breakers open, and Tech Spec LCO 3.3.1 is entered in the LCO Tracking Log, how the entry will be designated per OPDP-8, 'Limiting Conditions for Operation Tracking?'

a. Immediately
as an 'Active' LCO

b. Immediately
as an 'Information Only' LCO

c. 7 hours
as an 'Active' LCO

d.✓ 7 hours
as an 'Information Only' LCO

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. Licensed Operator Initial Training

B. Licensed Operator Requalification Training

III. TITLE

OPDP-8, LIMITING CONDITION FOR OPERATION TRACKING

IV. LENGTH OF LESSON

A. Licensed Operator Requalification Training 1 Hour

B. Initial License Operator Training 2 Hours

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
	X	X	X	01. Describe the purpose of OPDP-8, Limiting Condition for Operation Tracking
	X	X	X	02. Identify the responsibilities of the Shift Manager described in OPDP-8, Limiting Condition for Operation Tracking.
	X	X	X	03. Identify the responsibilities of the Unit Supervisor described in OPDP-8, Limiting Condition for Operation Tracking.
	X	X	X	04. Identify the responsibilities of the Unit Operator described in OPDP-8, Limiting Condition for Operation Tracking.

V. TRAINING OBJECTIVES (Continued)

A U O	R O	S R O	S T A	
	X	X	X	05. Identify where the official entry time/date for an LCO action statement will be recorded.
				06. DELETED
				07. DELETED
				08. DELETED
	X	X	X	09. Explain the Unique Identifier numbering scheme used in the LCO Tracking Log.
				10. DELETED

VI. TRAINING AIDS

A. Marker Board

B Assorted Markers

C. Overhead Projector

VII. MATERIALS

One copy of each of the following for each participant:

Attachment 1 - OPDP-8, Limiting Condition for Operation Tracking. (latest revision)

P. Safety Function Determination Program (SFDP)	
1. Conducted after OPERABILITY has been determined	Note in Appendix explains that OPERABILITY determinations are conducted separate and apart from the LOSF determination
2. The SFDP ensures that an LOSF is detected and appropriate actions taken.	In the context of the SFDP, references to the TS also include the TRM, FPR and ODCM.
3. The SFDP contains: - Provisions for cross-train checks to ensure a loss of capability to perform the safety function assumed in the accident analysis does not go undetected. - Provisions for ensuring the plant is in a safe condition if an LOSF condition exists. - Provisions for ensuring that an inoperable supported system's completion time is not inappropriately extended as a result of multiple support system inoperabilities; and - Other appropriate limitations and remedial or compensatory action.	
4. Scope - The SFDP includes those SSCs required to be OPERABLE. If an SSC is declared inoperable, then an LOSF evaluation shall be performed. An LOSF evaluation should also be performed before MODE changes (e.g., MODE 4 to 3, MODE 3 to 4) if multiple LCOs or inoperabilities are being tracked.	A note explains LOSF may occur through a combination of structures, systems and components (SSCs) inoperabilities addressed in the TS, as well as SSCs that are not addressed in TS.
5. In accordance with TS 5.7.2.18, an LOSF exists when, assuming no concurrent single failure, a safety function assumed (explicitly or implicitly) in the accident analysis cannot be performed.	

6. An LOSF may exist when a support system is inoperable, and:
   - A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
   - A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
   - A required system redundant to the support system(s) for the supported systems A and B above is also inoperable.

Use 3 legged table to explain Table is supported system, legs are the support systems

7. Initiating an LOSF determination

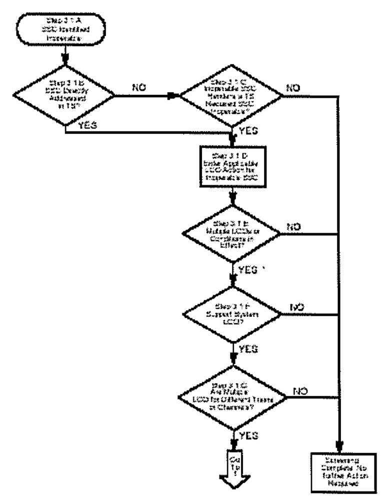
   a. Upon determining that an SSC is in a degraded or nonconforming condition and has been declared inoperable, an LOSF evaluation shall be initiated. A flowchart of the process is depicted in FIGURE 1.

   b. Determine if the inoperable SSC is directly addressed by a TS LCO. If the SSC is addressed in TS, then go to Step 3.1.D, otherwise continue with Step C.

   c. If the inoperable SSC is not addressed in TS, then determine if it renders another SSC directly addressed in TS inoperable. If the initially inoperable SSC does not render a SSC addressed in TS inoperable, then no further action is required, and no further LOSF evaluation is required.

   d. Enter the applicable TS conditions and required actions for the inoperable TS SSCs determined in Steps 3.1.B and/or C, above.

Work through the flow chart in OPDP-8, Appendix B



e. Determine if other inoperable SSCs exist for a TS LCO. If only one TS LCO action has been entered, then follow the required actions for that LCO, and no further LOSF evaluation is required. If more than one LCO is entered, then go to Step 3.1.F.

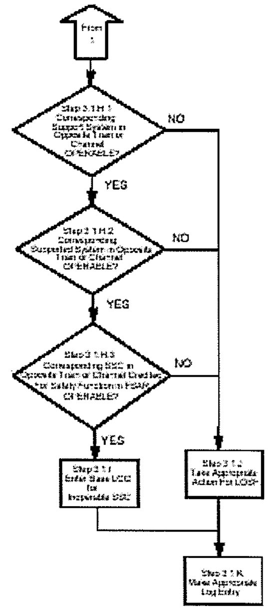
f. Determine if the LCO whose required actions were entered is a support system. If the inoperable SSC is a support system SSC, then continue to Step 3.1.G. LCOs which are not support systems will generally not result in an LOSF when taken in combination with additional inoperabilities. Therefore, no further LOSF evaluation is required and the required actions of the applicable LCOs shall be met in accordance with LCO 3.0.2.

g. If more than one LCO required actions were entered, then determine if all the LCOs have been entered for the same train or channel. If all LCOs were entered for the same train or channel, then no further LOSF evaluation is required. If more than one train or channel LCOs were entered, then continue to Step 3.1.H.

h. Determine if inoperabilities within redundant trains or channels result in the loss of capability to fulfill a safety function. The LOSF verification can be performed through evaluation of the impact of all inoperabilities being tracked. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
3. A required system redundant to the support system(s) for the supported systems (1.) and (2.) above is also inoperable.

i. If no LOSF is identified, then the required actions for the LCOs address the condition (some actions for supported systems are addressed in the support system LCOs, e.g., LCOs 3.7.7 and 3.7.8).



Discuss notes

NOTE 1 The inoperable SSC must be evaluated against each other inoperable SSC and appropriate combinations of inoperable SSCs, to determine if an LOSF exists.

NOTE 2 For support system inoperabilities, LCO 3.0.6 requires declaring the support system LCO not met, but suspends performing of the supported systems ACTIONS.

j. If an LOSF is identified, ensure that the most appropriate action is taken considering the current MODE and plant conditions: 1. The appropriate conditions and required actions of the LCO in which the LOSF exist are entered, or 2. If no appropriate LCO condition and required actions exist for the LOSF, then LCO 3.0.3 shall be evaluated for entry and other appropriate limitations and remedial or compensatory actions are to be taken commensurate with the LOSF, or 3. If no LCO exist for the LOSF and the plant is in a MODE where LCO 3.0.3 is not applicable, then appropriate limitations and remedial or compensatory actions are to be taken commensurate with the LOSF. k. An Unit Log entry is to be made documenting the LCO or the LOSF entry, and the actions taken. Q. Operating Experience - WBN - EGTS tagging 1. 1-HO-98-1306 was prepared and reviewed with an error. The Hold Order written on 10/14/98, eight months prior to the event, with an incorrect train hand switch. Scope of work grew and clearance as written expanded to cover the work. Clearance writer and reviewer both failed to detect the error and the Hold Order was placed on 6/3/99, without detecting the error, and LCO 3.6.9 was entered. 2. A second HO, 1-HO-99-0036, on EGTS was authorized by the SRO simultaneously, and the error not detected.	Discuss OE in Operating Experience - Control Building EBR AHU or train of Control Building EBR air-conditioning system.

95. G 2.1.29 095

Given the following:

- Unit 1 is in Mode 1.

- To facilitate on-going work, Maintenance request that 'Status Control' be relaxed on a section of safety related piping inside an existing clearance boundary that contains a locked throttle valve.

Which ONE of the following identifies...

(1) the lowest position in the Operations Organization that can authorize the relaxing of 'Status Control' on the locked throttle valve

and

(2) when ready to restore status control, what type of verification is required to restore the locked throttle valve to the correct position?

	Lowest position	Verification
A.	Shift Manager or designee	Independent
B.✓	Shift Manager or designee	Concurrent
C.	Operations Superintendent	Independent
D.	Operations Superintendent	Concurrent

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because the Shift Manager/designee being authorized for relaxing status control on equipment inside a clearance boundary is correct and Independent Verification is used in other applications (but it is not the correct verification method in this case.)*

B. *Correct, NPG-SPP-10.1 identifies that the Shift Manager/designee is authorized to approve relaxing status control within a clearance boundary and NPG-SSP-10.3 identifies that Concurrent Verification is the correct verification method for a locked throttle valve.*

C. *Incorrect, Plausible because the Operations Sup't is the individual authorized for relaxing status control on systems and Independent Verification is used in other applications (but it is not the correct verification method in this case.)*

D. *Incorrect, Plausible because the Operations Sup't is the individual authorized for relaxing status control on systems and concurrent Verification is the correct verification method for positioning a locked throttle valve.*

Question Number: 95

Tier: 3 **Group** n/a

K/A: G 2.1.29
Conduct of Operations
Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Importance Rating: 4.1 / 4.0

10 CFR Part 55: 41.10 / 45.1 / 45.12

10CFR55.43.b: Plant Specific Exemption - SRO objective

K/A Match: Applicant must know the requirements for relaxing Status Control within a clearance boundary and the requirements for restoring Status control when the work is finished. SRO because the process for relaxing status control is an SRO function (supported by the procedure and the SRO only objective in the lesson plan)

Technical Reference: NPG-SPP-10.1, System Status Control, Revision 0001
NPG-SPP-10.3, Verification Program, Revision 0000

Proposed references to be provided: None

Learning Objective: 3-OT-SSP1001
SRO Only Objective
03. Explain the responsibilities of the SM/ designee as directed by SPP-10.1, System Status Control.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	X
Modified Bank	
Bank	

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

Comments:

1.0 PURPOSE

This procedure establishes the responsibilities and programmatic methods for obtaining, maintaining and documenting control of equipment and system status in accordance with design, license and regulatory requirements.

2.0 SCOPE

This procedure applies to all TVA Nuclear Power Group (NPG) personnel and contractors performing activities affecting:

- Nuclear safety related and quality related systems and equipment.
- Non-safety related systems and equipment necessary to support the production of electricity.
- Fire Protection systems and equipment.

3.0 PROCESS

3.1 Roles and Responsibilities

A. Responsible Managers shall:

1. Ensure status control for equipment within their areas of responsibility.
2. Ensure procedures and processes for equipment within their area of responsibility meets the status control requirements of this procedure.

B. The Operations Superintendent is responsible for the following:

DISTRACTOR

1. Determining the systems and components requiring status control.
2. Authorizing relaxation of status control.

C. The Shift Manager (SM)/designee is responsible for the following:

1. Determining when re-performance of all or part of an Equipment Alignment Checklist is needed for procedure revisions and/or minor outages.
2. Maintaining status control.

CORRECT

3. Authorizing relaxation of status control within a clearance boundary when necessary.
4. Ensuring procedures restore system and equipment to the correct status.
5. Ensuring all activities that change the status of plant equipment are authorized by an approved plant procedure or work document.

3.1 Roles and Responsibilities (continued)

6. Ensuring the equipment and components within a clearance boundary are properly aligned before release of clearance.

7. Ensuring components that make up a clearance boundary are returned to their correct status when the clearance is released.

8. Reviewing and verifying Equipment Alignment Checklists are completed correctly before transmittal to the Operations Superintendent.

9. Evaluating any discrepancies between the actual field status of a system or component and the status assumed by the prerequisites of a procedure before implementing procedure.

10. Authorizing all activities affecting system or equipment status.

11. Minimizing the impact of clearances and open procedures on checklist completion.

12. Caution tags shall not be used to authorize, control or establish equipment status changes.

13. Ensuring that Caution tags are placed on valves when a valve dog is placed in an off normal status (engaged or disengaged). Caution tags should be placed on other devices as deemed necessary to identify that an off normal status exists.

D. The Responsible Individual shall:

1. Ensure procedures and work documents restore systems and equipment to the correct status.

2. Ensure all activities that change the status of plant equipment are authorized by an approved plant procedure, clearance, work order or TACF.

3. Identify any discrepancy between the actual field status of a system or component and the status assumed by the prerequisites of a procedure.

4. Notify the SM/designee of any discrepancy identified above.

3.2 Instructions

3.2.1 Manipulation of Systems and Equipment

A. Qualified Operations personnel shall operate equipment as directed by the SM, US, UO in accordance with an approved plant procedure or work document.

B. Qualified Maintenance personnel shall operate equipment in accordance with an approved plant procedure or work document as authorized by the SM/US.

C. Qualified Radcon and Chemistry personnel shall operate equipment in accordance with an approved plant procedure as authorized by the SM/US.

3.3.2 Independent Verification Standard: (continued)

E. Verifier verifies the as-found configuration or condition matches the required position, without changing it, using one or more of the following means:

1. Hands-on verification that configuration is correct (e.g., manually checking valve position)

2. Observing remote indication

3. Observing correct system/equipment/component response

F. Verifier confirms new configuration or condition agrees with guiding document and signs signature space provided in guiding document.

G. If as-found configuration or condition is incorrect, report the condition to supervision immediately.

3.3.3 Valves

A. Valves that are to be verified open shall be manipulated in the closed direction only as necessary to remove any slack from the operating mechanism and verify valve stem movement. The valve shall then be fully opened, subject to normal precautions on backseating valves.

B. Valves that are to be verified closed shall be manipulated in the closed direction only as necessary to verify the valve is fully closed, and not binding or difficult to operate. Care must be exercised, however, to avoid over torquing the valve operator and damaging the valve seat. If any doubt exists, SM should be contacted for resolution.

C. To determine the position of a throttled valve, the total number of turns until the handwheel stops moving in the open/closed direction shall be counted. To set the position of a throttled valve, open/close the valve the required number of turns from the full closed/open position (handwheel will no longer move in the closed/open direction).

D. Reach rod valve position indicators shall not be used as the sole method of position verification.

E. Locked valve and throttled valve position cannot be independently verified since these operations require the verifier to observe actions while they are being performed. CV shall be used to verify the position of locked and throttled valves in those cases where IV would normally be required.

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. LICENSE TRAINING
B. NOTP
C. LICENSE REQUALIFICATION
D. NAUO REQUALIFICATION

III. TITLE

SPP-10.1, SYSTEM STATUS CONTROL

IV. LENGTH OF LESSON

A. LICENSE TRAINING 1.0 HOUR
B. NOTP 1.0 HOUR

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

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	01. Identify the purpose of and explain the need for SPP-10.1, System Status Control.
X	X	X	X	02. Identify the scope of the equipment that must be controlled to the requirements of SPP-10.1, System Status Control.
		X	X	03. Explain the responsibilities of the SM/designee as directed by SPP-10.1, System Status Control.
X	X	X	X	04. Explain the responsibilities of the Responsible Individual as directed by SPP-10.1, System Status Control.

X. LESSON BODY	INSTRUCTOR NOTES
c. When necessary authorizes relaxation of status control within a clearance boundary.	
d. Ensures procedures restore system & equipment to the Correct Status.	SOER 81-16
e. Ensure all activities that change status of plant equipment are authorized by an approved plant procedure or work document.	
f. Ensure the equipment & components within a clearance are properly aligned prior to release of a clearance.	
g. Ensure components that makeup a clearance are returned to their correct status when the clearance is released.	Correct Status is defined as the correct position or state of a component based on present plant conditions, procedures currently in effect, & active equipment clearances as determined by the SM/designee.
h. Review and Verify Equipment Alignment Checklists are completed correctly before transmittal to the Operations Superintendent.	
i. Evaluate any discrepancies between actual field status of a system or component and the status assumed by the prerequisites of a procedure before implementing the procedure.	
j. Authorize all activities affecting system or equipment status.	
k. Minimize the impact of clearances and open procedures on checklist completion.	

III. Justification for Plant Specific Exemptions

The 25 SRO-only questions ***shall*** evaluate the additional knowledge and abilities required for the higher license level in accordance with 10 CFR 55.43(b). [NUREG 1021, Section ES-401D.2.d]

The fact that a facility licensee trains its ROs to master certain 10 CFR 55.43 knowledge, skills, and abilities does NOT mean that they can no longer be used as a basis for SRO-only questions. [Operator Licensing Feedback Web page Item 401.36 @ http://www.nrc.gov/reactors/operator-licensing/op-licensing-files/ol-feedback.pdf]

The SRO-only test item is required to be tied to one of the 10 CFR 55.43(b) items. However, if a licensee desires to evaluate a knowledge/ability that is not tied to one of the 10 CFR 55.43(b) items, then the licensee can classify the knowledge/ability as "*unique to the SRO position*" provided that there is documented evidence that ties the knowledge/ability to the licensee's SRO job position duties in accordance with the systematic approach to training (SAT).

- **Justification**: A question that is <u>not</u> tied to one of the 10 CFR 55.43(b) items can still be classified as "SRO-only" provided the licensee has documented evidence to prove that the knowledge/ability is "*unique to the SRO position*" at the site. An example of documented evidence includes:
    - The question is linked to a learning objective that is specifically labeled in the lesson plan as being SRO-only (e.g., some licensee lesson plans have columns in the margin that differentiate AO, RO, and SRO learning objectives) [NUREG 1021, ES-401, Section D.2.d]

      AND/OR

    - A question is linked to a task that is labeled as an SRO-only task, and the task is NOT listed in the RO task list.

96. G 2.2.1 096

Which ONE of the choices below completes the following two statements relative to the performance of 1-SI-85-10, "Rod Drop Time Measurement Using CERPI Rod Drop Test Computer?"

(1) Tech Spec LCO 3.1.5, Rod Group Alignment Limits, requires the performance prior to a reactor criticality following ___________.

(2) Performance of 1-SI-85-10 requires ______of the Reactor Coolant Pumps to be in service.

	(1)	(2)
A.	a Mode 5 entry	all 4
B.	a Mode 5 entry	at least 3
C.✓	reactor head removal	all 4
D.	reactor head removal	at least 3

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because there are additional SIs that are required to be performed after each Mode 5 entry and all RCPs being required to be running is correct.*

B. *Incorrect, Plausible because there are additional SIs that are required to be performed after each Mode 5 entry and there are Tech Spec conditions allowed with three RCPs running. (6 hours to be in Mode 3 with 3 RCPs running)*

C. *Correct, In accordance with the surveillance requirement SR-3.1.5.3 (see below) the SI is required prior to reactor criticality after initial fuel loading and each removal of the reactor head and all reactor Coolant pumps are required to be in service.*

D. *Incorrect, Plausible because being required prior to reactor criticality after each removal of the reactor head is correct and there are Tech Spec conditions allowed with three RCPs running. (6 hours to be in Mode 3 with 3 RCPs running)*

Rod Group Alignment Limits
3.1.5

SURVEILLANCE REQUIREMENTS (continued)

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

Question Number: 96

Tier: 3 **Group** n/a

K/A: G 2.2.1
Equipment Control
Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Importance Rating: 4.5 / 4.4

10 CFR Part 55: 41.5 / 41.10 / 43.5 / 43.6 / 45.1

10CFR55.43.b: 6, 2

K/A Match: K/A is matched because the question requires the knowledge of a test of a plant equipment that affects reactivity and how plant equipment must be aligned for the test. The question is SRO because it requires knowledge of Tech Spec that is below the line and is associated with surveillance requirements and well as being a procedure involved in alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Technical Reference: Tech Spec LCO 3.1.5, Rod Group Alignment Limits
1-SI-85-10, Rod Drop Time Measurement Using CERPI Rod Drop Test Computer, Revision 0002

Proposed references to be provided: None

Learning Objective: 3-OT-T/S0301
3. Given plant parameters/conditions, correctly determine the compliance with the LCOs or TRs in the Reactivity Control sections of T/S and T/R manuals.

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

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

Comments:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Rod Group Alignment Limits

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

ACTIONS

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

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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

TVA

Watts Bar Nuclear Plant

Unit 1

Surveillance Instruction

1-SI-85-10

Rod Drop Time Measurement Using CERPI Rod Drop Test Computer

Revision 0002

Quality Related

Level of Use: Continuous Use

Effective Date: 10-11-2009

Responsible Organization: SIE, System Eng - I&C Elect

Prepared By: R. B. Rieger

Approved By: J. E. Couch

Revision Log			
Revision or Change Number	Effective Date	Affected Page Numbers	Description of Revision/Change
Rev 0	02/28/05	All	Initial Issue.
Rev. 1	02/22/08	All	This procedure has been converted from Word 95 to Word XP using Rev. 0 by the Conversion Team. Added bank overlap switch check, RDTC power up and shutdown, and added reference points for frequency check.
Rev. 2	10/11/09	All	Added Performance Reference and Precaution for Jumper Control Process; see PER 140641. Incorporate DCN 52957.

Table of Contents

Table of Contents (continued)

1.0 INTRODUCTION

1.1 Purpose

A. This Surveillance Instruction (SI) provides detailed steps to verify that the drop time of each shutdown and control rod from the fully withdrawn position is ≤ 2.7 seconds from the beginning of decay of stationary gripper voltage to dashpot entry. The SI is performed after each removal of the reactor head before returning the reactor to criticality with:

1. $T_{avg} \geq 551°F$.

2. All reactor coolant pumps operating.

B. Measuring the rod drop times before reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanisms do **NOT** interfere with rod motion or adversely impact the drop times during operation in Modes 1 and 2.

1.2 Scope

Rod drop times for shutdown and control rods are measured by withdrawing all shutdown and control rod banks to the fully withdrawn position (i.e., 231 steps), and then tripping the reactor to simultaneously measure the drop times for all rods using the Rod Drop Test Computer (RDTC) in 1-R-44. The RDTC receives an input from two steam dump relays which are operated by interposing relays operated by the opening of the reactor trip breakers. Either of the relays provides a start signal to measure the rod drop times. The RDTC then measures the induced voltage on the Rod Position Indication (RPI) detector secondary coils by the motion of the falling Control Rod Drive Mechanism (CRDM) drive shaft.

This SI conservatively defines "rod drop time" as being the time from when the Reactor Trip Breakers change status (i.e. the breakers open following a reactor trip) until dashpot entry occurs. The CERPI Rod Drop Test Computer (RDTC) relies on a contact change of state for a starting time. The contact change of state occurs slightly after the Reactor Trip Breakers open (<100 msec)(Reference 2.2.D). Therefore, acceptance criteria in this SI is different from acceptance criteria in 1-SI-85-1 which uses the actual Reactor Trip Breakers opening as a start signal.

1.2.1 Operability Tests To be Performed

This SI demonstrates rod freedom of movement (i.e., tripability) by withdrawing all shutdown and control banks, tripping the reactor, and simultaneously measuring the drop time for all rods to verify that the drop times are within the bounds (i.e., less than or equal to the drop times) of the cycle specific transient safety analysis assumptions.

1.2.2 Surveillance Requirements Fulfilled and Modes

Performance of this SI satisfies the following Surveillance Requirements (SRs):

SURVEILLANCE REQUIREMENT	APPLICABLE MODES	PERFORMANCE MODES
SR 3.1.5.3	1, 2	3
SR 3.3.2.11-8.a*	1, 2, 3	ALL

* Operability of Reactor Trip P-4 ESFAS Interlock is demonstrated for Trains A and B.

1.3 Frequency and Conditions

A. This SI is to be performed before reactor criticality after each removal of the reactor head.

B. This SI is to be performed during the following Mode 3 conditions:

1. $T_{avg} \geq 551^{\circ}F$.
2. All reactor coolant pumps operating.

C. Reactor Trip P-4 ESFAS Interlock Operational Test (Appendix C) is performed once per Reactor Trip Breaker cycle.

I. PROGRAM

WATTS BAR OPERATOR TRAINING

II. COURSE

A. LICENSE OPERATOR TRAINING
B. LICENSED OPERATOR REQUAL

III. TITLE

T/S 3.1, "REACTIVITY CONTROL SYSTEM," AND BASES

IV. LENGTH OF LESSON

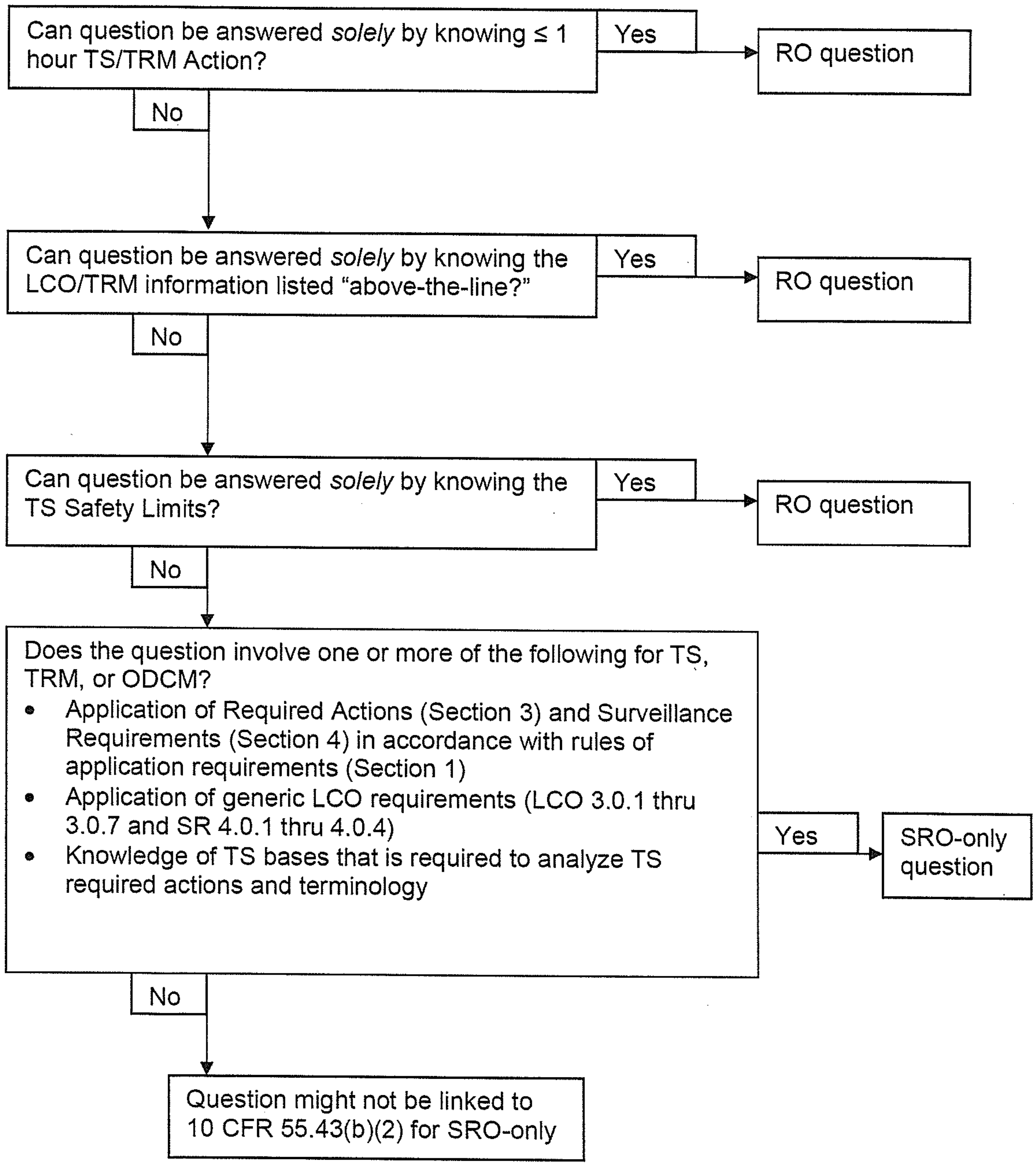
A. LICENSE OPERATOR TRAINING
B. LICENSED OPERATOR TRAINING TIMES WILL BE DETERMINED AFTER OBJECTIVES ARE IDENTIFIED.

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA		
	X	X	X	1.	Define the term SHUTDOWN MARGIN, and know the applicable limiting values for Modes 1-5.
	X	X	X	2.	Determine the bases for the limits placed on reactor core measured parameters (SDM, MTC).
	X	X	X	3.	Given plant parameters/conditions, correctly determine the compliance with the LCOs or TRs in the Reactivity Control sections of T/S and T/R manuals.
	X	X	X	4.	Given plant parameters/conditions, correctly determine applicable Action Conditions, Required Actions, and Completion Times for the Reactivity Control sections of T/S and T/R manuals.

A U O	R O	S R O	S T A	
	X	X	X	5. Determine the bases for the limits placed on control rod positioning and position monitoring equipment (Rod Insertion Limits, Alignment Limits, and Rod Position Indicating Systems).
	X	X	X	6. Determine the bases for the limits on boration systems (Borated Water Sources, Boration Flow Paths, and Charging Pumps).

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



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

Some examples of SRO exam items for this topic include:

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

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

Some examples of SRO exam items for this topic include:

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

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

97. G 2.2.7 097

Given the following:

- Unit 1 reactor is critical below the 'Point of Adding Heat' in accordance with GO-2, "Reactor Startup," during startup following a refueling outage.
- Physics Testing has been declared, and PET-201, "Initial Criticality and Low Power Physics Testing," is in progress.
- The Plant Manager assigned a senior line manager (CIPTE Manager) to exercise continuous responsibility for the oversight of PET-201.
- Isothermal Temperature Coefficient Measurement is being performed.

In accordance with NPG-SPP-06.9.1, "Conduct of Testing," which ONE of the following identifies...

(1) how the Shift Manager's responsibility for conduct of the test is affected by having a CIPTE Manager assigned

and

(2) who is responsible for conducting the CIPTE PRE-TEST BRIEFING CHECKLIST on management expectations for the test?

A. (1) The Shift Manager retains the responsibility for the control of the performance of PET-201.
(2) Test Director

B.✓ (1) The Shift Manager retains the responsibility for the control of the performance of PET-201.
(2) Plant Manager's designee

C. (1) The CIPTE Manager assumes the responsibility for the control of PET-201 from the Shift Manager.
(2) Test Director

D. (1) The CIPTE Manager assumes the responsibility for the control of PET-201 from the Shift Manager.
(2) Plant Manager's designee

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because the Shift Manager retaining the responsibility for controlling the performance is correct and the Test Director is responsible for conducting the briefings required by the SPP for the test except for the Management Expectations Briefing.*

B. *Correct, PET 201 is a CIPTE (Complex or Infrequently Performed Test or Evolution) and NPG-SPP-06.9.1, Conduct of Testing, requires the Plant Manager or designee to determine the need to assign a Senior Manager to provide continuous oversight of the test. However, the Shift Manager retains the responsibility for the control of the test. The SPP identifies that the assigned manager does not reduce the Shift Managers authority/responsibility. Briefing requirements change when a procedure is a CIPTE. The Plant Manager or his designee is required to conduct the Management Expectation Briefings Checklist using the Table in the SPP.*

C. *Incorrect, Plausible because the manager assigned as the CIPTE manager does have responsibility for controlling the pace of the test and the resolution (or escalation) of problems encountered during the test and the Test Director is responsible for conducting the briefings required by the SPP for the test except for the Management Expectations Briefing.*

D. *Incorrect, Plausible because the manager assigned as the CIPTE manager does have responsibility for controlling the pace of the test and the resolution (or escalation) of problems encountered during the test and the Plant Manager or designee is responsible for conducting the briefings the Management Expectations Briefing.*

3.8 Complex Infrequently Performed Tests or Evolutions

E. For CIPTEs, the test director will conduct pre-test formal briefings as required [i.e., one before the test crew assumes shift duties (a general test overview), usually at the Operations Shift Turnover Meeting, and a second before commencing the test (a detailed briefing)].

F. At the Test Director's pre-test formal briefing, the Plant Manager or designee for the test shall conduct a briefing for Operations and testing personnel on management expectations for the test utilizing Form NPG-SPP-06.9.1-3.

G. The Plant Manager or his designee shall determine the need to designate a senior line manager to advise the Shift Manager or Unit Supervisor, who has the authority and experience to exercise continuous responsibility for the oversight of a particular test or evolution. This authority includes control of the pace of the CIPTE and the resolution (or escalation) of problems encountered.

NOTE

This is an oversight position and shall not interfere with or reduce the Shift Manager's responsibility for control of the test.

H. For CIPTEs, the test director will conduct a post test briefing, as required by CIPTE Manager, to discuss lessons learned to include as a minimum potential procedure changes and training.

Question Number: 97

Tier: 3 **Group** n/a

K/A: G 2.2.7
Equipment Control
Knowledge of the process for conducting special or infrequent tests.

Importance Rating: 2.9 / 3.6

10 CFR Part 55: 41.10 / 43.3 / 45.13

10CFR55.43.b: 6

K/A Match: K/A is matched because the question requires knowledge of the process for conducting special or infrequent tests including recognizing that an evaluation is a special test and the additional management responsibilities during the test. The question is SRO because it involves administrative requirements associated with low power physics testing processes.

Technical Reference: PET-201, Initial Criticality and Low Power Physics Testing, Revision 0024
NPG-SPP-06.9.1, Conduct of Testing, Revision 0002

Proposed references to be provided: None

Learning Objective: 3-OT-SPP0801A
3. Describe the responsibilities of the supervisor, test director, and senior manager assigned to a Complex, Infrequently Performed Test or Evolution (CIPTE).

Cognitive Level:
Higher ___
Lower X

Question Source:
New X
Modified Bank ___
Bank ___

Question History: WBN bank question SPP0305 010 modified for the 10/2011 NRC exam

Comments:

TVA Watts Bar Nuclear Plant

Unit 1

Power Escalation Test

PET-201

Initial Criticality and Low Power Physics Testing

Revision 0024

Quality Related

Level of Use: Continuous Use

Complex Infrequently Performed Test or Evolution

Effective Date: 03-09-2011

Responsible Organization: RXE, Reactor Engineering

Prepared By: Erik Swanson

Approved By: Johnathan Pope

2.0 SCOPE (continued)

3. Simple tests performed on a routine basis such as RadioChem Lab sampling instructions or surveillances that do not operate plant equipment.

4. Short duration tests or activities (less than one shift) where actions or data are documented in the test/activity documentation.

3.0 INSTRUCTIONS

3.1 Preparing to Test

A. Responsible supervisor shall determine if the test is a Complex Infrequently Performed Test or Evolution (CIPTE) as defined in NPG-SPP-01.2, and if so, ensure the requirements of Section 3.8 are met.

B. Responsible supervisor shall determine if the test requires CTL and prejob brief. This is to be documented on Form NPG-SPP-06.9.1-1.

C. Responsible supervisor shall sign Form NPG-SPP-06.9.1-1, Test Director Assignment Sheet to attest the Test Director possesses the qualification to perform or oversee the conduct of the test. The qualification to be a Test Director includes 1) knowledgeable in the requirements and expectations of the Test Director as described in this procedure; 2) has the experience, knowledge, and skills to perform or oversee the conduct of the test being assigned; 3) shall be familiar with the tools and equipment to be employed; 4) shall be capable of determining that the calibration status of inspection and measuring equipment is current and that the proper measurement and test equipment is being used. Also the responsible supervisor is responsible for ensuring that assigned test performers possess the qualifications, knowledge, experience, and skills necessary to perform the test. The qualification can be demonstrated through a formal training class or by an evaluation of performance by the responsible supervisor. Test Director shall sign Form NPG-SPP-06.9.1-1, Test Director Assignment Sheet, to attest that: the Test Director is familiar with the test to be performed and has an understanding of the requirements and expectations of being a Test Director as described in this procedure.

3.2 Pre-test Briefing

A. The pre-test briefing shall be done using NPG Pre-Job Briefing Checklist in TVA-SPP-18.005, Plan Jobs Safely Form 40897.

B. The tests listed in Section 2.0B should have formal briefings performed prior to the conduct of the test. The responsible supervisor may elect not to require a formal briefing after considering the following: the complexity of the test, the number of people/organizations required to run the test, the experience of the Test Director with the test, the frequency with which the test is run, and whether or not Operations requires a formal briefing. The requirement for a pre-test formal brief will be annotated on Form NPG-SPP-06.9.1-1 by the responsible supervisor.

C. A manager assigned by the Plant Manager will cover items addressed in Form NPG-SPP-06.9.1-3, for CIPTEs.

3.6 Problems During Testing (continued)

B. Problems identified during the test shall be annotated on the CTL including a description of the problem, the procedure step when/where the problem was identified, corrective action steps taken to resolve the problem, and the number of the corrective action document, if one was required.

C. For continuing open test data packages, the Test Director shall determine when corrective actions warrant re-testing and the appropriate steps necessary to re-establish the conditions required for continuing the test. The Test Director will evaluate if Operations needs to be notified before the test is restarted. For details on stopping, exiting and/or reentering a test refer to Section 3.9.

D. Test data packages can be closed by the Test Director with open problems provided that:

1. TS (including Technical Requirements (TRs), ISFSI CoC, Offsite Dose Calculation Manual (ODCM), and Fire Protection Report, if applicable) operability is not affected, or the problem (if the TSs and/or ISFSI CoC are affected) is being tracked in the LCO Tracking Log or on a Fire Protection Impairment Permit, and

2. The open problems are being tracked by a corrective action document.

E. Problems involving not meeting testing acceptance criteria should be addressed in accordance with NPG-SPP-03.1.

3.7 Out-of-Sequence Testing

If specifically allowed by the instruction, steps may be performed out of sequence.

A. If the test requires sequential step execution and it is discovered steps have been executed out of sequence, Test Director shall stop the test, document on the CTL, Form NPG-SPP-06.9.1-2 and notify the Shift Manager and responsible supervisor.

B. Responsible supervisor and Shift Manager approve the corrective actions and document on the CTL, Form NPG-SPP-06.9.1-2. Problems involving steps executed out of sequence should be addressed in a PER.

3.8 Complex Infrequently Performed Tests or Evolutions

The following additional requirements shall be implemented for CIPTEs.

A. The responsible supervisor shall consider the temporary assignment of additional personnel to assist in the conduct of these tests. This includes assignment of personnel to exercise continuous responsibility for the oversight of a particular test, including controlling the pace and resolving problems.

B. The responsible supervisor shall consider the need for Just-In-Time training. If Just-In-Time training is required, the training should be conducted in the simulator if applicable.

3.8 Complex Infrequently Performed Tests or Evolutions (continued)

C. The responsible supervisor shall consider the temporary assignment of additional personnel under the direction of the Shift Manager to augment the shift personnel. For example, assignment of an engineer or coordinator for the test or evolution, or the assignment of an additional senior reactor operator during control rod manipulations. Another example may include additional data takers when the data required is not readily available to the assigned shift at their normal shift location. The duties, authority, and responsibility of any extra personnel should be included on the organization chart and made clear in the test briefings.

D. The responsible supervisor shall ensure the test has been reviewed by an individual knowledgeable of the test before performance of the test.

E. For CIPTEs, the test director will conduct pre-test formal briefings as required [i.e., one before the test crew assumes shift duties (a general test overview), usually at the Operations Shift Turnover Meeting, and a second before commencing the test (a detailed briefing)].

F. At the test director's pre-test formal briefing, a Senior Manager will discuss management expectations with test personnel.

G. The Plant Manager or his designee shall exercise responsibility for the oversight of a particular test or evolution. This authority includes control of the pace of the CIPTE and the resolution (or escalation) of problems encountered.

> **NOTE**
>
> This is an oversight position and shall not interfere with or reduce the Shift Manager's responsibility for control of the test.

H. For CIPTEs, the test director will conduct a post test briefing, as required by CIPTE Manager, to discuss lessons learned to include as a minimum potential procedure changes and training.

3.9 Test Stoppage/Exiting Test/Reentering Test

Operations and the responsible supervisor shall be notified whenever any test cannot be completed. Operations, the Test Director, and the responsible supervisor will determine how to resolve the situation and continue the testing or how to safely exit the test. Tests can only be reentered when continued performance does not jeopardize personnel safety, nuclear safety, or equipment performance. When the test is suspended, reverification of the initial conditions will be coordinated with Operations.

SPP0305 010

WBN BANK QUESTION

Preparations are in progress for an RCS drain down to establish mid-loop conditions. The _______(1)_______ must conduct a pretest formal briefing with Operations personnel.

The individual assigned responsibility for conducting the management expectations briefing for the test is the _______(2)________?

	1	2
a.	Test Director	Shift Manager
b.	Plant Manager	Shift Manager
c.✔	Test Director	CIPTE Manager
d.	Plant Manager	CIPTE Manager

I. PROGRAM

Watts Bar Operator Training

II. COURSES

A. Non-licensed Operator Training, NOTP
B. Non-licensed Operator Requalification, NLOR
C. Initial License Training, ILT
D. Licensed Operator Requalification, LOR

III. TITLE

SPP-8.1, Conduct of Test

IV. LENGTH OF LESSON

A. NOTP 2.0 hours
B. ILT 2.0 hours

LOR and NLOR times will be determined after objectives are identified.

V. TRAINING OBJECTIVES

AUO	RO	SRO	STA	
X	X	X	X	1. Explain the purpose of SPP-8.1, Conduct of Test.
X	X	X	X	2. Identify the applicability of SPP-8.1 to other tests or instructions.
X	X	X	X	3. Describe the responsibilities of the supervisor, test director, and senior manager assigned to a Complex, Infrequently Performed Test or Evolution (CIPTE).
X	X	X	X	4. Determine the contents of a formal pretest briefing.
X	X	X	X	5. Identify items required in the chronological test log.

V. TRAINING OBJECTIVES (continued)

AUO	RO	SRO	STA	
X	X	X	X	6. State when operations notification is required.
X	X	X	X	7. Describe actions required when test problems are discovered
X	X	X	X	8. Describe action required to close a test with open problems.
X	X	X	X	9. Determine the applicability of SOER 91-01 to a Complex Infrequently Performed Test or Evolution.
X	X	X	X	10. Differentiate the requirements of SPP-8.1 Conduct of Test and SMP-9.0Conduct of Test.

X. LESSON BODY	INSTRUCTOR NOTES
2. Test Director a. If practical, walks down the equipment being tested before the test begins. b. Maintains awareness of any Limiting Condition for Operation (LCO) time limits created and controlled by the test and, if requested, keeps the Shift Manager informed of progress. c. Researches to identify and document any expected Engineered Safety Features (ESF) actuations prior to performance of the test. Provides explicit details and precautions to minimize the possibility of an unexpected ESF. d. Verifies test prerequisites for each section are complete before testing of that section. e. Ensures that practices which "precondition" a test toward successful completion will be avoided. f. Test Director shall sign Form SPP-8.1-1, "Test Director Assignment Sheet" to attest that: the Test Director is familiar with the test to be performed and has an understanding of the requirements and expectations of being a Test Director as described in this procedure.	**Obj 3** PPT slide 12-16 **The Qualification to be a Test Director includes:** 1) Knowledgeable in the requirements and expectations of the Test Director as described in this procedure; 2) Has the experience, knowledge, and skills to perform or oversee the conduct of the test being assigned; 3) Shall be familiar with the tools and equipment to be employed; 4) Shall be capable of determining that the calibration status of inspection and measuring equipment is current and that the proper measurement and test equipment is being used
g. Test Director is considered a supervisor and shall not perform Line Verification (QC function) activities as outlined in SPP-3.8.	**Performance of Ops SI and other procedures does not fall under the SPP-3.8, Inspection program.**
3. CIPTE Senior Manager This is an advisory position to the Shift Manager or Unit Supervisor, who has the authority and experience to exercise continuous responsibility for the oversight of a particular test or evolution. This authority includes control of the pace of the CIPTE and the resolution (or escalation) of problems encountered. This is an oversight position and shall not interfere with or reduce the Shift Manager's responsibility for control of the test.	PPT slide 18

X. LESSON BODY	INSTRUCTOR NOTES
I. Complex Infrequently Performed Tests or Evolutions	PPT slide 37-39
The following additional requirements shall be implemented for CIPTEs.	**Refer to SPP-8.1, Sect. 3.8**
1. The responsible supervisor shall consider the temporary assignment of additional personnel to assist in the conduct of these tests. This includes assignment of personnel to exercise continuous responsibility for the oversight of a particular test, including controlling the pace and resolving problems.	
2. The responsible supervisor shall consider the temporary assignment of additional personnel under the direction of the Shift Manager to augment the shift personnel. For example, assignment of an engineer or coordinator for the test or evolution, or the assignment of an additional senior reactor operator during control rod manipulations. Another example may include additional data takers when the data required is not readily available to the assigned shift at their normal shift location. The duties, authority, and responsibility of any extra personnel should be included on the organization chart and made clear in the test briefings	
3. The responsible supervisor shall ensure the test has been reviewed by an individual knowledgeable of the test before performance of the test.	
4. For CIPTEs, the test director will conduct pre-test formal briefings as required [i.e., one before the test crew assumes shift duties (a general test overview), usually at the Operations Shift Turnover Meeting, and a second before commencing the test (a detailed briefing)].	
5. At the Test Director's pre-test formal briefing, the Plant Manager or designee for the test shall conduct a briefing for Operations and testing personnel on management expectations for the test utilizing Form SPP-8.1-3.	

X. LESSON BODY	INSTRUCTOR NOTES
6. The Plant Manager or his designee shall determine the need to designate a senior line manager to advise the Shift Manager or Unit Supervisor, who has the authority and experience to exercise continuous responsibility for the oversight of a particular test or evolution. This authority includes control of the pace of the CIPTE and the resolution (or escalation) of problems encountered.	PPT slide 40
J. Test Stoppage/Exiting Test/Reentering Test	
Operations and the responsible supervisor shall be notified whenever any test cannot be completed. Operations, the Test Director, and the responsible supervisor will determine how to resolve the situation and continue the testing or how to safely exit the test. Tests can only be reentered when continued performance does not jeopardize personnel safety, nuclear safety, or equipment performance. When the test is suspended, reverification of the initial conditions will be coordinated with Operations.	PPT slide 41

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

Some examples of SRO exam items for this topic include:

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

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

Some examples of SRO exam items for this topic include:

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

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

98. G 2.3.11 098

Given the following conditions:

- Unit 1 is in Mode 3 with RCS at normal operating temperature and pressure awaiting secondary plant equipment repair to continue the startup.

- RCS Activity was determined to be 0.28 microcuries/gram DOSE EQUIVALENT I-131 and Technical Specification LCO 3.4.16, RCS Specific Activity, is entered.

- The DOSE EQUIVALENT I-131 was unable to be restored to within the LCO limit in the required action time.

Which ONE of the following identifies an action required by Tech Specs and the bases for the action?

RCS Tavg below...

A. 350°F to limit doses at the site boundary in the event of a LOCA in conjunction with the L_a value of 0.25%/day leakage from containment.

B. 500°F to limit doses at the site boundary in the event of a LOCA in conjunction with the L_a value of 0.25%/day leakage from containment.

C. 350°F to limit doses at the site boundary in the event of a Main Steam Line Break in conjunction with an existing SG tube leakage of 150 gpd.

D.✓ 500°F to limit doses at the site boundary in the event of a Main Steam Line Break in conjunction with an existing SG tube leakage of 150 gpd.

DISTRACTOR ANALYSIS:

A. *Incorrect, reducing Tavg below 350°F is not correct but the basis is being to limit doses in the event of a SGTR is correct. Plausible because lowering Tavg to 350°F would allow RHR to be placed in service and discontinue the use of the SGs as the heat sink and because a LOCA with leakage from containment could cause elevated doses at the site boundary. The distractor for a LOCA in conjunction with 0.25%/day leakage from containment is from wording in the Bases for Tech Spec 3.6.1, Containment, and 3.6.2, Containment Air Locks.*

B. *Incorrect, reducing Tavg below 500°F is correct but the bases is not due to a LOCA with assumed containment leakage. Plausible because the action stated is correct and a LOCA with leakage from containment could cause elevated doses at the site boundary. The distractor for a LOCA in conjunction with 0.25%/day leakage from containment is from wording in the Bases for Tech Spec 3.6.1, Containment, and 3.6.2, Containment Air Locks.*

C. *Incorrect, reducing Tavg below 350°F is not correct and the basis is not due to a LOCA with assumed containment leakage. Plausible because lowering Tavg to 350°F would allow RHR to be placed in service and discontinue the use of the SGs as the heat sink and because the basis being to limit the dose rate at the site boundary is correct.*

D. *Correct, with the activity above the 0.265 microcuries/gram limit in the Tech Spec 3.4.16 for 48 continuous hours, Tavg is required to be reduced to less than 500°F within 6 hours in accordance with the Tech Spec. The T/S bases indicate that reducing Tavg below 500°F prevents the release of activity should a steam line break occur since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The limit on activity is based on the resulting 2-hour doses at the site boundary not exceeding a small fraction of the 10 CFR 100 limits following a SGTR or a Main Steam Line Break in conjunction with an assumed steady state SG tube leak of150 gpd.*

Question Number: 98

Tier: 3 **Group** n/a

K/A: G 2.3.11
Radiation Control
Ability to control radiation releases.

Importance Rating: 3.8 / 4.3

10 CFR Part 55: 41.11 / 43.4 / 45.10

10CFR55.43.b: 2

K/A Match: K/A is matched and the question is SRO because the question requires knowledge of actions required by Tech Specs to control/prevent a radiation release and the Tech Spec bases for the actions.

Technical Reference: Tech Spec LCO 3.4.16, RCS Specific Activity, Amendment 55
Tech Spec 3.4.16 Bases, Revision 68
Tech Spec Bases for 3.6.1 and 3.6.2, Revision 10

Proposed references to be provided: None

Learning Objective: 3-OT-T/S0304
2. Determine the bases for each specification, as applicable, to the RCS.
4. Given plant conditions and parameters correctly determine the applicable Limiting Conditions for Operations or Technical Requirements for the various components of the RCS.

Cognitive Level:

Higher	___
Lower	X

Question Source:

New	___
Modified Bank	___
Bank	X

Question History: WBN Bank question 2.3.11 098 with choices rearranged to relocate the correct answer and the wording changed in C and D to reflect the words in the Tech Spec Bases.

Comments:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 0.265 μCi/gm.	--------------------NOTE-------------------- LCO 3.0.4.c is applicable. --	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 21 μCi/gm	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Perform SR 3.4.16.2.	4 hours
	AND	
	B.2 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 > 21 μCi/gm.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/ $\bar{E}$ μCi/gm.	7 days
SR 3.4.16.2	-------------------------------NOTE--------------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.265 μCi/gm.	14 days <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	-------------------------------NOTE------------------------------- Required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. -- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	 184 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The maximum dose to the whole body and the thyroid that an individual occupying the Main Control Room can receive for the accident duration is specified in 10 CFR 50, Appendix A, GDC 19. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits and within the 10 CFR 50, Appendix A, GDC 19 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite and Main Control Room radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) or main steam line break (MSLB) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits, and ensure the Main Control Room accident dose is within the appropriate 10 CFR 50, Appendix A, GDC 19 dose guideline limits.

The evaluations showed the potential offsite and Main Control Room dose levels for a SGTR and MSLB accident were within the appropriate 10 CFR 100 and GDC 19 guideline limits.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary and Main Control Room accident doses will not exceed the appropriate 10 CFR 100 dose guideline limits and 10 CFR 50, Appendix A, GDC 19 dose guideline limits following a SGTR or MSLB accident. The SGTR and MSLB safety analysis (Ref. 2) assume the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gallons per day (GPD). The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.14, "Secondary Specific Activity."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR and MSLB accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analyses are for two cases of reactor coolant specific activity. One case assumes specific activity at 0.265 μCi/gm DOSE EQUIVALENT I-131 with an iodine spike immediately after the accident that increases the iodine activity in the reactor coolant by a factor of 500 times the iodine production rate necessary to maintain a steady state iodine concentration of 0.265 μCi/gm DOSE EQUIVALENT I-131. The second case assumes the initial reactor coolant iodine activity at 21 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR and MSLB event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. The MSLB results in a reactor trip due to low steam pressure.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of a SGTR and MSLB accident are within the appropriate 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 21 μCi/gm DOSE EQUIVALENT I-131, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 21 μCi/gm DOSE EQUIVALENT I-131.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specific iodine activity is limited to 0.265 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary and accident dose to personnel in the Main Control Room during the Design Basis Accident (DBA) will be within the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary and accident dose to personnel in the Main Control Room during the DBA will be within the allowed whole body dose.

The SGTR and MSLB accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels and Main Control Room accident dose are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SGTR or MSLB, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits, or Main Control Room accident dose that exceed the 10 CFR 50, Appendix A, GDC 19 dose limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an accident to within the acceptable Main Control Room and site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limit of 21 μCi/gm is not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1 and B.2

With the gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is greater than 21 μCi/gm, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following rapid power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," 1973.

2. Watts Bar FSAR, Section 15.4, "Condition IV - Limiting Faults."

BASES

BACKGROUND (continued)

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves."

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break (SLB), and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) related to the design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25% per day in the safety analysis at P_a = 15.0 psig which bounds the calculated peak containment internal pressure resulting from the limiting design basis LOCA (Ref. 3).

DISTRACTOR WORDING

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a significant release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate (L_a)of 0.25% of containment air weight per day (Ref. 2), at the calculated peak containment pressure of 15.0 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

DISTRACTOR WORDING

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

(continued)

G 2.3.11 098

WATTS BAR BANK QUESTION

Given the following conditions:

- Unit 1 is in Mode 3 with RCS at normal operating pressure and temperature awaiting secondary plant equipment repair to continue the startup.
- At 1300 on 12/05/09 RCS Activity was determined to be 0.28 microcuries/gram DOSE EQUIVALENT I-131 and Technical Specification LCO 3.4.16, RCS Specific Activity is entered.

If the DOSE EQUIVALENT I-131 cannot be restored to within the LCO limit in the required action time, Which of the following identifies an action required and a bases for the action?

a. RCS Tavg below 350°F to limit doses at the site boundary in the event of a Main Steam Line Break in conjunction with steady state SG tube leakage of 1 gpm.

b. RCS Tavg below 350°F to limit doses at the site boundary in the event of a LOCA in conjunction with 0.25% L_a leakage from containment.

c.✓ RCS Tavg below 500°F to limit doses at the site boundary in the event of a Main Steam Line Break in conjunction with steady state SG tube leakage of 1 gpm.

d. RCS Tavg below 500°F to limit doses at the site boundary in the event of a LOCA in conjunction with 0.25% L_a leakage from containment.

I. **PROGRAM**

WATTS BAR OPERATOR TRAINING

II. **COURSE**

A. License Training
B. Licensed Requalification

III. **TITLE**

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

IV. **LENGTH OF LESSON**

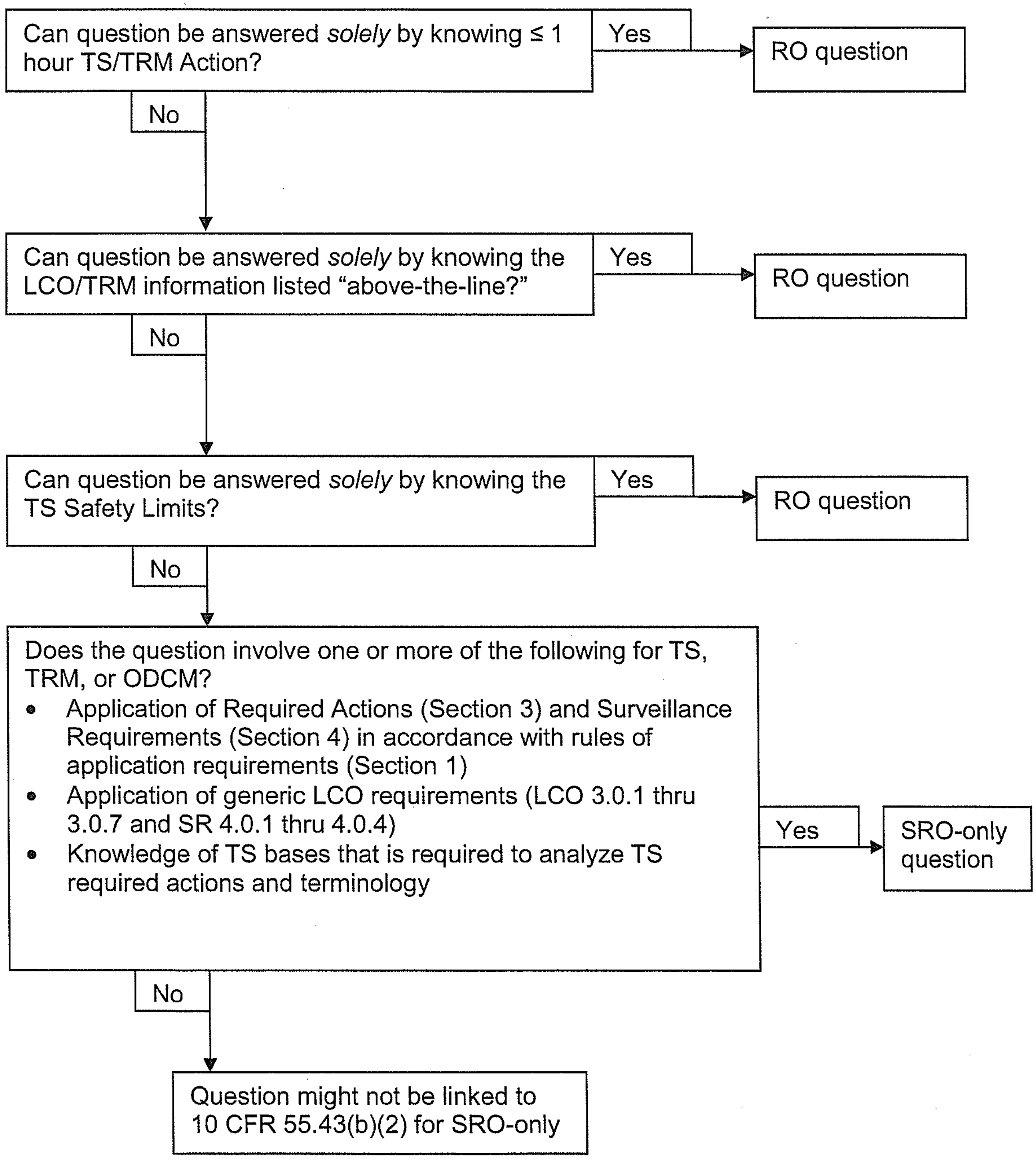
A. License Training 1 Hour

Licensed Requalification time will be determined after objectives are identified.

V. **TRAINING OBJECTIVES**

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

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



99. G 2.4.27 099

Given the following:

- Unit 1 was operating at 100% power when a fire occurred outside of the 125v Vital Battery I and II rooms.

- AOI-30.2, "Fire Safe Shutdown," has been entered.

- Using AOI-30.2, Appendix B, "Fire Safe Shutdown Elevation Diagrams," the determination is made that the fire involves both of the following rooms:

  - 480v Rx MOV Bd Rm 1B (East)
  - 480v Rx MOV Bd Rm 1B (West)

- While the fire is in progress, a Safety Injection signal is generated.

Which ONE of the following identifies...

(1) the required actions for performing the AOI-30.2 C.Series procedure(s) based on fire location

and

(2) the procedure implementation requirements after the Safety Injection occurs?

REFERENCE PROVIDED

A. (1) Performance of <u>both</u> AOI-30.2 C.3 and C.4 procedures is required.
(2) AOI-30.2 action takes precedence over the Emergency Operating Procedures.

B. (1) Performance of <u>both</u> AOI-30.2 C.3 and C.4 procedures is required.
(2) E-0, "Reactor Trip or Safety Injection," actions take precedence over AOI-30.2 actions.

C.✓ (1) Performance of <u>either</u> AOI-30.2 C.3 or C.4 procedure is sufficient.
(2) AOI-30.2 action takes precedence over the Emergency Operating Procedures.

D. (1) Performance of <u>either</u> AOI-30.2 C.3 or C.4 procedure is sufficient.
(2) E-0, "Reactor Trip or Safety Injection," actions take precedence over AOI-30.2 actions.

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because the procedure directs "IF fire spreads from one room to an adjacent room, THEN REFER to AOI-30.2 APP B again, AND PERFORM applicable AOI-30.2 C.Series procedure associated with the newly involved room" and the AOI taking precedence over the Emergency Operating Procedures is correct.*

B. *Incorrect, Plausible because the procedure directs "IF fire spreads from one room to an adjacent room, THEN REFER to AOI-30.2 APP B again, AND PERFORM applicable AOI-30.2 C.Series procedure associated with the newly involved room" and normally the Emergency Procedures take precedence over Abnormal Operating Procedures.*

C. *Correct, AOI-30.2 identifies that "For a fire that touches a soft interface (no physical wall or barrier), as indicated by heavy dashed lines in AOI-30.2 APP B, the actions of either room are sufficient" and that "For an Appendix R fire, this procedure takes precedence over the Emergency Operating Procedures" in notes in the procedure. The soft interface can be determined using the reference provided.*

D. *Incorrect, Plausible because while there is a separate AOI-30.2 C.series procedure for each of the rooms the note identifies that performance of either procedure is sufficient due to the soft interface between the areas and normally the Emergency Procedures take precedence over Abnormal Operating Procedures.*

Question Number: 99

Tier: 3 **Group** n/a

K/A: G 2.4.27
Emergency Procedures / Plan
Knowledge of “fire in the plant” procedures.

Importance Rating: 3.4 / 3.9

10 CFR Part 55: 41.10 / 43.5 / 45.13

10CFR55.43.b: 5

K/A Match: K/A is matched because the question requires knowledge of “fire in the plant” procedures and is SRO because it requires knowledge of assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed and the knowledge of hierarchy, implementation, and/or coordination of plant normal, abnormal, and

K/A Match: K/A is matched because the question requires knowledge of "fire in the plant" procedures and is SRO because it requires knowledge of assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed and the knowledge of hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Technical Reference: AOI-30.2, Fire Safe Shutdown, Revision 0031
AOI-30.2 APP B, Fire Safe Shutdown Elevation Diagrams, Revision 0000

Proposed references to be provided: AOI-30.2 APP B, Fire Safe Shutdown Elevation Diagrams 2.0 AB EL 772.0, 776.0 & 763.5 ELEVATION DIAGRAM page 5 (1 page)

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 X
Lower ___

Question Source:

New X
Modified Bank ___
Bank ___

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

Comments:

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.

4.0 OPERATOR ACTIONS (continued)

NOTE
For a fire that touches a soft interface (no physical wall or barrier), as indicated by heavy dashed lines in AOI-30.2 APP B, the actions of either room are sufficient.

ACTION/ EXPECTED RESPONSE | **RESPONSE NOT OBTAINED**

7) **IF** fire spreads from one room to an adjacent room,
   THEN
   REFER to AOI-30.2 APP B again, AND
   PERFORM applicable AOI-30.2 C. Series procedure associated with the newly involved room.

NOTE
Table 2 of the appropriate AOI-30.2 C-Series procedure provides a summary of AUO assignments.

8) **ASSIGN** AUOs appropriate local manual actions of applicable AOI-30.2 C-Series procedure.

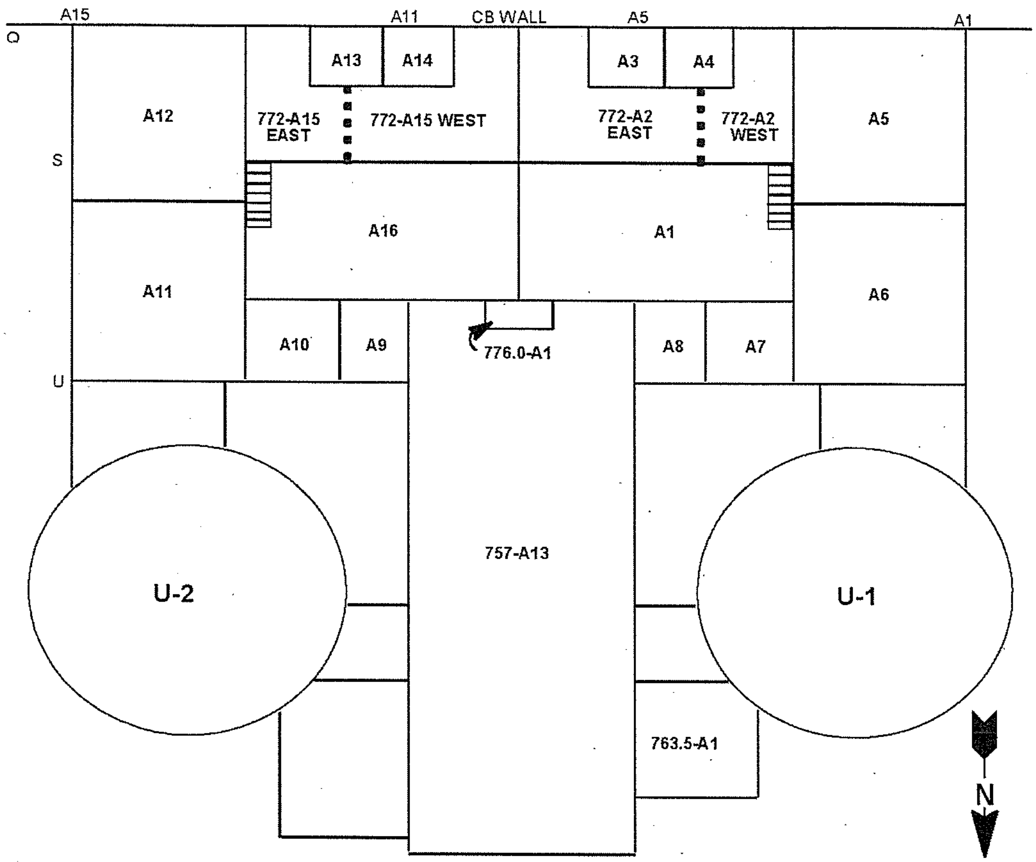
NOTE
Section 5.4 provides reference to normal plant procedures to consider while taking action to handle Appendix R fires.

9) **IF** CO_2 is discharged into U-1/U-2 Aux Instrument Rooms or Computer Room [El 708], **THEN**
   NOTIFY Nuclear Security of the need to EVACUATE Secondary Alarm Station (SAS).

10) **REFER TO** SOI-13.01 for ventilation systems required and USE AOI-30.2 APP A, as necessary, for further guidance.

2.0 AB EL 772.0, 776.0 & 763.5 ELEVATION DIAGRAM

Auxiliary Building El 772.0, 776.0 & 763.5 Diagram



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

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

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)

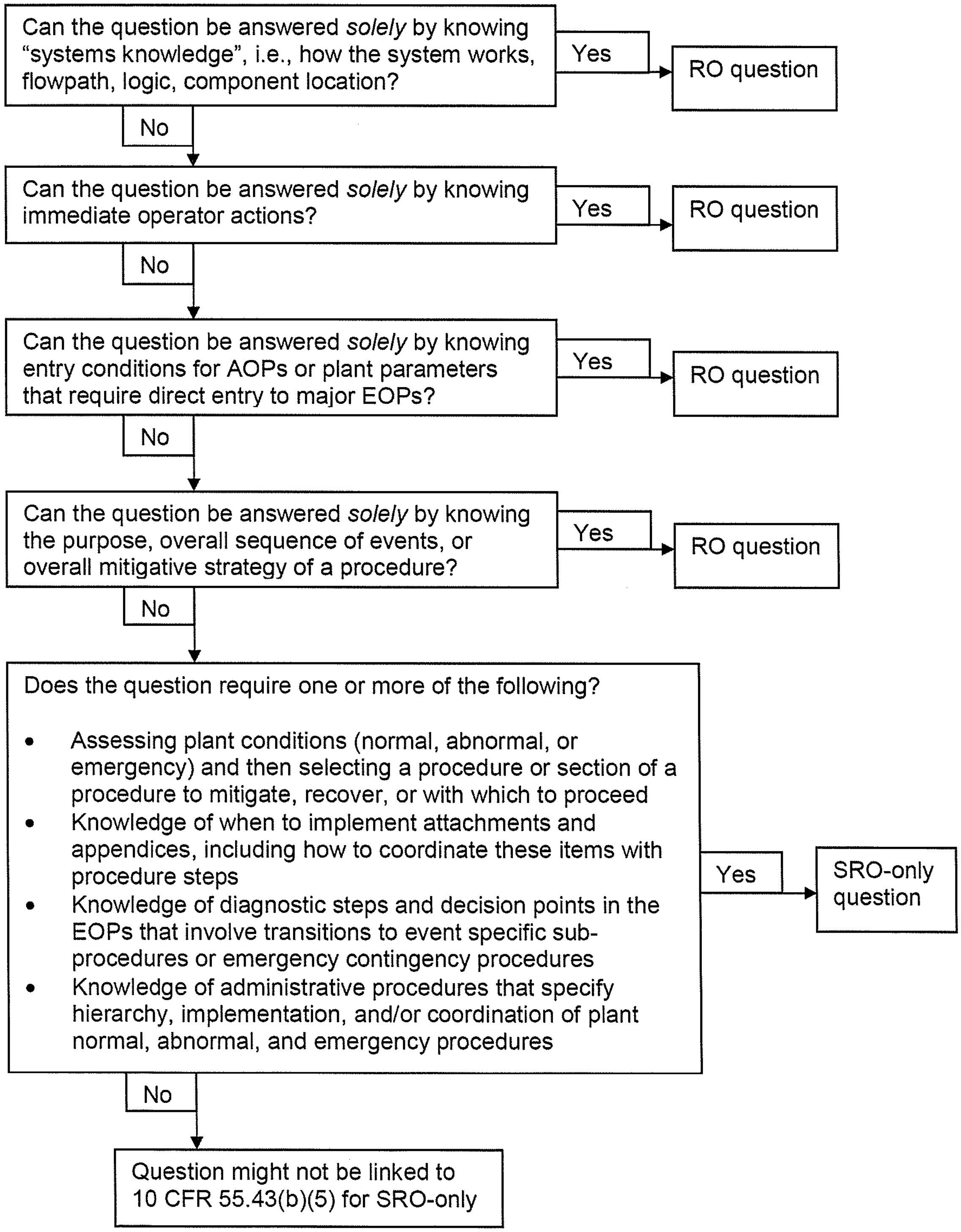
AUO	RO	SRO	STA	
X	X	X	X	13. For the following systems, describe the required system configuration and basis for the alignment as described in AOI-30.2 a. Main/Reheat Steam b. Main/Auxiliary Feedwater c. Control Air System d. Fuel Oil System e. Ventilation (HVAC) f. CVCS g. Safety Injection System h. 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

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



100. G 2.4.42 100

Given the following:

0825 - Shift Manager/SED declares a Site Area Emergerency and initiates the Emergency Paging System to staff the TSC and OSC.

0830 - The Assembly and Accountability process is initiated.

In accordance with the Radiological Emergency Plan, which ONE of the following identifies the latest time the Shift Manager would expect...

(1) to receive a report from Security on the results of accountability

and

(2) to be notified that the Technical Support Center had been activated?

	(1)	(2)
A.	0900	0925
B.✓	0900	0955
C.	0915	0925
D.	0915	0955

DISTRACTOR ANALYSIS:

A. *Incorrect, Plausible because within 30 minutes is the time expected to receive the report and 60 minutes the activation time for the CECC.*

B. *Correct, EPIP-6 identifies "Target activation time for the TSC is approximately 90 minutes." and EPIP-8, Appendix E (Nuclear Security Actions) states "**REPORT** the results of accountability to the SM/SED within 30 minutes after the assembly and accountability sirens have sounded."*

C. *Incorrect, Plausible because 45 minutes is the time that search teams would be assembled for locating individuals unaccounted for and to 60 minutes the activation time for the CECC.*

D. *Incorrect, Plausible because 45 minutes is the time that search teams would be assembled for locating individuals unaccounted for and the 90 minutes for the TSC activation time is correct.*

Question Number: 100

Tier: 3 **Group** n/a

K/A: G 2.4.42
Emergency Procedures / Plan
2.4.42 Knowledge of emergency response facilities.

Importance Rating: 2.6 / 3.8

10 CFR Part 55: 41.10 / 45.11

10CFR55.43.b: 7

K/A Match: K/A is matched and is SRO because the question requires knowledge of the time periods following implementation of the emergency plan when an emergency response facility would be expected to be activated and when a report on the status of actions taken by the Shift Manager/SED to establish accountability isexpected to be available.

Technical Reference: EPIP-6, Activation and Operation of the Technical Support Center (TSC), Revision 0039
EPIP-8, Personnel Accountability and Evacuation, Revision 0025

Proposed references to be provided: None

Learning Objective: 3-OT-PCD0048C
3. Identify the functions of the onsite emergency response facilities.

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

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

Comments:

3.2 Initiating Conditions (continued)

2. At the discretion of the Shift Manager (SM)/Site Emergency Director (SED).

B. The Shift Manager will initiate activation of the TSC by announcing the emergency condition using one or more of the following methods.

1. Plant Public Address (PA) announcement.

2. Emergency Paging System (EPS) activation.

NOTE
The Radiological Emergency Response Organization Call List is handled in accordance with the TVA Fitness For Duty Program.

3. Utilizing the Radiological Emergency Response Organization Call List and Appendix V of this procedure.

3.3 Activation of the TSC

A. Target activation time for the TSC is approximately 90 minutes. Once notified, Emergency Responders are expected to report to the TSC without delay.

B. The SED will declare the TSC activated once the Initial Activation Checklists for the required minimum staffing positions are complete in the TSC and shall inform the Shift Manager of the final transfer of responsibilities.

C. Once the TSC is activated a plant-wide brief will be given announcing the facility activation, transfer of SED responsibilities from the SM to the TSC and updating personnel on the status of the emergency.

D. Positions that are in communication with the CECC will contact their CECC counterparts as soon as practical. TSC personnel should not let contacting the CECC delay activation of the TSC.

E. Upon arrival in the TSC, personnel will perform the following

1. Card in using the accountability card readers inside the TSC hallway

2. Sign in on the roster and the TSC Staffing chart.

3. Refer to their appropriate position notebook and appendix to this procedure and perform the activities listed in the appropriate Initial TSC Activation Checklist (appendices D-R).

4. Turn on and perform function checks of equipment they will be using.

Appendix E
(Page 3 of 5)

Nuclear Security - Assembly And Accountability Actions

Date ________

1.0 GENERAL (continued)

C. **IF** Employees having emergency response assignments are located:

THEN NS will: ☐

- **Warn** and advise individuals of current actions to conduct assembly and accountability,
- **Advise** personnel to report immediately to their designated emergency response center.

D. **IF** Employees not having emergency response assignments are located

THEN NS will: ☐

- **Warn** and advise individuals of current actions to conduct assembly and accountability.
- **Advise** personnel to report immediately to a designated assembly area location outside the Protected Area. (See Appendices B and C).

[9] **REPORT** the results of accountability to the SM/SED within 30 minutes after the assembly and accountability sirens have sounded.

Time ___________ ________ Initial

[10] **UNACCOUNTED FOR INDIVIDUALS** ☐

IF Individuals remain unaccounted for **(45)** minutes following the activation of the assembly and accountability sirens,

THEN NOTIFY the TSC Security Manager or SM/SED that search teams will be needed to locate the missing individual(s),

AND RP will assist search teams (as needed).

V. TRAINING OBJECTIVES

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

V. TRAINING OBJECTIVES (continued)

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

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

Some examples of SRO exam items for this topic include:

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

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

Some examples of SRO exam items for this topic include:

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

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

III. Justification for Plant Specific Exemptions

The 25 SRO-only questions ***shall*** evaluate the additional knowledge and abilities required for the higher license level in accordance with 10 CFR 55.43(b). [NUREG 1021, Section ES-401D.2.d]

The fact that a facility licensee trains its ROs to master certain 10 CFR 55.43 knowledge, skills, and abilities does NOT mean that they can no longer be used as a basis for SRO-only questions. [Operator Licensing Feedback Web page Item 401.36 @ http://www.nrc.gov/reactors/operator-licensing/op-licensing-files/ol-feedback.pdf]

The SRO-only test item is required to be tied to one of the 10 CFR 55.43(b) items. However, if a licensee desires to evaluate a knowledge/ability that is not tied to one of the 10 CFR 55.43(b) items, then the licensee can classify the knowledge/ability as "*unique to the SRO position*" provided that there is documented evidence that ties the knowledge/ability to the licensee's SRO job position duties in accordance with the systematic approach to training (SAT).

- **Justification**: A question that is not tied to one of the 10 CFR 55.43(b) items can still be classified as "SRO-only" provided the licensee has documented evidence to prove that the knowledge/ability is "*unique to the SRO position*" at the site. An example of documented evidence includes:

    - The question is linked to a learning objective that is specifically labeled in the lesson plan as being SRO-only (e.g., some licensee lesson plans have columns in the margin that differentiate AO, RO, and SRO learning objectives) [NUREG 1021, ES-401, Section D.2.d]

      AND/OR

    - A question is linked to a task that is labeled as an SRO-only task, and the task is NOT listed in the RO task list.