Draft Submittal
(Pink Paper)

1. Senior Reactor Operator Written Exam

SURRY EXAM 2002-301

50-280, 281/2002-301
MARCH 18 - 28, 2002

Surry Initial SRO Exam 03/2002

1. 001AA2.03 001/T1G2/T1G1//4.5/4.8//SR02301/S/

During a Reactor Startup with power stable at 1 x 10E-8 amps, the control rods begin to withdraw in an uncontrolled manner (without operator action).

Which ONE of the following is the appropriate course of action?

A. Manually trip the reactor.
B. Allow the control rods to step out until power reaches the POAH where FTD and MDT will turn power.
C. Commence an Emergency Boration to compensate for the continuous rod withdrawal.
D. Place the Bank Selector switch from "MAN" to "Shutdown Bank A" since it is already fully withdrawn.

Ref: Source SR EB #3361
Surry Lesson Plan ND-93.3-LP-3 Rev. 14 objective I
Surry Lesson Plan ND-93.3-LP-3 Rev. 14 p. 30
Surry abnormal procedure 0-AP-1.00 step 2 RNO. Note stem places them in startup where the Rod Control Mode selector switch is in MAN, which is step 2.

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	A C A D A C A D D D	Scramble Range: A - D

G. Explain the purpose of the following Main Control Board reset pushbuttons, including the alarms/components affected by use:

- Start-Up Pushbutton
- Alarm Reset Pushbutton
- Reactor Trip Breakers' Reset Pushbutton

H. Using a simplified one-line diagram for illustration, explain how the Insertion Limit annunciators are generated.

I. **Explain the operator actions taken in AP-1.00, Rod Control System Malfunction, and AP-1.01, Control Rod Misalignment, to mitigate problems in the Rod Control System.**

J. Summarize the Technical Specifications associated with the Rod Control System.

K. Reproducing simplified one-line diagrams for illustration purposes, explain the overall integrated operation of the Rod Control System.

<u>Presentation</u>

Distribute all handouts. Refer to/display H/T-3.1, Objectives.

A. Purpose and Design

1. The Rod Control System serves two (major) purposes:

   a. Provides emergency shutdown (trip) of the reactor in response to signals from the Reactor Protection System or the Reactor Operator.

a. Review procedure entry conditions.

b. Review AP-1.00 for each type of failure that can occur using the following step sequence and highlighting the specified steps:

(1) Continuous rod withdrawal or insertion.

(a) Steps 1 and 2 are immediate action steps. If rod movement cannot be stopped, the reactor is tripped.

(b) If rod motion is stopped, the team checks for urgent failure, stabilizes the unit, and makes notifications to repair the problem and to management.

(2) Dropped rod

(a) Step 1 RNO sends team to step 4 to check for a dropped rod. Dropped rod indication from the IRPI is considered to be more reliable than from the NIS.

(b) Review indications of a dropped rod IAW step 4.

(c) If more than one rod has dropped, the reactor is subcritical, or the reactor is less than 25% power, trip the reactor.

(d) Reactor power is reduced to ≤ 70% power within one hour, and transition is made to AP-1.01, Control Rod Misalignment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

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<u>CAUTION</u>: The minimum temperature for criticality is 522°F. If Tave decreases below this temperature, Tech Spec 3.1.e must be reviewed.

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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]	__CHECK FOR EITHER OF THE FOLLOWING: • Continuous rod withdrawal • Continuous rod insertion	GO TO Step 4.
[2]	__STOP ROD MOTION:	
	a) Put ROD CONT MODE SEL switch in MANUAL ←	Stem of question starts w/ Rod select in manual
	b) Verify rod motion - STOPPED	b) Trip Reactor and GO TO ()-E-0, REACTOR TRIP OR SAFETY INJECTION.
3.	__GO TO STEP 13	

1. 005A2.03 001/T2G3/T2G3/CAVITATION/C/A 2.9/3.1/N/SR02301/S/RLM

The following conditions exist:
-Unit 1 has been shutdown for 200 hours
-RCS temperature is 140 degrees F
-RCS level is at mid-loop
-1A RHR pump is operating with 1B RHR pump in standby

While adjusting RHR flow to lower RCS temperature, annunciator 1B-G6, RHR HX LO FLOW, alarms. Attempts by the Reactor Operator to stabilize RHR flow rate ~~has~~ have been unsuccessful.

Which ONE of the following is the correct course of action?

A. Secure 1A RHR pump and verify RCS level in the acceptable region.
B. Start 1B RHR pump and verify RCS level in the acceptable region.
C. Verify RCS level in the acceptable region and start 1B RHR pump
D. Verify RCS level in the acceptable region and secure 1A RHR pump.

Ref: Surry lesson plan ND 95.2-LP-12, Rev. 9 objective D.
Lesson plan p.44
1-AP-27.00, LOSS OF DECAY HEAT REMOVAL CAPABILITY, steps 9 through 11.
Answer D is the correct sequence as specified in the AP.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D A D C B C B B A Scramble Range: A - D

D. [For each of the major categories of Loss of RHR events, explain how Surry's Abnormal Procedures direct the operator to respond to the events, including the importance of proper procedural sequence where applicable. SOER 85-04, Rec. 1; SOER 88-03, Rec. 3.]

E. Given a loss of RCS inventory with RCS pressure less than 1000 psig (SI Accumulators isolated) and RCS temperature greater than 200°F, describe the use of AP-16.01 to address this event.

F. Given either hypothetical or actual situations involving a loss of RHR event (or the potential thereof), differentiate between appropriate and inappropriate operator actions, including why certain actions would aggravate a Loss of Decay Heat Removal event.

Presentation

Distribute all handouts and AIAs. Refer to/display H/T-12.1, Objectives, and discuss with the trainees.

A. Event Synopses, Lessons Learned, Procedural and Design Features

Refer to AIA-12.1, INPO 88-018, Case Study on Loss of Decay Heat Removal. Have the trainees discuss the conditions and factors which caused the event and contributed to the severity of the event. Add any of the following as necessary to enhance the discussion and to ensure all areas are discussed.

1. The following is a summary of the events that took place at San Onofre.

    a. Initial plant conditions

b. If the running RHR pump has failed, Step 4 RNO will establish the standby pump to service. It is important to note that both the RHR FCV 1605 and the heat exchanger outlet HCV-1758 are closed prior to the start of the standby pump. This is done to prevent run out or vortexing if reduced inventory conditions exist. Assuming the standby pump is started, the 1605 and 1758 valves are positioned to pre-event conditions. At this point, the operator will go to Step 5 where proper RHR operation will be verified in a series of steps that will lead to procedure termination. If the pump start was unsuccessful or could not be accomplished due to electrical failures, the operator is advanced to Step 16 where actions are initiated to employ steps that will lead to establishing alternate methods of decay heat release.

c. If a successful pump start was accomplished in Step 4 RNO, RHR flow should be satisfactory IAW Step 5.

d. In Step 6, vortexing most likely would not be occurring advancing the operator to Step 12.

The procedure covers what to do if vortexing continues

e. Steps 12, 13 and 14 close out the procedure at this point provided stable temperature control is achieved.

6. Loss of inventory

a. Step 1 states 9 bulleted parameters that may be indicators associated with a loss of inventory.

b. Step 2 takes actions to stop any inventory loss including isolating letdown. Eventually Step 3 is entered which advances the operator to Step 15.

Refer to AIA-12.6, ARP BG8, Shutdown Cooling Low Level.

Review ARP with trainees.

e. RVLIS

(1) Confidence in RVLIS is lacking and the system will not be available if the outage is a refueling outage or if maintenance on the system is being performed.

(2) RVLIS readout is calibrated in % level and each percent accounts for approximately 6 inches of water level. This accuracy is not acceptable for use of level monitoring near mid-nozzle. So, RVLIS would only be good for "trending" of water level. In addition, if the RCS goes into a vacuum, as sensed by the PTs associated with RVLIS, all RVLIS channels will read "INVALID."

(3) RVLIS would provide an indication or trend of lowering level if it were operable.

Ask trainees: What would most likely be the first indications of the loss of level while at mid-nozzle?

Answer: Shutdown cooling low level alarm and RHR pump amps fluctuating.

(4) By the time the RHR pumps start vortexing, there may not be enough time to respond to the event to prevent vapor binding of the RHR pumps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	__CHECK RHR PUMP - VORTEXING • Flow indication on 1-RH-FI-1605 - OSCILLATING • Amperage indication - OSCILLATING	GO TO Step 12.

* *

CAUTION: RCS temperature may increase if RHR flow rate is less than required based on time after shutdown. (Attachment 1)

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	__REDUCE RHR FLOW TO STOP VORTEXING • Use 1-RH-FCV-1605 in MANUAL OR • Use 1-RH-HCV-1758	
8.	__CHECK RHR PUMP - STILL VORTEXING	GO TO Step 12.
9.	__CHECK RCS LEVEL - WITHIN ACCEPTABLE REGION • 1-RC-LI-100A (Attachment 2) OR • 1-RC-LR-105 (Attachment 3)	Restore RCS level to Acceptable Region of Attachment 2 or 3.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	__**VERIFY RHR PUMPS - BOTH AVAILABLE** Given in stem of question	Restore RHR pump: a) Stop pump. b) Verify RHR flow - NONE INDICATED. c) Vent pump. • 1-RH-P-1A, 1-RH-9 • 1-RH-P-1B, 1-RH-3 d) Restart pump. e) IF RHR pump can NOT be restored, THEN GO TO Step 16. f) IF RHR pump is restored, THEN GO TO Step 12.
11.	__RESTORE RHR PUMPS: a) Stop vortexing pump b) Verify RHR flow - NONE INDICATED c) Manually close 1-RH-FCV-1605 and 1-RH-HCV-1758 d) Start other RHR pump	
	e) Adjust RH control valves to return flow to pre-event rate: • 1-RH-FCV-1605 • 1-RH-HCV-1758	e) GO TO Step 16.

QUESTIONS REPORT
for Surry2002

1. 008A2.01 001/T2G3/T2G3/PUMP/C/A 3.3/3.6/N/SR02301/S/RLM

-Units 1 and 2 are at 100% power.
-1A component cooling water pump is tagged out for maintenance.
-No other activites are in progress on the component cooling water system for either unit.

Annunciator 1K-D6, CC PPS DISCH HDR B LO FLOW alarms. The Reactor Operator notes 1B component cooling water pump motor amps at minimum, but steady and greater than zero.

Which ONE of the following is the most probable cause of this alarm and what action should be taken?

✓A. 1B component cooling water pump has failed and the system should be crosstied the other unit.
B. The flow indicator has failed and a work request should be written.
C. There is a line rupture and makeup to system should aligned, the leak identified and isolated.
D. The discharge valve has been throttled and should be opened as required to clear the alarm.

Ref: Surry lesson plan ND-88.5-LP-1, objective G.
Annuciator response procedure 1K-D6, CC PPS DISCH HDR B LO FLOW, symptoms and actions.
Answer A is correct because the low flow in conjunction with the low, but greater than zero amps, indicates a pump shaft shear with the motor going to no load amps. With the other train pump unavailable, the procedure provides cross connect as the only option.
Answers B, C, and D are also possible symptoms of the alarm, as listed in the ARP. However, answer B is incorrect because a flow indicator failure would not explain pump amp decrease.
Answer C is incorrect because, per the ARP, the pump amps would oscillate.
Answer D is incorrect because the stem said no other activities on either unit's ccw system was in progress.

Note to self: The ARP's for the low flow annunciator (1K-D6 and 1K-C6) are asymetrical. 1K-C6 says the alarm is disabled when the outlet from the A RHR Hx is closed. 1K-D6 is silent. Need to know if the alarm in my scenario is disabled.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C C A C C B A A B Scramble Range: A - D

D. Summarize the contents of the normal and abnormal procedures associated with the component cooling system, including:

- Normal system operation
- AP-15.00, Loss of Component Cooling
- AP-16.00, Excessive RCS Leakage

E. State the technical specifications associated with the component cooling system, including for SRO candidates, the basis behind these specifications.

F. Describe the major system components and operation of the Chilled CC System, including:

- System purposes and components supplied
- Heat Exchangers and Valves
- Chilled CC pumps
- Indications and controls

G. Describe the overall integrated operations of the component cooling system.

Presentation

Distribute all handouts. Refer to/display H/T-1.1, Objectives, and review with trainees.

A. System Components

1. The component cooling system purpose is to provide a cooling medium for various heat loads of each reactor unit. It also acts as a barrier against the release of radioactivity to the environment.

2. CC Surge Tank

Level 2 Controlled Distribution
Maintained by this Department
Do not remove this document from work

VIRGINIA POWER
SURRY POWER STATION

ANNUNCIATOR RESPONSE PROCEDURE

PROCEDURE TITLE	REVISION
CC PPS DISCH HDR B LO FLOW	2
	PAGE

1K-30

AR 9.0

A48-ESK-10K

A48-FM-72

ch Spec 3.13

.DRP-005, Instrumentation Setpoints

CP 95-001.27, MI Setpoint Change

CAUSES

Alarm actuates when 1-CC-FS-100B senses CC Header B flow less than or equal to 4,000 gpm.

Low CC flow may be caused by one or more of the following:

- CC Pump failure.
- Flow indicator failure.
- Line rupture.
- Discharge valve throttled.

. Instrumentation failure has occurred.

ROVAL RECOMMENDED	APPROVED	DATE
		1-29-98
VIEWED S. DAVIS		

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__VERIFY ALARM - DUE TO PLANNED EVOLUTION	GO TO Step 3.
__RETURN TO PROCEDURE IN EFFECT	
__VERIFY ALARM - CC SUP HDR B FLOW - LESS THAN OR EQUAL TO 4,000 GPM	Initiate a Work Request <u>AND</u> GO TO Step 8.
__INCREASE SURVEILLANCE OF COMPONENTS SUPPLIED BY CC	
__REDUCE CC LOADS AS NECESSARY	

* *

<u>TION</u>: An LCO will be entered if Component Cooling is lost or if certain CC System components are inoperable IAW Tech Spec 3.13.

* *

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__CHECK CC SYSTEM FOR LEAKAGE • CC Surge Tank Level - DECREASING • Aux Building or CTMT Sump Level - INCREASING • CC Pump amps - OSCILLATING	Verify running or start a CC Pump. <u>IF</u> a CC Pump can <u>NOT</u> be started, <u>THEN</u> do the following: a) Verify crosstied or crosstie CC System. b) Review Tech Spec 3.13. c) GO TO Step 8. <u>IF</u> CC can <u>NOT</u> be restored, <u>THEN</u> GO TO 1-AP-15.00, LOSS OF COMPONENT COOLING.

QUESTIONS REPORT

for Surry2002

1. 008AA2.19 001/T1G2/T1G2/STUCK SPRAY VALVE/3.4/3.6/B/SR02301/S/RLM

initial

Which ONE of the following is an indication of a stuck open Pressurizer spray valve?

A. Pressurizer pressure decreasing and level decreasing.
B. Pressurizer pressure decreasing and level increasing.
C. Surge line temperature decreasing.
D. High temperature on either Pressurizer Spray Line temperature indicators 1-RC-TI-1451 or 1452

REF: SR EB # 32067

Answer C is incorrect based upon probable cause listed in ARP 1C-G8

Answer D is incorrect based upon 1-AP-31 entry condition 3, last bullet.

VIRGINIA POWER
SURRY POWER STATION

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-31.00	INCREASING OR DECREASING RCS PRESSURE (WITH 2 ATTACHMENTS)	5
		PAGE 1 of 6

PURPOSE

To provide guidance in the event of abnormal RCS pressure caused by a plant transient or equipment malfunction.

ENTRY CONDITIONS

1. Decreasing RCS pressure as indicated by any of the following:
   - PRESS LO PRESS annunciator, 1C-B8
   - Decreasing trend on PRZR PRESS Recorder, 1-RC-PR-1444 Pos 2
2. Increasing RCS pressure as indicated by any of the following:
   - PRESS HI PRESS annunciator, 1C-F8
   - Increasing trend on PRZR PRESS Recorder, 1-RC-PR-1444 Pos 2
3. Failure of RCS pressure control component(s) as indicated by any of the following:
   - PRZR PRESS CONTR HI OUTPUT annunciator, 1C-A8
   - PRZR HTRS CONT GP OL TRIP annunciator, 1C-H8
   - Increasing or decreasing trend on PRZR PRESS Recorder, 1-RC-PR-1444 Pos 1
   - Leaking PRZR Safety Valve(s) or PORV(s) as indicated by either of the following:
     a. PRZR SFTY VV LINE HI TEMP annunciator, 1C-C7
     b. PRZR PWR RELIEF LINE HI TEMP annunciator, 1C-D7
   - Leaking PRZR spray valve as indicated by low temperature on PRZR SPRAY LINE TEMP temperature indicators 1-RC-TI-1451 or 1-RC-TI-1452

APPROVAL RECOMMENDED	APPROVED	DATE
	WB Gross	6-13-01
REVIEWED		

NUMBER	PROCEDURE TITLE	REVISION / PAGE
1C-G8	PRZR SURGE LINE LO TEMP	REVISION 0
		PAGE 1 of 2

REFERENCES

1C-56

1. UFSAR 4.0
2. 11448-ESK-10C, 10AJ
3. 1-DRP-005, Instrumentation Setpoints

PROBABLE CAUSES

1. Alarm actuates when 1-RC-TC-1450 senses Pressurizer Surge Line temperature less than or equal to 500°F.
2. Low Surge Line temperature may be caused by one of the following:
   - Loss of continuous spray flow
   - Cooldown of RCS
3. Instrumentation failure has occurred.

APPROVAL RECOMMENDED	APPROVED	DATE
Roger W. Cross	Bryce O. Shriver	12/21/95
REVIEWED T. Kunkle Alice J. Swanson	CHAIRMAN STATION NUCLEAR SAFETY AND OPERATING COMMITTEE	

QUESTIONS REPORT
for Surry2002

1. 009EA2.04 001/T1G2/T1G2//3.8/4.0/B/SR02301/S/RLM

-Unit 1 has experienced a small break LOCA.
-The RCP's are tripped.
-A cooldown has been performed to ≈ 150 psig
-The plant has been depressurized in accordance with 1-ES-1.2, Post LOCA Cooldown and Depressurization.

Which ONE (1) of the following explains why pressurizer level will eventually stabilize?

A. Break flow equalizes with injection flow.
B. The void in the vessel head stops expanding.
C. ECCS injection flow has been heated and expanded and is now in the thermal equilibrium with decay heat generation.
D. Accumulators have partially injected to raise pressurizer level and are now at equal pressure with the RCS.

Source: FA EB# 44701
Surry Lesson Plan ND-95.3-LP-9, Rev 8
Learning objective B
Correct answer based on pp. b(1), p.5

(d) An explicit check of S/G levels is performed and is contained within the main cooldown loop. This ensures continuous monitoring for possible SGTRs.

(2) After these actions and checks are performed, a cooldown to CSD is initiated. With continued cooldown, subsequent actions can be performed when specified RCS subcooling criteria are satisfied.

b. **DEPRESSURIZE RCS TO REFILL PRESSURIZER.**

(1) This action is performed prior to RCP restart or before/after an SI reduction action. As RCS pressure decreases, injection flow will increase relative to break flow. Consequently, this depressurization action should be sufficient for restoring pzr level if the LOCA is small.

(2) A "small" LOCA is first ensured by requiring RCS subcooling before depressurization. If subcooling is lost during the depressurization, it should be restored as the cooldown continues. Prior to restoring pzr level, all pzr heaters are turned off.

c. **START ONE RCP/STOP ALL BUT ONE RCP.**

(1) Once RCS subcooling, pzr level, and other RCP support conditions are established, an RCP can be started if no RCPs are running. The RCP restarted (or left running) is used to provide normal pzr spray and mix the RCS.

(2) If more than one RCP is running, all but one are stopped to minimize RCS heat input.

LESSON PLAN

Introduction

ES-1.2, Post-LOCA Cooldown and Depressurization, provides guidance to cooldown and depressurize the RCS to cold shutdown conditions following a loss of reactor coolant. This procedure and supporting analyses are structured to deal primarily with small LOCAs where SI flow can keep up with break flow, at pressures above the shutoff head of the LHSI pumps.

In addition, if a LOCA occurs and the HHSI system fails, the procedure provides optimal recovery actions to try to prevent an inadequate core cooling condition while trying to restore SI flow.

After reaching and maintaining cold shutdown conditions (RCS temperature less than 200°F), the final step of ES-1.2 instructs the team and plant engineering staff to evaluate the long-term plant status. At this time, the RCS will be cooled by either RHR or the cold/hot leg recirculation mode.

This lesson plan on the post-LOCA cooldown and depressurization will present the procedure both from a "big-picture" perspective and from an "in-depth" perspective.

Objectives

After receiving this instruction, the trainee will be able to:

A. Given the major action categories associated with ES-1.2, Post-LOCA Cooldown and Depressurization, explain the purpose of ES-1.2, the transition criteria for entering and exiting ES-1.2 and the types of operator actions that will occur within each category.

B. Given a copy of ES-1.2, Post-LOCA Cooldown and Depressurization, explain the basis of each procedural step.

QUESTIONS REPORT
for Surry2002

1. 010A2.03 002/T2G2/T2G2/RCS LEAKAGE/C/A 4.1/4.2/B/SR02301/S/RLM

A pressurizer PORV is leaking by the seat to the PRT at a rate of 1 gpm. All other system components are normal. PRT remains intact.

Which ONE of the following describes the Technical Specification classification and required actions?

A. Unidentified leakage that requires shutdown.
B. Identified leakage that requires shutdown.
C. Unidentified leakage with no shutdown required.
D. Identified leakage with no shutdown required.

Ref: SR EB #1773
ND-88.1-LP-9H; SROUTP-SDS-1/C; TS 3.1.C

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D B D B D A A B C B Scramble Range: A - D

H. Describe the RCS Tech Specs, including for the SRO candidate, the basis behind each specification.

I. Prepare a general content outline of the subject matter in Surry Technical Specifications, specifying the major area to which each section is dedicated, including a detailed description of the RCS section of Tech Specs.

<u>**Presentation**</u>

Distribute all handouts. Refer to/display H/T-9.1, Objectives, and review with trainees.

A. Tech Spec Section 1.0, Definitions

This section presents a number of frequently used terms. However, looking at 10CFR50.36, "Definitions" is not a required section of Tech Specs.

Ask trainees: Why are definitions considered important enough to be a T.S. Section? Answer: To ensure consistency and set a standard for terminology. To provide for uniform interpretation of the specifications.

Review each of the definitions in Tech Spec Section 1.0.

Refer to/display H/T-9.2, 2.0 Safety Limits and Limiting Safety System Settings.

(2) 3.1.B - Requirements for RCS component Heatup and Cooldown limits.

(3) 3.1.C - Limits for RCS and associated component leakage.

(4) 3.1.D - Limits for the activity levels in the RCS.

(5) 3.1.E - Requirements for the Minimum Temperature for Criticality.

(6) 3.1.F - Limits for RCS chemical containment concentrations.

(7) 3.1.G - Requirements for RCS Overpressure Mitigation Operability.

f. Tech Spec Section 3.2 - This section describes the CVCS components that must be operable. This section also provides the definition of AVAILABLE.

g. Tech Spec Section 3.3 - This section provides the requirements for the SI system components.

h. Tech Spec Section 3.4 - This section provides the requirements for the CS and RS components.

i. Tech Spec Section 3.5 - This section provides the requirements for the RHR system.

j. Tech Spec Section 3.6 - This section provides the requirements for the SG Safety Valves and the AFW systems components.

age

Specifications

1. Detected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated. At least two means shall be available to detect reactor coolant system leakage. One of these means must depend on the detection of radionuclides in the containment.
2. If the leakage rate, from other than controlled leakage sources, such as the Reactor Coolant Pump Controlled Leakage Seals, exceeds 1 gpm and the source of the leakage is not identified within four hours of leak detection, the reactor shall be brought to hot shutdown. If the source of leakage is not identified within an additional 48 hours, the reactor shall be brought to a cold shutdown condition.
3. If the sources of leakage are identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage, other than leakage from controlled sources, not exceeding 10 gpm shall be permitted except as specified in C.4 below.
4. If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System component body, pipe well, vessel wall, or pipe weld, the reactor shall be brought to a cold shutdown condition and corrective action taken prior to resumption of unit operation.
5. If the total leakage,other than leakage from controlled sources, exceeds 10 gpm the reactor shall be placed in the cold shutdown condition.

1. 012A2.02 001

Procedures AP-10.02, AP-10.03, and AP-10.04 (Loss of Vital Bus II, III, or IV) direct the operator to trip the reactor prior to tripping the affected RCP.

Which one of the following is the basis for tripping the reactor before tripping the RCP?

A. To ensure a cooldown rate is initiated in the affected loop.

B. To prevent exceeding the linear heat generation rate limit.

C. To ensure SDM is present when backflow through the affected loop is initiated.

D. To prevent an unnecessary challenge to the Reactor Protection System.

Ref: Surry Exam Bank.
Lesson Plan ND-90.3LP-5E; AP-10.02, AP-10.03, AP-10.04.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D A D D B B D A A A Scramble Range: A - D

RO Tier:	T2G2	SRO Tier:	T2G2
Keyword:		Cog Level:	3.6/3.9 MEMORY
Source:	B	Exam:	SR02301
Test:	S	Misc:	GWL

B. [Describe the components and indications associated with an Uninterruptable Power Supply (UPS). SOER 83-03, Recommendation 11]

C. Describe the power sources and loads associated with the Appendix R distribution system.

D. Describe the power sources and loads associated with the Semi-Vital Bus distribution system.

E. [Given a loss of a Vital or Semi-Vital bus, describe the actions taken IAW AP-10.01, 10.02, 10.03, 10.04, and/or 10.05 to address this loss. SOER 83-03, Recommendation 11 and SOER 81-02, Recommendation 5]

F. Given a loss of a Vital or Semi-Vital bus, describe the effect on Plant indications and controls, including actions taken IAW applicable APs to address the loss.

Presentation

Distribute all handouts. Refer to/display H/T-5.1, Objectives, and review with trainees.

A. One-Line Diagram

1. The purpose of the Vital Bus Distribution System is to supply a stable, reliable source of power to vital instruments. It must remain uninterrupted to prevent spurious shutdowns and guarantee proper action when instruments or controls are required.

LESSON PLAN

Introduction

It was a normal shift on Surry Unit 2. The date was 10-10-82. Surry #2 was operating steady state 100% power with no evolutions planned. Suddenly, the Unit 2 annunciators sounded. A runback of the turbine started. The operators verify the steam dumps are opening and that the rods are inserting. They diagnosed that vital bus #3 had failed. Shortly after this and within a few seconds, the #2 reactor trips and then safety injects.

Each event listed above actually occurred. The loss of a vital bus is a major challenge to the plant instrumentation and to the operator.

A loss of power to a vital bus can result in either a runback or a reactor trip. This results in a loss of income for the company. To reduce the possibility of loss of power to a vital bus, uninterruptable power supplies were installed during an upgrade of the 120 VDC and vital electrical distribution system.

This lesson plan will provide the information for the trainee to identify and respond properly to a transient on any vital bus. It will discuss the operation and the location of major vital bus components.

Objectives

After receiving this instruction, the trainee will be able to:

A. [Using a one-line diagram drawn from memory, describe the components and current flowpaths of the Vital, Semi-Vital, and Appendix R Distribution Systems. SOER 83-03, Recommendation 11]

216 **ID: AOP0050** **Points: 1.00**

Procedures AP-10.02, AP-10.03, and AP-10.04 (Loss of Vital Bus II, III, or IV) direct the operator to trip the reactor prior to tripping the affected RCP.

Which ONE of the following is the basis for tripping the reactor before tripping the RCP?

A. To prevent exceeding the linear heat generation rate limit.

B. To prevent an unnecessary challenge to the Reactor Protection System.

C. To ensure a cooldown rate is initiated in the affected loop.

D. To ensure SDM is present when backflow through the affected loop is initiated.

Answer: B

Question 216 Details

Question Type:	Multiple Choice
Topic:	AOP0050 (AOP0049)
System ID:	72525
User ID:	AOP0050
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-90.3-LP-5E; AP-10.02, AP-10.03, AP-10.04 [S97-0497], [S95-1153], [S95-0431]

1. 033G2.4.21 001

-A feed line break in Containment is in progress.
-The reactor failed to trip and FR-S.1 has been entered.
-Containment Pressure is 10 psig. 21.0
-When Checking for subcriticality, power is still approximately15% and falling.

Which one of the following conditions must be present IAW FR-S.1 to satisfy the subcriticality criteria, and allow an exit from the procedure?

A. A negative Intermediate range start up rate and Tavg. trending down.

B. A negative gamma-metric wide range power decreasing and Gamma-metrics wide range power < 5%.

C. A negative intermediate range startup rate and power range channels less than 5%.

D. A negative intermediate range startup rate and gamma-metrics power range channels less than 15%.

Surry Lesson Plan ND-95.3-36 objective D. Lesson plan page four paragraph C.

A. Incorrect, Tavg is not used to determine subcriticality.

B. Correct, with adverse conditions in containment the FR directs the gamma-metrics to be used because the Excore NI's are not enviromentally qualified.

C. Incorrect, This would be used if no adverse conditions in containment existed.

D. Incorrect, again with adverse conditions in containment the Excores are not used.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B C C B A C B A A B	Scramble Range: A - D

RO Tier:	T1G2	SRO Tier:	T1G2
Keyword:		Cog Level:	C/A 3.7/4.3
Source:	N	Exam:	SR02301
Test:	S	Misc:	GWL

Presentation

Distribute all handouts. Refer to/display H/T-36.1, Objectives. Review objectives with trainees.

A. Subcriticality Status Tree

1. The Subcriticality status tree provides a systematic method to determine the status of the Subcriticality Critical Safety Function. It evaluates whether any challenges to this CSF exist or not.

2. General

   a. This tree requires no operator action other than monitoring a limited set of plant parameters and comparing them to reference values within the tree.

   b. This tree represents the highest priority Critical Safety Function and is <u>always</u> entered <u>first</u> anytime tree monitoring is initiated. The tree can direct operators to either of two subcriticality FRs.

   c. The Excore NI's are not environmentally qualified for adverse containment conditions. For this reason the note exits to use the Gamma-Metrics Excore neutron monitor system(Source and Wide Ranges) for monitoring the subcriticality status tree for the duration of the event. The Gamma-metrics are used once adverse containment numbers are exceeded.

   d. Since this tree is monitoring the reactivity state of the core, the parameters being evaluated are those characterizing neutron flux behavior (leakage) measured by the Ex-Core NIS and Ex-core Gamma-Metrics systems.

The Function Restoration procedure, FR-S.1, Response to Nuclear Power Generation/ATWS, provides guidance in the event of an unexpected nuclear flux condition following a Reactor Trip or SI actuation or if an ATWS has occurred.

The objective of the recovery/restoration technique of FR-S.1 is to add negative reactivity to restore the core to subcriticality; restoration of shutdown margin is desired, but is not a necessity to exit this procedure.

This lesson on FR-S.1, Response to Nuclear Power Generation/ATWS, will provide an in-depth look at the designed response to this challenge to the Subcriticality Critical Safety Function.

Objectives

After receiving this instruction, the trainee will be able to:

A. Given a simulated plant condition requiring the use of the critical safety function status trees, transition through the subcriticality status tree denoting, in accordance with the rules of priority, any applicable function restoration procedure needing implementation.

B. Given the Major Action Categories associated with FR-S.1, Response to Nuclear Power Generation/ATWS, explain the purpose of FR-S.1, the transition criteria for entering and exiting FR-S.1, and the types of operator actions that will occur within each category.

C. Given a copy of FR-S.1, Response to Nuclear Power Generation/ATWS, explain the basis of each procedural step.

D. Given actual or simulated plant conditions requiring implementation of FR-S.1, Response to Nuclear Power Generation/ATWS, successfully transition through the procedure, performing immediate operator actions from memory and applying step background knowledge as required, to address the Critical Safety Function challenge in progress.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12.	__CHECK CETCs - LESS THAN 1200°F	<u>IF</u> CETC temperature increasing, <u>THEN</u> GO TO 1-SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE.

<u>NOTE</u>: If adverse CTMT conditions have been exceeded, the Gamma-Metrics Excore Neutron Monitor system (Source and Wide Ranges) should be used to monitor neutron flux for the duration of the event.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	__VERIFY REACTOR SUBCRITICAL: a) Check power range channels - LESS THAN 5% [Gamma-Metrics Wide Range Power - LESS THAN 5%] b) Check the following: • Intermediate range channels - NEGATIVE STARTUP RATE [Gamma-Metrics Wide Range Power - DECREASING] <u>OR</u> • Shutdown margin IAW 1-OP-RX-002, SHUTDOWN MARGIN (CALCULATED AT ZERO POWER) - GREATER THAN 1.77%	Do the following: 1) Continue to borate. <u>IF</u> boration <u>NOT</u> effective, <u>THEN</u> allow RCS to heat up. 2) Do actions of other FRs in effect which do <u>NOT</u> cooldown or otherwise add positive reactivity to the core. 3) RETURN TO Step 4.

* *

<u>CAUTION</u>: Boration should be continued to obtain adequate shutdown margin during subsequent actions.

* *

14. __RETURN TO PROCEDURE AND STEP IN EFFECT

\- END -

1. 026AA2.01 001/T1G1/T1G1/CCWS LEAK/2.9/3.5/B/SR02301/S/RLM

Given the following plant conditions:

- The plant is shutdown.
- RCS cooldown is in progress.
- RCS temperature is 190 degrees F.
- The CC SURGE TK HI-LO LVL alarm has actuated.
- The CC surge tank indicates an increase in level.

Which one of the following could be the cause of the problem?

A. The automatic makeup valve has malfunctioned causing level to increase.
B. TV-CC-109A, CC RTN HDR A OTSD TRIP VLV, has closed isolating the "A" CC return header.
C.✓A leak is present in the RHR heat exchanger.
D. A leak is present in the Seal Water return cooler.

REF: SR EB # 43389

ND-88.5-LP-1, Rev. 16, pg. 8, Obj. G.

ND-88.5-AIA-1.1 p.1&2

Answer C is correct due to plant conditions being in a cooldown at 190 degrees F (RHR inservice)

Answer A is incorrect because makeup sources have only manuall operation capability. (see Lesson Plan p.5.

Answer B is incorrect because the hydraulics of the system do not support the answer.

Answers D is incorrect because they it is not listed in the lesson plan as a posible inleakage source.

D. Summarize the contents of the normal and abnormal procedures associated with the component cooling system, including:

- Normal system operation
- AP-15.00, Loss of Component Cooling
- AP-16.00, Excessive RCS Leakage

E. State the technical specifications associated with the component cooling system, including for SRO candidates, the basis behind these specifications.

F. Describe the major system components and operation of the Chilled CC System, including:

- System purposes and components supplied
- Chilled CC pumps
- Heat Exchangers and Valves
- Indications and controls

G. Describe the overall integrated operations of the component cooling system.

Presentation

Distribute all handouts. Refer to/display H/T-1.1, Objectives, and review with trainees.

A. System Components

1. The component cooling system purpose is to provide a cooling medium for various heat loads of each reactor unit. It also acts as a barrier against the release of radioactivity to the environment.

2. CC Surge Tank

Ask Trainees: **What are some of these possible leakage sources?** Write on chalkboard as trainees name the sources:

- RCP thermal barriers,
- Primary sample coolers,
- High Radiation Sample system coolers,
- Boron Recovery system heat exchangers and pump seals,
- Fuel pit coolers
- Non-regenerative heat exchangers,
- Excess letdown heat exchanger, and
- RHR heat exchangers and pump seals.

b. RM-SW-107A/B/C/D

Each CCHX has an in-line radiation monitor installed in a well on the SW discharge piping from each HX. The detectors are connected to individual modules on the Common Rad Monitor Panel in the MCR.

7. Chemical Addition

a. The CC system is provided with a 120-gallon chemical addition tank. Its original function was to provide a means of adding either potassium hydroxide for Ph control or potassium chromate (or dichromate) for corrosion control.

a. The CC surge tank provides the NPSH for the CC pumps. It is located approximately 30 feet above the pumps, ensuring that an adequate head exists at the pump suction to prevent cavitation. The surge tank allows for fluid expansion and contraction and provides a source of makeup to the system.

b. The surge tank has a capacity of 2810 gallons and is normally maintained approximately 60% full, allowing sufficient volume to accommodate minor system surges and thermal swell due to cooldown operations.

c. Makeup water is provided by the condensate system via the bearing cooling makeup pump (1-BC-P-2) or the high pressure condensate header. There is no automatic makeup control provided, therefore, both sources of makeup water require a manual valve lineup in the Turbine Building basement.

d. The tank is vented to the process vent system via HCV-CC-100. This vent valve will automatically close upon receipt of a CC radiation monitor alarm.

3. Component Cooling Pumps

a. The CC pumps provide the motive force for circulating cooling water through the CC heat exchangers, individual system loads, and back to the pump suction. Normally two pumps (one per unit) supply the required cooling water flow. The two standby pumps provide 100% backup capability. The standby pump will auto start on a low discharge header pressure of 55 psig.

b. Each pump is rated at 9000 gpm at 200 ft. head.

COMPONENT COOLING SYSTEM LOADS

Common BR Components

1. Stripper Overhead Condenser
2. PDT Vent Chiller Condenser
3. PDT Pump
4. High Level Waste Drain Tank Pump
5. Overhead Gas Compressor
6. Stripper Trim Cooler

BR Evaporator Components

7. HRSS Sample Coolers †
8. BR Evaporator Circ Pumps •
9. BR Distillate Coolers •
10. BR Overhead Condensers •
11. BR Evaporator Distillate Pumps •
12. Primary Sample Coolers †

Spent Fuel Pit Coolers/Cask

13. Spent Fuel Pit Coolers †
14. Spent Fuel Pit Cask

NRHX/Seal Water RTN

15. Nonregenerative Heat Exchanger †
16. Seal Water Return Cooler

Notes:

† **Possible source of leakage into Component Cooling**

• Abandoned in place equipment

COMPONENT COOLING SYSTEM LOADS

CARF/NST

17. Containment Instrument Air Compressor
18. Containment Air Recirc Fan Coolers
19. Neutron Shield Tank Coolers

CRDM Shroud Cooling/RCP

19. Shroud Cooling Coils
21. RCP Thermal Barrier Heat Exchangers †
22. RCP Motor Air Coolers
23. RCP Bearing Lube Oil Coolers

Hot Pipe Containment Penetration Cooling (>150^{0}F)

24. Containment Penetration Coolers
    a. Letdown
    b. Blowdown
    c. Main Steam
    d. Main Feed

Excess Letdown/RHR

25. Excess Letdown Heat Exchanger †
26. Primary Drains Cooler
27. RHR Heat Exchanger †
28. RHR Pump Seals †
29. Primary Shield Wall Coolers - for each loop penetration

Notes:

† Possible source of leakage into Component Cooling

1. 029G2.4.21 001/T1G2/T1G1/EVALUATE PERFORMANCE/C/A 3.7/4.3/B/SR02301/S/RLM
The following plant conditions exist:

-An ATWS is in progress.
-All feedwater to the steam generators has been lost.
-The turbine generator has remained loaded and running.

Which ONE of the following would be an indication of the above conditions several minutes after the ATWS occurred? (Assume all control systems are in AUTO and no operator action is taken.)

A. Reactor power increases; pressurizer pressure decreases; pressurizer level decreases; steam pressure increases.
B. Reactor power decreases; pressurizer pressure decreases; pressurizer level decreases; steam pressure decreases.
C.✓ Reactor power decreases; pressurizer pressure increases; pressurizer level increases; steam pressure decreases.
D. Reactor power remains stable; pressurizer pressure increases; pressurizer level increases; steam pressure increases.

Ref: Surry EB #TAA0081

Surry lesson plan: ND-95.1-LP-11, obj. B

RO Tier:	T1G2	SRO Tier:	T1G1
Keyword:	EVALUATE PERFORMANCE	Cog Level:	C/A 3.7/4.3
Source:	B	Exam:	SR02301
Test:	S	Misc:	RLM

have tripped but did not. The reactor was tripped manually at the SRO's direction approximately 25 seconds after the trip demand was generated. Subsequent testing of the reactor trip breakers indicated that both breakers had failed to open, apparently due to mechanical binding in the undervoltage trip mechanisms.

The failure of the RTBs to open automatically when required places total reliance on operator actions to terminate a plant transient. Failure to initiate a reactor trip during certain transients results in a potentially severe challenge to the integrity of the Primary Coolant System. If a manual reactor trip is delayed, permanent damage to components and systems may occur. The Safety Analyses that rely on automatic reactor trips also may be invalidated. The time available for operator actions necessary to mitigate the consequences of certain events is varied and dependent on initial plant conditions. This type of transient results in a severe challenge to the barriers associated with the prevention of radioactive materials released to the environment. This lesson plan will provide an insight into the most limiting analyzed ATWT event in order to provide the background knowledge level required to discuss associated Emergency Response Guideline Procedures associated with this events.

Objectives

After receiving this instruction, the trainee will be able to:

A. Differentiate between the "trip-demand" signal first out annunciators and the "trip-indication" first out annunciators.

B. Explain the sequence of events for the most limiting ATWT event.

C. **[Explain the two events that must occur following an ATWT in order to prevent the Reactor Coolant System pressure from exceeding the stress limitations. SOER 83-08, Recommendation 11].**

1. 037AA2.10 001/T1G2/T1G2/T S LEAKAGE/C/A 3.2/4.1/M/SR02301/S/RLM

Given the following:

-Unit 1 is operating at 100% power and the latest leak rate data shows:

8.6 GPM - Total RCS leakage rate
1.6 GPM - Leakage into the PRT (previously evaluated as permissible)
3.0 GPM - Leakage into the Reactor Coolant Drain Tank from RCP seals
0.35 GPM - From the 1A Steam Generator
0.34 GPM - From the 1B Steam Generator
0.32 GPM - From the 1C Steam Generator
2.0 GPM - Charging pump leakage (previously evaluated as permissible)

WHICH ONE (1) of the following identifies the RCS leakage that requires the plant to be shutdown?

A. PRESSURE BOUNDARY LEAKAGE
B. UNIDENTIFIED LEAKAGE
C. IDENTIFIED LEAKAGE
D. PRIMARY to SECONDARY LEAKAGE
Ref: HR EB # 44454
Answer A incorrect because no pressure boundary leakage was specified in the stem.
Answer B is incorrect because 8.6-(1.6+3.0+1.1+2.0) = .9 gpm unidentified is acceptable.
Answer C is incorrect because indentified leakage is less than 10 gpm.
Answer D is correct because S/G total leakage exceeds both 1 gpm and >500 gpd in the 1A S/G.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D C B D D C A C D A Scramble Range: A - D

Lesson plan ND-88.1-LP-9

Lesson plan says S/G leakage is ~~AND~~ OR for 1gpm + 500 gpd
Tech Specs says AND. Which is correct?

*QNUM 44454
*HNUM 45878 (Do NOT change If < 9,000,000)
*ANUM
*QCHANGED FALSE
*ACHANGED FALSE
*QDATE 1995/06/26
*FAC 400 Shearon Harris 1
*RTYP PWR-WEC3
*EXLEVEL S
*EXMNR
*QVAL
*SEC
*SUBSORT
*KA 002000G005
*QUESTION
Given the following:

-The plant is operating at 75% power and the latest leak rate data
shows:

11.3 GPM - Total RCS leakage rate
1.6 GPM - Leakage into the PRT
2.0 GPM - Leakage into the Reactor Coolant Drain Tank
1.5 GPM - Leakage past check valves from RCS to SI system
1.7 GPM - Leakage into Equipment Drain Tank
0.8 GPM - Total primary to secondary leakage (Assume distributed over all S/Gs)
2.0 GPM - Charging pump leakage

Source of Modified Question

WHICH ONE (1) of the following identifies the RCS leakage that requires
the plant to be shutdown?

a. PRESSURE BOUNDARY LEAKAGE

b. UNIDENTIFIED LEAKAGE

c. IDENTIFIED LEAKAGE

d. PRIMARY to SECONDARY LEAKAGE

C. Leakage

Specifications

1. Detected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated. At least two means shall be available to detect reactor coolant system leakage. One of these means must depend on the detection of radionuclides in the containment.
2. If the leakage rate, from other than controlled leakage sources, such as the Reactor Coolant Pump Controlled Leakage Seals, exceeds 1 gpm and the source of the leakage is not identified within four hours of leak detection, the reactor shall be brought to hot shutdown. If the source of leakage is not identified within an additional 48 hours, the reactor shall be brought to a cold shutdown condition.
3. If the sources of leakage are identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage, other than leakage from controlled sources, not exceeding 10 gpm shall be permitted except as specified in C.4 below.
4. If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System component body, pipe wall, vessel wall, or pipe weld, the reactor shall be brought to a cold shutdown condition and corrective action taken prior to resumption of unit operation.
5. If the total leakage,other than leakage from controlled sources, exceeds 10 gpm the reactor shall be placed in the cold shutdown condition.

6. If the primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System exceeds 1 gpm total and 500 gallons-per day through any one steam generator not isolated from the Reactor Coolant System, reduce the leakage rate to within limits within 4 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

7a. Prior to going critical all primary coolant system pressure isolation valves listed below shall be functional as a pressure isolation device, except as specified in 3.1.C.7.b. Valve leakage shall not exceed the amounts indicated.

	Unit 1	Unit 2	Max. Allowable Leakage (see note (a) below)
Loop A, Cold Leg	1-SI-79, 1-SI-241	2-SI-79, 2-SI-241	≤5.0 gpm for each valve
Loop B, Cold Leg	1-SI-82, 1-SI-242	2-SI-82, 2-SI-242	
Loop C, Cold Leg	1-SI-85, 1-SI-243	2-SI-85, 2-SI-243	

b. If Specification 3.1.C.7.a cannot be met, an orderly shutdown shall be initiated and the reactor shall be in hot shutdown within 6 hours and in the cold shutdown condition within the following 30 hours.

Notes

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

H. Describe the RCS Tech Specs, including for the SRO candidate, the basis behind each specification.

I. Prepare a general content outline of the subject matter in Surry Technical Specifications, specifying the major area to which each section is dedicated, including a detailed description of the RCS section of Tech Specs.

Presentation

Distribute all handouts. Refer to/display H/T-9.1, Objectives, and review with trainees.

A. Tech Spec Section 1.0, Definitions

This section presents a number of frequently used terms. However, looking at 10CFR50.36, "Definitions" is not a required section of Tech Specs.

Ask trainees: Why are definitions considered important enough to be a T.S. Section? Answer: To ensure consistency and set a standard for terminology. To provide for uniform interpretation of the specifications.

Review each of the definitions in Tech Spec Section 1.0.

Refer to/display H/T-9.2, 2.0 Safety Limits and Limiting Safety System Settings.

c. Pressurizer H/U less than 100°F/hr and C/D less than 200°F/hr. Maximum ΔT between pressurizer and spray water is 320°F.

Basis - Maintains thermal stresses at spray line nozzle below design limits.

9. Tech Spec 3.1.C - Leakage

a. Detected or suspected leakage shall be investigated and evaluated.

b. Two means of detecting RCS leakage shall be available, one of which must depend on detection of radionuclides in containment

Ask trainees: What indications are available to detect RCS leakage? List on chalkboard: 1. Increased make-up water (charging and/or VCT M/U) 2. High temp in Rx vessel flange leakoff 3. Containment sump level 4. Containment pressure, temperature, humidity 5. Containment particulate and gas RM 6. Other RMs - air ejector, CC, SGBD

c. If leak rate greater than 1 gpm from other than controlled leakage sources and not identified within 4 hours, proceed to HSD; if still not found after additional 48 hours, proceed to CSD.

d. If leak source identified and safe operations is verified, leakage may increase up to 10 gpm.

e. If leakage is through an unisolable fault in a component body, pipe well, vessel wall, or pipe weld, proceed to CSD and repair prior to restart.

f. If total leakage, other than controlled, exceeds 10 gpm proceed to CSD.

g. If primary-to-secondary leakage through all S/G not isolated from RCS exceeds 1 gpm total or 500 gallons per day (.35 gpm) through any one S/G, get leakage in spec within 4 hours or shutdown within the next six hours.

h. Prior to criticality, the listed Tc check valves shall be functional with leakage as follows:

(1) less than 1 gpm acceptable

(2) greater than 5 gpm unacceptable

(3) leak rate >1 gpm but <5 gpm are acceptable so long as the new leak rate does not reduce the margin to 5 gpm by ≥ 50% of the difference between the last leak rate and the present leak rate.

10. Tech Spec 3.1.D - Maximum Coolant Activity

a. Total specific activity due to nuclides with half-lives of greater than 15 minutes shall not exceed $100/\overline{E}$ µci/cc when critical or >500°F. If not met, shut down Rx and cool to <500°F within 6 hours. If exceeded limit by 25%, perform cooldown to 500°F within 2 hours.

b. Specific activity of RCS limited to ≤1.0 µci/cc Dose Equivalent I-131 whenever critical or >500°F

QUESTIONS REPORT
for SURRY2002

1. 062A2.04 001/T2G2/T2G2/EFFECT OF BUS LOSS/~~M~~ 3.4/3.1/N/SR02301/S/RLM

Plant conditions:

-Unit 1 Semi-Vital bus faulted 1300 1324

-Unit 1 tripped approximately ~~1 hour~~ ago during the downpower ~~required due to the faulted~~ ~~Semi-Vital bus~~

Which one of the following actions are required to maintain Tav at 547 °F during repair to the Semi-Vital bus?

A. Dump ~~steam~~ via the PORV's IAW 1-ES-0.1, Reactor Trip Response

B. Dump steam via the PORV's IAW 1-AP-10.05, Loss of Semi-Vital Bus

C. Dump steam via the steam dumps IAW 1-ES-0.1, Reactor Trip Response

D. Dump steam via the steam dumps IAW 1-AP-10.05, Loss of Semi-Vital Bus

Ref: ~~SR EB # 216~~
Surry Lesson Plan ND-90.3-LP-5, objective F
Note: After 1 hour, no power is available to either the steam dump or SG PORV controllers.
Only AP 10.05 provides guidance on local operation of SG PORV's

B. [Describe the components and indications associated with an Uninterruptable Power Supply (UPS). SOER 83-03, Recommendation 11]

C. Describe the power sources and loads associated with the Appendix R distribution system.

D. Describe the power sources and loads associated with the Semi-Vital Bus distribution system.

E. [Given a loss of a Vital or Semi-Vital bus, describe the actions taken IAW AP-10.01, 10.02, 10.03, 10.04, and/or 10.05 to address this loss. SOER 83-03, Recommendation 11 and SOER 81-02, Recommendation 5]

F. Given a loss of a Vital or Semi-Vital bus, describe the effect on Plant indications and controls, including actions taken IAW applicable APs to address the loss.

Presentation

Distribute all handouts. Refer to/display H/T-5.1, Objectives, and review with trainees.

A. One-Line Diagram

1. The purpose of the Vital Bus Distribution System is to supply a stable, reliable source of power to vital instruments. It must remain uninterrupted to prevent spurious shutdowns and guarantee proper action when instruments or controls are required.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- Power to the Steam Header Pressure Manual/Auto station has been lost, with the associated controller in Auto-Hold with fixed output demand.
- The Steam Dump system condenser interlocks will remain energized by an UPS in MB-8 for approximately 30 minutes. The Steam Dump system will continue to operate in Tave mode during this 30 minute period, after which the Steam Dumps will be unavailable due to MB-8 losing power.
- The SG PORVs will remain energized by an UPS in MB-8 for approximately 30 minutes. The PORVs will continue to control in automatic at 1035 psig, or may be operated manually during this 30 minute period.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	__VERIFY SEMI-VITAL BUS - NOT ELECTRICALLY FAULTED • Semi-Vital Bus lost as result of a loss of Emergency Bus OR • Electrical Department confirms Semi-Vital Bus NOT faulted	GO TO Step 11. WHEN fault corrected, THEN perform Step 9.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	__GO TO STEP 21	

NOTE: Breaker 15 on Unit 1 and Breaker 15 on Unit 2 Semi-Vital Bus should be opened before performing the following step.

11. __DIRECT THE ELECTRICAL DEPARTMENT TO SWAP THE GAI-TRONICS POWER SUPPLY TO UNIT 2 SEMI-VITAL BUS (JUNCTION BOX IN UNIT 1 ESGR)

* *

CAUTION: If the Semi-Vital Bus has been deenergized for greater than 30 minutes and a Reactor trip occurs, alternate steam release will be required to keep the Main Steam Safety valves from lifting.

* *

*12. __VERIFY REACTOR - NOT TRIPPED

RESPONSE NOT OBTAINED:

Maintain RCS temperature as necessary to prevent lifting Main Steam safety valves.

- Manually use SG PORVs if available

OR

- Use Steam Dumps if available

OR

- Locally use SG PORV(s) IAW Attachment 3, LOCAL OPERATION OF SG PORV(s).

__ 1. Consult with Shift Supervisor to determine which SG PORV(s) will be operated.

__ 2. Send Operator to Safeguards.

3. Close the isolation valve between the PORV positioner and actuator for the PORV(s) to be operated.

__ • 1-IA-1635A, 1-MS-RV-101A Positioner Isolation Valve

__ • 1-IA-1635B, 1-MS-RV-101B Positioner Isolation Valve

__ • 1-IA-1635C, 1-MS-RV-101C Positioner Isolation Valve

4. Open the bottled air supply valve for the PORV(s) to be operated.

__ • 1-IA-1638A, Bottled Air Supply for 1-MS-RV-101A

__ • 1-IA-1638B, Bottled Air Supply for 1-MS-RV-101B

__ • 1-IA-1638C, Bottled Air Supply for 1-MS-RV-101C

__ 5. Verify 1-IA-PCV-111 is backed-off. (no spring pressure)

6. Align air bottle to supply the SG PORV(s) by opening one of the following:

__ • 1-IA-1639, Air Bottled Manifold Isolation Valve

__ • 1-IA-1640, Air Bottled Manifold Isolation Valve

NOTE: • The SG PORV(s) will start to open when regulator output pressure is approximately six psig and will be fully open at approximately 30 psig.

• Close communication must be maintained between the MCR and Safeguards to control RCS cooldown rate.

• Vent valve 1-IA-1643, SG PORV Rapid Closure Vent Valve, may be opened as necessary for rapid closure of the SG PORVs.

__ 7. Adjust 1-IA-PCV-111, SG PORV Bottled Air System Pressure Regulator, to open SG PORV(s) for desired cooldown rate.

1. 062A2.04 001/T2G2/T2G2/LOSS OF BUS/M 3.4/3.1/B/SR02301/S/RLM

Procedures AP-10.02, AP-10.03, and AP-10.04 (Loss of Vital Bus II, III, or IV) direct the operator to trip the reactor prior to tripping the affected RCP.

Which ONE of the following is the basis for tripping the reactor before tripping the RCP?

A. To ensure SDM is present when backflow through the affected loop is initiated.

B. To ensure a cooldown rate is initiated in the affected loop.

C. To prevent an unnecessary challenge to the Reactor Protection System.

D. To prevent exceeding the linear heat generation rate limit.

Ref: SR EB # 216
Surry Lesson Plan ND-90.3-LP-5E; AP-10.02, AP-10.03, AP-10.04

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C B C A C C D D B Scramble Range: A - D

RO Tier: T2G2 SRO Tier: T2G2
Keyword: LOSS OF BUS Cog Level: M 3.4/3.1
Source: B Exam: SR02301
Test: S Misc: RLM

Replaced

B. [Describe the components and indications associated with an Uninterruptable Power Supply (UPS). SOER 83-03, Recommendation 11]

C. Describe the power sources and loads associated with the Appendix R distribution system.

D. Describe the power sources and loads associated with the Semi-Vital Bus distribution system.

E. [Given a loss of a Vital or Semi-Vital bus, describe the actions taken IAW AP-10.01, 10.02, 10.03, 10.04, and/or 10.05 to address this loss. SOER 83-03, Recommendation 11 and SOER 81-02, Recommendation 5]

F. Given a loss of a Vital or Semi-Vital bus, describe the effect on Plant indications and controls, including actions taken IAW applicable APs to address the loss.

Presentation

Distribute all handouts. Refer to/display H/T-5.1, Objectives, and review with trainees.

A. One-Line Diagram

1. The purpose of the Vital Bus Distribution System is to supply a stable, reliable source of power to vital instruments. It must remain uninterrupted to prevent spurious shutdowns and guarantee proper action when instruments or controls are required.

f. The principle plant effects, should vital bus 1 be lost, are the following:

(1) Loss of letdown

(2) Loss of CC to all RCP thermal barriers

(3) Loss of wide range loop "A" temperature

(4) Loss of S/G wide range level recorder

(5) Loss of AFW flow meter to "A" S/G

(6) Loss of HCV-FW-155A "A" S/G feed water bypass

(7) Loss of S/G blowdown

(8) Loss of Channel 1 NIs (SR, IR and PR)

4. AP-10.02, Loss of Vital Bus II

Ensure trainees have the latest revision of AP-10.02 to follow for this presentation. Perform a step-by-step discussion of this procedure highlighting applicable areas.

a. Initially a determination is made to see if VB 1-II or VB 1-IIA is lost.

b. If VB 1-II is lost, the reactor is tripped and "B" RCP is secured due to loss of CC to the RCP lube oil coolers. The team should initiate E-0 and continue with AP-10.02.

Ensure trainees have the latest revision of AP-10.03 to follow for this presentation. Perform a step-by-step discussion of this procedure highlighting applicable areas.

a. Initially a determination is made to see if VB 1-III or VB 1-IIIA is lost.

b. If VB 1-III is lost, the reactor is tripped and "A" RCP is secured due to loss of CC to the RCP lube oil coolers. The team should initiate E-0 and continue with AP-10.03.

c. Actions necessary to stabilize the plant for a loss of VB 1-III are listed in Attachment 1.

d. An attempt is made to re-energize the vital bus by pushing the alternate source to load button on the UPS or using the manual bypass switch.

e. The team must stop at this point until the vital bus is re-energized. After the bus is energized, the remainder of the procedure restores affected systems to pre-event conditions.

f. The principle plant effects, should vital bus 1-III be lost, are the following:

(1) Loss of CC to "A" RCP lube oil and stator coolers

(2) Loss of PR channel III (N-43)

(3) Loss of all main feed control bypass valve control

(4) Failure of steam dumps to control T_{ave} to required T_{ref}

(5) Loss of all steam generator feed regulating valve control (controllers are affected).

6. AP-10.04, Loss of Vital Bus IV

Ensure trainees have the latest revision of AP-10.04 to follow for this presentation. Perform a step-by-step discussion of this procedure highlighting applicable areas.

a. Initially a determination is made to see if VB 1-IV or VB 1-IVA is lost.

b. If VB 1-IV is lost, the reactor is tripped and "C" RCP is secured due to loss of CC to the RCP lube oil coolers. The team should initiate E-0 and continue with AP-10.04.

c. Actions necessary to stabilize the plant for a loss of VB 1-IV are listed in Attachment 1.

d. An attempt is made to re-energize the vital bus by pushing the alternate source to load button on the UPS or using the manual bypass switch.

e. The team must stop at this point until the vital bus is re-energized. After the bus is energized, the remainder of the procedure restores affected systems to pre-event conditions.

f. The principle plant effects, should vital bus 1-IV be lost, are the following:

(1) Loss of CC flow to "C" RCP stator and oil coolers

(2) Loss of PR channel IV (N-44)

(3) Loss of automatic pressure control of RCS

QUESTIONS REPORT

for Surry2002

#3

1. 065AA2.01 001/T1G3/T1G2/PRESSURE SWITCH/C/A 2.9/3.2/N/SR02301/S/RLM

-Unit 1 is at 100%
-Instrument Air Compressors are in AUTO and are not running.
-Annunciators 1B-E6, IA LO HEADER PRESS/1A COMPR 1 TRBL and 1B-G5, INST AIR DRYER TRBL illuminate.
-The RO reports that service air pressure appears steady at approximately 100 psig.
-The AO reports that the Instrument Air Dryer bypass valves have opened and the Instrument Air Compressors are NOT running.

Which ONE of the following is the reason Annunciator 1B-G5 alarmed?

A. The air dryer bypass trip valves opened.
B. The lag Service Air Compressor auto started.
C.✓ Failure of the Instrument Air Header pressure switch
D. The Instrument Air Compressor auto start pressure switch failed to actuate.

Ref: Surry Lesson Plans: ND-92.1-LP-1, obj E and ND-95.1-LP-9
ARP's 1B-E6 and 1B-G5
Answer A is incorrect because the bypass valve opens as a result of the PS failure and cannot account for the 1B-E6 alarm (ie does not feed that alarm)
Answer B is incorrect because auto start of the lag compressor is not an input to either alarm (input to 1B-E5)
Answer C is correct because both alarms actuate in response to an 80 psig PS (see note)
Answer D is incorrect because instrument air pressure is normal an therefore has not actuated the 90 psig switch that auto starts the IA compressors.

Note: This question is based on the assumption that 1-IA-PS-120 feeds both alarms. This needs to be verified. Current info insufficient.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C A A B D A C A C C Scramble Range: A - D

RO Tier:	T1G3	SRO Tier:	T1G2
Keyword:	PRESSURE SWITCH	Cog Level:	C/A 2.9/3.2
Source:	N	Exam:	SR02301
Test:	S	Misc:	RLM

SR Exam bank question 356 is a possible substitute if this question has technical problems too difficult to fix.

LESSON PLAN

Introduction

The Station Air Systems supply compressed air to operate tools, valves and components throughout the station. This lesson describes the systems, including flowpaths, system components, instrumentation and control, and annunciators or alarms associated with these systems.

Objectives

After receiving this instruction, the trainee will be able to:

A. Describe the system flowpaths and components associated with the Service Air System.

B. [Describe the system flowpaths and components associated with the Instrument Air System. SOER 88-01, Recommendation 2&3]

C. Describe the flowpaths and components associated with the Polishing Building Air System.

D. Describe the flowpaths and components associated with the Containment Instrument Air System.

E. Describe the flowpaths, components, indications, and controls associated with the Station Air Systems.

STATION AIR SYSTEM ALARMS

Service Air Compressor 1 Trouble (1B-E5)

- Compressor motor overload
- High oil temperature – 176°F
- Low oil pressure - 20 psig (22 sec TD) - byp 15 secs on a start for compr to reach operating speed.
- L.P. stage outlet high air temperature – 425°F
- H.P. stage outlet high air temperature – 425°F
- High intercooler air temperature – 190°F
- Loss of power
- Emergency backup running (lag compressor start): If 1-SA-C-1 in LEAD and 2-SA-C-1 auto starts in LAG - alarm received on Unit 1 (similar on Unit 2).

Instrument Air Low Header Pressure/Compressor 1 Trouble (BE6)

- Bearing cooling water outlet temperature high (140 degrees).
- Discharge air high temperature (444 degrees).
- Low lube oil pressure (5 psig) with a compressor run signal present.
- Emergency back-up running.
- Failure to continue running after initiation of start signal, due to low lube oil pressure (8 psig) at end of seven (7) second timing period.
- Compressor motor thermal overload.
- Loss of power.
- Low instrument air header pressure - 80 psig.

Instrument air dryer trouble: (BG5)

- Loss of power - causes auto bypass of dryer; dryer continues normal operation until battery depleted (~4.5 hours), then both chambers go into service.
- Chamber performance degrading/LC or RC AMLOC Failure
- Valve malf (U1), LC/RC REPRESS/DEPRESS FAILURE, ONLINE PRESSURE (U2)
- Low instrument air header bypass
- ARP indicates that Lo IA air pressure also gives alarm

(e) Filtration of the instrument air is provided by replaceable coalescing prefilter located upstream of the dryers as well as particulate after-filters positioned downstream. The prefilter removes liquid aerosols of water and oil and have a particulate performance rating of 100% of 0.6 microns and larger. The accumulated liquids are drained from the pre-filter housing through an automatic drain valve. The after-filters have a particulate performance rating of .9 microns absolute.

3. Instrumentation and controls

a. Instrument Air compressors

(1) The controls are three position switches, hand-off-auto. In HAND, the compressor motor runs continuously, the compressor loads and unloads at 100 and 110 psig, respectively. In AUTO, the compressor starts if pressure reaches 90 psig; load and unload setpoints are the same.

(2) Following an auto start, the auto start mercoid switch must be reset manually; it will not reset itself on high pressure.

(3) On any start signal, interlocks must be satisfied to start the compressor as follows:

- No motor overload.
- Discharge air temperature less than 444°F.
- Cooling water temperature less than 140°F.

ANNUNCIATOR RESPONSE PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1B-E6	IA LOW HDR PRESS/IA COMPR 1 TRBL	9
		PAGE
		1 of 7

REFERENCES 1B-38

1. UFSAR - Sections 9.8.2 and 10.3.9.3
2. 11448-ESK-6DA, 10B, 10AE
3. EWRs 89-329, 89-547, EWR 89-557B
4. 1-DRP-005, Instrumentation Setpoints
5. DCP-86-03-C, IA DRYER MODIFICATION (Step 10)

PROBABLE CAUSE

1. 1-IA-PS-120 senses IA header pressure less than or equal to 80 PSIG.

   Low header pressure may be caused by one or more of the following:

   - Compressor failure
   - Line rupture
   - Excessive demand
   - Dryer control malfunction

2. Local annunciator Panel 01-IA-ANN-PNL receives a trouble signal from one or more of the following:

   - Low oil PRESS less than or equal to 5 PSIG
   - Motor overload
   - EMERG backup IA compressor running
   - High DISCH air TEMP greater than 444 °F
   - High cooling water outlet TEMP greater than 140°F

3. Instrumentation failure has occurred.

APPROVAL RECOMMENDED		DATE
W.R. Benthall	W.B. Hort	02/25/00
REVIEWED		
[signature]		

ANNUNCIATOR RESPONSE PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1B-G5	INST AIR DRYER TRBL	3
		PAGE 1 of 4

1B-53

REFERENCES

1. UFSAR 9.8
2. 11448-FM-075A
3. 11548-FE-18AW
4. 11448-ESK-10B, 10AX
5. DCP-86-03A-3
6. DCP-86-03C-3
7. PAR 93-0445
8. DR S-98-1572

PROBABLE CAUSE

1. Alarm actuates when one or more of the following conditions exist:
   a. Instrument Air dryer discharge pressure less than or equal to 80 PSIG.
   b. Loss of power to dryer.
   c. Dryer bed too wet. (Chamber performance degrading)
   d. Moisture probe cable disconnected.
   e. Exhaust valve malfunction.
   f. Inlet valve malfunction.
   g. Inlet isolation trip valve (1-IA-TV-125) closed.
   h. Bypass trip valve (1-IA-TV-126) open.
2. Instrumentation failure has occurred.

APPROVAL RECOMMENDED	APPROVED	DATE
W. R. [illegible]	[illegible]	4/6/99
REVIEWED Neil Turner [illegible]		

1. 103A2.05 001

-Unit 2 is making preperations for a reactor startup.
-An RCP low oil level is recieved.
-An Entry into containment is required to add oil to the RCP.

Which one of the following describes what requirements must be met to allow entry?

A. An SCBA with 19% to 23% oxygen by volume, a confined space entry permit, and permission of the Station Manager.

B. An SCBA with 19% to 23% oxygen by volume, a VPAP-0106 attachment 1, and permission of the HP Supervisor.

C. An SCBA with 33% to 37% oxygen by volume, a confined space entry permit, and permission of the Operations Manager.

D. An SCBA with 33% to 37% oxygen by volume, a VPAP-0106 attachment 1, and permission of the Site Vice President.

VPAP-0106 Subatmospheric Containment Entry.
Surry Lesson Plan ND-88.4-LP-7 Objective E.

A. Incorrect, SCBA must have 33 to 37 % oxygen by volume, and no confined space entry permit is required.

B. Incorrect, SCBA must have 33 to 37 % oxygen by volume, and permission cannot be granted by the HP Supervisor.

C. Incorrect, A confined space entry permit is not required and the ops manager cannot grant permission.

D. Correct, An SCBA with 33 to 37% oxygen by volume, a VPAP-0106 attachment 1 and permission of the Site Vice President allows entry.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: Scramble Range: A - D

RO Tier:	T2G3	SRO Tier:	T2G2
Keyword:		Cog Level:	C/A 2.9/3.9
Source:	N	Exam:	SR02301
Test:	S	Misc:	GWL

6.0 INSTRUCTIONS

6.1 Subatmospheric Containment Entry Hazards

6.1.1 Subatmospheric containment entry will expose Containment Entry Team members to four distinct hazards:

- Ionizing radiation
- Heat stress
- Differential pressure
- Oxygen deficiency due to subatmospheric pressure

6.1.2 Personnel Air-Lock entry and exit may cause personnel discomfort due to the air pressure changes. Personnel experiencing discomfort during pressure changes should notify the Containment Entry Team Leader to prevent severe pain and potential damage to the ear.

6.1.3 Containment entries shall not be made if the containment pressure is less than 9.0 psia.

6.1.4 If containment pressure is greater than or equal to 9.0 psia and less than 12.0 psia, SCBA with 33 to 37 percent oxygen by volume shall be used.

6.1.5 If required to change SCBA cylinders inside containment, brief exposures to 9.0 psia to 12.0 psia containment atmosphere, without enriched oxygen breathing gas mixture, is permissible provided there are no radiological concerns listed on the RWP.

6.1.6 A subatmospheric containment meets the conditions for being a Confined Space (non permit required) as defined in 29 CFR 1910.146. All of the applicable requirements for entry into a non permit-required confined space are met or exceeded by this VPAP, therefore this VPAP shall be used in lieu of the Confined Space Entry Program. Attachment 1 shall be completed for each containment entry except in cases of emergencies. Confined Space requirements for equipment within containment apply as required by the Confined Space Entry Program.

Containment Entry Checklist

VIRGINIA POWER

VPAP-0106 - Attachment 1

Part 1 - Completed by Responsible Supervisor

☐ Unit 1 ☐ Unit 2	Date	Estimated Time of Entry	Radiation Work Permit (RWP) Number

List personnel designated for Containment Entry Team

Note: A Containment Entry Team minimum composition is two and maximum composition is fifteen people.

Name (Please Print)	Signature	Badge Number	Containment Entry Training Satisfactorily Completed
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No
			☐ Yes ☐ No

Containment Entry Team Leader (Name - Please Print)

Permission granted by Site Vice President or Station Manager (Name - Please Print)

If any Containment Entry Team Member is not Trained, List Reason Why and Designate Escort

R[illegible]on for Entry and Work to be Performed

Responsible Supervisor (Signature)	Date

Form No. 722650(Dec 99)

64 **ID: ADM0109** **Points: 1.00**

Which ONE of the following individuals by title is the MINIMUM authorization that must be obtained before containment entry during subatmospheric conditions?

A. Shift Supervisor

B. Site Vice President

C. Immediate Supervisor

D. HP Supervisor

Answer: B

Question 64 Details

Question Type:	Multiple Choice
Topic:	ADM0109
System ID:	72335
User ID:	ADM0109
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-88.4-LP-7E; SROUTP-SDS-2/P; VPAP-0106 [S99-0136], [S97-0830], [S95-0039]

QUESTIONS REPORT

for Surry2002

1. G2.1.4 001/T3/T3/STAFFING/M 2.5/3.3/B/SR02301/S/RLM

The following plant conditions exist:

-Unit 1 is in HOT SHUTDOWN.
-Unit 2 is in COLD SHUTDOWN.

Which ONE of the following is the MINIMUM Shift Manning requirement for the Station under the conditions shown above per Tech Spec 6.1, Table 6-1-1, "Minimum Shift Crew Composition"?

	SSU	SRO	RO	AOs	STAs
A.	1	1	2	4	0
B.	1	1	3	4	1
C.	1	0	3	3	1
D.	1	0	2	4	1

Ref: SR EB # TS00126

ND-88.1-LP-9, obj. F

Tech Spec 6.1, Table 6-1-1, "Minimum Shift Crew Composition"

RO Tier:	T3	SRO Tier:	T3
Keyword:	STAFFING	Cog Level:	M 2.5/3.3
Source:	B	Exam:	SR02301
Test:	S	Misc:	RLM

Test Name	Test Date	rpb	p(Diff)	Time	Equ	User Values	
<Cumulative>		0.000	0.000	0	N	1: 0	2: 0
						3: 0	4: 0

	--- A ---			--- B ---			--- C ---			--- D ---					
	Resp	%	p	Resp	%	p	Resp	%	p	Resp	%	p	Resp	%	p
<Cumulative>						Total:	0	100				Omits:	0	0	
	0	-1	0.00	0	-1	0.00	0	-1	0.00	0	-1	0.00			

LESSON PLAN

Introduction

10CFR50.36 requires applicants for a nuclear facility operating license to submit and comply with Technical Specifications. These specifications are derived from the analyses and evolutions included in the Final Safety Analysis Reports. Since these Tech Specs are required by law and are approved by the NRC, they are a legal document. This lesson will provide a general description of each section of Tech Specs and a detailed description of the RCS Tech Specs. The knowledge gained from this lesson will provide an understanding of the content and layout of the Technical Specifications.

Objectives

After receiving this instruction, the trainee will be able to:

A. Summarize the purpose of Tech Spec Section 1.0 including the definition of applicable terms in this section.

B. Summarize the purpose of Tech Spec Section 2.0.

C. Summarize the purpose of Tech Spec Section 3.0.

D. Summarize the purpose of Tech Spec Section 4.0.

E. Summarize the purpose of Tech Spec Section 5.0.

F. Summarize the purpose of Tech Spec Section 6.0.

G. Describe the purpose and specification for the Safety Limits IAW section 2 of Tech Specs including for SRO candidates, the basis behind these specifications.

TABLE 6.1-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	ONE UNIT OPERATING	TWO UNITS OPERATING	TWO UNITS IN COLD SHUTDOWN OR REFUELING
SS	1	1	1
SRO	1	1	None
RO	3	3	2
AO	4	4	4
STA	1	1	None

TABLE 6.1-1 (Continued)

SS - Shift Supervisor with a Senior Reactor Operators License.
SRO - Individual with a Senior Reactor Operators License.
RO - Individual with a Reactor Operators License.
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.1-1 for a period of time not to exceed 2 hours in order to accomodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.1-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in operation, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is shutdown or refueling, an individual with a valid RO license (other than the Shift Technical Advisor) shall be designated to assume the Control Room command functions.

QUESTIONS REPORT
for Surry2002

1. G2.1.34 001/T3/T3/PRIMARY CHEMISTRY/M 2.3/2.9/B/SR02301/S/RLM

A-6

Think up new Question

The following plant conditions exist:

- The reactor has been at 100% power for 30 days.
- Chemistry reports that RCS activity is 1.5μCi/cc DOSE EQUIVALENT I-131.

WHICH ONE (1) of the following actions will reduce RCS activity?

A. Vent the Volume Control Tank (VCT) to the Waste Gas System.

B. Place cation demineralizer in service and maximize letdown.

C. Maximize letdown through the mixed bed demineralizer.

D. Reduce letdown to minimum and establish an RCS ph between 6.5 and 7.5 by chemical injection.

Ref: SM EB # 41617

Note: Need procedure that specifies the actions that the operating crew would take.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C B D D C A A D A D	Scramble Range: A - D

RO Tier:	T3	SRO Tier:	T3
Keyword:	PRIMARY CHEMISTRY	Cog Level:	M 2.3/2.9
Source:	B	Exam:	SR02301
Test:	S	Misc:	RLM

Letdown Rad Monitor AR

CPS Does

*QNUM 41617
*HNUM 42864 (Do NOT change If < 9,000,000)
*ANUM 41627
*QCHANGED FALSE
*ACHANGED FALSE
*QDATE 1992/12/07
*FAC 395 V. C. Summer 1
*RTYP PWR-WEC3
*EXLEVEL S
*EXMNR
*QVAL
*SEC
*SUBSORT
*KA 000076K305
*QUESTION
The following plant conditions exist:

- The reactor has been at 100% power for 30 days.
- Chemistry reports that RCS activity is 1.5 microCuries per gram DOSE EQUIVALENT I-131.

WHICH ONE (1) of the following actions will reduce RCS activity?

a. Vent the Volume Control Tank (VCT) to the Waste Gas System.

b. Place cation demineralizer in service and maximize letdown.

c. Maximize letdown through the mixed bed demineralizer.

d. Reduce letdown to minimum and establish an RCS ph between 6.5 and 7.5 by chemical injection.

*ANSWER
c. (+1.0)

*REFERENCE
1. VCS: GS-6, Primary Chemistry and Sampling, Objectives 6, 9, 10, p.22, 29, 30
2. VCS: CR-2, Plant Chemistry Control, Objectives, 7, 12, p. 17
3. VCS: Technical Specifications 3.4.8
4. KA 000076K305 (2.9/3.6)

LESSON PLAN

Introduction

Primary chemistry limits provide for the safety and health of the public if an accident should occur at the plant. They also ensure the integrity of primary materials by minimizing corrosion. Since the licensed operator is responsible for plant performance, he/she should be able to recognize these limits and realize what processes these specifications are limiting. Fuel integrity is a major concern to the licensed operator. He/she should be able to use primary isotopic concentrations to determine if a change in fuel integrity has occurred. Operators are responsible for primary chemical additions and should know the purpose of these additions. Besides the chemistry limits themselves, this lesson plan will also include a discussion of the rationale behind them.

Objectives

After receiving this instruction, the trainee will be able to:

A. Explain the primary coolant reactions and chemical controls.

B. Describe the Tech Spec limits for RCS chemistry control.

C. Explain Surry Power Stations's technical specification and recommended primary chemistry limits.

7/17

QUESTIONS REPORT
for Surry2002

1. G2.2.6 001/T3/T3/PROCEDURE CHANGE/M 2.3/3.3/N/SR02301/S/RLM

Which ONE of the following are required approval authorities for a change to a telephone area code in EPIP-2.01, NOTIFICATION OF STATE AND LOCAL GOVERENMENTS?

A. The Director of Nuclear Security and Emergency Preparedness and the Shift Supervisor

B. The Shift Supervisor and the Site Vice President

C. The Director of Nuclear Security and Emergency Preparedness and the SNSOC

D. The SNSOC and the Site Vice President

Ref: VPAP-0502, Procedure Process Control, p.85&86

No lesson plan or learning objective.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D D B A A B D D C B		Scramble Range: A - D
RO Tier:	T3			SRO Tier:	T3	
Keyword:	PROCEDURE CHANGE			Cog Level:	M 2.3/3.3	
Source:	N			Exam:	SR02301	
Test:	S			Misc:	RLM	

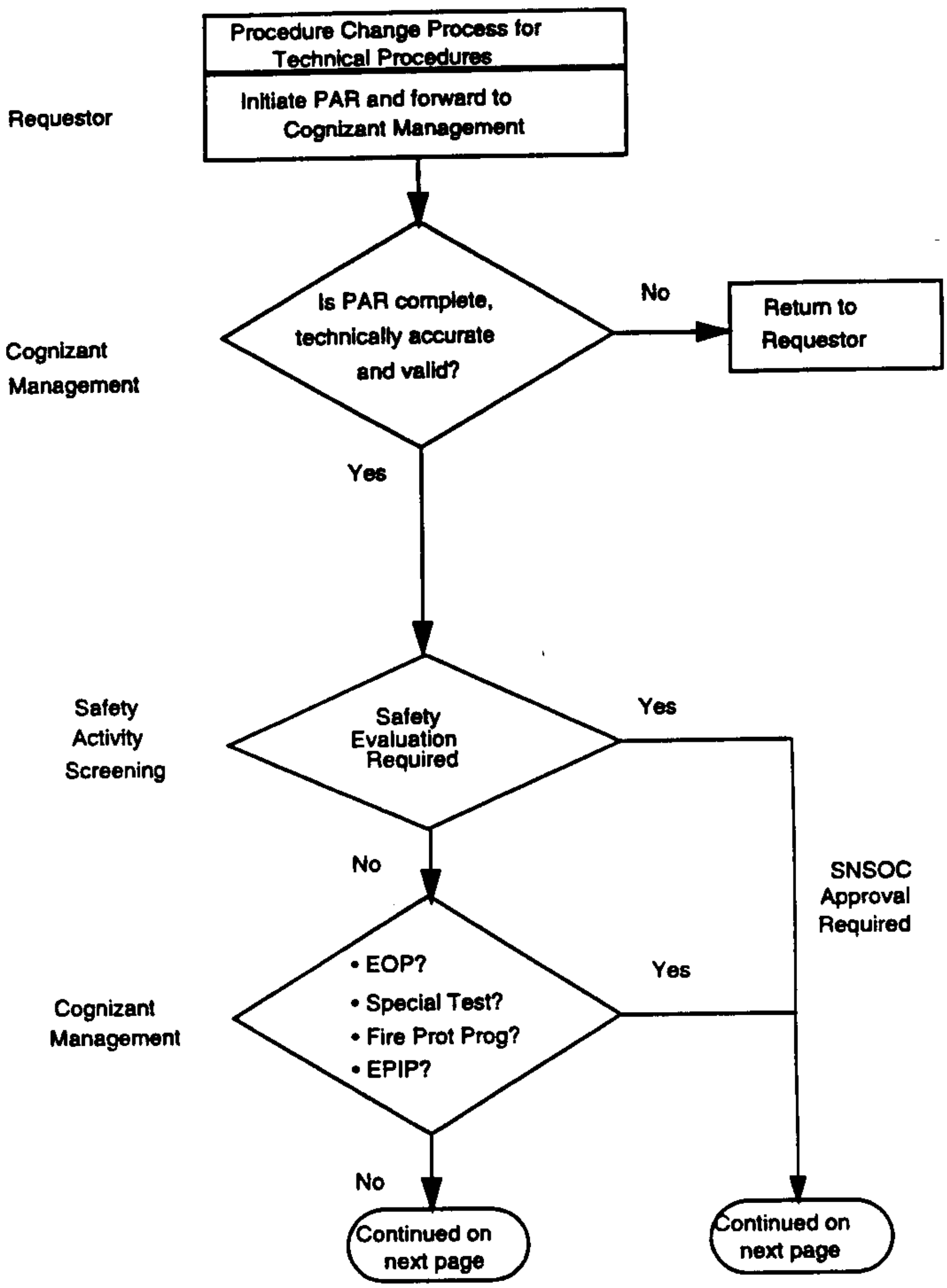
Look at Question

They didn't change Procedures

ATTACHMENT 1

(Page 3 of 4)

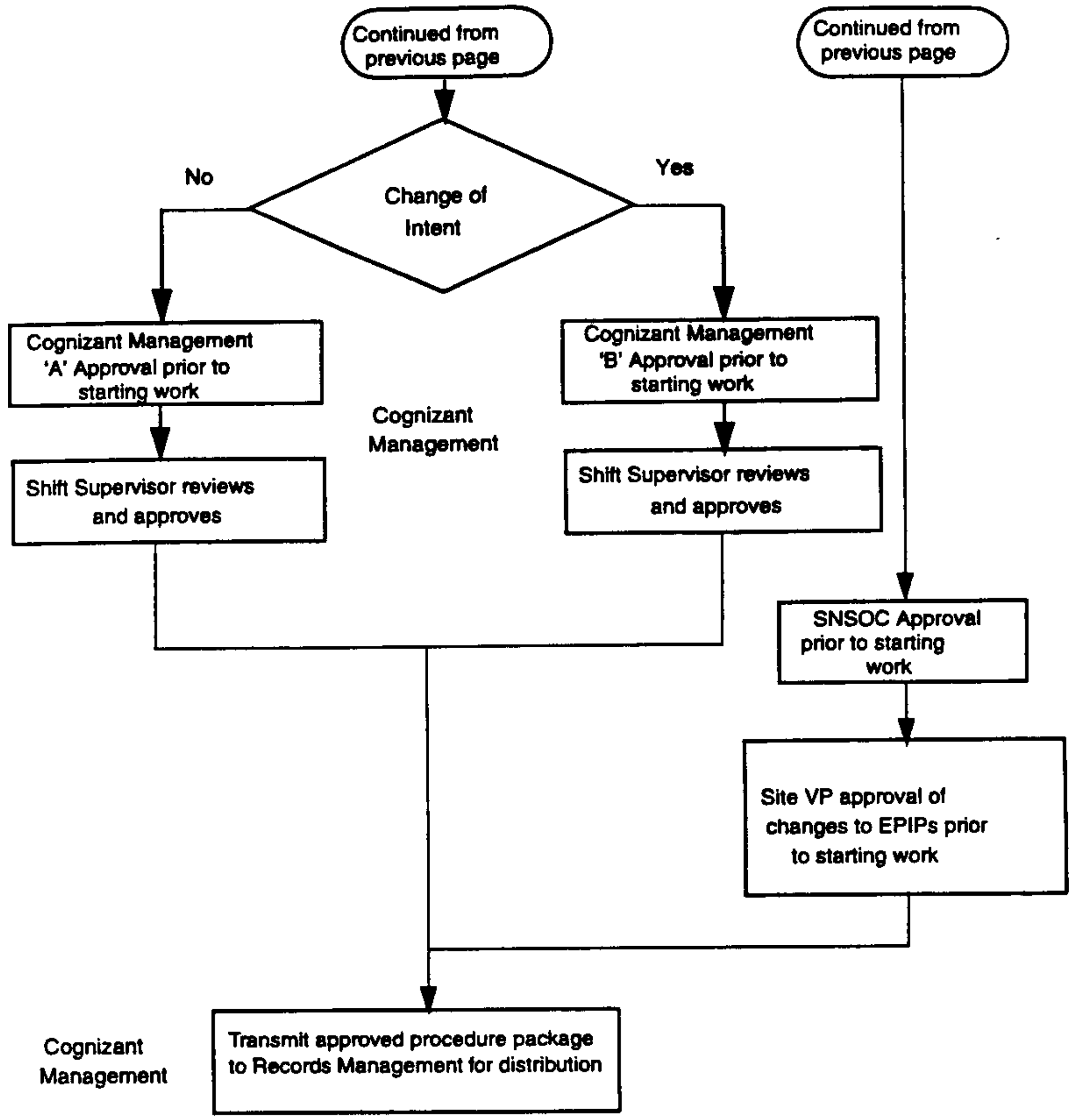
Procedure Process Flow Chart



ATTACHMENT 1

(Page 4 of 4)

Procedure Process Flow Chart



#18

QUESTIONS REPORT

for Surry2002

1. G2.3.10 001/T3/T3/TEMP SHIELDING/M 2.1/3.1/B/SR02301/S/RLM

Which ONE of the following describes the Shift Supervisors responsibilities IAW VPAP-2105, Temporary Shielding, concerning the Temporary Shielding request form?

A. The Shift Supervisor must acknowledge installation and removal of temporary shielding.
B. The Shift Supervisor must acknowledge installation and approve removal of temporary shielding.
C. The Shift Supervisor must approve installation and removal of temporary shielding.
D.✓ The Shift Supervisor must approve installation and acknowledge the removal of temporary shielding.

Ref: SR EB # ADM0174

VPAP-2105

No specific learning objective found

Note: Reference document not included in materials received.

RO Tier:	T3	SRO Tier:	T3
Keyword:	TEMP SHIELDING	Cog Level:	M 2.1/3.1
Source:	B	Exam:	SR02301
Test:	S	Misc:	RLM

Thinks it too hard.

19

1. WE02G2.4.6 001

-Unit 1 has had a Reactor Trip and SI. In E-1 graded

-Subcooling on CETC's 45^{o} F.

-RCS pressure is stable at 1530 psig.

-Containment Pressure is 6psig.

-Pressurizer level is 100 %.

-AFW flows (to each S/G): 120gpm, 120 gpm, 100 gpm.

Steam Generator narrow range levels: 11%, 15%, 8%.

Which one of the following is the appropriate status concerning SI Termination Criteria?

A. SI Termination Criteria will be met if AFW flow is adjusted to greater than 350 gpm.

B. SI Termination Criteria are NOT met due to RCS subcooling, continue ECCS pumps running.

C. SI Termination Criteria is met, Transition should be made to ES-1.1 "SI Termination".

D. SI Termination Criteria is NOT met; due to Pressurizer level being high ECCS pumps should be reduced.

Surry Lesson Plan; ND-95.3-LP8 objective A.

A. Incorrect, SI termination criteria would still not be met.

B. Correct, SI termination criteria is not met. ECCS should remain running.

C. Incorrect, SI termination criteria is not met.

D. Incorrect, SI termination criteria is not met, but ECCS should remain running even though Pressurizer level is high.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B D C A A D A D D A	Scramble Range: A - D
RO Tier:	T1G2		SRO Tier:	T1G1	
Keyword:			Cog Level:	C/A 4.0/4.0	
Source:	M		Exam:	SR02301	
Test:	S		Misc:	GWL	

LESSON PLAN

Introduction

This lesson plan will provide classroom training for ES-1.1, SI Termination. The material will be presented first as an overall "big picture" of the procedure which will then be followed up by an in-depth presentation of the step backgrounds and required knowledges of the procedure. Shortly after the classroom presentation, the simulator will be used to reinforce this material and allow practice of the techniques incorporated into ES-1.1.

In its entirety, ES-1.1 provides the necessary instructions to terminate SI and stabilize plant conditions.

It is entered from E-0, Reactor Trip or Safety Injection, from E-1, Loss of Reactor or Secondary Coolant, or FR-H.1, Loss of secondary Heat Sink, when the SI termination criteria are satisfied. The goal of ES-1.1 is to stop SI pumps in a prescribed sequence while maintaining control of the RCS, until makeup is by charging flow alone. Following termination of SI, the operator will exit to normal procedures for either startup or cooldown.

Objectives

After receiving this instruction, the trainee will be able to:

A. Given the major action categories associated with ES-1.1, SI Termination, explain the purpose of ES-1.1, the transition criteria for entering and exiting ES-1.1 and the types of operator actions that will occur within each category.

B. Given a copy of ES-1.1, SI Termination, explain the basis of each step of the procedure.

*QNUM 43875
*HNUM 45211 (Do NOT change If < 9,000,000)
*ANUM
*QCHANGED FALSE
*ACHANGED FALSE
*QDATE 1995/04/17
*FAC 456 Braidwood 1 & 2
*RTYP PWR-WEC4
*EXLEVEL S
*EXMNR
*QVAL
*SEC
*SUBSORT
*KA 000009A204
*QUESTION
The following conditions exist on Unit 1 following a reactor trip and SI:

- Wide range RCS pressure is 1375 psig and stable.
- Average of 10 highest core-exit TCs is 565 degrees F.
- The Subcooled Margin Monitor (SMM) Iconics display indicates a red 17.
- Pressurizer level is 10% and increasing.
- Containment radiation is 2 Rem/hr
- Containment pressure is 4 psig.
- SG narrow range levels: 2%, 8%, 5%, 2%.
- AFW flows (to each SG): 125 gpm, 125 gpm, 125 gpm, 100 gpm.

The Unit Supervisor is trying to determine if ECCS flow should be reduced per step 6 of BwEP-1 "Loss of Reactor or Secondary Coolant".

WHICH ONE of the following is the appropriate status concerning SI termination criteria?

(Attached Figure 1BwEP ES 1.1-1 may be used for reference.)

a. SI termination criteria are met and transition should be made to BwEP ES-1.1 "SI Termination".

b. SI termination criteria are NOT met due to pressurizer level being low, continue ECCS pumps running.

c. SI termination criteria are NOT met due to RCS subcooling margin being low, continue ECCS pumps running.

d. SI termination criteria are met if total AFW flow is adjusted to 500 gpm.

*ANSWER

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6.	__CHECK IF SI FLOW SHOULD BE REDUCED:	
	a) RCS subcooling based on CETCs - GREATER THAN 30°F [85°F]	a) GO TO Step 7.
	b) Secondary heat sink:	b) GO TO Step 7.
	• Total feed flow to INTACT SGs - GREATER THAN 350 GPM [450 GPM]	
	OR	
	• Narrow range level in at least one intact SG - GREATER THAN 11% [22%]	
	c) RCS pressure - STABLE OR INCREASING	c) GO TO Step 7.
	d) PRZR level - GREATER THAN 22% [43%]	d) Try to stabilize RCS pressure with normal PRZR spray. GO TO Step 7.
	e) GO TO 1-ES-1.1, SI TERMINATION	
* 7.	__CHECK IF HI HI CLS INITIATED:	GO TO Step 13.
	• RS pump(s) - RUNNING	
	OR	
	• Any Hi Hi CLS annunciator - LIT	

QUESTIONS REPORT
for Surry2002

1. G2.4.14 001

While in the Emergency Respose procedures the team is directed to "Go To" another procedure.

Which one of the following is correct way to implement this direction?

A. The "GO TO" implies that the procedure in use is no longer applicable, and any tasks that were in progress need not be completed.

B. Tasks still in progress must be completed prior to the transition directed by the "GO TO" step.

C. The "GO TO" implies that the procedure in use is no longer applicable, but any tasks that were in progress and should completed.

D. Tasks still in progress need not be completed prior to the transition directed by the "GO TO" step, unless preceeded by a bullet.

Surry Lesson Plan ND-95.3-LP-2 objectives # Dand F.

A. Incorrect, The tasks should be completed.

B. Incorrect, Tasks in progrees do not have to be completed prior to the transition.

C. Correct, The previous procedure is nolonger applicable and the tasks that were in progress should be completed.

D. Incorrect. Tasks in progress need not be completed prior to the transition, a bulleted step can be performed in any order, and does not have to be performed prior to transition.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9

Answer: Scramble Range: A - D

RO Tier:	T3	SRO Tier:	T3
Keyword:		Cog Level:	M 3.3/3.9
Source:	N	Exam:	SR02301
Test:	C	Misc:	GWL

B. Explain the two-column format of the Emergency Response Guideline Procedures, including the placement criteria for cautions and notes.

C. Explain the method by which "Immediate Operator Action" steps are identified in the body of the ERG Procedures.

D. Describe the intended overall usage of the Emergency Response Guidelines Network.

E. Given various plant conditions during which an emergency event occurs, evaluate the application of the "Modes of Applicability" as described in the ERG User's Guide.

F. Given actual or simulated EOP implementation, apply the management standards and other good practices applicable to EOP usage.

G. Explain the format design of the Emergency Response Guideline Procedures.

Presentation

Distribute all handouts. Refer to/display H/T-2.1, Objectives, and review objectives with trainees

A. Action Verb Identification

Direct trainees to turn to AIA-2.1, Action Verbs. Review various action verbs with trainees.

b. If a particular task MUST BE COMPLETED prior to proceeding, the step containing the task or an associated NOTE will explicitly state that requirement.

11. Transitions to other procedures or to different steps in the same guideline may be made from either column. Such transitions should be made realizing that preceding NOTES or CAUTIONS are applicable.

    a. Any tasks still in progress need not be completed prior to making a transition; however, the requirement to complete the tasks is still present and must not be neglected.

    b. A transitional "GO TO..." to some other procedure implies that the procedure in use is now no longer applicable and the procedure referred to is now in effect.

EXAMINATION ANSWER KEY

RO/SRO Exam Bank

839 **ID: EOP0088** **Points: 1.00**

Which ONE of the following indicates when substeps of an Emergency Operating Procedure must be performed in order?

A. Substeps designated by numbers only.

B. Substeps designated by bullets.

C. Substeps designated by asterisks.

D. Substeps designated by letters or numbers.

Answer: D

Question 839 Details

Question Type:	Multiple Choice
Topic:	EOP0088
System ID:	73378
User ID:	EOP0088
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-95.3-LP-2B; OPAP-0002
	[S95-1096]

SUBSTEPS Designated by letters or Numbers 1, A.
MUST BE Performed In ORDER.
Immediate Action Steps always have a Bracket. [ ]
Asterik - Continuing action Steps
any Sequence Bullets

Question 813 Details

Question Type:	Multiple Choice
Topic:	EOP0020
System ID:	73317
User ID:	EOP0020
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-95.3-LP-38C and D; ND-95.4-LP-3A, B, and D; FR-C.1 [S97-0047], [S96-1021], [S96-1350]

814 **ID: EOP0022** **Points: 1.00**

Which ONE of the following action steps must be performed in sequence in accordance with the rules for Emergency Operating Procedure (EOP) usage?

A. All immediate action steps of E-0, Reactor Trip or Safety Injection, and FR-S.1, Response to Nuclear Power Generation/ATWS.

B. All immediate action steps for ECA-0.0, Loss of All AC Power, and FR-S.1, Response to Nuclear Power Generation/ATWS.

C. All immediate action steps for E-0, Reactor Trip or Safety Injection and ES-0.1, Reactor Trip Response.

D. All immediate action steps of E-0, Reactor Trip or Safety Injection, and ECA-0.0, Loss of All AC Power.

Answer: A

1. WE03G2.4.6 001

-Unit 1 has experienced a SBLOCA.
-ES-1.2, Post LOCA Cooldown and Depressurization is in progress.
-Three RCPs are running.
-An RCS cooldown to place RHR on service has been initiated by dumping steam to the atmosphere.

Which one of the following describes the optimum RCP configuration, and the basis for this configuration?

A. One RCP should be secured to produce effective heat transfer, provide boron mixing for RHR operations, and provide RCS pressure control.

B. All RCPs should be stopped to minimize RCS inventory loss when the break uncovers.

C. Two RCPs should be secured to minimize RCS heat input, and still produce effective heat transfer and RCS pressure control.

D. Three RCPs should be left running to ensure symeteric heat transfer to the S/Gs, to aid in RCS pressure control, and prevent steam voiding in the Reactor vessel head.

Surry Exam Bank, question # 3384 slightly modified.
Surry Lesson Plan ND-95.3-LP-9 Objective B.

A. Incorrect, Two RCPs should be secured. Mixing for placing RHR on service is not a reason for running RCPs.

B. Incorrect, The procedure directs the operator to leave an RCP operation if possible.

C. Correct only one pump should be left running to minimize heat input, control RCS pressure and provide effective heat transfer.

D. Incorrect, The procedure directs only one RCP to be left running.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: Scramble Range: A - D

RO Tier:	T1G2	SRO Tier:	T1G2
Keyword:		Cog Level:	C/A 4.0/4.3
Source:	B	Exam:	SR02301
Test:	S	Misc:	GWL

OBJECTIVES

After receiving this instruction, the trainee will be able to:

A. Given the major action categories associated with ES-1.2, Post-LOCA Cooldown and Depressurization, explain the purpose of ES-1.2, the transition criteria for entering and exiting ES-1.2 and the types of operator actions that will occur within each category.

B. Given a copy of ES-1.2, Post-LOCA Cooldown and Depressurization, explain the basis of each procedural step.

C. Given actual or simulated plant conditions requiring ES-1.2, Post-LOCA Cooldown and Depressurization, implementation, successfully transition through the procedure, applying step background knowledge as required, to safely bring the plant to a cold shutdown condition.

e. The value of pressurizer level chosen for this step is that indication with water level just above the top of the heaters, including allowance for normal channel accuracy and reference leg heating. This value is used to verify that sufficient liquid is present to allow operation of the pzr heaters.

f. **It is not critical to maintain level at 35% [55%]. In many cases, the level (and pressure) will increase after the depressurization is stopped until injection flow balances break flow and loss due to cooldown shrink. (rk)**

18. **STEP 13: CHECK IF AN RCP SHOULD BE STARTED.**

a. The purpose of this step is to establish forced circulation flow in the RCS from one RCP.

b. Forced flow is the preferred mode of operation to allow for normal RCS cooldown and provide pzr spray.

c. If RCPs had not been tripped, all but one are stopped to minimize heat input to the RCS.

(1) The RCP started or left running should be the one that can provide normal pzr spray.

(2) The normal spray valve associated with any stopped RCP should be closed. This maximizes spray flow from the active loop by preventing backflow through the spray lines of inactive loops.

d. With no RCP running, depressurization of the RCS may generate a steam bubble in the upper head. This bubble could rapidly condense during pump startup, drawing liquid from the pzr and reducing RCS subcooling. If pzr

3384 **ID: EOP0338** **Points: 1.00**

Given the following plant conditions:

- A SBLOCA has occurred.
- The team is in ES-1.2, Post-LOCA Cooldown and Depressurization.
- An RCS cooldown has been initiated by dumping steam to the atmosphere.

Which ONE of the following describes the optimum RCP configuration, and the basis for this configuration?

A. One RCP should be run to produce effective heat transfer and RCS pressure control, yet minimize RCS heat input.

B. All RCPs should be stopped to minimize RCS inventory loss when the break uncovers.

C. Two RCPs should be run to ensure symetric heat transfer to the S/Gs, to enhance RCS pressure control, and to prevent steam voiding in the vessel head during RCS depressurization.

D. One RCP should be run to produce effective heat transer and RCS pressure control, yet minimize RCS inventory loss.

Answer: A

Question 3384 Details

Question Type:	Multiple Choice
Topic:	EOP0338
System ID:	107196
User ID:	EOP0338
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-95.3-LP-9/B [S00-0306]

Question 1073 Details

Question Type:	Multiple Choice
Topic:	EOP0332 (provide CSFSTs)
System ID:	73615
User ID:	EOP0332
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-95.3-LP-26F; ND-95.3-LP-48C [S99-0176]

1074 **ID: EOP0333** **Points: 1.00**

Unit 1 has experienced a loss of coolant accident and the team is presently in 1-ES-1.2, Post LOCA Cooldown and Depressurization. The team has initiated a cooldown and is depressurizing the RCS to refill the Pressurizer. Subcooling is lost during the depressurization.

Which ONE of the following identifies the method used to regain subcooling?

A. Deenergize the Pressurizer heaters.

B. Energize the Pressurizer heaters.

C. Continue the RCS cooldown.

D. Stop the RCS cooldown.

Answer: C

21

QUESTIONS REPORT
for Surry2002

1. WE04G2.4.5 001/T1G2/T1G1/PROCEDURE USAG/C/A 2.9/3.6/B/SR02301/S/RLM

The following conditions exist:

-A manual Rx trip was initiated 10 minutes ago based on AP-16.00 criteria
-Pressurizer level is off-scale low
-Pressurizer pressure is 1500 psig and decreasing
-All SG levels are 5% NR and slowly increasing
-All SG pressures are 1005 psig
-All main steam line radiation monitors are reading .02 mr/hr
-Vent-Vent radiation monitor is reading 4.3 E6 cpm
-Containment pressure is 9.2 psia
-Containment sump level is 47%
-Safeguards Area Sump high level alarm is locked in

Upon exiting E-0, which ONE of the following is the correct procedure transitions for the event in progress, if the leak is unisolable?

A.✓ ECA-1.2 (LOCA Outside Containment), ECA-1.1, (Loss of Emergency Coolant Recirculation)
B. E-1, ECA-1.1 (Loss of Emergency Coolant Recirculation), ECA-1.2 (LOCA Outside Containment)
C. ECA-1.1 (Loss of Emergency Coolant Recirculation), ECA-1.2 (LOCA Outside Containment)
D. E-1, ECA-1.2 (LOCA Outside Containment), ECA-1.1 (Loss of Emergency Coolant Recirculation)

Ref: SR EB # EOP0263
Surry lesson plans ND-95.3-LP-20, obj A&C; ND-95.3-LP-21, obj A&C;
ND-95.4-LP-12, obj A&C

RO Tier:	T1G2	SRO Tier:	T1G1
Keyword:	PROCEDURE USAG	Cog Level:	C/A 2.9/3.6
Source:	B	Exam:	SR02301
Test:	S	Misc:	RLM

Objectives

After receiving this instruction, the trainee will be able to:

A. Given the major action categories associated with ECA-1.1, Loss of Emergency Coolant Recirculation, explain the purpose of ECA-1.1, the transition criteria for entering and exiting ECA-1.1, and the types of operator actions that will occur within each category.

B. Given a copy of ECA-1.1, Loss of Emergency Coolant Recirculation, explain the basis of each step of the procedure.

C. Given actual or simulated plant conditions requiring implementation of ECA-1.1, Loss of Emergency Coolant Recirculation, successfully transition through the procedure, applying step background knowledge as required, to safely place the plant in the required optimal recovery condition.

Presentation

Distribute all handouts. Refer to/display H/T-20.1, Objectives, and review objectives with trainees.

A. Major Actions of ECA-1.1, Loss of Emergency Coolant Recirculation

1. Purpose

To provide guidance to restore emergency coolant recirculation capability, to delay RWST depletion by adding makeup and reducing outflow, and to depressurize the RCS to minimize break flow.

LESSON PLAN

Introduction

The Loss Of Coolant Accident, in itself, is a serious plant accident. However, the level of severity can be compounded by the fact that the LOCA is outside of the FINAL fission product barrier - Containment. Now, there is no protective shield enveloping the spilled reactor coolant water and fission products carried out of the RCS. This type of accident poses both a serious threat to the post-accident cooling capability of the plant and a potential hazard to the general public in the form of radioactive releases.

This lesson on the Emergency Response Guideline for LOCA Outside Containment is designed to provide an introduction to the accident and an in-depth analysis of the procedure associated with combatting this event.

Objectives

After receiving this instruction, the trainee will be able to:

A. Given the major action categories associated with ECA-1.2, LOCA Outside Containment, explain the purpose of ECA-1.2, the transition criteria for entering and exiting ECA-1.2, and the types of operator actions that will occur within each category.

B. Given a copy of ECA-1.2, LOCA Outside Containment, explain the basis of each step of the procedure.

C. Given actual or simulated plant conditions requiring implementation of ECA-1.2, LOCA Outside Containment, successfully transition through the procedure, applying step background knowledge as required, to address the challenge to plant and public safety.

LESSON PLAN

Introduction

The Reactor Safety Study, WASH-1400, identified Event V Sequences (Interfacing System LOCAs) as a significant contributor to the risk of core melt and high activity release. Some recent events have highlighted the need for greater attention to this potentially disastrous Loss of Coolant Event. This lesson plan will outline some of the concerns and methods of mitigating the probability of an Event V Sequence.

Objectives

After receiving this instruction, the trainee will be able to:

A. Describe an Interfacing System LOCA.

B. Describe the possible means of limiting the probability and consequences of an Interfacing System LOCA.

C. Describe the significance of the EVENT V Sequence.

Presentation

Distribute all handouts and copies of all the AIAs. Refer to/display H/T-12.1, Objectives, and review with trainees.

A. Interfacing LOCAs

1. Event V Sequence

Question 1007 Details

Question Type:	Multiple Choice
Topic:	EOP0262
System ID:	73547
User ID:	EOP0262
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-95.3-LP-21B; ND-95.4-LP-12B [S96-1030], [S96-1341]

1008 **ID: EOP0263** **Points: 1.00**

The following conditions exist:

- A manual Rx trip was initiated 10 minutes ago based on AP-16.00 criteria
- Pressurizer level is off-scale low
- Pressurizer pressure is 1500 psig and decreasing
- All SG levels are 5% NR and slowly increasing
- All SG pressures are 1005 psig
- All main steam line radiation monitors are reading .02 mr/hr
- Vent-Vent radiation monitor is reading 4.3 E6 cpm
- Containment pressure is 9.2 psia
- Containment sump level is 47%
- Safeguards Area Sump high level alarm is locked in

Which ONE of the following is the correct procedure transitions for the event in progress if the leak is unisolable?

A. E-0, E-1, ECA-1.2 (LOCA Outside Containment), ECA-1.1 (Loss of Emergency Coolant Recirculation)

B. E-0, ECA-1.1 (Loss of Emergency Coolant Recirculation), ECA-1.2 (LOCA Outside Containment)

C. E-0, E-1, ECA-1.1 (Loss of Emergency Coolant Recirculation), ECA-1.2 (LOCA Outside Containment)

D. E-0, ECA-1.2 (LOCA Outside Containment), ECA-1.1, (Loss of Emergency Coolant Recirculation)

Answer: D

22

1. WE05EA2.1 001

-Unit 1 has had a loss of Both Feedwater Pumps.
-SG lo-lo level alarms come in and the Reactor fails to trip.
-Actions of S.1 " Response to Nuclear Power Generation / ATWS are performed.
-Reactor Power is < 5%, with a negitive start up rate.
-All AFW pumps faied to start.

Which one of the following procedures should the SRO transistion to?

A. Re-enter E-0 Reactor Trip/SI at step 1, complete immediate operator actions and then transition to FR-H.1 "Response to Loss of Secondary Heat Sink."

B. Re-enter E-0 Reactor Trip/SI at the begining and transition to ES-0.1" Reactor Trip Response" ~~at the appropriate~~ step. when directed

C. Directly Enter ES-0.1, "Reactor Trip Response".

D. Directly Enter FR-H.1, "Response to Loss of Secondary Heat Sink".

Surry Exam Bank Question # 862 slightly modified.
ND-95.3-LP-2D; ND-95.3-LP-26 objectives D and F; ND-95.3-LP-41 objective A

A. Incorrect, E-0 has been exited from and CSFs apply FR-H.1 has a red path and should be entered.

B. Incorrect, E-0 has been exited from and CSFs apply FR-H.1 has a red path and should be entered.

C. Incorrect, E-0 has been exited from and CSFs apply FR-H.1 has a red path and should be entered.

D Correct FRP H.1 should be entered.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: Scramble Range: A - D

RO Tier:	T1G2	SRO Tier:	T1G2
Keyword:		Cog Level:	C/A 3.4/4.4
Source:	B	Exam:	SR02301
Test:	S	Misc:	GW;

Question 861 Details

Question Type:	Multiple Choice
Topic:	EOP0110
System ID:	73400
User ID:	EOP0110
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-95.3-LP-13B; E-3 [S96-1333], [S95-1085]

862 **ID: EOP0111** **Points: 1.00**

Due to a loss of feedwater pumps, the SGs go below the LO-LO setpoint and no reactor trip occurs. The RO carries out the actions of FR-S.1, Response to Nuclear Power Generation/ATWS. Reactor power is < 5% with a negative SUR. At the completion of this procedure, a "red" path exists on heat sink.

Which ONE of the following procedures should the SRO go to next?

A. Enter ES-0.1, Reactor Trip Response.

B. Re-enter E-0, Reactor Trip/SI, at step 1, complete immediate action steps, and then go to FR-H.1.

C. Re-enter E-0, Reactor Trip/SI, at the step in effect and complete E-0 up to step 14, which transitions the team to FR-H.1.

D. Enter FR-H.1, Response to Loss of Secondary Heat Sink.

Answer: D

Objectives

After receiving this instruction, the trainee will be able to:

A. [Given a simulated plant condition requiring the use of the Critical Safety Function Status Trees, transition through the Heat Sink status tree denoting, in accordance with the rules of priority, any applicable Function Restoration Procedure needing implementation. SOER 86-01, Recommendation 7]

B. Given the Major Action Categories associated with FR-H.1, Response to Loss of Secondary Heat Sink, explain the purpose of FR-H.1, the transition criteria for entering and exiting FR-H.1, and the types of operator actions that will occur within each category.

C. Given a copy of FR-H.1, Response to Loss of Secondary Heat Sink, explain the basis of each procedural step.

D. Given actual or simulated plant conditions requiring implementation of FR-H.1, Response to Loss of Secondary Heat Sink, successfully transition through the procedure, applying step background knowledge as required, to address the Critical Safety Function challenge in progress.

Presentation

Distribute all handouts. Refer to/display H/T-41.1, Objectives, and review with trainees.

B. State, in order of priority sequence, the six critical safety functions.

C. Explain the four-color, color-coding "Rules of Priority" as they apply to the CSF Status Trees.

D. Explain the prioritization of challenges within and between the Critical Safety Function Procedures.

E. Explain the points at which, during the course of a transient, CSF Status Tree monitoring is to be implemented.

F. Explain the use, including the function, of the CSF Status Trees during a Control Room emergency event.

<u>Presentation</u>

Distribute all handouts. Refer to/display H/T-26.1, Objectives, and review with trainees.

A. CSF/Barrier Associations

1. The second category of guideline procedures contained in the ERG Procedures set are called the **<u>FUNCTION RESTORATION</u> procedures (FRs).** The "FUNCTIONS" referred to in the title are those which must be satisfied to assure the physical barrier maintenance to prevent radioactive material release.

B. Explain the two-column format of the Emergency Response Guideline Procedures, including the placement criteria for cautions and notes.

C. Explain the method by which "Immediate Operator Action" steps are identified in the body of the ERG Procedures.

D. Describe the intended overall usage of the Emergency Response Guidelines Network.

E. Given various plant conditions during which an emergency event occurs, evaluate the application of the "Modes of Applicability" as described in the ERG User's Guide.

F. Given actual or simulated EOP implementation, apply the management standards and other good practices applicable to EOP usage.

G. Explain the format design of the Emergency Response Guideline Procedures.

<u>Presentation</u>

Distribute all handouts. Refer to/display H/T-2.1, Objectives, and review objectives with trainees

A. Action Verb Identification

Direct trainees to turn to AIA-2.1, Action Verbs. Review various action verbs with trainees.

23

1. WE06EA2.1 001

-Unit 2 has had a LOCA.
-E-1, "Loss of Reactor or Secondary Coolant" is in progress.
-RCPs are secured.
-Containment Pressure is 47 psia, and slowly increasing.
-Total AFW flow is 485 gpm.
-SG WR levels are: A-48%; B-40%; C-39%.
-RCS Pressure 920 psig.
-IR NIs indicate 2 X10-11 amps, with a SUR of 0.
-CETCs indicate 600 degrees F.
-RVLIS Full Range indicates 45%.

Which of the following is the correct procedure for the team to transistion to?

A. FR-S.2, "Response to Loss of Core Shutdown".

B. FR-C.2, "Response to Degraded Core Cooling".

C. FR-Z.1, "Response to Containment High Pressure".

D. FR-H.5, " Response to Steam Generator Low Level".

Surry Exam Bank Question # 1066.
Surry Lesson Plans.ND-95.3-LP-26 objective D ; ND-95.3-LP-39 objective A.

A. Incorrect, S.2 would be entered on a yellow path.

B. Correct, the conditions to enter C.2 are met with RVLIS < 46%.

C. Incorrect, Z.1 would be entered on an Orange path and C.2 is a higher priority.

D. Incorrect , H.5 is a yellow path, C.2 would be entered first,

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: Scramble Range: A - D

RO Tier:	T1G1	SRO Tier:	T1G1
Keyword:		Cog Level:	C/A 3.4/4.2
Source:	B	Exam:	SR02301
Test:	S	Misc:	GWL

Objectives

After receiving this instruction, the trainee will be able to:

A. Given the Major Action Categories associated with FR-C.2, Response to Degraded Core Cooling, explain the purpose of FR-C.2, the transition criteria for entering and exiting FR-C.2, and the types of operator actions that will occur within each category.

B. Given a copy of FR-C.2, Response to Degraded Core Cooling, explain the basis of each procedural step.

C. Given actual or simulated plant conditions requiring implementation of FR-C.2, Response to Degraded Core Cooling, successfully transition through the procedure, applying step background knowledge as required, to address the Critical Safety Function challenge in progress.

Presentation

Distribute all handouts. Refer to/display H/T-39.1, Objectives. Review objectives with trainees.

A. Major Actions of FR-C.2, Response to degraded Core Cooling

1. The purpose of FR-C.2, Response to Degraded Core Cooling, is to provide guidance to restore adequate core cooling.

2. This guideline is entered from an ORANGE priority from the CSF status tree upon symptoms of degraded core cooling.

B. State, in order of priority sequence, the six critical safety functions.

C. Explain the four-color, color-coding "Rules of Priority" as they apply to the CSF Status Trees.

D. Explain the prioritization of challenges within and between the Critical Safety Function Procedures.

E. Explain the points at which, during the course of a transient, CSF Status Tree monitoring is to be implemented.

F. Explain the use, including the function, of the CSF Status Trees during a Control Room emergency event.

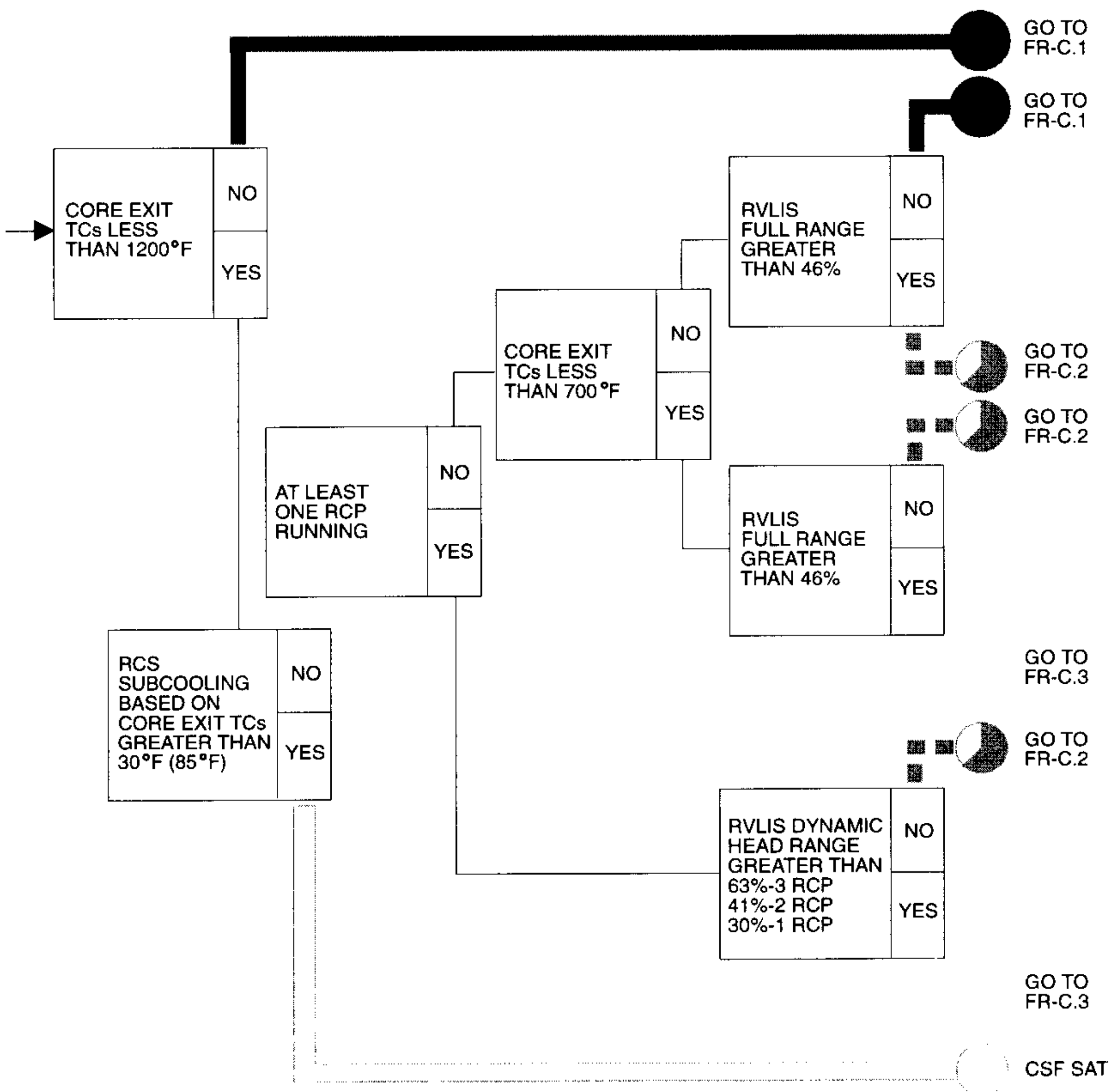
Presentation

Distribute all handouts. Refer to/display H/T-26.1, Objectives, and review with trainees.

A. CSF/Barrier Associations

1. The second category of guideline procedures contained in the ERG Procedures set are called the **FUNCTION RESTORATION procedures (FRs).** The "FUNCTIONS" referred to in the title are those which must be satisfied to assure the physical barrier maintenance to prevent radioactive material release.

Number:	Title:	Revision:
F-2	CORE COOLING	



Drawing No. CB380

SNSOC CHAIRMAN

DATE

Question 1065 Details

Question Type:	Multiple Choice
Topic:	EOP0324
System ID:	73607
User ID:	EOP0324
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-83-LP-5B; ND-89.1-LP-2B [S99-0176]

1066 **ID: EOP0325** **Points: 1.00**

(Refer to CSFSTs)

A LOCA has occurred and the team is presently in 1-E-1, Loss of Reactor or Secondary Coolant.

The following conditions exist:

ù RCPs are secured.
ù Containment pressure is 47 psia and slowly increasing.
ù Total AFW flow is 485 gpm.
ù SG WR levels: A-48%, B-40%, C-39%.
ù RCS pressure 920 psig.
ù IR NIs indicate 2 x 10^{-11} amps with a SUR of 0.
ù CETCs indicate 530øF.
ù RVLIS Full Range indicates 45%.

Based on the above conditions, the team should transition to _____.

A. 1-FR-H.2

B. 1-FR-Z.1

C. 1-FR-S.1

D. 1-FR-C.2

Answer: D

24

1. WE08EA2.1 001

Which one of the following conditions would require entering FR-P.1 "Response to imminent Pressurized Thermal Shock Condition" on an orange or red path? (CSF status trees are attached.)

A. Cooldown Greater than 100 degrees F. in 60 minutes,Temperature 290 degrees F. RCS pressure 1800 psig.

B. Cooldown Less than 100 degrees F. in 60 minutes, Temperature 250 degrees F. RCS pressure 350 psig.

C. Cooldown Greater than 100 degrees F. in 60 minutes,Temperature 270 degrees F. RCS pressure 520 psig.

D. Cooldown less than 100 degrees F. in 60 minutes,Temperature 290 degrees F. RCS pressure 1800 psig.

Bank Question, Several bank questions used to develop. From Farley, and Surry base question. ND-95.3-LP-46 Objectives A, and D.

A. Incorrect, Does not meet the criteria for entry in to FR-P.1 on a orange or red path.

B. Incorrect, Does not meet the criteria for entry in to FR-P.1 on a orange or red path.

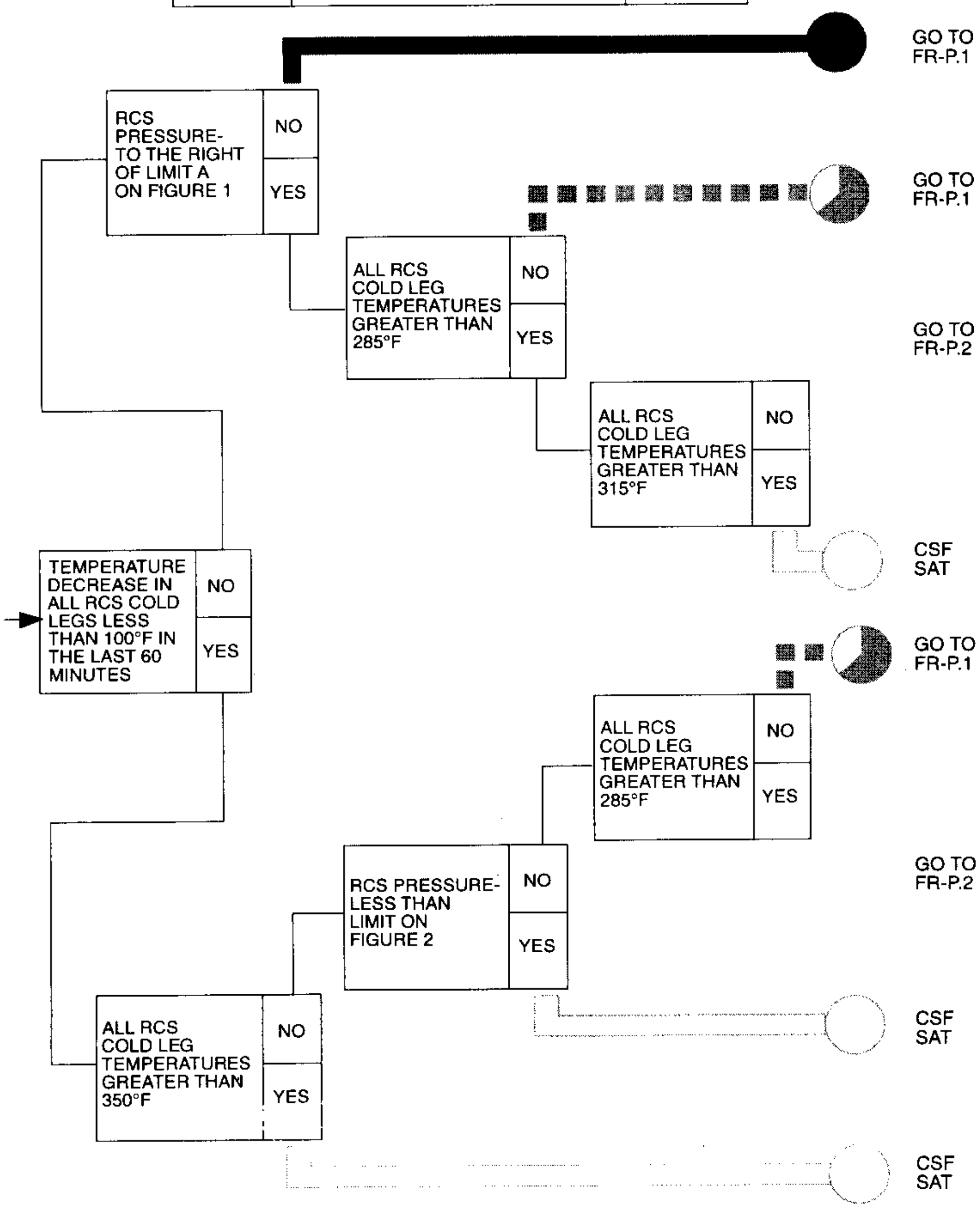
C. Correct, Meets the entry requiremnent for an orange path.

D. Incorrect, Does not meet the criteria for entry in to FR-P.1 on a orange or red path.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C B D D C B C A B C Scramble Range: A - D

RO Tier:	T1G1	SRO Tier:	T1G1
Keyword:		Cog Level:	C/A 3.4/4.2
Source:	B	Exam:	SR02301
Test:	S	Misc:	GWL

Number:	Title:	Revision:
F-4	INTEGRITY	2

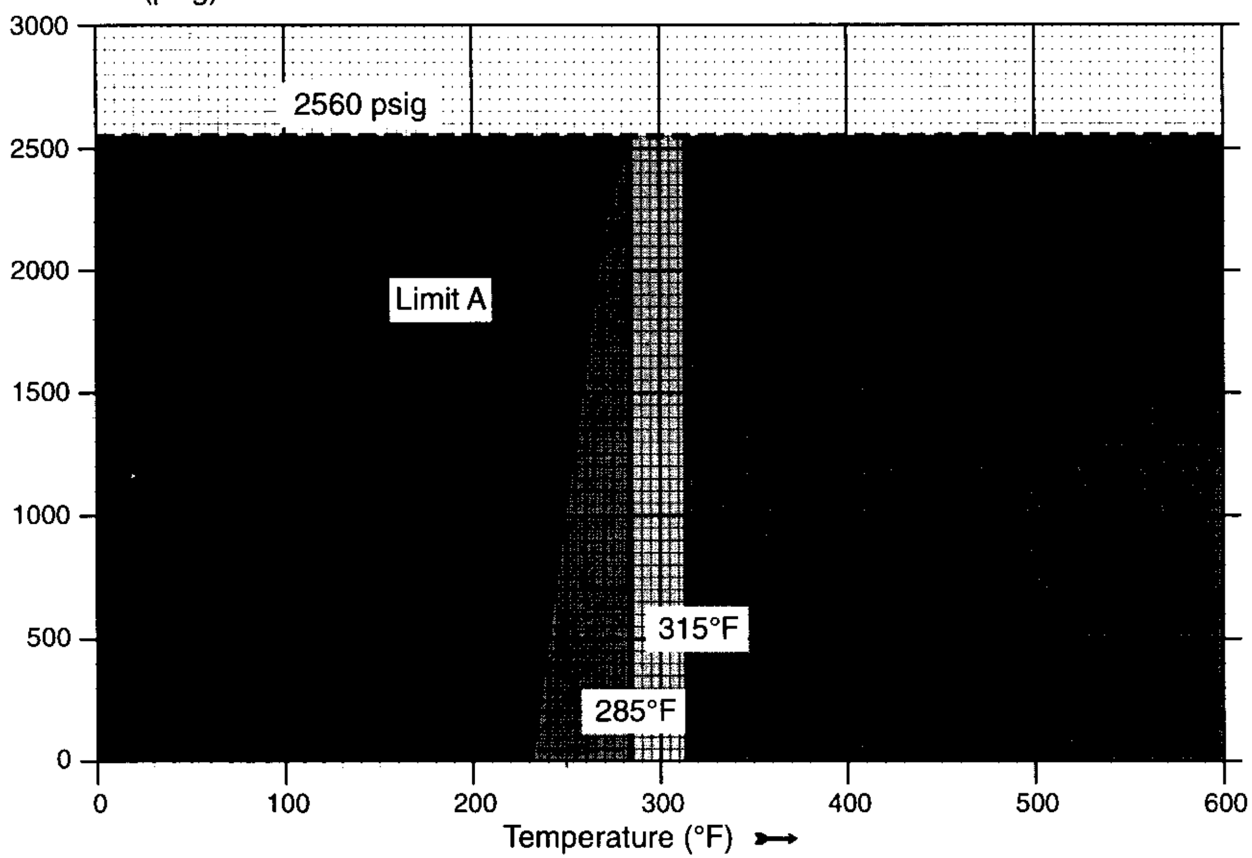


Graphics No. CB383

SNSOC CHAIRMAN DATE

Number:	Title:	Revision:
F-4	INTEGRITY	2

FIGURE 1 - OPERATIONAL LIMITS CURVE
SURRY UNIT 1 AND UNIT 2



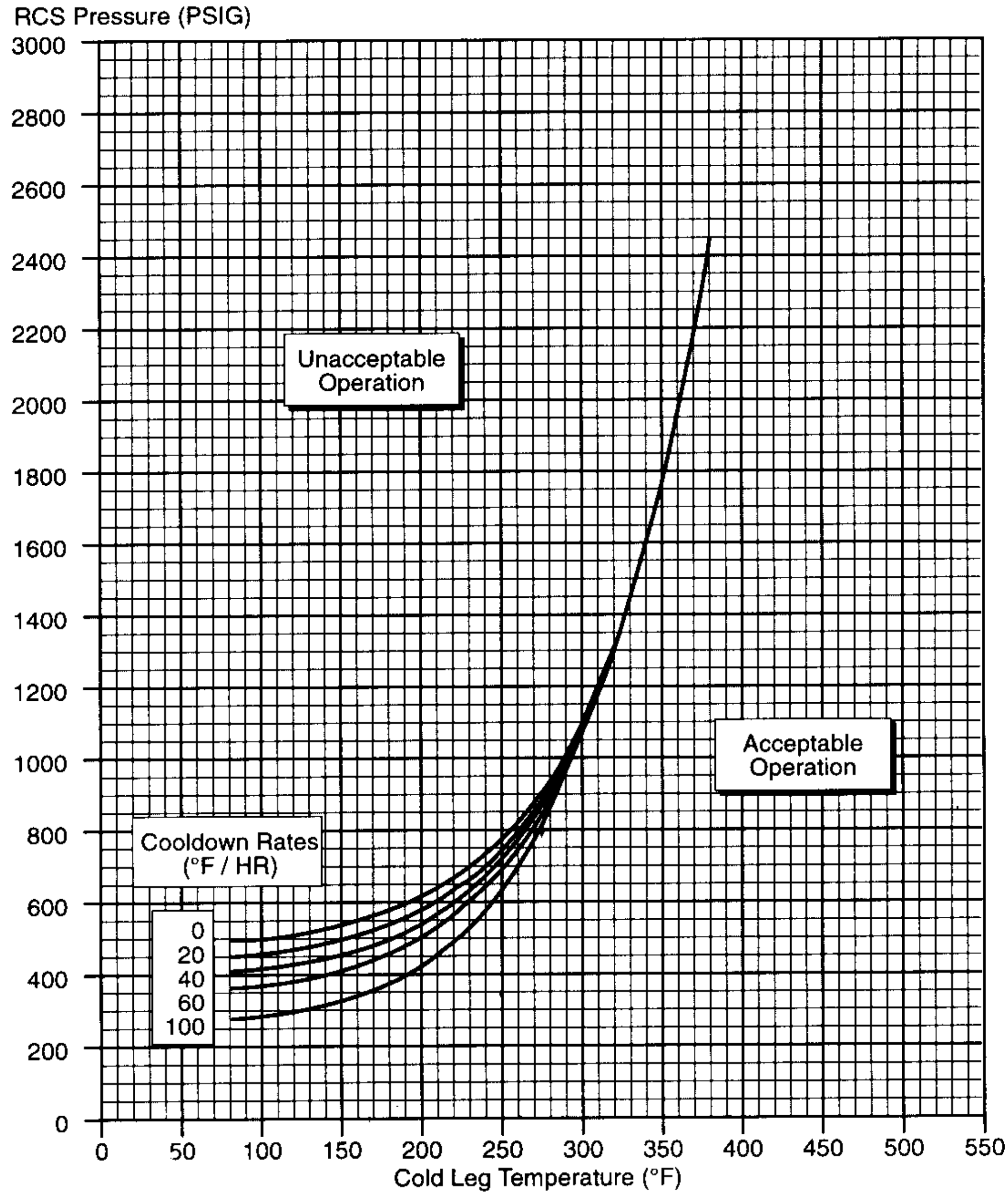
SNSOC Chairman

Date

Drawing No: WT316

Number:	Title:	Revision:
F-4	INTEGRITY	2

Figure 2
RCS COOLDOWN RESTRICTIONS



SNSOC Chairman

Date

Graphics No. CS1392

O52533K13013;

Which of the following conditions would require entering FRP-P.1 on a red or orange path? (Circle the correct response.)

A. Greater than 100° cooldown in last 60 minutes to a temperature of 250° and 100 psig.

B. Less than 100° cooldown in last 60 minutes to a temperature of 250° and 100 psig

C. Greater than 100° cooldown in last 60 minutes to a temperature of 285° and 1800 psig

D. Less than 100° cooldown in last 60 minutes to a temperature of 285° and 1800 psig

ANSWER: A. Point Value: 1.0 Answer Time: 4.0 Mins. Part B. 100
Static Sim Scenario Nos. ___ ___ ___ ___ ___ ___ ___
S&K No. 240205023020 ___________ ___________
K/A No. 002000A0.15G 000009EA2.14 ___________
RO/SRO Impf. 4.1 /4.3 3.8 /4.4 ___/___
Objective O52533K13
Reference O52533K, FRP-P.1 CSF-0

*QNUM 33696
*HNUM 34319 (Do NOT change If < 9,000,000)
*ANUM
*QCHANGED FALSE
*ACHANGED FALSE
*QDATE 1992/10/19
*FAC 348 Farley 1 & 2
*RTYP PWR-WEC3
*EXLEVEL S
*EXMNR
*QVAL
*SEC
*SUBSORT
*KA 000011G012
*QUESTION
WHICH ONE (1) of the following conditions would require entering FNP-1-FRP-P.1, "Response to Imminent Pressurized Thermal Shock Condition"? FNP-1-CSF-0.4, "Integrity" is attached.

a. Cooldown less than 100 degrees F. in 60 minutes, temperature 250 degrees F, pressure 520 psig.

b. Cooldown less than 100 degrees F. in 60 minutes, temperature 250 degrees F, pressure 350 psig.

c. Cooldown greater than 100 degrees F. in 60 minutes, temperature 275 degrees F, pressure 520 psig.

d. Cooldown greater than 100 degrees F. in 60 minutes, temperature 275 degrees F, pressure 350 psig.

*ANSWER
a. [+1.0]

*REFERENCE
1. Farley: OPS-52533K, "FRP-P.1, Response to Imminent Pressurized Thermal Shock Condition", Objective 13 and FNP-1-CSF-0.4, "Integrity".
2. Farley: License Retraining exam bank question 052533K13015, question #360.
3. KA 000011G012 (4.0/4.1)

947 **ID: EOP0196** **Points: 1.00**

The operator entered FR-C.2, Response to Degraded Core Cooling, in response to an ORANGE path condition.

Which ONE of the following statements is correct with regard to transitioning out of this procedure?

A. The operator may leave this procedure at any step as soon as the Core Cooling adverse condition has cleared.

B. The operator must leave this procedure before completion and go to FR-Z.1, Response to Containment High Pressure, if the status tree indicates an ORANGE path.

C. The operator may leave this procedure before completion and go to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, if the status tree indicates an ORANGE path.

D. The operator must leave this procedure before completion and go to FR-S.1, Response to Nuclear Power Generation/ATWS, if the subcriticality status tree indicates an ORANGE path.

Answer: D

Question 947 Details

Question Type:	Multiple Choice
Topic:	EOP0196
System ID:	73486
User ID:	EOP0196
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-95.3-LP-26D and F; ND-95.3-LP-39A [S96-0989], [S96-1351]

QUESTIONS REPORT
for Surry2002

75

1. WE14EA2.1 001

A large steam break accident occurred 45 minutes ago, the crew transitioned to E-1 and the following plant conditions now exist:

-The faulted S/G has blown dry.
-The SI is still in progress.
-RCS Th and Tc are 260 degrees F.
-RCS Pressure is 1500 psig and rising
-Containment pressure is 47 psig.
-Containment sump level is 6 feet.
-Containment Rad levels are pre-event.

Which one of the following describes the appropriate procedure flowpaths that the crew should take.

A. FR-Z.1 should be implemented until the entry condition is restored to a yellow or green path.

B. FR-P.1 shold be implemented until completion and then FRZ.1 should be implemented.

C. FR-Z.1 should be impemented until completion, and then FR-P.1 should be implemented.

D. FR-P.1 should be implemented until the entry condition is restored to a yellow or green path.

Ref: from exam bank (Farley).
Surry Lesson Plan ND-95.3-LP-48 Objectives A and D.

A. Incorrect, A red path does exist on Z.1, it should be finished through to completion, and then FR-P.1 should be entered. Tfhe team should not wait for the conditions to become green or yellow.

B. Incorrect, FR-P.1 should be entered, but it is an orange path, and a red pathe exists on Z.1, so Z.1 should also be entered and it should be entered first.

C. Correct, Per the CSF's and entry conditions.

D. Incorrect, A red path exists on Z.1, it should be finished through to completion, and then FR-P.1 should be entered.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C B D D A A D D C B Scramble Range: A - D

RO Tier:	T1G1	SRO Tier:	T1G1
Keyword:		Cog Level:	C/A 3.3/3.8
Source:	B	Exam:	SR02301
Test:	S	Misc:	GWL

O52533K13012;

A large steam break accident occurred 50 minutes ago, and the following plant conditions now exist:

- The faulted SG has blown dry.
- The SI is still in progress.
- RCS Th and Tc are 255°.
- RCS pressure is 1600 psig and rising.
- PRZR level is 95% and stable.
- Containment pressure is 56 psig.
- Containment sump level is 6 feet.
- Containment rad monitors at pre-event values.

Based on the above conditions: (Circle the correct response.)

A. FRP-Z.1 is the only procedure which should be implemented until entry condition is restored to yellow or green path.

B. FRP-P.1 is the only procedure which should be implemented until entry condition is restored to yellow or green path.

C. FRP-Z.1 should be implemented until completion, and then FRP-P.1 should be implemented.

D. FRP-P.1 should be implemented until completion, and then FRP-Z.1 should be implemented.

ANSWER: C. Point Value: 1.0 Answer Time: 6.0 Mins. Part B. 100
Static Sim Scenario Nos. ___ ___ ___ ___ ___ ___ ___
S&K No. 240205023020 ___________ ___________
K/A No. 002000A015G 000009A2.14 ___________
RO/SRO Impf. 4.1 /4.3 3.8 /4.4 ___/___
Objective O52533K13
Reference O52533K, FRP-P.1
Rev. Date 8/24/94

Objectives

After receiving this instruction, the trainee will be able to:

A. Given a simulated plant condition requiring the use of the critical safety function status trees, transition through the Containment Status Tree denoting, in accordance with the rules of priority, any applicable function restoration procedure needing implementation.

B. Given the Major Action Categories associated with FR-Z.1, Response to Containment High Pressure, explain the purpose of FR-Z.1, the transition criteria for entering and exiting FR-Z.1, and the types of operator actions that will occur within each category.

C. Given a copy of FR-Z.1, Response to Containment High Pressure, explain the basis of each procedural step.

D. Given actual or simulated plant conditions requiring implementation of FR-Z.1, Response to Containment High Pressure, successfully transition through the procedure, applying step background knowledge as required, to address the Critical Safety Function challenge in progress.

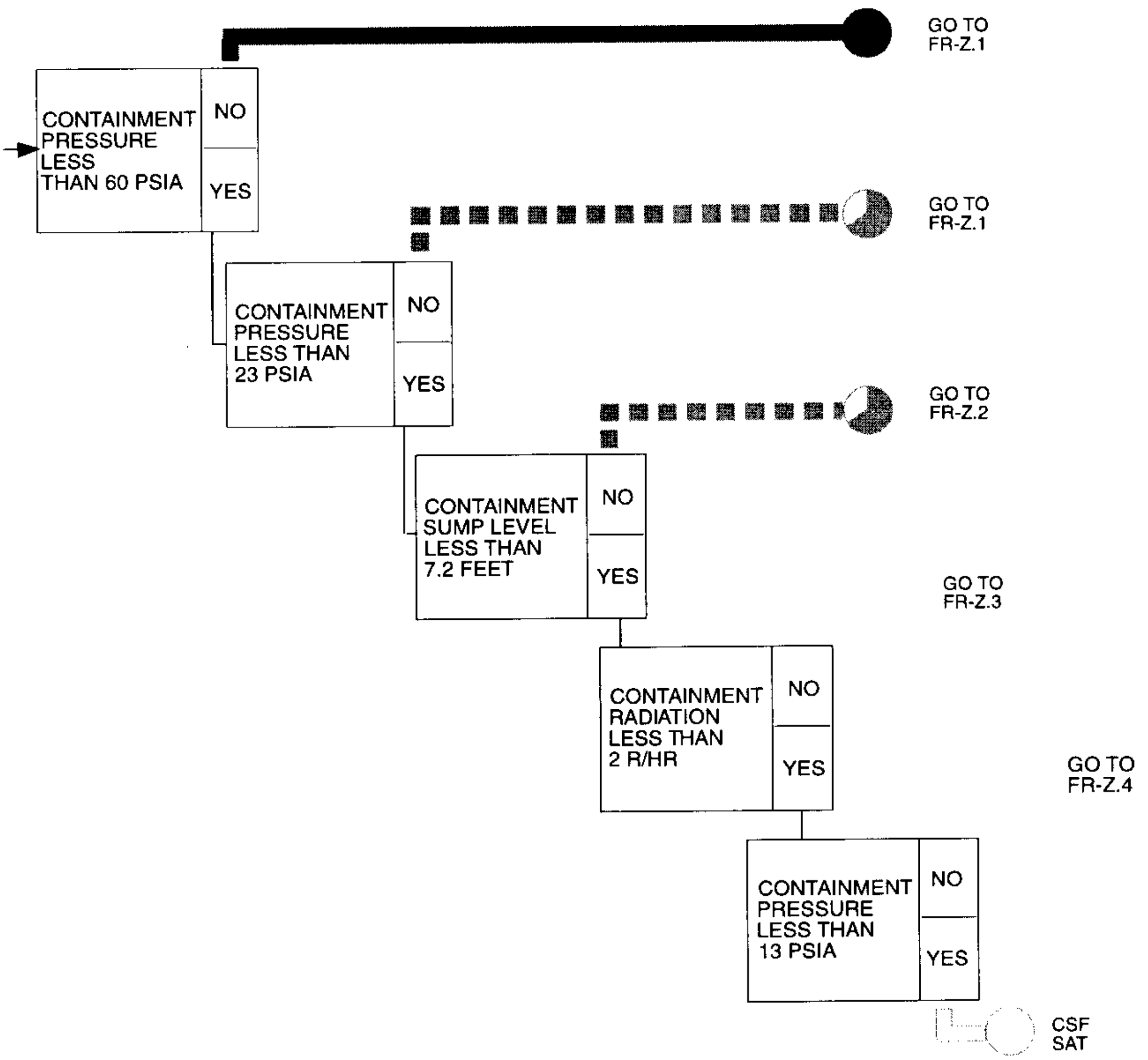
Presentation

Distribute all handouts. Refer to/display H/T-48.1, Objectives and review objectives with trainees.

A. Containment Status Tree

1. The Containment status tree provides a systematic method to determine the status of the Containment Critical Safety Function.

Number:	Title:	Revision:
F-5	CONTAINMENT	

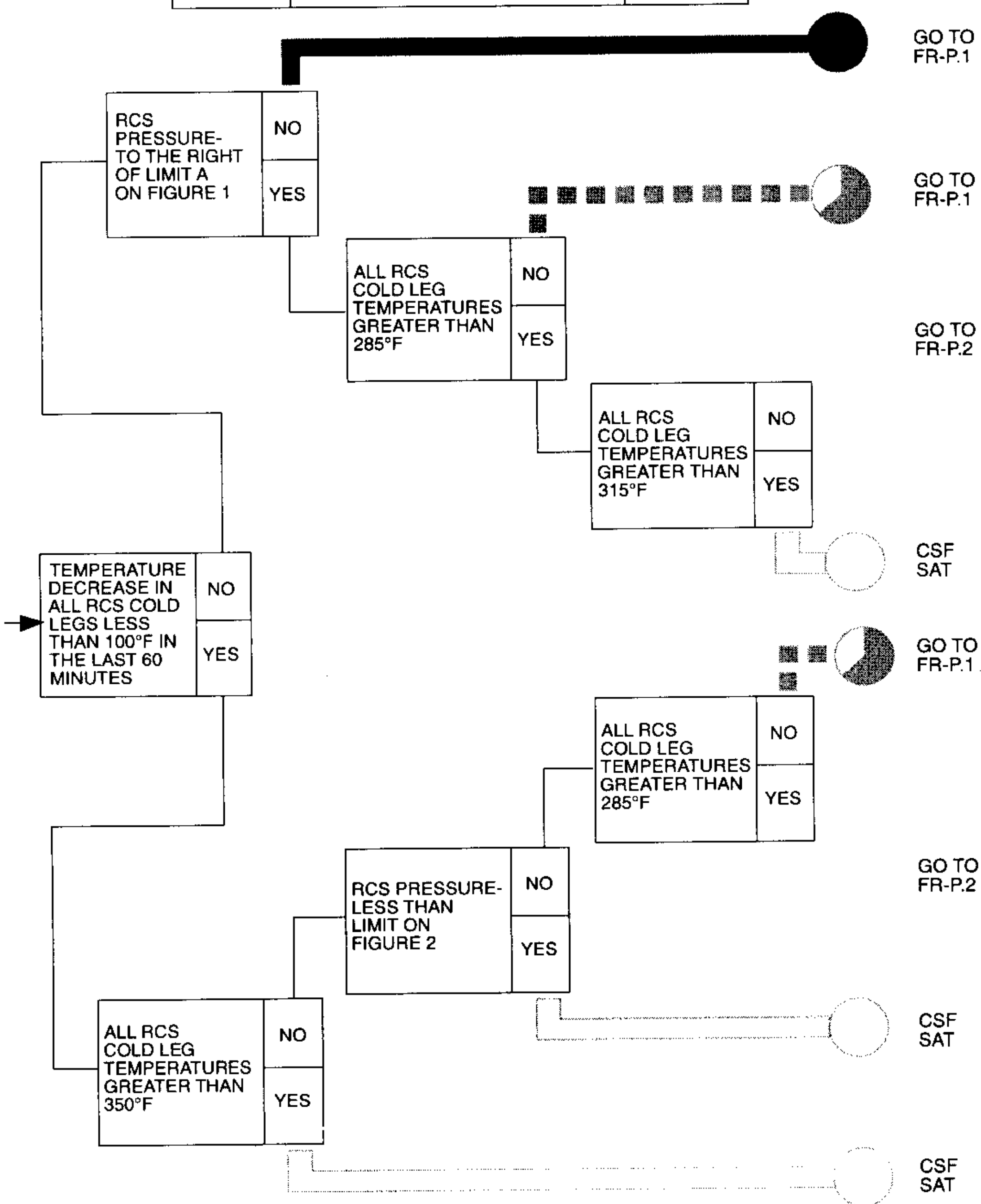


Drawing No. CB382

SNSOC CHAIRMAN

DATE

Number:	Title:	Revision:
F-4	INTEGRITY	2



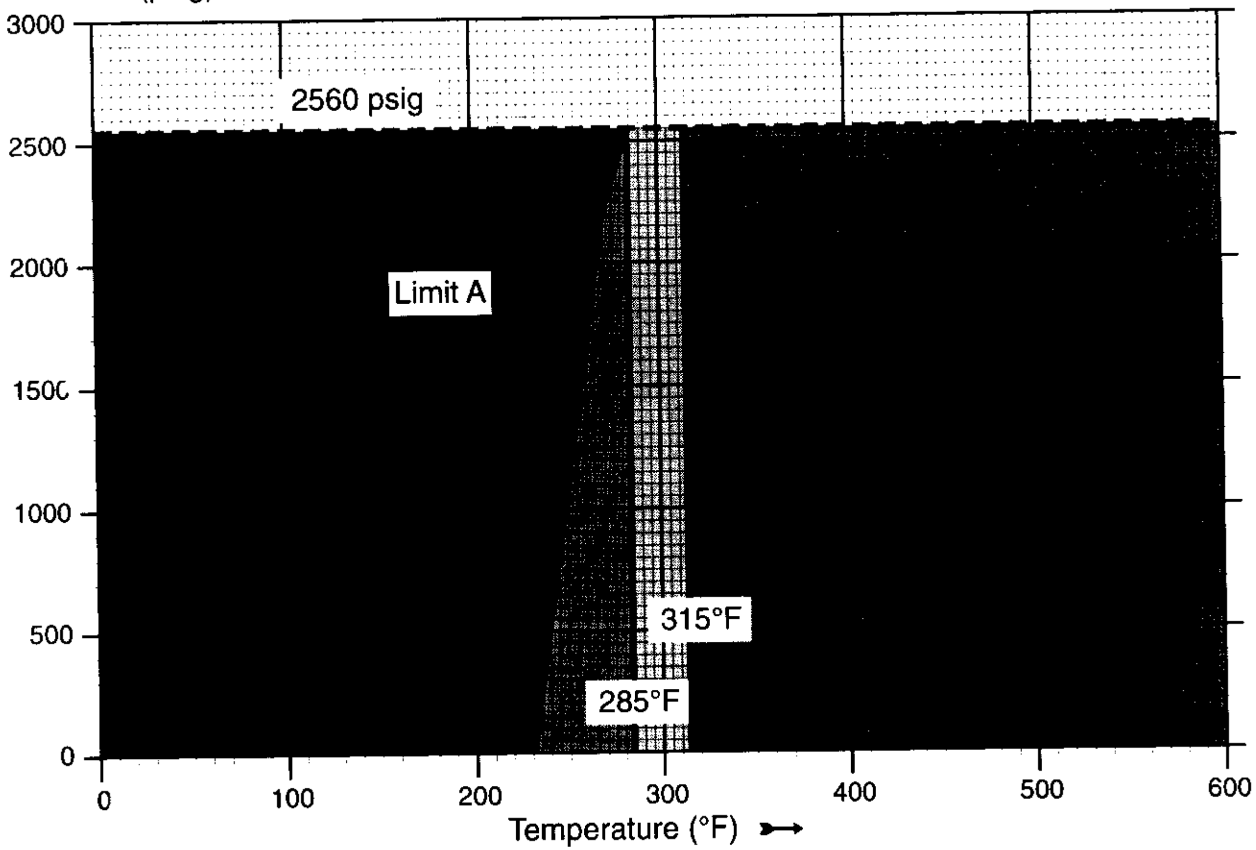
Graphics No. CB383

SNSOC CHAIRMAN

DATE

Number:	Title:	Revision:
F-4	**INTEGRITY**	**2**

FIGURE 1 - OPERATIONAL LIMITS CURVE
SURRY UNIT 1 AND UNIT 2



SNSOC Chairman

Date

Drawing No: WT316

896 **ID: EOP0145** **Points: 1.00**

The following conditions exist:

- In response to a large break LOCA, a transition from 1-E-0, Reactor Trip or Safety Injection, to 1-E-1, Loss of Reactor or Secondary Coolant, has been performed.
- Due to a RED path on the Core Cooling Status Tree, a transition to 1-FR-C.1, Response to Inadequate Core Cooling, has been performed.
- During performance of 1-FR-C.1, you observe that the Core Cooling Status Tree has changed from a RED to a YELLOW condition while you identify a RED path on the Containment Status Tree.

Which ONE of the following is the proper procedural transition, and why?

A. Immediately transition to 1-FR-Z.1, Response to Containment High Pressure, since a RED path is a higher priority than a YELLOW path.

B. Complete 1-FR-C.1; since once ANY FR is entered, it must be completed before any other transition can be made.

C. Complete 1-FR-C.1; since it was entered due to a RED path, it must be completed unless a higher priority path occurs, then transition to FR-Z.1.

D. Perform the actions of 1-FR-C.1 and 1-FR-Z.1 simultaneously, since FR procedures of the same priority can be executed together.

Answer: C

Question 896 Details

Question Type:	Multiple Choice
Topic:	EOP0145
System ID:	73435
User ID:	EOP0145
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	1.00
User Number 1:	0.00
User Number 2:	0.00
Comment:	ND-95.3-LP-26D and F; ND-95.3-LP-48B [S96-0989], [S96-1360]