INITIAL SUBMITTAL

OCONEE EXAM 2000-301
50-269, 270, AND 287/2000-301

JULY 10 - 14, 18, 19, AND 20, 2000

INITIAL SUBMITTAL

JPMS - RO

JPM's - RO

Oconee
2000

NRC Copy

Every JPM should:

1. ___ be supported by facility licensee's job task analysis.
2. ___ be operationally important (meets NRC K/A Catalog threshold criterion of 2.5 (3 for requalification exams) or as determined by the facility and agreed to by the NRC).
3. ___ be designed as either SRO only, RO/SRO or AO/RO/SRO.
4. include the following, as applicable:
   a. ___ initial conditions
   b. ___ initiating cues
   c. ___ references and tools, including associated procedures
   d. ___ validated time limits (average time allowed for completion) and specific designation of those JPMs that are deemed to be time-critical by the facility operations department
   e. ___ specific performance criteria that include:
      (1) ___ expected actions with exact control and indication nomenclature and criteria (switch position, meter reading), even if these criteria are not specified in the procedural step
      (2) ___ system response and other cues that are complete and correct so that the examiner can properly cue the examinee, if asked
      (3) ___ statements describing important observations that should be made by the examinee
      (4) ___ criteria for successful completion of the task
      (5) ___ identification of those steps that are considered critical
      (6) ___ restrictions on the sequence of steps

H - Bobby Ayers

Facility: Oconee
Exam Level: **RO**

Date of Examination: 7-10/17-00
Operating Test No.: 1

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
a. CRO-12A, Recover a Dropped Control Rod; (20 min.) AP/1/A/1700/15, OP/0/A/1105/09, Encl 4.10 [KA: 005AA2.03 (3.4/4.4)] Note: This JPM conducted with Admin A.1 CRO-40, (Calculate SDM) (RO ONLY) (5 min.)	D, S, A	1
b. NRC-001, Establish PZR Steam Bubble (IPE) (20 min.) OP/1/A/1103/002, Encl. 4.14 [KA: 004A4.09 (3.5/3.3)]	N, S, L	2
c. NRC-002, PORV Stroke Test (10 min.) PT/0/A/0201/004 [KA: 010A4.03 (4.0/3.8)] Note: Performed following completion of NRC-001, (Establish Steam Bubble)	N, S, L, A	3
d. CRO-013, Swap TD EFDWP Suction to Hotwell; (10 min.) AP/1/A/1700/19 [KA: APE054AA1.01 (4.5/4.4)]	D, C/S, L	4S
e. CRO-095, Swap RBCU's (Inadvertent Ch. 5 ES actuation); (10 min.) OP/0/A/1104/15 [KA: 022A4.01 (3.6/3.6)]	M, C/S	5
f. CRO-009, Following Keowee Emergency Start, transfer from CT-4 to CT-5; (10 min.) OP/0/A/1106/19, Encl. 3.13 [KA: 062A4.01 (3.3-3.1)]	D, C/S, L	6
g. JPM NRC-1998 – RIA-57 Operability Check; (RIA-57 fails to meet acceptance criteria) (10 min.) PT/0/1/0230/001, Encl. 13.9 and Encl. 13.3 &4 [KA: 072A2.02 (2.8/2.9)]	D, C/S, A	7

B.2 Facility Walk-Through

System / JPM Title	Type Code*	Safety Function
a. NLO-017, Align Cooling Water to HPIP's from Station ASWP; (16 min.) AP/1,2,3/A/1700/11, Encl 6.3; OP/0/A/1102/06, Encl. 3.3 [KA: 076A2.01 (3.5/3.7)]	D, R, L	4S
b. NLO-004, Manually Bypass the KI Inverter; (5 min.) AP/1,2,3/A/1700/23, Encl 6.1 [KA: 063K4.01 (2.7/3.0)]	D, L	6
c. NLO-041, Restart the Primary Instrument Air Compressor following a Compressor trip; (10 min.) OP/0/1106/27, Encl. 4.11; APO/1/A/1700/22 [KA: 078G2.1.30 (3.9/3.4)]	D, A	8

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM CRO-012A/SIM

RECOVERY OF A DROPPED CONTROL ROD

CANDIDATE ____

EXAMINER ____

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

RECOVERY OF A DROPPED CONTROL ROD

Alternate Path:

Unit is tripped upon receipt of second dropped CR

Facility JPM #:

CRO-12A

K/A Rating(s):

005-AA2.03 3.5/4.4

Task Standard:

Control Rod recovery
Unit is tripped upon receipt of second dropped CR

Preferred Evaluation Location: Simulator __X__ In-Plant ______

Preferred Evaluation Method: Perform __X__ Simulate ______

References:

AP/1/A/1700/15, Dropped Control Rods

Validation Time: 20 min. **Time Critical:** NO

==

Candidate: ________________________________ Time Start : _________
NAME Time Finish: _________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: _________________________ ______________________________/_______
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC # _______
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/0/A/1105/009, Control Rod Drive System

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.6.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

START TIME: _______

STEP 1: Take manual control of rods at the Diamond Control Station by performing the following: Place the Diamond Station in MANUAL STANDARD: The AUTO/MANUAL pushbutton on the Diamond Control Panel is depressed, the MANUAL half of the Push Button is backlighted. Location 1UB1 ***Cue: Inform candidate time compression has taken place and the Control Rod has been repaired and should be withdrawn.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 2: Obtain copy of Enclosure 4.10, Recovery of Dropped/Misaligned Regulating Control Rod, of OP/0/A/1105/009, Control Rod Drive System. STANDARD: Obtain a copy of OP/0/A/1105/009, Control Rod Drive System and determine that Enclosure 4.10, Recovery of Dropped/Misaligned Regulating Control Rod, is the proper enclosure for this condition and obtain a copy from the procedure file. COMMENTS:	___ SAT ___ UNSAT

STEP 3: Take manual control of rods at the Diamond Control Station by performing the following: Place the SG Master in HAND Place the Diamond Station in MANUAL (**Diamond is already in HAND per step 1**) STANDARD: The manual pushbutton for the SG Master hand/auto station is depressed, The White Hand light comes ON and the Red Auto light Goes OFF. Location 1UB1 COMMENTS:	___ SAT ___ UNSAT
STEP 4: SELECT group with dropped/misaligned rod on the Group Select Switch STANDARD: GROUP SELECT SWITCH on 1UB1 is located by the student and rotated to Group 6. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: Press selector for SEQ OVERRIDE. STANDARD: The SEQ/SEQ OR pushbutton is located on the Diamond Control panel on 1UB1 and depressed. "SEQ OR" is backlighted. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6: Select JOG on the Speed Selector STANDARD: The SPEED Selector is located by the student on the Diamond Control panel on 1UB1 and rotated to the JOG position. COMMENTS:	___ SAT ___ UNSAT

STEP 7: Press selector for LATCH switch and insert group for approximately 15 seconds or until the group OUT LIMIT lamp on the Diamond Panel goes off. Release LATCH switch. STANDARD: The IN LIMIT (LATCH) BYPASS pushbutton is located by the student and depressed and held while the INSERT/WITHDRAW joystick is used to insert Group 6 until the Group 6 Out limit lamp, located on the Diamond Control Panel on 1UB1, extinguishes. The LATCH pushbutton is then released and the INSERT/WITHDRAW joystick returned to neutral. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: TRANSFER the dropped/misaligned rod to the auxiliary power supply. Select dropped/misaligned rod on the Single Select Switch Press selector for SEQ OVERRIDE Press selector for AUXILIARY Press selector for CLAMP Press selector for MANUAL TRANSFER switch until TRANSFER CONFIRM lamp and the CONTROL ON lamp on the PI panel light Press selector for CLAMP RELEASE STANDARD: On the CRD Panel on 1UB1: SELECT dropped/misaligned rod on the SINGLE SELECT SWITCH. VERIFY SEQ OR is backlit **(Not Critical).** Depresses GROUP/AUXIL pushbutton to make transfer to AUXIL. Verifies SYNC is backlit on MAN TRANS/SY/TR CF pushbutton **(Not Critical)** Depresses CLAMP/CLAMP REL pushbutton to make transfer to CLAMP. CLAMP will be backlit. Depresses MAN TRANS/SY/TR CF pushbutton. TR CF will become backlit. White CONTROL ON lights will illuminate for the Dropped Rod on the Position Indication panel. Depresses CLAMP/CLAMP REL pushbutton and verifies CLAMP REL is backlit. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 9: Perform PI alignment on the dropped/misaligned rod as follows: Press selector for the LATCH switch and insert rod for 15 seconds. Release LATCH switch. Compare absolute and relative readings on the PI panel. Adjust RPI to equal API with POSITION RESET RAISE/LOWER switch. STANDARD: Absolute and relative indications on the PI panel, on 1UB1, are compared using toggle switch to make comparison. RPI is selected with the select toggle switch. The POSITION RESET RAISE/LOWER toggle switch is then placed in the lower position and RPI indication is matched to API position. When matched the RAISE/LOWER toggle is released to neutral. The select toggle switch is returned to the API position. COMMENTS:	___ SAT ___ UNSAT

STEP 10: SELECT RUN on the Speed Selector. STANDARD: SPEED SELECTOR is located by the student on 1UB1 and rotated to the run position. ***CUE: Rod has been misaligned for less than 24 hours.*** COMMENTS:	___ SAT ___ UNSAT
STEP 11: Withdraw dropped/misaligned rod until power begins to increase and then stop withdrawal. STANDARD: Rod is withdrawn while monitoring reactor power for an increase. ***NOTE: When rod is 50% withdrawn, booth operator drop second rod.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 12: Manually trip the reactor STANDARD: The student recognizes the second control rod inserting and manually trips the reactor by depressing the Reactor Trip pushbutton and performs IMAs. COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	The Diamond is taken to manual before repairs on the CR begins.
4	Instructs the rod logic as to which group the rod is in that the operator wants to recover.
5	Allows the operator to withdraw the dropped rod.
7	The latching of the group to clear the out limit is necessary so that the individual rod can be withdrawn.
8	Places the dropped rod on the auxiliary power supply for withdrawal while leaving the group on the group power supply
11	Necessary to withdraw dropped CR.
12	The second dropped rod places the unit in an unanalyzed condition and this is a direction, which is given by OMP 1-18, Operator memory Items.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.6.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods

1. Initial Conditions

____ 1.1 Refer to ITS 3.1.4.

____ 1.2 Reactor Power < 60% of the allowable thermal power for the existing RCP combination.

____ 1.3 Pre-job briefing completed including review of OMP 2-12 (Reactivity Management).

____ 1.4 Reactor Engineer has been contacted.

____ 1.5 Review all Limits and Precautions.

2. Procedure

CAUTION:	When ICS is placed in MANUAL, Rx power, RCS temperature, RCS pressure and feedwater flow should be monitored for proper response

NOTE:	If a regulating rod is dropped and the Diamond is in MANUAL, to drive the group that has the dropped rod in it, the IN LIMIT BYPASS switch on the Diamond will have to be depressed due to the group "in" limit stopping rod "in" motion. If the Diamond is in AUTO, the Asymmetric Rod Fault (9") will bypass the "in" limit automatically.

2.1 Take manual control of rods at the Diamond Control Station by performing the following:

____ 2.1.1 Place the SG Master in HAND.

____ 2.1.2 Place the Diamond Station in MANUAL.

____ 2.2 Select group with dropped/misaligned rod on the Group Select Switch.

____ 2.3 Press selector for SEQ OVERRIDE.

____ 2.4 Select JOG on the Speed Selector.

NOTE:	Omit this step if group is not at OUT LIMIT. Step 2.5 is performed to clear the group OUT LIMIT lamp on the Diamond Panel for the Regulating Rod group with the dropped/misaligned rod to allow outward motion of the dropped/misaligned rod.

____ 2.5 Press selector for LATCH switch and insert group for approximately 15 seconds or until the group OUT LIMIT lamp on the Diamond Panel goes off. Release LATCH switch.

___ 2.6 Transfer the dropped/misaligned rod to the Auxiliary Power Supply as follows:

2.6.1 Select dropped/misaligned rod on the Single Select Switch.

2.6.2 Press selector for SEQ OVERRIDE.

2.6.3 Press selector for AUXILIARY.

NOTE:	Ensure Manual Transfer Sync light is lit before pressing clamp.

2.6.4 Press selector for CLAMP.

NOTE:	Transfer from Normal to Auxiliary power will be confirmed by both the TRANSFER CONFIRM and the CONTROL ON lights lit. These lights will be OFF, if the transfer switch sticks with both power supplies energized.

2.6.5 Press selector for MANUAL TRANSFER switch until TRANSFER CONFIRM lamp and the CONTROL ON lamp on the PI panel light.

2.6.6 Press selector for CLAMP RELEASE.

___ 2.7 Perform PI alignment on the dropped/misaligned rod as follows:

2.7.1 Press selector for the LATCH switch and insert rod for 15 seconds.

2.7.2 Release LATCH switch.

2.7.3 Compare absolute and relative readings on the PI panel.

2.7.4 Adjust RPI to equal API with POSITION RESET RAISE/LOWER switch.

___ 2.8 Select RUN on the Speed Selector.

CAUTION:
- If rod is known to have been dropped/misaligned for <u>less than 24 hours</u>, then rod can be withdrawn at 30 in./min. (RUN Speed). {1}
- If rod has been dropped/misaligned for <u>greater than 24 hours</u>, then rod can be withdrawn in 10% withdrawal increments spaced 30 minutes apart at 30 in./min. (RUN Speed). {1}

2.9 Withdraw dropped/misaligned rod until power begins to increase and then stop withdrawal.

2.10 Perform the following to stop power increase:

2.10.1 If the dropped rod is in the controlling group:

_____ A. Transfer the dropped rod from the Auxiliary Power Supply to its Normal Power Supply per Enclosure "Transfer Rods Between Normal and Auxiliary Power Supply."

_____ B. Insert controlling group regulating rods to stop power increase.

_____ C. Transfer the dropped rod from its Normal Power Supply to the Auxiliary Power Supply per Enclosure "Transfer Rods Between Normal and Auxiliary Power Supply."

_____ D. Repeat steps 2.9 - 2.10.1 until dropped rod is realigned with its group.

2.10.2 If the dropped rod is <u>not</u> in the controlling group:

_____ A. Press selector for GROUP.

_____ B. Place ROD SELECTOR switch to ALL.

_____ C. Place GROUP SELECTOR switch to controlling group.

_____ D. Insert controlling group regulating rods to stop power increase.

E. Reselect the dropped rod as follows:

_____ 1. Press selector for AUXILIARY.

_____ 2. Place ROD SELECTOR switch to rod being recovered.

_____ 3. Place GROUP SELECTOR switch to group containing the dropped rod.

_____ F. Repeat steps 2.9 - 2.10.2 until dropped rod is realigned with its group.

_____ 2.11 If the dropped/misaligned rod is not on its Normal Power Supply, place dropped/misaligned rod on its Normal Power Supply as follows:

2.11.1 Select JOG on the Speed Selector.

NOTE:	Ensure Manual Transfer Sync Light is lit before pressing clamp.

2.11.2 Press selector for CLAMP.

2.11.3 Press selector for MANUAL TRANSFER switch until the TRANSFER CONFIRM lamp and the CONTROL ON lamp at the PI panel go out.

2.11.4 Press selector for CLAMP RELEASE.

2.11.5 Press selector for GROUP.

2.11.6 Press selector for TRANSFER RESET.

2.11.7 Press selector for FAULT RESET.

2.11.8 Select RUN on the Speed Selector.

_____ 2.12 If the group with the dropped/misaligned rod was initially at its OUT LIMIT, withdraw group until its OUT LIMIT light is lit on the Diamond panel.

_____ 2.13 Press selector for SEQUENCE.

NOTE:	The Diamond Control station can be placed in AUTO anytime provided the neutron error is within ±1.0%, auto power available, and Safety Groups at the out limit.

2.14 Place ICS back to Automatic Control by performing the following:

_____ 2.14.1 Place the Diamond Station in AUTO.

_____ 2.14.2 Place the SG Master in AUTO.

2.15 If the rod was recovered in less than 8 hours:

THEN

the normal power escalation rate should be used after recovery of the rod to return to 100% power. {1}

2.16 If the rod was recovered within 8 to 24 hours:

THEN

the deconditioned power escalation rate should be used after recovery of the rod to return to 100% reactor power. {1}

2.17 If the rod recovery took greater than 24 hours:

THEN

the initial cycle startup power escalation rate should be used after recovery of the rod to return to 100% reactor power. {1}

(R06-97)

WHC/JMB/JPP

Duke Power Company TRNS. (1) ID No AP/1/A/1700/015

PROCEDURE PROCESS RECORD

Revision No 4

LAN Location: SAROS

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Dropped Control Rods

(4) Prepared By Dennis Gordon. Date 2/17/99

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)
☐ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By Walter M. Barber (QR) Date 2/25/99

Cross-Disciplinary Review By ____ (QR)NA WB Date ____

Reactivity Mgmt. Review By Walter M. Barber (QR)NA ____ Date 2/25/99

(7) Additional Reviews

Reviewed By ____ Date ____

Reviewed By ____ Date ____

(8) Temporary Approval *(if necessary)*

By ____ (SRO/QR) Date ____

By ____ (QR) Date ____

(9) Approved By [signature] Date 3/8/99

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

(11) Date(s) Performed ____

Work Order Number (WO#) ____

COMPLETION

(12) Procedure Completion Verification

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ____ Date ____

(13) Procedure Completion Approved ____ Date ____

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Dropped Control Rods**	Procedure No. AP/1/A/1700/015
	Revision No. 004
Continuous Use **Reactivity Management Related**	Electronic Reference No. OX002RGS

DROPPED CONTROL RODS
Reactivity Management Related

TABLE OF CONTENTS

OCONEE NUCLEAR STATION

Dropped Control Rods

1. Purpose

This procedure provides the actions necessary to maintain the plant in a safe condition following a dropped control rod(s).

2. Symptoms

- "CRD POSITION ERROR" statalarm (1SA-2, B-10)
- "CRD SAFETY RODS NOT AT UPPER LIMIT" statalarm (1SA-2, C-10)
- Control Rod "IN-LIMIT" light on (Green light on Position Indication Panel)
- Control Rod "ASYMM. RODS" light on (Yellow light on Diamond).

3. Automatic Systems Actions

3.1 **IF** ICS is in Auto,

AND an "ASYMM. RODS" (Yellow Light on Diamond) occurs,

THEN an "OUT" inhibit at 60% power is established and the Reactor will runback to 55% power.

3.1.1 **IF** the "ASYMM. RODS" (Yellow Light on Diamond) clears,

THEN runback may stop before reaching 55% power.

4. Immediate Manual Actions

_____ 4.1 **IF** more than one Control Rod has dropped,

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 4.2 **IF** more than one Control Rod is misaligned > 9" (6%),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.3 **IF** due to a malfunction, a Control Rod Group is misaligned > 9" (6%) from the acceptable region of PT/1/A/0600/001, (Periodic Instrument Surveillance),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.4 **IF** a Control Rod has dropped on an approach to criticality,

OR a dropped Control Rod results in a return to subcriticality from a critical condition,

THEN manually insert all Control Rods to Group 1 at 50% WD.

5. Subsequent Actions

_____ 5.1 **IF** the Reactor has tripped,

THEN **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 5.2 Verify Reactor runback <60% Full Power is in progress.:

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

NOTE 5.2.1:	If a control rod drops while the ICS Diamond Panel is in MANUAL, an "Auto Inhibit" (safety rod) or "Sequence Enable" (regulating rod) alarm will prevent placing the ICS in automatic.

_____ 5.2.1 **IF** the Reactor has **NOT** runback,

THEN commence manual Reactor Power reduction to < 60% of the allowable thermal power for the RCP combination.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.3 **IF** operating with only three (3)RCPs,

THEN commence manual Reactor Power reduction to < 45% Full Power.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.4 Notify I&E to begin investigation for the cause of the Dropped Control Rod.

_____ 5.5 Initiate actions to meet the Required Action of ITS 3.1.4 by performing the following:

_____ 5.5.1 Within one hour verify > 1% SDM
with allowance for the inoperable control rod(s):

- Perform PT/1/A/1103/15, (Reactivity Balance Calculation).

_____ 5.5.2 Within two hours reduce Reactor Power < 60%
of the allowable thermal power for the RCP combination.

NOTE 5.5.3:	Notification to I&E should be made as soon as possible due to the complexity of resetting the RPS trip setpoints and the short ITS allowable time limits.

_____ 5.5.3 Notify I&E to reduce RPS Flux/Flow-Imbalance trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.5.4 Notify I&E to reduce RPS High Flux trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.6 **WHEN** Reactor Power is < 60%
of the allowable thermal power for the RCP combination,

THEN notify I&E to begin repair of the Dropped Control Rod.

_____ 5.7 **WHEN** I&E is ready to begin repairs on the Dropped Control Rod,

THEN Place the ICS Diamond control station in MANUAL,

AND permit I&E to repair Dropped Control Rod.

Dropped Control Rods

CAUTION:5.8	The Duty Reactor Engineer must evaluate the effects of local power distribution and the necessity for special maneuvering limits prior to the recovery of a dropped or an asymmetric Control Rod.

_____ 5.8 **WHEN** I&E has repaired the Dropped Control Rod,

THEN recover the Dropped Control Rod per OP/0/A/1105/009, (Control Rod Drive System).

END

Dropped Control Rods

Appendix

1. PIP # 0-O98-2734

END

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM NRC-001/SIM
Establish PZR Steam Bubble

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:
Establish a PZR Steam Bubble

Alternate Path:

N/A

Facility JPM #:

NEW

K/A Rating(s):

004 A4.09 (3.5 / 3.3)

Task Standard:

Per OP/1103/002 Encl 4.14, properly operate PZR heaters and venting to establish a PZR steam bubble.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:
OP/1103/002 Encl 4.13 and 14

Validation Time: 20 min. **Time Critical:** NO

==

Candidate: __________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: __________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall SNAP _____
2. IMPORT NRC-001
3. Set LPSW to both LPI Coolers to ≈ 900 gpm/cooler
4. Ensure both GWD compressors are operating
5. Place QT in recirc (Open CS-5 and 6 then start the Component Drain Pump)
6. Override QT press to 0 psig (prevent pressure increase)

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/1103/002 Encl 4.14

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC
Unit 1 startup is in progress
Establishing a PZR bubble is in progress

INITIATING CUES:

The SRO in the Control Room directs you to complete operations to establish a PZR bubble using the in-progress procedure OP/1103/002 Encl 4.14 starting at step 2.2

START TIME: _______

STEP 1: Obtain a copy of the appropriate procedure. STANDARD: Operator obtains a copy of OP/1103/002 Encl 4.14. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Energize PZR heaters to complete PZR temperature increase STANDARD: Group 1, 2, 3, and 4 are energized as necessary to control PZR temperature increase < 90°F/hr COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: Ensure 1GWD-17 (PZR Vent) is closed STANDARD: 1GWD-17 (PZR Vent) observed to be closed. *****NOTE: Red OPEN light OFF; Green CLOSED light ON.*** COMMENTS:	___ SAT ___ UNSAT

STEP 4: Place 1RC-1 (PZR Spray) in automatic STANDARD: 1RC-1 (PZR Spray) push button is depressed and placed in automatic COMMENTS:	___ SAT ___ UNSAT
STEP 5: Ensure the following valve are properly positioned: 1N-119 Closed 1N-161 Closed 1GWD-146 Open 1N-107 Locked Closed 1N-110 Closed NOTE: This JPM step covers procedure steps 2.5-2.7 ***CUE: All the above valves are positioned correct and SG blanketing is not required*** STANDARD: Procedure steps 2.5 – 2.7 are verified COMMENTS:	___ SAT ___ UNSAT
STEP 6: ENSURE 1GWD-12 (QT Vent inside RB) open STANDARD: 1GDW-12 is opened COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: ENSURE 1GWD-13 (QT Vent outside RB) open STANDARD: 1GWD-13 is opened COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: Start a second GWD Compressor STANDARD: Verify the second GWD Compressor is operating NOTE: The second GDW compressor is operating COMMENTS:	___ SAT ___ UNSAT
STEP 9: As RCS pressure increases, throttle 1GWD-17 (PZR Vent) to maintain pressure between 38-45 psig STANDARD: PZR heaters are cycled to maintain RCS pressure within the procedure band while venting the PZR by throttling open GWD-17. When QT pressure high alarm (1SA-6/B7) is actuated then the operator closes GWD-17. **NOTE: At the Examiners request and after the candidate has cycled QT pressure by venting PZR N2 to the QT the simulator will be placed into freeze and new PZR conditions established.** **Freeze simulator at Examiners request.** **Recall SNAP 104** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 10: Closed 1GWD-13 STANDARD: 1GWD-13 is closed *Cue:* ***New PZR conditions: all N2 has been vented, GDW-13 and 17 have been open for 35 minutes (Step 2.15)*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 11:	Ensure QT level increases with minimal pressure change.	**CRITICAL STEP**
STANDARD:	QT pressure, level, and temperature is monitored to ensure a steam bubble is established in the PZR. Temperature and level should increase as steam is being vented under water in the QT. If all the N2 has been vented then pressure will not increase much.	___ SAT
COMMENTS:		___ UNSAT
	END TASK	

STOP TIME: ___________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
2	PZR temperature increase required to form steam bubble
6	Establishes QT vent flow path to vent header
7	Establishes QT vent flow path to vent header
9	1GWD-17 is throttled to prevent over pressurizing the QT.
10	Isolates PZR vent path.
11	Determine a steam bubble is established in the PZR by verifying QT pressure will not increase much as the PZR is vented.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the unit 1 OATC
Unit 1 startup is in progress
Establishing a PZR bubble is in progress

INITIATING CUES:

The SRO in the Control Room directs you to complete operations to establish a PZR bubble using the in-progress procedure OP/1103/002 Encl 4.14 starting at step 2.2.

Establishing Pzr Steam Bubble And RCS Final Vent

1. Initial Conditions

____ 1.1 RCS PZR level 80" to 120", loops full with 38 to 45 psig nitrogen bubble in Pzr.

____ 1.2 Notify Chemistry to add hydrazine to Pzr per Chemistry Procedure.

____ 1.3 Verify a M/U flow path to RCS available.

____ 1.4 Verify 1A BHUT boron ≥ 1% ΔK/K SDM per PT/1/A/1103/015 (Reactivity Balance Procedure).

____ • Verify 1A BLEED TRANSFER PUMP Suction aligned to 1A BHUT. {5}

____ 1.5 Verify QT level ≥ 80" **OR** fill QT to ≥ 80" from 1A BHUT.

____ 1.6 Review Limits and Precautions.

2. Procedure

NOTE:	RCS boron may change affecting core reactivity. {5}

____ 2.1 **WHEN** Enclosure "Requirements To Form Pzr Steam Bubble" is complete, continue this enclosure to form a Pzr Steam Bubble.

NOTE:	Maximum allowable heatup rate for Pzr is 90°F/hr

____ 2.2 Energize Pzr Heaters to increase Pzr Temperature. {9}

____ 2.3 Ensure 1GWD-17 (PRESSURIZER VENT) closed.

____ 2.4 Place 1RC-1 (PZR SPRAY) in "AUTO". {9}

2.5 Position the following valves: {2}

____ • Ensure 1N-119 (SG Pzr Supply Blk) closed. (R-2G-E)

____ • Ensure 1N-161 (Pzr Supply Blk) closed. (R-2G-E)

____ 2.6 Ensure 1GWD-146 (Pzr Vent to RV Line) open. (R-3G-E. Side at Stairs)

2.7 **IF NOT** required for SG blanketing, perform the following: {2}

____ • Lock closed 1N-107 (LP Nitrogen Htr Byp). (AB-4-E Penet)

____ • Close 1N-110 (Rx Bld Low Press Hdr Supply). (AB-2-Hallway)

_____ 2.8 Ensure 1GWD-12 (QUENCH TANK VENT (INSIDE RB)) open. {9}

_____ 2.9 Ensure 1GWD-13 (QUENCH TANK VENT (OUTSIDE RB)) open. {9}

_____ 2.10 **IF** required, start second GWD Compressor per OP/1&2/A/1104/018 (GWD System) **OR** OP/3/A/1104/018 (GWD System).

NOTE:	• Pzr Heaters may require cycling to maintain RCS pressure 38 - 45 psig. • Vent Header should **NOT** be over pressurized.

_____ 2.11 As RCS pressure increases, throttle 1GWD-17 (PRESSURIZER VENT) maintain RCS pressure 38 - 45 psig. {17}

- _____ **IF** required, cycle 1GWD-13 (QUENCH TANK VENT (OUTSIDE RB)) as required for Vent Header pressure control. {17}

NOTE:	Opening 1GWD-13 and throttling 1GWD-17 to maintain RCS pressure constant is better than cycling 1GWD-17 and 1GWD-13.

2.12 Continue to throttle open 1GWD-17 (PRESSURIZER VENT) to maintain RCS pressure between 38 - 45 psig. {17}

_____ 2.13 Ensure 1GWD-13 (QUENCH TANK VENT (OUTSIDE RB)) open. {17}

_____ 2.14 Ensure 1GWD-17 (PRESSURIZER VENT) open. {17}

cycle

ISA-6/B7 QT Press high

NOTE:	When 1GWD-17 is open and QT pressure is only changing due to QT level increase, Pzr Steam Bubble Formation is complete. {17}

IC4

SNAP IC4 → 2.15 **WHEN** 1GWD-13 **AND** 1GWD-17 have been open for > 30 minutes, perform the following to ensure Pzr steam bubble complete: {17}

_____ 2.15.1 Close 1GWD-13 (QUENCH TANK VENT (OUTSIDE RB)). {17}

NOTE:	QT level change because of steam condensing from Pzr is required.

_____ 2.15.2 Ensure QT level increases with minimal pressure change. {17}

2.16 WHEN Pzr venting is complete, ensure the following: {9}

_____ • 1GWD-17 (PRESSURIZER VENT closed

_____ _____ • 1GWD-13 (QUENCH TANK VENT (OUTSIDE RB)) closed.

_____ _____ • 1GWD-12 (QUENCH TANK VENT (INSIDE RB)) closed.

_____ _____ • 1GWD-146 (Pzr Vent to RV Line) closed. (R-3G-E. Side at Stairs)

_____ _____ • 1GWD-18 (Pzr Vent and N_2 Isolation) closed. (R-3G-E. Side at Stairs)

_____ 2.17 Perform PT/1/A/0201/004 (1RC-66 Stroke Test).

_____ 2.18 Begin RCS pressure increase to a band of 50-70 psig.

_____ • Place new RCS pressure band on Turnover sheet.

_____ • Notify STA of new RCS pressure band.

NOTE:	CRDs should **NOT** be vented if RCS > 225°F

2.19 After RCS ≥ 50 psig, vent the following CRDs per Enclosure "CRDM Vent Procedure" to detect any gases:

_____ • Center CRD.

_____ • Two (2) adjacent CRDs.

_____ • CRD #34

_____ 2.19.1 IF any sign of gases is detected during vent, vent all CRDs per Enclosure "CRDM Vent Procedure".

CAUTION: 1RC-19 and 1RC-38 access requires lifting deck grating. Safety should be considered.

2.20 Verify the following valves are open:

_____ _____ • 1RC-19 (Loop 1A High Point Vent Blk) (Top of 1A Hot Leg) (SV/Encl. 4.14)

_____ _____ • 1RC-38 (Loop 'B' High Point Vent) (Top of 1B Hot Leg) (SV/Encl. 4.14)

_____ _____ • 1RC-184 (Reactor Vessel Head Vent) (RB-3F-W) (SV/Encl. 4.14)

_____ _____ • 1RC-84 (Post Accident Sample Line Blk) (R-B-W) (SV/Encl. 4.14)

_____ _____ • 1HP-268 (RCP A2 Seal #1 Byp Flow Inst. Outlet) (R-1-E/ Near Incore Tk) (SV/Encl. 4.10)

_____ _____ • 1HP-265 (RCP A1 Seal #1 Byp Flow Inst. Outlet) (R-1-E/ Near Incore Tk) (SV/Encl. 4.10)

_____ _____ • 1HP-274 (RCP B2 Seal #1 Byp Flow Inst. Outlet) (R-1-SW/Near 1LP-1&2) (SV/Encl. 4.10)

_____ _____ • 1HP-271 (RCP B1 Seal #1 Byp Flow Inst. Outlet) (R-1-SW/Near 1LP-1&2) (SV/Encl. 4.10)

_____ _____ 2.21 Verify 1HP-330 (RCP B1 & B2 Seal #1 Byp Drn) closed. (R-1-E/ Near Incore Tk)

_____ 2.22 Inspect Incore Tube Caps inside Incore Tank for leaks.

2.23 After CRD final vent is complete:

_____ • Remove venting equipment from RxV Head **AND** place (except Torque Wrenches) in Vent Equipment Locker. (RB-3F-E)

NOTE: Place faulty Vent equipment in metal cabinet located in Hot Machine Shop Storage Room.

_____ • Bring faulty Vent equipment out of R.B.

_____ • Issue an R005 **AND** place deficiency tags on faulty equipment.

_____ • Return Torque Wrenches to Decon.

_____ • Notify Ops OWPG how many vent hoses are in equipment locker and any equipment with R005s issued (include WR/WO numbers).

Problem Investigation Process
Oconee Nuclear Station

PIP Serial No:	Action Category:	LER No:	Other Report:
O-00-00255	3		

Problem Identification

Discovered Time/Date: 15:10 01/20/2000 **Occurred Time/Date:** 19:00

Unit(s) Affected:

Unit	Mode	%Power	Unit Status Remarks
2	1		

System(s) Affected:

CS Coolant Storage

Affected Equipment

(No Equipment Affected)

Location of Problem:

Bldg: Column Line: Elev:

Location Remarks:

Method Used to Discover Problem:

Brief Problem Description:
RCS Hydrogen concentration is on a abnormal decreasing trend, requiring periodic LDST venting and Hydrogen Makeup. (FIP Team Established)

Detail Problem Description:
Establishing and maintaining Unit 2 RCS Hydrogen concentration has been abnormal since unit startup. PIP's 99-5212 (LDST pressure increase) and 99-5232 (Unit 2 RCS hydrogen concentration Low) were generated to address this problem. Via these PIP's, it was determined that isolating nitrogen lines to Unit 2 LDST and returning Unit 2 RCS hydrogen concentration to normal, by venting and hydrogen makeup, would correct this problem. However, Unit 2 RCS hydrogen concentration continues to decrease and warrants establishing a FIP team.

Events.

On 12-18-99 PIP 99-5212 was generated due to LDST pressure increase to 20 lbs from an unknown source. This pressure spike occurred at the same time hydrogen additions were made to Unit 1 and Unit 3. Enclosure 4.1 of BTO-10 was initiated to allow chemistry sampling of the LDST to determine if hydrogen gas in-leakage was the cause of the pressure increase. Chemistry was able to analyze the LDST gas space for hydrogen (<.2%), but was not able to analyze for nitrogen (ONS chemistry does not have the equipment to perform this analysis). Hydrogen in-leakage was ruled out by the sample results.

Unit 2 LDST had to be vented to maintain pressure within guidelines.

On 12-19-99 at 0325 Chemistry sampled Unit 2 RCS for Hydrogen concentration. Hydrogen concentration was out-of-spec low (20.6 :/kg). Chemistry issued PIP 99-5232 to document this problem. Operations worked to increase RCS hydrogen levels by LDST venting and Hydrogen additions. The concentration did not increase as would be expected based on the number of hydrogen additions and the pressure of the LDST. Subsequent sampling proved hydrogen concentration continued to decrease.

Problem Investigation Process
Oconee Nuclear Station

Nitrogen lines to Unit 2 LDST were isolated due to possible in-leakage flow paths. This seemed to stop the LDST pressure increase. LDST nitrogen concentration could not be determined via sampling, but other parameters indicated nitrogen in-leakage was likely. Nitrogen can react with hydrogen via radioloysis in the RCS to form ammonia. Unit 2 ammonia levels were abnormally high (>2ppm). Also, if nitrogen in-leakge was occurring, hydrogen concentration would be low due to stratification. Through the investigation (PIP 99-5212) it was determined nitrogen valves 2N-1, 2N-3, N-5, 1N-69, and 3N-28 had seat leaks.

Due to the low RCS Hydrogen concentration and the low LDST gas space Hydrogen purity, it was determined several LDST vents and hydrogen makeup would be required to return the system to normal RCS hydrogen concentrations. "Henry's Law" was used as a basis for this determination.

Vents/hydrogen makeups were performed and the RCS hydrogen concentrations returned within normal range. Over a 2-week period, Unit 2 RCS hydrogen concentration decreased from approx. 35 cc/kg to 26.8 cc/kg.

A FIP team was formed.

Originated By: WAP7317: PERRY JR, WILLIAM A Team: BLN7354 Group: CHM Date: 01/20/2000

Other Units/Components/Systems/Areas Affected(Y,N,U): N

Industry Plants Affected(Y,N,U): U

Immediate Corrective Actions:
'P team formed.

Originated By: WAP7317: PERRY JR, WILLIAM A Team: BLN7354 Group: CHM Date: 01/20/2000

Immediate Corrective Action Documents / Work Orders:

	Indiv	Team	Group	Date
Problem Identified By:	WAP7317	BLN7354	CHM	01/20/2000
Problem Entered By:	WAP7317	BLN7354	CHM	01/20/2000

Screening
Is the Problem Significant? No Action Category: 3

Significance Codes:
N/A N/A

OEP No:

Other Report Nos:

Event Codes:
F3 Equipment Out of Norm
F7a System Chemical Specification/Internal

Screening Remarks:
"This PIP was originally classified as a Category 2 PIP at management's description to assure a prompt and thorough investigation and

corrective action by a FIP Team that was organized. This team determined the physical and chemical root causes for the initial event and established corrective actions to prevent recurrence. At this point, the appropriate management representatives determined that no further root cause analysis was needed and this effort was terminated without completion of all aspects of the analysis as required by the root cause process. For this reason, Rick Bond agreed to downgrade the PIP to a Category 3 which is in conformance with the event classification criteria."

Last Updated By: RWV1470: VASSEY, RAY W Team: RTB7310 Group: SRG Date: 04/04/2000

This event has been reviewed by the CST and found to meet the MSE significance criteria.

Screening members present for this review: RD Burns (MNT & WCG), and Mike Pruitt (OPS)

Originated By: RWV1470: VASSEY, RAY W Team: RTB7310 Group: SRG Date: 01/24/2000

Assignments:

Responsible Groups(s) for Problem Evaluation:	CHM	Chemistry
Responsible Group for Present Operability:	N/A	
Responsible Group for Past Operability:	N/A	
Responsible Group for Reportability:	N/A	
Responsible Group for Overall PIP Approval:	CHM	Chemistry

Signature Type	Indiv	Team	Group	Date
Screened By:	RWV1470	RTB7310	SRG	04/04/2000

Present Operability

Responsible Group: RGC Status: NotRequired

Sys/Comp Operable? (Y,N,C,E,T):

Required Mode:

Comments:

No Current Signatures For This Section

Past Operability:

Responsible Group: RGC Status: NotRequired

Sys/Comp Operable?(Y,N,C,E,T):

Required Mode:

Comments:

Problem Investigation Process
Oconee Nuclear Station

No Current Signatures For This Section

Reportability

Responsible Group: Status:

Problem Reportable(Y,N,E):

Reportable Per:

Comments:

No Current Signatures For This Section

Investigation Report:

Responsible Group: Act Date:

Investigator: Group:

Due Date:
Date Due to VP or Sta. Mgr:
Date Regulatory or Agency Rpt Due:
Date Investigation Report Approved:

NRC Cause Codes:

Problem Evaluation

Event	Cause Code	Cause Description	Primary	Causing Groups
F7a	B4d	Inadequate documentational provisions	Yes	OPS
F7a	I4f	Relation of task to overall plant operations	Yes	OPS

Problem Evaluation From: Resp. Group: CHM Status: Closed OEDB Checked: Yes

Last Updated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Root Cause Executive Summary

Title of Event or Issue: Unit 2 RCS Hydrogen Control

PIP Number: O-00-00255

Date of Event or Identification of Issue: January 19, 2000

Problem Investigation Process
Oconee Nuclear Station

Date Report Completed:

Root Cause Investigator: Dean Cantrell / Bryon Norris

Description of Event/Issue:

At 0325 on December 19, 1999, the Unit 2 hydrogen concentration was reported to be out-of-specification (low). The hydrogen concentration in the RCS (Reactor Coolant System) was analyzed to be 20.6 cc/kg. The operating limit for hydrogen during MODES 1 and 2 is 25-50 cc/kg. The required corrective actions (venting of the LDST (letdown storage tank) and additions of hydrogen to the LDST) restored the hydrogen concentration within specification; however, the hydrogen concentration continued to follow an abnormal decreasing trend from January 11, 2000 until January 19, 2000.

Summary of Root Cause:

The root cause of the decreasing hydrogen concentration in the RCS was due to a large fixed volume of nitrogen in the RCS and PZR. The large amount of nitrogen present in the RCS and PZR was the result of not sufficiently purging nitrogen from the PZR during steam bubble formation on startup due to inadequate procedural guidance.

Planned Corrective Actions:

Remedial/Immediate Corrective Actions

1. Review offsite dose impact by venting (possibly multiple vents) the LDST through the GWD system and subsequent waste gas tank releases.
2. Vent the LDST through the GWD System and add hydrogen to maintain concentration > 30 cc/kg in the RCS.
3. Maintain LDST pressure > 40 psig.
4. Verify hydrogen analysis accuracy.

Interim Corrective Actions

1. Evaluate the presence of a fixed inventory of nitrogen in the PZR.
2. Evaluate the potential for nitrogen leakage into the LDST/RCS/PZR.
3. Evaluate source terms for air in-leakage into the LDST/RCS.
4. Review hydrogen removal 'terms'.
5. Verify LDST pressure indication accuracy.
6. Review outage modifications and maintenance activities associated with the nitrogen header.
7. Evaluate the potential for other gas in-leakage.
8. Evaluate lithium-6 producing tritium (hydrogen generation or reduction).
9. Develop and implement a plan for nitrogen removal.

Corrective Actions to Prevent Recurrence:

1. Revise Operations procedure OP/1,2,3/A/1103/02, "Filling and Venting the RCS", to include additional indicators (e.g. Quench Tank level) to inform the operator when PZR is adequately vented.
2. Revise Operations procedure PT/1,2,3/A/0201/004, "2RC-66 Stroke Test" to include a caution (or note) as to the expected pressure increase on the Quench Tank during PORV operation (to validate adequate PZR venting occurs during steam bubble formation).
3. Train all Operators on indications for ensuring proper PZR venting and bubble formation.
4. Revise Operations lesson plan based on # 3.
5. Change Chemistry procedure to require PZR liquid sample to verify nitrogen concentration - needs to be timed with bubble formation.
6. Transmit operating experience from this event (INPO, Nuclear Network, etc.).

Additional PIPs created as a result of the Problem Evaluation

None

Problem Investigation Process
Oconee Nuclear Station

Problem Identification

Establishing and maintaining Unit 2 RCS Hydrogen concentration had been abnormal since unit startup. PIPs 99-5212 (LDST pressure increase) and 99-5232 (Unit 2 RCS hydrogen concentration low) were generated to address this problem. The Unit 2 RCS hydrogen concentration was returned to specification by venting and hydrogen makeup, but the hydrogen concentration continued to trend downward over the next several days.

Data Collection

On 1/19/00, a FIP Team was assembled to investigate the depressed RCS hydrogen concentration on Oconee Unit 2. The team consisted of the following members:

Bryon Norris (Team Leader) - Chemistry
Mike Leighton - Engineering
Dean Cantrell - Chemistry
Mike Garrison - Chemistry
Bob Dobson - SRG
Doug Berkshire - Radiation Protection
Larry Atkinson - Work Control
Joe Price - Operations
Ken Johnson - General Office Chemistry

Since the Unit 2 start-up in December of '99, RCS hydrogen gradually trended down and on several occasions dropped below the minimum specification of 25 cc/kg. Venting of the LDST only temporarily raised the RCS hydrogen concentration. Operation below 15 cc/kg would require the unit to be brought to cold shutdown.

Sequence of Events:

Unit 2 Startup

12/10/99, Operations closes the vent valve on the LDST. Chemistry initiates venting of the LDST through their sample rig to increase hydrogen purity in the LDST. Operations adds hydrogen to the LDST. LDST purity (liquid) is analyzed to be 82% hydrogen.

12/11/99, Venting of the LDST through the Chemistry sample rig continues. Operations continues makeup with hydrogen. Hydrogen is maintained at 82-86% in the LDST (liquid).

12/12/99, Chemistry continues to vent the LDST. Operations continues hydrogen additions. LDST hydrogen is analyzed to be 79% in the liquid. RCS dissolved hydrogen is 30.5 cc/kg. The operating specification is 25-50 cc/kg.

12/13/99, Operations makes the final hydrogen addition to the LDST prior to zero power physics testing (ZPPT).

12/14/99, Operations has initiated continuous PZR spray. Hydrogen in the LDST is analyzed to be 48% (liquid). The RCS hydrogen is 21 cc/kg. The RCS specification is > 15 cc/kg prior to going critical. Venting of the LDST by Chemistry is secured prior to ZPPT which begins at 2312.

12/15/99, At 0407 Operations vents the LDST due to approaching the pressure limit. Operations vents the LDST again at 0741 with no corresponding hydrogen addition. ZPPT is complete at 1630. Chemistry initiates venting of the LDST through the sample rig to raise the hydrogen purity in the LDST. No measurable increase was noted. The RCS hydrogen concentration was analyzed to be 21 cc/kg. Operations makes up to the LDST with hydrogen.

12/16/99, Operations makes 2 hydrogen additions to the LDST. RCS hydrogen is measured at 25 cc/kg. The specification for steady-state power operation is 25-50 cc/kg.

Problem Investigation Process
Oconee Nuclear Station

12/17/99, Operations adds hydrogen to the LDST. RCS is measured at 25 cc/kg.

12/18/99, Operations begins PZR spray. Continuous spray is maintained from 1752 until 0400 on 12/19/99. Operations vents the LDST at 2023 due to increasing pressure. PIP 99-5212 was generated due to pressure approaching the upper limit of the LDST vs. level curve. RCS hydrogen is measured at 25 cc/kg.

12/19/99, Operations vents the LDST and adds hydrogen. Pressurizer spray is terminated. RCS hydrogen drops to 20 cc/kg. At 0523 the nitrogen header is isolated due to suspected nitrogen leakage into the LDST. At 1726 Operations vents the LDST. At 2259 Operations vents the LDST again and adds hydrogen. Unit 2 is at 100% power. PIP 99-5232 is generated due to hydrogen being out-of-specification (low).

12/20/99, Operations adds hydrogen to the LDST. RCS hydrogen is 21 cc/kg.

12/21/99, LDST purity (liquid) is analyzed to be 43-45 %. Operations vents the LDST at 1942 to maintain pressure limits.

12/22/99, At 0112 Operations vents the LDST to maintain pressure limits. The reactor trips at 0403. The hydrogen purity in the LDST (liquid) trends at ~ 435 through 12/23.

12/23/99, Operations vents the LDST and adds hydrogen.

12/24/99, Operations vents the LDST and 2 hydrogen adds are made. The reactor trips.

12/25/99, For the period of 12/25/99 until 1/2/00, 9 hydrogen additions were made to the LDST. No LDST venting was done. The unit reached 100% power during this period. The hydrogen was maintained just above 25 cc/kg.

'4/00, LDST hydrogen purity is 41% (liquid).

1/5/00, LDST hydrogen purity is 34% (liquid). Operations vents the LDST and adds hydrogen.

1/6/00, Operations adds hydrogen to the LDST. The purity of the LDST (liquid) is measured to be 40%.

1/7/00, Operations vents the LDST and adds hydrogen.

1/8/00, Operations vents the LDST. Two hydrogen additions are made.

1/9/00, For the period of 1/9/00 until 1/19/00, there were 12 hydrogen additions to the LDST. The RCS hydrogen concentration was raised to 32 cc/kg but trended down to 28 cc/kg (and decreasing) by 1/19/00.

1/19/00, FIP team established.

Background:

RCS hydrogen concentration is normally controlled by the pressure of hydrogen gas in the LDST gas space. Letdown from the RCS is sprayed into the LDST. Henry's law determines the concentration of the dissolved hydrogen in the LDST liquid as follows:

Xi = Pi/H

Where: Xi is the mole fraction of the gas dissolved in the liquid,
Pi is the partial pressure of the gas in the vapor, and
H is the temperature dependant henry's law constant for the gas,

This equation is simplified by calculating a henry's law coefficient, h as follows:

Ci = Pi/H

Where Ci is the dissolved gas concentration in the liquid in cc/kg,
Pi is the partial pressure of the gas in psia, and
H is the henry's law coefficient with units of psia/(cc/kg).

For Hydrogen at 100 degrees F, h = 0.9058 psia/(cc/kg).

Additional facts gathered during the investigation:

On 1/19/00 at 00:15, RCS hydrogen was measured at 28.0 cc/kg. Using Henry's law to calculate the hydrogen pressure in the LDST yields 25.4 psia; however the LDST pressure was approximately 44 psig (58.7 psia). This indicates a significant quantity of dilution gas present in the LDST.

Data for LDST pressure and level was retrieved from the PI Server and used to calculate the total inventory of gas in the LDST vapor space using the following equations:

V (LDST,liq (gal)) = (31.3344 X L (LDST, liq (in)) + 677.4262

V (LDST,gas (gal)) = (600 X 7.4805) - V (LDST,liq (gal))

V (LDST,gas (cc)) = 3785.412 X V (LDST,gas (gal))

INV (LDST,gas (cc)) = V (LDST,gas (cc)) X ((P (LDST,psig) + 14.7)/14.7)

INV (LDST,gas (SCF)) = INV (LDST,gas (cc)) X 0.00003531467

Plotting the LDST gas inventory revealed that the rate of decrease was significantly less than the rate of decrease in either Unit 1 or 3, or in Unit 2 during a comparable time frame during the start of the previous fuel cycle. It was concluded that there appeared to be an input of nitrogen into the LDST, either from nitrogen in-leakage into the system, or from a fixed inventory already within the RCS (i.e., the Pressurizer steam space).

Collection of stripped gas samples from the RCS was restarted beginning on 1/21/99. The percent hydrogen values confirmed that the hydrogen purity in RCS was much less than the normal approximate 85%. Base on Henry's Law, the purity of hydrogen in the LDST should be approximately equal to the % hydrogen measured in the stripped gas samples. The stripped gas sample collected on 1/21/00 at 12:55 was 53.22% hydrogen.

The gas stripping procedure was changed to allow collection of stripped gas samples from the pressurizer liquid. A sample was collected on 1/21/00 at 13:25. The dissolved hydrogen concentration was 21.8 cc/kg at 48.22 % hydrogen. This indicated that the hydrogen purity in the pressurizer was lower than the purity in the rest of the reactor coolant system and the LDST gas space.

No credible mechanisms for leakage of nitrogen into the pressurized RCS or PZR were found.

The trend of the LDST gas inventory was reviewed from 12/10/99 when the vent to the LDST (GWD-19) was closed. The major features were 1) drops in inventory that coincided with venting of the LDST via the chemistry vent rig early in the period, or venting by Operations to the GWD header, 2) sudden increases in inventory coinciding with hydrogen additions made by operations, and 3) several periods of steady significant increasing inventory that occurred during the following approximate time frames:

12/13/99 05:00 to 12/13/99 09:00
12/13/99 17:00 to 12/13/99 19:00
12/13/99 20:00 to 12/13/99 23:00
2/14/99 11:00 to 12/15/99 08:00
12/18/99 12:00 to 12/19/99 00:00

Pressurizer spray data from OILS/PI (for RC-1) was reviewed as well and all of the above time periods match with the periods during which PZR spray was maximized.

An Excel spreadsheet was developed to model the pressurizer gas space nitrogen concentration during venting to establish a steam bubble. The initial conditions were:

PZR Vapor Space Volume : 962 cu. ft.
PZR Vapor pressure: 54.7 psia
PZR Vapor temperature : 100 F

The vapor pressure of water at the vapor space temperature was calculated using the steam table library. The balance of the pressure was assumed to be the starting nitrogen inventory, the temperature was increased to induce a pressure step increase of 0.1 psia. The additional water vapor was mixed with the nitrogen inventory and then enough of the resulting gas was removed to return the original pressure. The amount of nitrogen removed was subtracted from the PZR inventory, and the water vapor was assumed to be condensed and totaled. This step was repeated to yield the SCF of nitrogen inventory in the PZR as a function of gallons of condensed steam that would be added to the Quench Tank inventory. For conservatism, the nitrogen dissolved in the PZR and RCS liquid inventory was ignored.

(see graph in original FIP report)

Data for the positions of valves GWD-13 and GWD-17 was retrieved to find the time that the purging of the PZR steam space to the Quench Tank was performed. This started on 12/9/99 at 14:51 and ended on 12/9/99 at 17:25. Data for Quench Tank level was retrieved from the PI Server and indicate that the Quench Tank level increased from 83.3" on 12/9/99 at 14:51 to 83.86" at 17:25. This would indicate 19.6 gallons of level increase. Using the Excel model, this would leave at least 400 SCF of nitrogen in the steam space.

The Quench Tank volume increases for previous steam bubble purges have been much larger than the 19.6 gallons for this one.

uench Tank pressure data is not available on the PI Server; however a strip chart was retrieved that included the pressure during the venting of the pressurizer. A PORV operability test was performed later on 12/9/99 at about 1830. At that time, the Quench Tank pressure increased approximately 14 psig (23% of 60 psig full scale). The Quench Tank pressure increase from the same test on Unit 1 on 6/30/99 was less than 1 psig (<1% of 60 psig full scale). This would indicate that the nitrogen concentration in the Pressurizer steam space was still high after the steam bubble formation.

A total system inventory of hydrogen and nitrogen was calculated using the chemistry samples from the RCS on 1/25/99 at 11:50 and the PZR liquid on 1/25/99 at 12:25 with the following results:

	Nitrogen (SCF)	Hydrogen (SCF)
RCS Liquid	152.9	270.7
LDST Gas	203.5	346.5
PZR Vapor	362.5	303.3
Total	718.9	920.5

Failure Modes and Failure Scenario

Using facts derived from field observations, work history, PIP Database, OEDB searches, and work experience, the FIP Team developed a list of potential failure modes. Each failure mode was evaluated for credibility.

Potential Failure Modes Matrix - See original FIP report for Table with specific data.

By evaluation of probable sources of Nitrogen in the RCS, the FIP Team developed the following failure scenario.

Failure Scenario for the Low Hydrogen concentration in Unit 2 RCS:

Unit 2 was in the startup mode of operation after refueling. The startup sequence had progressed to the point in which the RCS was filled with a Nitrogen overpressure of ~ 35 psig on the Pressurizer (PZR). This overpressure allowed the RCS to remain filled until the procedure for Pressurizer Operation was performed that replaced the Nitrogen overpressure with Steam pressure. The sequence of events in forming a steam bubble in the Pressurizer involved both Operator knowledge in determining when a steam bubble existed and

Nitrogen was removed, and guidance of the procedure in aligning Pressurizer heaters and valves to perform the task. The objective in forming a steam bubble is to allow the heaters to form steam in the Pressurizer (indicated by an increase in pressure) and vent the increase in pressure to the Quench Tank. The increase in pressure in the Quench Tank is then vented to the Gaseous Waste Header. The initial venting to the Quench Tank results in a pressure increase with little to no water level increase since mainly Nitrogen is the gas removed from the Pressurizer. As the process continues, the contents removed from the Pressurizer to the Quench Tank contain an increasing amount of steam and a decreasing amount of Nitrogen. The evolution is completed when the venting is all steam pressure in the Pressurizer and no Nitrogen. This is indicated by the Quench Tank trending a water level increase (due to condensing steam) and no pressure increase indicating no Nitrogen in the Pressurizer gas (since Nitrogen is a non-condensable gas and would only increase pressure in the Quench Tank).

The procedure provides a note to the Operator to throttle open the vent from the Pressurizer and open the vent from the Quench Tank as a preferred method of performing the task however; it is also states cycling of the valves can also be used. The sequence of performing the task states to throttle the Pressurizer vent and open the Quench Tank vent. Information is provided to the Operator that when the Pressurizer vent and the Quench Tank vents are both open with Quench Tank level increasing and only pressure increasing due to level increase in the Quench Tank, the task is completed and a steam bubble is present in the Pressurizer.

The data obtained from the computer for operation of the Pressurizer vent and Quench Tank vent during the evolution indicated the Pressurizer vent was throttled open and the Quench tank vent was cycled open and closed. During the final repetition of forming a steam bubble the Pressurizer vent was only throttled open with the Quench Tank vent fully open. At no time was the Pressurizer vent fully open with the Quench Tank vent also fully open. In addition, the increase in Quench Tank level obtained by data from the computer and paper recorder indicated a total level increase of 0.5 inches.

An analytical model was used to determine the required volume of gases (Nitrogen and steam) which would be required to enter the Quench Tank in order to remove all the Nitrogen from the Pressurizer. The results indicated approximately 2 inches of water increase for all Nitrogen to be removed. At 0.5 inches of water level increase, the Nitrogen remaining in the Pressurizer would be at atmospheric pressure (0 psig) which would not have caused an increase in Quench Tank pressure above 0psig since the Quench Tank was connected to the Gaseous Waste Header maintained at approximately 0 psig.

Supporting the scenario that all Nitrogen was not removed from the Pressurizer during formation of the steam bubble was an operability test performed later on the Pressurizer Power Operated Relief valve (PORV). During this test the PORV is opened which directs steam from the Pressurizer to the Quench Tank in order to insure the PORV can be opened and closed. The test is completed successfully when a level increase is observed in the Quench Tank. By retrieving the recorder chart from the Quench Tank during the PORV test, it is recorded a pressure increase of greater than 15 psig occurred during the test. This increase in pressure would occur if Nitrogen gas were still present in the Pressurizer since steam would condense raising water level and Nitrogen would increase pressure (the Quench Tank vent is closed during the test). After the test, the pressure was removed by cycling the Quench Tank vent. As observed on other startups when Pressurizer steam bubbles were formed and the PORV was tested later, pressure increased 1 to 3 psig during the test.

Finally, the evolution of forming a Pressurizer steam bubble normally requires approximately 6 hours as obtained using data from the computer and the Unit logbook. The evolution on the Unit 2 startup was performed in approximately 2 hours.

Cause Determination

The root cause for the improperly formed steam bubble in the pressurizer was determined to be:

Inadequate Procedure guidance. Enclosure 4.14 (Establishing PZR Steam Bubble And RCS Final Vent) of procedure OP/2/A/1103/02 is used to establish a steam bubble in the pressurizer. Step 2.9 directs the Operator to throttle open 2GWD-17, but the procedure never directs the Operator to fully open the valve. The only guidance given to the Operator to determine when a steam bubble is formed is contained in a note. The note states: With 2GWD-17 and 2GWD-13 open and QT pressure changing only due to QT level increase, PZR Steam Bubble Formation should be complete. If 2GWD-17 is not FULLY open, it is possible to meet the conditions of this note and not have vented all the nitrogen from the pressurizer.

A contributing cause to this incident is Lack of Operator Training. The proper method to form a steam bubble is not taught in Operator Training. Training does not cover the indications to use to verify a proper steam bubble has been formed.

Benchmarking

Oconee Unit 2 Hydrogen Concentration FIP
Operating Experience Report
January 25, 2000

Included in this report are summaries of OEDB and PIP searches for operating experience pertinent to the ONS U2 problem documented initially in PIP O-99-5212. The following bullets capture the essential elements of these reviews:
· At least two other utilities have experienced problems maintaining hydrogen concentration within specifications, due to excess nitrogen in the RCS. Both of these utilities resolved their problem by purging the LDST gas space and replacing it with hydrogen.
· Oconee Unit 2 had a similar problem in 1997 documented in PIP O-97-1133.
· No operating experience indicating a potential source for such a problem other than nitrogen was found.

Oconee Unit 2 Hydrogen Concentration FIP
Operating Experience Report
Telephone Reports
January 25, 2000

· Larry Wilson, GO Chemistry, surveyed other B&WOG chemistry contacts for information about problems maintaining hydrogen concentration. As of the date of this writing, the only problem noted was at ANO - a leak on one of the RCP's went to their quench tank and displaced nitrogen (used as a blanket in the quench tank) into the RCS. Contact at ANO is Larry MCCollum (501-558-5492).
· Harry Williams of FTI provided the name of Merl Bell as a contact at FTI.
Merl Bell (804-832-3516) indicated that Davis Besse had experienced a problem following a startup after a mid-cycle outage - there was nitrogen in the LDST due to an inadequate purge of the LDST. Several vents and hydrogen additions resolved the problem. Contact at Davis Besse is Rich Edwards (419-832-3516) - attempts to contact have been unsuccessful.
· Henry Lowery provided information from W Stanley of Amergen Energy indicating that TMI has experienced problems with maintaining hydrogen concentration, solved by purging the makeup tank gas space and replacing the hydrogen and keeping the pressure toward the high end of the band. Attempts to contact Gary Chevalier (717-948-8136) have been unsuccessful.

Operability, Generic Applicability, and Transportability

There are no present or past operability concerns associated with this event. The only issue related to operability would be the quality of the steam bubble in the PZR and the ability of the PZR to perform it's intended function. Sample analyses to calculate the percent hydrogen and percent nitrogen in the RCS and PZR showed no operability concerns. The total gas of the RCS and PZR was maintained well below 100 cc/kg.

This event affects all 3 Units at Oconee since the fixed volume of nitrogen in the PZR and RCS could be produced due to inadequate formation of the PZR steam bubble. This event is also applicable to McGuire and Catawba Nuclear Stations. Similar problems could result by entrapment (or input) of nitrogen into the LDST or VCT (volume control tank).

Recent data trending indicates that the hydrogen concentration in the Unit 2 RCS has stabilized. The concentration has been maintained in the 30-35 cc/kg range.

Indications for future events would be the inability to maintain hydrogen concentration. During this event many vents of the LDST by Operations were required to maintain hydrogen concentration and this is not typical.

Planned Corrective Actions

Problem Investigation Process
Oconee Nuclear Station

Remedial/Immediate Corrective Actions

1. Review offsite dose impact by venting (possibly multiple vents) the LDST through the GWD system and subsequent waste gas tank releases.
2. Vent the LDST through the GWD System and add hydrogen to maintain concentration > 30 cc/kg in the RCS.
3. Maintain LDST pressure > 40 psig.
4. Verify hydrogen analysis accuracy.

Interim Corrective Actions

1. Evaluate the presence of a fixed inventory of nitrogen in the PZR.
2. Evaluate the potential for nitrogen leakage into the LDST/RCS/PZR.
3. Evaluate source terms for air in-leakage into the LDST/RCS.
4. Review hydrogen removal 'terms'.
5. Verify LDST pressure indication accuracy.
6. Review outage modifications and maintenance activities associated with the nitrogen header.
7. Evaluate the potential for other gas in-leakage.
8. Evaluate lithium-6 producing tritium (hydrogen generation or reduction).
9. Develop and implement a plan for nitrogen removal.

Corrective Actions to Prevent Recurrence:

1. Revise Operations procedure OP/1,2,3/A/1103/02, "Filling and Venting the RCS", to include additional indicators (e.g. Quench Tank level) to inform the operator when PZR is adequately vented.
2. Revise Operations procedure PT/1,2,3/A/0201/004, "2RC-66 Stroke Test", to include a caution (or note) as to the expected pressure crease on the Quench Tank during PORV operation (to validate adequate PZR venting occurs during steam bubble formation).
3. Train all Operators on indications for ensuring proper PZR venting and bubble formation.
4. Revise Operations lesson plan based on # 3.
5. Change Chemistry procedure to require PZR liquid sample to verify nitrogen concentration - needs to be timed with bubble formation.
6. Transmit operating experience from this event (INPO, Nuclear Network, etc.).

List of Attachments

1. Possible Causes Matrix
2. FIP Organization
3. FIP Handbook
4. LDST Inventory Calculation
5. Hydrogen Data
6. LDST Pressure, Level, and Inventory Data
7. Gas Inventory Data
8. RCS Ammonia Data
9. Operating Experience
10. PIPs
11. Autolog Records
12. Work History
13. Miscellaneous Data
14. Quench Tank Temperature, Level, and Pressure Strip Charts
15. Timeline

Originated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Problem Investigation Process
Oconee Nuclear Station

OEDB Comments:
Operating experience is documented in the detailed FIP report shown in the text of the Problem Evaluation.

Originated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Remarks Comments:

Signature Type	Indiv	Team	Group	Date
Accepted By:	WAP7317	BLN7354	CHM	01/24/2000
Assigned To:		BLN7354	CHM	01/24/2000
Due Date:	03/17/2000			
Ready For Approval:	HDC7317	BLN7354	CHM	03/15/2000
Approval Assigned To:	WAP7317	BLN7354	CHM	03/15/2000
Approved By:	BLN7354	BLN7354	CHM	03/16/2000
Concurrence Assigned To:	MWA7315	RTB7310	SRG	03/16/2000

Corrective Actions

CA Seq. No: 1

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	CHM	F7a	A3	B4d

Proposed Corrective Action:

Last Updated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/15/2000

Revise Operations procedure OP/1,2,3/A/1103/02, "Filling and Venting the RCS", to include additional indicators (e.g. Quench Tank level) to inform the operator when PZR is adequately vented.

This corrective action was discussed with Bluford Jones.

Originated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	HDC7317	BLN7354	CHM	03/15/2000
Approval Assigned To:	WAP7317	BLN7354	CHM	03/15/2000
Approved By:	BLN7354	BLN7354	CHM	03/16/2000

General: Outage: Mode:

Other Tracking Processes

Problem Investigation Process
Oconee Nuclear Station

Type Number Text

Actual Corrective Action:
Actual CAC: Status: Open Due Date: 07/18/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/18/2000			
Accepted By:	MAP7314	HRL7353	OPS	03/21/2000
Assigned To:	GBJ1009	DBC7309	OPS	03/22/2000

CA Seq. No: 2

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	CHM	F7a	A3	B4d

Proposed Corrective Action:

Last Updated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/15/2000

Revise Operations procedures PT/1,2,3/A/0201/004, "2RC-66 Stroke Test", to include a caution (or note) as to the expected pressure increase on the Quench Tank during PORV operation (to validate adequate PZR venting occurs during steam bubble formation).

This corrective action was discussed with Bluford Jones.

Originated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	HDC7317	BLN7354	CHM	03/15/2000
Approval Assigned To:	WAP7317	BLN7354	CHM	03/15/2000
Approved By:	BLN7354	BLN7354	CHM	03/16/2000

General: Outage: Mode:

Other Tracking Processes
Type Number Text

ctual Corrective Action:
Actual CAC: Status: Open Due Date: 07/18/2000

Problem Investigation Process
Oconee Nuclear Station

Signature Type	Indiv	Team	Group	Date
Due Date:	07/18/2000			
Accepted By:	MAP7314	HRL7353	OPS	03/21/2000
Assigned To:	DYP7314	DBC7309	OPS	04/03/2000

CA Seq. No: 3

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	CHM	F7a	C4	I4f

Proposed Corrective Action:

Once the Operations procedures have been revised and approved, Operator Training should evaluate the need to train all licensed Operators on venting the pressurizer.

This corrective action was discussed with Cam Eflin.

Per Bill Caudills Not accepted comments. This Action is being redirected to the Operations TPRC. Once training determination is completed,another corrective action will be developed to perform training.

Originated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Signature Type	Indiv	Team	Group	Date
Approval Assigned To:	WAP7317	BLN7354	CHM	03/15/2000
Ready For Approval:	WAP7317	BLN7354	CHM	03/27/2000
Approved By:	WAP7317	BLN7354	CHM	03/27/2000

General: Outage: Mode:

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: Status: Open Due Date: 07/18/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/18/2000			
Accepted By:	MAP7314	HRL7353	OPS	03/27/2000
Assigned To:	ASH0555	RDL3572	OPS	03/27/2000

Problem Investigation Process
Oconee Nuclear Station

CA Seq. No: 4

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	CHM	F7a	C7	I4f

Proposed Corrective Action:

Once the Operations procedures have been revised and approved, Operator Training should revise the associated training material for venting the pressurizer.

This corrective was discussed with Cam Eflin.

Per Bill Caudills Not accepted comments. This Action is being redirected to the Operations TPRC. Once training determination is completed,another corrective action will be developed to perform training.

Originated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Signature Type	Indiv	Team	Group	Date
Approval Assigned To:	WAP7317	BLN7354	CHM	03/15/2000
Ready For Approval:	WAP7317	BLN7354	CHM	03/27/2000
Approved By:	WAP7317	BLN7354	CHM	03/27/2000

General: Outage: Mode:

Other Tracking Processes

Type Number Text

Actual Corrective Action:

Actual CAC: Status: Open Due Date: 07/18/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/18/2000			
Accepted By:	MAP7314	HRL7353	OPS	03/27/2000
Assigned To:	ASH0555	RDL3572	OPS	03/27/2000

CA Seq. No: 5

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
CHM	Closed	CHM	F7a	A3	I4f

Proposed Corrective Action:

Problem Investigation Process
Oconee Nuclear Station

Change Chemistry procedures to require PZR liquid sample to verify nitrogen concentration (needs to be timed with bubble formation).

Originated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	HDC7317	BLN7354	CHM	03/15/2000
Approval Assigned To:	BLN7354	BLN7354	CHM	03/15/2000
Approved By:	BLN7354	BLN7354	CHM	03/16/2000

General: Outage: Mode:

Other Tracking Processes

Type Number Text

Actual Corrective Action:

Actual CAC: A3Status: Closed Due Date: 07/18/2000

CSM 3.10 (Primary Chemistry Sample Frequencies and Specifications) was changed to include sampling requirements for nitrogen on the PZR during Unit startup. This change was approved on 3/29/00.

riginated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/30/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/18/2000			
Accepted By:	WAP7317	BLN7354	CHM	03/16/2000
Assigned To:		HDC7317	CHM	03/16/2000
Ready For Approval:	HDC7317	BLN7354	CHM	03/30/2000
Approval Assigned To:	WAP7317	BLN7354	CHM	03/30/2000
Approved By:	WAP7317	BLN7354	CHM	03/30/2000
Concurrence Assigned To:	MWA7315	RTB7310	SRG	03/30/2000
Concurrence By:	RWVASSEY	RTB7310	SRG	04/03/2000

CA Seq. No: 6

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
INP	Closed	CHM	F7a	C3	B4d

Proposed Corrective Action:

Last Updated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/15/2000

Problem Investigation Process
Oconee Nuclear Station

Transmit operating experience from this event (INPO, Nuclear Network, etc.).

This corrective action was discussed with Wendell Barron.

Originated By: HDC7317: CANTRELL, HOYLE D Team: BLN7354 Group: CHM Date: 03/13/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	HDC7317	BLN7354	CHM	03/15/2000
Approval Assigned To:	WAP7317	BLN7354	CHM	03/15/2000
Approved By:	BLN7354	BLN7354	CHM	03/16/2000

General: Outage: Mode:

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: Status: Open Due Date: 07/18/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/18/2000			
Accepted By:	WHB7121	RTB7310	INP	03/29/2000
Assigned To:	WHB7121	RTB7310	INP	03/29/2000

Final and Overall PIP Approval

Responsible Group: CHM Status: Screened

Signature Type	Indiv	Team	Group	Date
Assigned To:			CHM	01/24/2000
Accepted By:	WAP7317	BLN7354	CHM	01/24/2000

Any Supplemental Concurrence Signatures Above Do Not Affect PIP Closure.

Closure Document Type **Closure Document No**

Attachments

Generic Applicability

Problem Investigation Process
Oconee Nuclear Station

Responsible Group: OEA   Status: NotRequired
GO PIP No:

Assessment Remarks:

Signature Type	Indiv	Team	Group	Date
Assigned To:			OEA	01/24/2000

Failure Prevention Investigation

Quality of CA: Quality of Cause: Resp Group: SRG   Status: Closed

Special Codes:

1) Event   Inapp.Action #   F3   001

Description: Operator did not fully open the valve.

Process:	PO	Process:	
Group:	OPS	Group:	
Sub-Group:	OPS-PROC	Sub-Group:	

O and P Failure Mode: P2
HE Failure Mode: SK5
HE Type: Knowledge Based
Key Activity: ca

Associated Corrective Actions: None

Comments

Signature Type	Indiv	Team	Group	Date
Assigned To:			SRG	01/24/2000
Ready For Approval:	RWVASSEY	RTB7310	SRG	04/04/2000
Approval Assigned To:	RTB7310	RTB7310	SRG	04/04/2000
Approved By:	RWVASSEY	RTB7310	SRG	04/04/2000

Remarks

No Remarks for this PIP.

Maintenance Rule

Responsible Group: MSE   Status: ReadyForApprove

Maintenance Rule SSC

SSC	Description	Risk Significant	Primary System
RC	Reactor Coolant System	No	Yes

Problem Investigation Process
Oconee Nuclear Station

Equipment Group:
Applicable Unit: Unit 2
Functional Failure: No MPFF: No Repetitive MPFF: No

Functional Failure Comments:
This event was not a Maintenance Rule Functional Failure.

There are no Maintenance Rule Functions for Hydrogen overpressure. Additionally, the limits for RCS Hydrogen were not exceeded at any time.

Originated By: MDL0422: LEIGHTON, MICHAEL D Team: SDC3511 Group: MSE Date: 04/06/2000

MPFF Comments:
N/A

Originated By: MDL0422: LEIGHTON, MICHAEL D Team: SDC3511 Group: MSE Date: 04/06/2000

Repetitive MPFF Comments:
N/A

Originated By: MDL0422: LEIGHTON, MICHAEL D Team: SDC3511 Group: MSE Date: 04/06/2000

Reactor Trip: No Safety System Actuation: No Loss of Heat Decay Removal: No
Force Outage Rate or Plant Transient: No Loss Of Spent Fuel: No

Comments:

Signature Type	Indiv	Team	Group	Date
Assigned To:	MDL0422	SDC3511	MSE	02/14/2000
Due Date:	05/04/2000			
Ready For Approval:	MDL0422	SDC3511	MSE	04/06/2000
Approval Assigned To:	SDC3511	SDC3511	MSE	04/06/2000

Problem Investigation Process
Oconee Nuclear Station

End of the Document for PIP No:	O-0-255
The status of this PIP is:	Screened
The duration of this PIP was:	75 days

Problem Investigation Process
Oconee Nuclear Station

PIP Serial No:	Action Category:	LER No:	Other Report:
O-00-00010	2	287-00-01	

Problem Identification

Discovered Time/Date: 18:54 01/03/2000 **Occurred Time/Date:** 12:58 01/03/2000

Unit(s) Affected:

Unit	Mode	%Power	Unit Status Remarks
3	1	100	

System(s) Affected:
SC Stator Cooling

Affected Equipment

(No Equipment Affected)

Location of Problem:
Bldg: TB Column Line: Elev: Basemen

Location Remarks:
Unit 3 Stator Coolant Skid

Method Used to Discover Problem:
Plant observation

Brief Problem Description:
Unit 3 Main Turbine/Reactor Trip due to high Stator Coolant Temp.

Detail Problem Description:
At approximately 12:55 on 1-3-00 a Stator Coolant Panel alarm was received in the Unit 3 Control Room. The basement NLO was paged to investigate the cause of the alarm. Approximately one minute later, generator stator coil temperature alarms were received on the OAC. An SRO from WCC was immediately dispatched to the stator coolant skid and another SRO from WCC was dispatched to the Unit 3 Control Room. The cause for these alarms was later determined to be a failed low temperature signal to valve 3SC-5. This resulted in this valve positioning to bypass the stator coolers. The unit quickly experienced a Stator Coolant Runback due to the high stator coolant temperature. The CR crew responded to the rapidly changing plant conditions by verifying that the plant was running back properly. At approximately 12:58 the CR crew noted that Generated MWe were indicating 0 while Generator MVARs indicated 260. This indication prompted the crew to initiate a manual trip of the Main Turbine which resulted in an automatic trip of the Reactor (Rx power was approximately 55% at the time of the Main Turbine Trip). The crew initiated EP/1800/01 Emergency Operating Procedure at this time. Plant response was normal and no major problems were encountered in stabilizing the plant. The following notifications were made following the trip:

Station Manager, Human Resource Manager, Human Performance Manager, Regulatory Compliance, NRC Residents, 4 hour Non-Emergency Notification made to the NRC, Operations Duty Superintendent, Shift Operations Manager, STA, Site Vice President.

Problems encountered with plant equipment during the trip included the following:

1. Main Turbine Emergency Bearing Oil Pump started unexpectedly
2. 3RIA-35 sample pump tripped and had to be restarted manually (WR# 98110747)
3. 3RIA-42 sample pump tripped and had to be restarted manually (WR# 98110746)
4. 3RIA-54 sample pump tripped and had to be restarted manually. Also the Turb. Building Sump did NOT interlock when this occurred (WR# 98110748).

5. 3A ESV pump tripped and had to be restarted manually (WR# 98110749).
6. 3B CCW Booster Pump tripped and had to be restarted.
7. AHUs 3-32 and 3-33 (Unit 3 East Penetration Room) tripped and had to be restarted.
8. A previous stator coolant leak that had been temporarily fixed began leaking again (WO# 98147578)

An investigation is on-going concerning the root cause of the failure that lead to the unit trip.

Originated By: TRL5928: LEE, TONY R Team: MJD4241 Group: OPS Date: 01/03/2000

Other Units/Components/Systems/Areas Affected(Y,N,U): N

Industry Plants Affected(Y,N,U): N

Immediate Corrective Actions:
The CR crew responded to the rapidly changing plant conditions by verifying that the plant was running back properly. At approximately 12:58 the CR crew noted that Generated MWe were indicating 0 while Generator MVARs indicated 260. This indication prompted the crew to initiate a manual trip of the Main Turbine which resulted in an automatic trip of the Reactor (Rx power was approximately 55% at the time of the Main Turbine Trip). The crew initiated EP/1800/01 Emergency Operating Procedure at this time. Plant response was normal and no major problems were encountered in stabilizing the plant. The following notifications were made following the trip:

Station Manager, Human Resource Manager, Human Performance Manager, Regulatory Compliance, NRC Residents, 4 hour Non-Emergency Notification made to the NRC, Operations Duty Superintendent, Shift Operations Manager, STA, Site Vice President.

Problems encountered with plant equipment during the trip included the following:

1. Main Turbine Emergency Bearing Oil Pump started unexpectedly
2. 3RIA-35 sample pump tripped and had to be restarted manually (WR# 98110747)
3. 3RIA-42 sample pump tripped and had to be restarted manually (WR# 98110746)
4. 3RIA-54 sample pump tripped and had to be restarted manually. Also the Turb. Building Sump did NOT interlock when this occurred (WR# 98110748).
5. 3A ESV pump tripped and had to be restarted manually (WR# 98110749).
6. 3B CCW Booster Pump tripped and had to be restarted.
7. AHUs 3-32 and 3-33 (Unit 3 East Penetration Room) tripped and had to be restarted.
8. A previous stator coolant leak that had been temporarily fixed began leaking again (WO# 98147578)

An investigation is on-going concerning the root cause of the failure that lead to the unit trip.

Originated By: TRL5928: LEE, TONY R Team: MJD4241 Group: OPS Date: 01/03/2000

Immediate Corrective Action Documents / Work Orders:

	Indiv	Team	Group	Date
Problem Identified By:	TRL5928	MJD4241	OPS	01/03/2000
Problem Entered By:	TRL5928	MJD4241	OPS	01/03/2000

Screening
Is the Problem Significant? Yes Action Category: 2

Significance Codes:

3	Any Unit unplanned outage, oper at sign.

Problem Investigation Process
Oconee Nuclear Station

OEP No:

Other Report Nos:

Event Codes:

F2	Equipment Failure (Important Component)
G1	Reactor Trip
L	Power Reduction - Unscheduled
N3	Reactivity Management Precursor

Screening Remarks:
Per TA Ledford, SD Capps is leading the Root Cause Team and should have the problem evaluation. This PIP is rescreened to MSE.

Last Updated By: SNS3927: SEVERANCE, SANDRA N Team: TDC7309 Group: MSE Date: 01/05/2000

This event has been reviewed by the CST and found to meet the MSE significance criteria.

Screening members present for this review: Sandy Severance (ENG), Sammy Oates (MNT & WCG), and Jean Miller (OPS).

riginated By: RWV1470: VASSEY, RAY W Team: RTB7310 Group: SRG Date: 01/04/2000

Assignments:

Responsible Groups(s) for Problem Evaluation:	MSE	Mech. Sys/Equip
RGC Regulatory Compliance		
Responsible Group for Present Operability:	N/A	
Responsible Group for Past Operability:	N/A	
Responsible Group for Reportability:	RGC	Regulatory Compliance
Responsible Group for Overall PIP Approval:	OPS	Operations

Signature Type	Indiv	Team	Group	Date
Screened By:	SNS3927	TDC7309	MSE	01/05/2000

Present Operability

Responsible Group: RGC Status: NotRequired

Sys/Comp Operable? (Y,N,C,E,T):

Required Mode:

'omments:

No Current Signatures For This Section

Problem Investigation Process
Oconee Nuclear Station

Past Operability:

Responsible Group: RGC Status: NotRequired

Sys/Comp Operable?(Y,N,C,E,T):

Required Mode:

Comments:

No Current Signatures For This Section

Reportability

Responsible Group: RGC Status: Closed

Problem Reportable(Y,N,E): Y

Reportable Per: 10CFR 50.72 and 73

Comments:

This is a reportable reactor trip (actuation of RPS) per 10CFR 50.72(b)(2)(ii) and 50.73(a)(2)(iv). The 50.72 notification was made by OPS.

Originated By: RPT7314: TODD, RANDALL P Team: LEN2127 Group: RGC Date: 01/04/2000

Signature Type	Indiv	Team	Group	Date
Assigned To:	RPT7314	LEN2127	RGC	01/04/2000
Ready For Approval:	RPT7314	LEN2127	RGC	02/03/2000
Approval Assigned To:	LEN2127	LEN2127	RGC	02/03/2000
Approved By:	LEN2127	LEN2127	RGC	02/08/2000

Investigation Report:

Responsible Group: RGC Act Date: 01/03/2000

Investigator: JVW2844 Group:RGC

Due Date:
Date Due to VP or Sta. Mgr:
Date Regulatory or Agency Rpt Due: 02/02/2000
Date Investigation Report Approved: 02/02/2000

NRC Cause Codes:

Problem Evaluation

Problem Investigation Process
Oconee Nuclear Station

Event	Cause Code	Cause Description	Primary	Causing Groups
F2	M	Design Configuration and Analysis The design layo	Yes	CEN
F2	M	Design Configuration and Analysis The design layo	Yes	MSE

Problem Evaluation From: Resp. Group: MSE Status: Closed OEDB Checked: Yes

Root Cause Failure Analysis Report

Unit 3 Stator Cooling Temperature Controller Failure

Revision # 0

PIP# O-00-00010

Root Cause Executive Summary

Title of Event or Issue: Unit 3 Stator Cooling Temperature Controller Failure

PIP Number: O-00-00010

Date of Event or Identification of Issue: January 3, 2000
Date Report Completed: January 20, 2000

Root Cause Investigator: Ted Royal, Mentored by Milton Addis

Description of Event/Issue:

At approximately 12:55 hrs. on 1-3-00, a Stator Cooling (SC) trouble alarm was received in the Unit 3 Control Room. Approximately one minute later, generator stator coil temperature alarms were received on the OAC. The unit quickly experienced a Stator Coolant runback (Setpoint = 176 deg. F) due to the high stator coolant temperature. The maximum stator coolant outlet temperature reached was 187 deg. F. At approximately 12:58 hrs., the control room crew noted that Generated MWe were indicating 0 while Generator MVARS indicated 260 MVARS. This was an unexpected condition and in order to protect the electric generator from high temperatures, at approximately 12:58 hrs., the control room crew initiated a manual trip of the Main Turbine. This resulted in an automatic trip of the Reactor (Rx power was approximately 55% at the time of the Main Turbine Trip). Initial investigations showed that the temperature controlled, three way proportioning valve, 3SC-5, was positioned such that the SC coolers were bypassed. This caused the stator coolant and generator temperatures to increase.

This root cause report will document the cause for the increase in the stator coolant temperature that led to the Unit 3 runback.

Summary of Root Cause:

A stator coolant runback can occur due to either low generator winding inlet pressure or high generator winding outlet temperature. An

exhaustive review of all the potential failure modes that could lead to either of these runback conditions resulted in a conclusion that the cause of the Unit 3 runback was a failure of temperature controller 3SCTT0098. Specifically, the failure was due to a problem with the capillary tube for the transmitter. This controller failed low, outputting a low stator coolant temperature signal. This low SC temperature signal caused 3SC-5 to go to the full SC cooler bypass position, thus removing cooling from the stator cooling system. Additional testing of the controller and follow up with the controller manufacturer (General Electric and Fischer-Porter) will be used to determine the exact cause for temperature controller capillary failure and to specify any future additional corrective actions.

Planned Corrective Actions:

IMMEDIATE: The following immediate actions were taken as a result of this failure:

A detailed review of the Unit 3 Electric Generator's response to the loss of stator cooling was performed on 1/3/2000 to determine if the response was as expected and within limits. Generator temperatures, rectifier temperatures, stator cooling conductivity, and the runback response were examined. The conclusion of this review was that there were no thermal/mechanical concerns with the U3 Generator response or reliability.

The faulty temperature controller, 3SCTT0098, was replaced and calibrated per vendor recommendations and Duke Tubing Installation Specifications which included proper tubing bend radii.

A tubing tray was installed to support the temperature element bulb and capillary line. NOTE: The original installation of the controller capillary line did not have a tubing tray installed.

Signage was posted near the temperature controllers on all three Oconee units stating Unit Trip Potential.

A vibration inspection was performed at 100% full power. Vibration amplitudes were small and there was no resonance at the stator cooling pump vane pass frequency.

LONG-TERM:

The faulty stator coolant temperature controller, 3SCTT0098, will be tested to determine the ultimate cause for the failure.

After testing is completed, any resulting recommended actions will be evaluated and implemented.

Determine if any changes can be made to the temperature controller or 3SC-5 to prevent 3SC-5 from going to the full bypass position for similar instrument failures.

Review and correction of any operating procedures or alarm response guides associated with a stator cooling runback event. Provide any needed OPS training on this issue.

Problem Identification

At 12:55 hrs. on 1-3-2000, Unit 3 experienced a stator coolant runback that ultimately led the control room operators to trip the Main Turbine to protect the generator from rapidly rising temperatures. Initial investigations showed that the three way temperature controlled proportioning valve, 3SC-5, was positioned such that the SC coolers were bypassed. This caused the high stator coolant and generator temperatures to increase, ultimately resulting in the stator coolant runback. A plot of the Unit 3 generator, stator coolant inlet and outlet temperatures, generator gas pressure, and power is provided as attachment # 1.

Data Collection

Problem Investigation Process
Oconee Nuclear Station

Event Narrative

At approximately 12:55 hrs. on 1-3-00 a Stator Cooling Panel trouble alarm was received in the Unit 3 Control Room. Approximately one minute later, generator stator coil temperature alarms were received on the OAC. The unit quickly experienced a SC (Stator Coolant) runback (Setpoint = 176 deg. F) due to the high stator coolant temperature. The maximum stator coolant outlet temperature reached was 187 deg. F. In order to protect the electric generator from high temperatures, at approximately 12:58 hrs. the control room crew initiated a manual trip of the Main Turbine. This resulted in an automatic trip of the Reactor (Rx power was approximately 55% at the time of the Main Turbine Trip). Initial investigations showed that the temperature controlled, three way proportioning valve, 3SC-5 was positioned such that the SC coolers were bypassed. This caused the stator coolant and generator temperatures to increase.

The Failure Investigation Process began after the trip of Unit 3 on January 3, 2000 with the Post Trip Review interviews with key plant personnel. An root cause investigation team was formed with representatives from Operations, CEN, and MSE. The investigation team collected key data/parameters associated with the stator cooling system and the electric generator. They also performed a detailed walkdown of the Stator Cooling skid in the turbine building basement. The vendor, General Electric was contacted to provide input into the root cause investigation.

Facts/Information Associated With the Stator Coolant Runback/Trip

On 1/3/2000, at 1255 hours, Generator Stator Coolant Panel alarms 3SA3-5 (A5), "GN Stator Coolant Panel Trouble" and 3SA4-25 (C-7)7, "MT Turbine Panel Trouble" were received in the Unit 3 Control Room.

At approximately 1256 hours, generator stator coil temperature alarms were received on the OAC. A NLO was dispatched to the Stator Cooling panel in the turbine building basement. Unit 3 had already begun an automatic stator coolant runback.

At 1258 hours, the Operations crew initiated a manual trip when 260 MVARS and 0 Mwatts were indicated in the control room. From interviews, this generator output condition was unexpected and prompted the OPS crew to manually trip the turbine to protect the generator from rising temperatures. The turbine trip resulted in an automatic trip of the reactor.

The NLO dispatched to the Stator Coolant skid and panel in the turbine building observed the following conditions:

· Stator Coolant header flow (locally read) was high (> 650 gpm).
· Stator coolant inlet and outlet temperatures were high.
· Both Stator Coolant pumps were running.

I & E technicians were working on a failed conductivity probe on the stator coolant skid near the capillary tube for the 3SC-5 temperature controller 3SCTT0098. Interviews with the technicians revealed that they could have brushed against the capillary tube, but they did not bend, move or step on the tube (A key point, as the capillary tube was unsupported by hangers or a tubing tray).

The interviews with the I&E technicians also revealed that the capillary tube was specified to have a 3" minimum bend radius. Certain sections of the capillary tube had bends that were close to ½" radius. This required bend radius was confirmed through a review of the General Electric Generator Manual, OM # 2200-0070.

The controller and capillary tube were removed and tested, using an ice bath and heat gun. There was no movement of the controller helical coils or the temperature indicator when the capillary tube-sensing bulb was exposed to the temperature change. The controller linkages did function when manually moved.

Further inspection of 3SCTT0098 by a member of the investigation team confirmed that the capillary tube was not supported and was bent in a tighter radius than the General Electric recommended 3" radius. Copper filings were found on the bulkhead fitting where the capillary enters the controller enclosure.

Discussions with General Electric were held on 1/4/2000. General Electric stated that the described symptoms were indicative of capillary depressurization.

An OEDB review revealed there have been several incidences in the industry associated with instrument failure due to loss of capillary

fill, damage due to unsupported capillary tubing, and damage due to bent controller features. General Electric was contacted to see if they had any specific experience with failures of the stator cooling temperature controller. GE stated that they did not have any specific knowledge of any events.

On 1/4/2000, a new controller, capillary tube, and sensing bulb were installed. A tubing tray was installed using Specification # OSS-0060.00-00-0001. This tray provided support/protection for the new capillary tube. Unit 3 started up on 1/5/2000 and stator cooling temperature control was restored to normal.

Failure Modes and Failure Scenario

Background: Stator Cooling System Overview

The stator cooling (SC) system is designed to remove heat from the generator stator windings and from the rotating fields' excitation diodes. A temperature controlled three-way valve (3SC-5) on the discharge of the stator cooling coolers proportions part of the pump discharge around the coolers to control the temperature of the stator coolant going to the generator windings and the rectifiers. The temperature controller for 3SC-5 uses a sealed, liquid filled seamless beryllium copper tube. When the liquid warms or cools, a corresponding volume change occurs. The volume change turns the bourdon tube and ultimately provides the signal to reposition 3SC-5. The three way valve, 3 SC-5 is normally set to maintain the stator cooling inlet temperature at about 46 degrees C (115 deg. F). Normally, as the stator coolant inlet temperature increases, more flow is directed through the stator cooling heat exchangers. When stator coolant temperatures are lower than the set temperature of 46 degrees C, 3SC-5 repositions to allow less flow to move through the coolers.

'ailure Modes

There are two conditions that can cause a Stator Coolant runback. These conditions are low stator winding inlet pressure and high winding outlet temperature. Failure modes associated with both of these conditions were investigated thoroughly by the investigation team. A summary of this investigation along with supporting and refuting evidence is listed in the table below.

Credible, Potential Failure Modes Evaluation Method Does Evaluation Support or Refute Failure Mode? Actual Failure Mode (yes/no) Contrb Factor (yes/no)

Low Winding Inlet Pressure

L1 - Loss of SC Pump & Failure of Standby Pump:
Refute: Both SC pumps 3A & 3B were found running with flow pegged high. No low pressure alarm was received.

L2 - Large System Leak:
Refute: A high generator liquid level alarm was not received. No significant leaks were found during system walkdown. SC pumps were operating & SC discharge press. was acceptable. No loss of inventory from the SC tank was observed.

L3 - Clogged Filter
Refute: Two stator coolant pumps were found operating with pump discharge flow pegged high. Filter differential pressure is normal with pumps operating. 76 psig filter inlet pressure/ 74 psig filter discharge pressure.

L4 - SC-6 Failed Closed
Refute: Two stator coolant pumps were found operating with pump discharge flow pegged high. Valve has a low flow travel stop that would prevent flow from being completely isolated. No low pressure alarm was received.

L5 - 3PS-216 runback pressure switch failure
Refute: If run back were due to a faulted pressure signal, temperature increases observed in the stator cooling system would not have been present.

L6 - SC-5 Failed in 100 % Cooler Alignment
Refute: Increasing stator coolant inlet temperatures were observed. This would not have occurred if the flow rate through the stator coolers had been increased. No low pressure alarm was received.

L7 - Clogged Y Strainer
Refute: Two stator coolant pumps were found operating with stator inlet flow pegged high. Therefore, the strainer is not restricting flow through the system.

L8 - Leaking Pump Discharge Stop Check Valve
Refute: Flow rate and pump discharge pressure were approx. the same and at expected values when the 3A pump was operating with the 3B pump stopped as it was with the 3B pump operating with the 3A pump stopped. Flow indication is measured downstream of where 3A & 3B headers are cross connected. If flow were short circuited through the check valves, indicated flow and pressure would have been lower.

High Winding Outlet Temperature

H1 - 3SC-5 Controller Failure:
Support: 3SCTT0098 Temperature control indication was found in the off-scale low temperature position. Stator coolant temperature increases were observed. Two I & E technicians were working conductivity cell near the SC skid at the time of the trip. Techs reported that capillary tube (part of 3SCTT0098) interaction did occur. The bulb of the capillary tube was later placed in an ice bath to determine if the controller was responding properly. No change in position was observed. The GE representative agreed that symptoms of capillary depressurization were present.

H2 - Winding Discharge Temperature Indicator:
Refute: Individual stator coil outlet temperatures increased. Responded to correct temperature observed by other temperature indications.

H3 - Loss of Secondary Cooling:
Refute: Prior to the run back, there was not an upset of the condensate system. Data indicates cooling remained available but heat load went away.

H4 - SC-5 Failed in Bypass:
Refute: The instrument calibration procedure verified that 3SC-5 responded correctly to temperature changes (Prior to returning system to service).

H5 - Increased Generator Heat Production:
Refute: Inlet, outlet, and stator coil outlet temperatures increased at a similar rate.

H6 - Loss of Flow (Leaking Disch. Check, SC-6 Closure, Clogged Strainer):
Refute: See L3, L7, & L8.

Cause Determination

The cause of the U3 runback is determined to be temperature controller 3SCTT0098 failing low, causing the three way proportioning valve, 3SC-5 to go into the full bypass position, thus removing cooling from the stator cooling system. The reasons for this determination are listed below:

The 3SC-5 temperature controller temperature indication was found in the off-scale low position. This position caused the controller to bypass flow to move the three way proportioning valve, 3SC-5 to a position such that flow was diverted around the stator cooling coolers.

Stator coolant temperature increases were observed. The full bypass mode of 3SC-5 caused the stator coolant temperature and therefore the generator coil temperatures to increase.

Two I & E technicians were working on a conductivity cell on the SC skid at the time of the trip. The technicians reported that capillary

tube interaction did occur (Part of 3SC TT0098). The physical layout of the unsupported capillary tube made it virtually impossible to work in the area without coming into contact with the capillary tube.

The capillary tube bend radius of 3" minimum was not maintained.

The capillary tube and it's sensing bulb (part of 3SCTT0098) were tested with an ice bath and a heat gun to determine if the controller was responding properly. No change in position was observed. Controller linkages did work properly when manually manipulated by hand. This specifically points to a root cause of a failure of the controller capillary tube. GE representatives agreed that symptoms of capillary depressurization were present.

Note: A follow up corrective action has been written to have the temperature controller tested/evaluated for determination of specific cause of the controller failure and to implement further recommendations based on the results of this testing.

Benchmarking

As a part of this investigation, benchmarking was performed. A search of the OEDB database was performed using the keywords "temperature transmitter" and/or "capillary. Several items were uncovered.

The OEDB review revealed there have been several incidences in the industry associated with instrument failure due to loss of capillary fill, damage due to unsupported capillary tubing, and damage due to bent controller features. General Electric was contacted to see if they had any specific experience with failures of the stator coolant temperature controller. GE stated that they did not have any specific knowledge of any events.

Operability, Generic Applicability, and Transportability

Operability:

The stator cooling system is a non-safety related secondary system. Operability determinations are not applicable to this failure.

Generic Applicability:

The failed temperature controller is of a design that is used on stator cooling systems of all three units at Oconee. As a result, the controllers on all three Stator Coolant systems should be assessed for vulnerability to a failure similar to the 1/3/2000 failure. Any further instrumentation assessments on other plant systems will be determined based on the testing results on the failed transmitter/capillary.

Transportability:

Catawba and McGuire have both been notified of this failure. Catawba, in particular has a Stator Coolant system that is very similar to Oconee's.

Planned Corrective Actions

1) Immediate/remedial: The following immediate actions were taken as a result of this failure:

A detailed review of the Unit 3 Electric Generator's response to the loss of stator cooling was performed on 1/3/2000 to determine if the response was as expected and within limits. Generator temperatures, rectifier temperatures, Stator Coolant conductivity, and the runback response were examined. The conclusion of this review was that there were no thermal/mechanical concerns with the U3 Generator response or reliability.

The faulty temperature controller, 3SCTT0098 was replaced and calibrated per vendor recommendations and Duke Tubing Installation

Specifications which included proper tubing bend radii. The replacement included the sensing bulb, capillary, helical coil, and the controller enclosure containing components converting signal to air. (Addresses Root Cause).

A tubing tray was installed to support the temperature element capillary line. NOTE: The original installation of the controller capillary line did not have a tubing tray installed to support the temperature element capillary line (Addresses Root Cause).

2) Interim

Signage was posted near the temperature controllers on all three Oconee units stating Unit Trip Potential (Addresses Root Cause).

A vibration inspection was performed at 100% full power. Vibration amplitudes were small and there was no resonance at the stator cooling pump vane pass frequency (Addresses Root Cause).

3) Corrective Actions to Prevent Recurrence

The faulty stator coolant temperature controller, 3SCTT0098, will be tested to determine the ultimate cause for the capillary failure (Responsible Group: CEN).

After testing is completed, any resulting recommended actions will be evaluated and implemented (Responsible Group: CEN).

Determine if any changes could be made to the temperature controller or 3SC-5 to prevent 3SC-5 from going to the full bypass position for similar instrument failures (Responsible Group: MSE).

Review and correction of any operating procedures or alarm response guides associated with a stator cooling runback event. Provide OPS training as necessary.
(Responsible Group: OPS)

List of Attachments

Attachment # 1: Plot of Oconee Unit 3 Generator Parameters

Attachment # 2: Controller Drawing From GE OM # 2200-0070

Attachment # 3: Temperature Controller (Pages 1-4)

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 01/26/2000

OEDB Comments:
See OEDB section of the root cause analysis report for details (Located in this Problem Evaluation for this PIP).

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 01/27/2000

Remarks Comments:

Signature Type	Indiv	Team	Group	Date
Due Date:	02/02/2000			
Accepted By:	GKM7309	GKM7309	MSE	01/06/2000
Assigned To:	TKR7315	RJF2111	MSE	01/21/2000

Problem Investigation Process
Oconee Nuclear Station

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	TKR7315	GKM7309	MSE	01/27/2000
Approval Assigned To:	RJF2111	RJF2111	MSE	01/27/2000
Approved By:	RJF2111	RJF2111	MSE	01/27/2000
Concurrence Assigned To:	MWA7315	RTB7310	SRG	01/27/2000
Concurrence By:	MWA7315	RTB7310	SRG	02/24/2000

Problem Evaluation From: Resp. Group: RGC Status: Closed OEDB Checked: No

February 2, 2000

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-287
Licensee Event Report 50-287/00-01, Revision 0
Problem Investigation Process No.: 0-O-00-0010

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 50-287/00-01, concerning a Reactor Trip that resulted when the Main Turbine was manually tripped due to equipment failure.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

W. R. McCollum, Jr.

ABSTRACT
At approximately 1254 on January 3, 2000, with Unit 3 at approximately 100 percent power, a Stator Cooling runback occurred. Operations verified that plant equipment was responding appropriately. It was observed that Generator MWatts had run back to 0 with Generator MVARs indicating 260 and that the Generator breakers were closed. At approximately 1257, these conditions along with HiHi Generator Stator winding temperatures led Control Room personnel to manually trip the Main Turbine, which resulted in an anticipatory Reactor trip. Reactor Power was approximately 56 percent at the time of the Main Turbine trip. This event involved equipment failures on the secondary side only. Primary and secondary equipment and parameters responded as expected. There were no major problems encountered in placing the plant in Mode 3.

Investigations revealed the Stator Cooling runback resulted from high Stator Cooling temperature. This condition was caused by the failure of the temperature controller for the Stator Cooling System proportioning valve. The cause of the reactor trip was inadequate installation of the temperature controller tubing. Corrective actions to prevent recurrence include replacement and analysis of the failed temperature controller. This event is considered of no significance with respect to the health and safety of the public.

EVALUATION:

BACKGROUND

The Stator Cooling (SC) [EIIS:TJ] Water System is a closed-loop system used to cool the generator's stationary armature windings and the excitor rectifying diodes. The major system components are the Stator Cooling Panel, a storage tank, two cooling pumps, two coolers, a proportioning valve, a de-ionizer and a differential pressure-regulating valve. The SC Panel contains the various instrumentation and controls used for monitoring and controlling SC system operation and an alarm panel consisting of 12 alarm windows. An alarm on this panel causes an alarm on 3SA3/A5, SC Panel Trouble, in the Control Room. The proportioning valve (3SC-5)[EIIS:FCV] is a temperature-controlled three-way valve on the discharge of the SC coolers that proportions part of the pump discharge around the coolers to control the temperature of the SC System going to the generator windings and rectifiers. Valve 3SC-5 is normally set to maintain the stator winding inlet temperature at about 46 C (115 F).

The Integrated Control System (ICS) [EIIS:JA] is designed to match reactor thermal power with core thermal power demand.

The Electro-Hydraulic Control System (EHC) [EIIS:TG] is divided into three major functional control areas: speed control, load control, and flow control. Monitoring circuits are provided to annunciate conditions of the system. Protection circuitry is imbedded in each control section to protect the turbine generator from mechanical and electronic failures in the system. Loss of the ability to adequately cool the stator windings causes the EHC to reduce MW load in order to decrease heat load on the stator windings.

The Reactor Protective System (RPS) [EIIS:JC] monitors several important system parameters and initiates a reactor trip when any trip setpoint is reached using two-of-four channel logic. One reactor trip parameter is a Main Turbine (MT) [EIIS:MT] trip which will initiate a reactor trip when power is greater than 30 percent full power by actuating RPS turbine anticipatory trip channels. The purpose of this ip is to limit Reactor Coolant System (RCS) [EIIS:AB] pressure and to minimize challenges to the Power Operated Relief Valve.

EVENT DESCRIPTION

On January 3, 2000, Unit 3 was in Mode 1 (Power Operation) at 100 percent power. Maintenance personnel were working on a failed SC system conductivity probe that had previously initiated a "Stator Cooling Panel Trouble" alarm at 0047 hours.

At approximately 1253, the Unit 3 Control Room received Statalarms for Stator Cooling Panel Trouble, Turbine Panel Trouble, and shortly afterwards, the GEN Loss of Stator Coolant alarm. An Operator was dispatched to the SC panel to investigate.

Control Room personnel continued to monitor SC graphics on the Operator Aid Computer. SC system temperatures were increasing as indicated by numerous Control Room alarms.

Control Room personnel monitored plant parameters and confirmed that the ICS was responding properly to reduce Reactor power and balance RCS heat input verses output. At approximately 1255, Control Room personnel observed that Generator MVARS were at approximately 260, Generator MWe had run back to 0, Stator winding temperature was still high and the Generator breakers were closed. This was not an expected indication for the turbine. Control Room personnel were concerned about the Generator being damaged under these conditions. At approximately 1257, a decision was made to trip the Turbine. When the Turbine tripped, the Reactor tripped, as expected, due to the anticipatory reactor trip.

Control Room personnel confirmed that the Reactor and Turbine had tripped and monitored the unit for proper operation in accordance with the Emergency Operating Procedures. As normally required after a Reactor trip, Operators started High Pressure Injection (HPI) [EIIS:BG] pump 3A and throttled 3HP-26 (3A High Pressure Injection Valve) to restore RCS Pressurizer level. The 3A Essential Siphon Vacuum (ESV) Pump tripped and was restarted per procedure. The Unit 2 ESV Headers were fully operable and available to supply Unit 3 while the 3A ESV pump was off.

Except as previously mentioned, all primary and secondary equipment and parameters responded as expected. There were no Engineered Safety Features actuations and Main Feedwater was maintained following the Reactor trip.

Following the Reactor trip, the Failure Investigation Process was initiated and a root cause investigation team was formed with

representatives from Operations and Engineering. The team determined that the initiating transient for the event was caused by the failure of the temperature controller for 3SC-5 (SC Proportioning Valve). The temperature controller temperature indication for valve 3SC-5 was found to be off-scale low. This caused valve 3SC-5 to reposition to the full cooler bypass position, thus removing the ability of the SC system to adequately cool the Generator. This condition was caused by a failure of the controller capillary tube. A new controller, capillary tube and sensing bulb were installed. A tubing tray to provide support/protection for the new capillary tube was installed using Specification # OSS-0060.00-00-0001. After the unit reached 100 percent full power, the controller and tubing were inspected for vibration/fretting interaction. The inspection for the controller and tubing revealed no vibration/fretting interaction. Signs were installed on the capillary tubing for all 3 units SC skids warning of a Unit Trip potential. The failed controller was sent to the Duke Power Metallurgical laboratory for a detailed failure analysis.

The Unit 3 Reactor was returned to critical on January 5, 2000 at 0141.

CAUSAL FACTORS

The cause of the Reactor Trip is equipment failure. The initiating transient was caused by the failure of the temperature controller for valve 3SC-5. The temperature controller failed low, causing valve 3SC-5 to reposition to the full cooler bypass position, thus removing the ability of the SC system to adequately cool the Generator. The root cause of the failed controller is attributed to an Installation deficiency. The capillary tube for the controller was not properly installed nor supported when the control panel was originally manufactured by the equipment vendor for construction of the plant. Maintenance personnel working in the area at the time of the Reactor trip stated that they did contact or move the capillary tube while performing their work; however, the physical layout of the unsupported capillary tube made it impossible to work in the area without coming in contact with the capillary tube. The required capillary tube bend radius of 3 inches minimum was not maintained. Certain sections of the capillary tube had bends that were close to ½ inch. The failed temperature controller has been sent to the Duke Power Metallurgical laboratory for a detailed analysis.

ORRECTIVE ACTIONS

Immediate:

Control Room personnel took appropriate actions to bring the unit to stable conditions per the Emergency Operating procedures.

Subsequent:

1. A new controller, capillary tube and sensing bulb were installed. A tubing tray was installed using Specification # OSS-0060.00-00-0001.

2. Inspected the Units 1 and 2 temperature controllers and capillary tubes and work requests were written for minor deficiencies on Unit 2 that will be repaired during a unit outage.

3. Signs were installed on the capillary tubing for all 3 units SC skids warning of a Unit Trip potential.

4. A vibration inspection was performed at 100% power and the inspection revealed no vibration/fretting interaction.

Planned:

The temperature controller has been sent to the Duke Power Metallurgical laboratory for a failure analysis. Depending upon the results of the analysis, more corrective actions may be recommended.

There are no NRC commitments contained in this LER.

SAFETY ANALYSIS

Problem Investigation Process
Oconee Nuclear Station

This event is not considered to be significant. At no time were the health or safety of the public or plant personnel compromised.

The Unit 3 Reactor automatically tripped due to a manual trip of the Main Turbine. The Main Turbine was manually tripped to protect the Generator which was experiencing increasing Stator winding temperatures.

Stator winding temperature exceeded 176 F, which initiated the Loss of SC runback circuit. The high temperature condition did not clear and the SC runback occurred as expected. The EHC system reduced MW load; consequently, the heat load was removed from the Generator, thus decreasing the need for Stator Winding cooling.

The unit post-trip response was normal. No Engineered Safety Features or Emergency Feedwater actuations were either required or received after the trip. All safety systems required for maintaining Reactor Core and Containment protection remained fully available.

The core damage significance of this event has been evaluated to be very low. No significant systems were unavailable at the time of the trip. The conditional core damage probability for a Reactor trip for this condition is expected to be less than 1E-07.

ADDITIONAL INFORMATION

A review of LERs and Operating Experience within the past two years indicates that there have not been any Reactor trips associated with equipment failures of this type.

The failed component is not EPIX reportable.

'his event did not result in personnel injuries, radiation overexposures, or releases of radioactive materials.

CAUSE CODES:

*B - The root cause of the failed controller is attributed to an Installation deficiency. The capillary tube for the controller was not properly installed nor supported. Maintenance personnel working in the area at the time of the Reactor trip stated that capillary tube interaction did occur; however, the physical layout of the unsupported capillary tube made it impossible to work in the area without coming in contact with the capillary tube. The required capillary tube bend radius of 3 inches minimum was not maintained. Certain sections of the capillary tube had bends that were close to ½ inch. The failed temperature controller has been sent to the Duke Power Metallurgical laboratory for a detailed analysis.

Originated By: JVW2844: WEAST,JAMES V Team: LEN2127 Group: RGC Date: 02/03/2000

OEDB Comments:

Remarks Comments:

Signature Type	Indiv	Team	Group	Date
Due Date:	02/02/2000			
Accepted By:	RVGAMBRE	LEN2127	RGC	01/04/2000
Assigned To:	JVW2844	LEN2127	RGC	01/04/2000
Ready For Approval:	JVW2844	LEN2127	RGC	02/03/2000
Approval Assigned To:	LEN2127	LEN2127	RGC	02/03/2000
Approved By:	LEN2127	LEN2127	RGC	02/08/2000
Concurrence By:	MWA7315	RTB7310	SRG	03/01/2000

Corrective Actions

CA Seq. No: 1

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
CEN	Closed	CEN	F2	B9	YYY

Proposed Corrective Action:
Replace valve 3SC5 temperature controller, 3SCTT0098.

Originated By: TAL8382: LEDFORD, TERENCE A Team: CAL7344 Group: CEN Date: 01/04/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	TAL8382	CAL7344	CEN	01/04/2000
Approval Assigned To:	CAL7344	CAL7344	CEN	01/04/2000
Approved By:	TAL8382	CAL7344	CEN	01/04/2000

General: Outage: N/A Mode: N/A

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: B9Status: Closed Due Date: 07/01/2000
Temperature controller 3SCTT0098 was replaced with a Fischer & Porter model 51C1451TFCXX under work request 98110726. This controller consist of the series 51-1451 enclosure with a series 53PF pneumatic controller and a series 12T-2000 code TF liquid filled type temperature element with a 9 foot capillary line. The old controller was basically the same unit but had a code TG temperature element with a 15 foot capillary line. The code TG is case compensated only where the code TF is fully compensated. The code TF basically gives better accuracy where the capillary is exposed to temperature extremes. The new temperature controller was calibrated per procedure IP/0/B/0290/001A which was changed to show the new part number. The new controller was installed per NPP-212 acceptable sub criteria where the replacement unit meets form, fit, and function for a like-for-like replacement.

The EQDB data for 3SCTT0098 has a Foxboro unit listed which is incorrect. In addition, the OM manual (OM 2200-0070) did not address the code TF temperature element. These documents will be updated by a minor mod which is covered under a separate corrective action.

The original installation did not have tubing tray installed to support the temperature element capillary line. This may have contributed to the failure of the original controller. As a result, tubing tray is being installed under work request 98110830 per OSS-0060.00-00-0001. Oconee Engineering has given prior approval for this non-QA installation as allowed by OSS-0060.00-00-0001.

Originated By: JHS7316: SMITH, JAMES H Team: TAL8382 Group: CEN Date: 01/04/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/01/2000			
Accepted By:	TAL8382	CAL7344	CEN	01/04/2000
Assigned To:	JHS7316	TAL8382	CEN	01/04/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	JHS7316	TAL8382	CEN	01/04/2000
Approval Assigned To:	TAL8382	TAL8382	CEN	01/04/2000
Approved By:	TAL8382	CAL7344	CEN	01/04/2000
Concurrence By:	RWVASSEY	RTB7310	SRG	01/05/2000

CA Seq. No: 2

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
CEN	Closed	CEN	F2	B1b	YYY

Proposed Corrective Action:
Update the EQDB for 3SCTT0098 and also update OM-2200-0070 to show the information for 3SCTT0098.

Originated By: JHS7316: SMITH, JAMES H Team: TAL8382 Group: CEN Date: 01/04/2000

Signature Type	Indiv	Team	Group	Date
Assigned To:		TAL8382	CEN	01/05/2000
Ready For Approval:	TAL8382	CAL7344	CEN	01/10/2000
Approval Assigned To:	CAL7344	CAL7344	CEN	01/10/2000
Approved By:	TAL8382	CAL7344	CEN	01/10/2000

General: Outage: N/A Mode: N/A

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: B1aStatus: Closed Due Date: 07/01/2000
Minor modification ONOE-14742 has been approved and released for processing.

Originated By: JHS7316: SMITH, JAMES H Team: TAL8382 Group: CEN Date: 03/14/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/01/2000			
Accepted By:	TAL8382	CAL7344	CEN	01/10/2000
Assigned To:	JHS7316	TAL8382	CEN	01/10/2000
Ready For Approval:	JHS7316	TAL8382	CEN	03/14/2000
Approval Assigned To:	TAL8382	TAL8382	CEN	03/14/2000
Approved By:	TAL8382	CAL7344	CEN	03/14/2000
Concurrence By:	RWVASSEY	RTB7310	SRG	03/15/2000

Problem Investigation Process
Oconee Nuclear Station

CA Seq. No: 3

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
CEN	Closed	CEN	F2	B3	YYY

Proposed Corrective Action:
Civil engineering is to inspect the tubing tray installation used to support the capillary line for the temperature element on 3SCTT0098 and verify no concerns exist related to vibration and the installation.

Originated By: JHS7316: SMITH, JAMES H Team: TAL8382 Group: CEN Date: 01/04/2000

Signature Type	Indiv	Team	Group	Date
Assigned To:		TAL8382	CEN	01/05/2000
Ready For Approval:	TAL8382	CAL7344	CEN	01/10/2000
Approval Assigned To:	CAL7344	CAL7344	CEN	01/10/2000
Approved By:	TAL8382	CAL7344	CEN	01/10/2000

General: Outage: Mode:

Other Tracking Processes
Type Number Text

ctual Corrective Action:
Actual CAC: Status: Open Due Date: 07/01/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/01/2000			
Accepted By:	RAH8344	RAH8344	CEN	01/13/2000
Assigned To:	PCC2458	RAH8344	CEN	01/13/2000

CA Seq. No: 4

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
CEN	Closed	MSE	F2	B3	M

Proposed Corrective Action:
The faulty stator coolant temperature controller, 3SCTT0098, will be tested to determine the ultimate cause for the capillary failure. After testing is completed, any resulting recommended actions will be evaluated and implemented (Responsible Group: CEN).

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 01/26/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	TKR7315	GKM7309	MSE	01/27/2000
Approval Assigned To:	RJF2111	RJF2111	MSE	01/27/2000
Approved By:	RJF2111	RJF2111	MSE	01/27/2000

General: Outage: N/A Mode: N/A

Other Tracking Processes

Type Number Text

Actual Corrective Action:

Actual CAC: B9Status: Open Due Date: 07/01/2000

The faulty stator coolant temperature controller, 3SCTT0098, was tested to determine the ultimate cause for the capillary failure. The capillary tube broke in two at the bulb connection. The tube had been bent sharply at this point. The combination of the capillary tube being unsupported; sharply bend at it entrance point to the bulb; and this point being subject to repetitive loading throughout its life, as the bulb is removed and reinserted for calibration, have contributed to the growth of a fatigue crack, which ultimately fractured the tube. See the Metallurgical Analysis Report below.

The following are actions are being taken to preclude future controller failures:

1. Provide protection (like rubber insulation) around the new Unit 3 SC temperature controller capillary tube at the point it enters the controller enclosure (through the bulkhead fitting) on the inside, to eliminate potential tube wearing.
------An additional task was added to Unit 3 PM Work Order 98190922 to do this work during the Unit 3 refueling outage. No further action is required.

2. Replace the Unit 1 and Unit 2 SC temperature controller capillary line and bulb. The controller designations are 1SC TT0098 and 2SC TT0098.
------Maintenance has written a work order, 98234502, to replace the Unit 2 controller, including capillary line and bulb, for the next Unit 2 refueling outage. I wrote work order 98252171 to replace the Unit 1 controller, including capillary line and bulb, for the next Unit 1 refueling outage. No further action is required.

3. Add instruction or precautions to the Maintenance temperature controller calibration procedure, IP/0/B/0290/001A, to ensure proper handling of the capillary line and bulb.
------Maintenance (PRT) will incorporate this in revision 52 of the procedure. No further action is required.

4. Provide training specific to handling of the capillary line and bulb.
------???? (((((Talk to Maintenance to ask them if they will accept a corrective action for this.)))))

5. Evaluate the need to use an alternative SC temperature controller, which uses something other than a capillary tube.
------This has been looked at; it is concluded that the capillary tube is actually a substantial component of the controller if supported according to the installation specifications. Other options would include as many failure modes or more than the present controller. No further action is required.

6. Evaluate the option to bend the bulb (if a bendable bulb) 90 degrees downward to keep the capillary tube from bending excessively and to eliminate interfernce for repalcing the SC outlet conductivity probe.
------This will be done as part of replacing the 1SC TT0098 and 2SC TT0098 in item 2, above. This will be done for 3SC TT0098 as part of the Unit 3 PM Work Order 98190922, Task ___________, during the Unit 3 refueling outage. No further action is required.

7. Research and evaluate other applications, which are of a critical/generation risk, that use the temperature capillary tube. Then make recommendations, accordingly.
------See corrective action ______ for this.

The DE&S, Testing and Laborary Services, Metallurgical Analysis Report (without figures) follows:

Problem Investigation Process
Oconee Nuclear Station

Equipment Description:
Temperature controller 3SCTT0098 on valve 3SC-5 (generator stator cooler system)

Background Information:
On 1/3/00, a failed low temperature signal to valve 3SC-5 precipitated a series of events which resulted in the manual trip of the turbine and reactor (see PIP O-00-00010). The temperature controller was found to have failed; it was suspected that its 15-foot capillary line had broken internally. Additionally, there was no tubing tray installed to support its capillary line. The Fischer & Porter Series 1450 controller dated to the 1970's.

Description/Macro-Examination:
The profile of the 0.061"-diameter copper capillary line where the copper tubing armor attaches to the bulb is shown in Figure 1. With the armor removed, the break in the capillary line is evident (Figure 2). The break occurred in the copper just beyond the attachment point to the stainless steel fitting. The capillary which entered the bulb was stainless steel, 0.0635" diameter. Greenish deposits were present at and around the fracture, as well as on the copper armor. The armor looked as though it had been manually polished sometime previously.
The mating fractured halves are shown in Figures 3-4. There was some parallel circumferential cracking visible near the edge of the silver solder. The fracture plane was 0.065" from the edge of the solder. A single drop of solder was present 0.03" from the edge of the solder pool, and the fracture occurred at one edge of this drop. A crack was visible along the other edge of the drop, further still from the attachment point.
Copper shavings were visible on the beveled surface where the capillary line passed through the bulkhead fitting inside the housing (Figures 5-6). The capillary line in that area was rubbed shiny and showed both axial scratches and circumferential gouge marks (Figures 7-8). The copper shavings were wear debris from the capillary line. No cracking was found in this area.
Fractography/SEM/EDS:
Scanning electron microscope (SEM) examination of the fracture surface showed it to be significantly rubbed from contact between the mating fractures (Figure 9). Fatigue striations were found over half of the fracture, traveling from the OD inward (Figure 10). The remainder of the fracture showed signs of ductile overload, as is expected after a crack progresses to a critical depth.
Qualitative EDS analysis of the green deposits on the capillary line showed elevated phosphorus, sulfur, and chloride, all of which can be corrosive to copper. The solder was confirmed to be silver-based. A test for ammonia content in the green deposit on the armor was performed by Duke Power's EH&S Laboratories; results showed an average of 4.9 ppm ammonia and 8.6 ppm chloride in the deposits.

Metallography:
A cross-section through the capillary end shown in Figure 4 is shown in Figure 11. The crack at the toe of the solder bead had been pulled wide open, either during overload fracture or from subsequent handling. The initial area of the fracture was unusually blocky and had several very shallow secondary cracks (Figure 12). No other significant cracks were present elsewhere along the OD of the capillary line. The fracture on the mating half of the fracture had been significantly damaged by rubbing, and also showed no major secondary cracking.
In cross-section, the circumferential cracking on the surface of the solder was found to be superficial and related to subsurface pores in the solder. It was not relevant to the fracture.

Discussion/Conclusions:
The capillary line in the temperature controller had fractured in two near the point where it bent sharply prior to attaching to the bulb. Fatigue striations were identified on the fracture surface.
Green deposits containing potentially corroding species (sulfur and chloride) were present on both the armor and the capillary line. There was no significant corrosive attack (i.e. pitting) on either component, however.
Rubbing wear had occurred on the capillary line where it entered the housing, to the point

where an accumulation of copper shavings was visible. This wear indicated that the capillary line had been subjected to some type of repetitive loading throughout its life, whether vibrational or the result of being bumped while unsupported. This repetitive loading contributed to the growth of the fatigue crack which ultimately fractured the line. After fracture occurred, the two halves of the capillary line continued to rub against each other.
It appears that the cracking was limited to the highest-stress location along the capillary line -- that is, where it was sharply bent to enter the bulb. The stresses at the housing entry point were not as severe; so, despite having numerous circumferential gouges present there, no cracking initiated. Use of a strain-relief fitting next to the bulb may reduce the potential for cracking.
There may have been some contribution of ammonia-induced stress-corrosion cracking in initiating the crack, but the crack progressed under a fatigue mechanism. Even if SCC had acted initially, the stress at the bend point was a primary factor, because there is essentially no branching to the crack and there is no other similar cracking elsewhere. The amount of ammonia detected in the armor could potentially have induced SCC if the local stress were high enough.
If the DE&S Metallurgy Lab can be of further assistance, please call us at (704) 875-5275.

Originated By: TMD7360: DWYER, TERRY M Team: TAL8382 Group: CEN Date: 02/29/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/01/2000			
Accepted By:	TAL8382	CAL7344	CEN	02/01/2000
Assigned To:	TMD7360	TAL8382	CEN	02/01/2000

CA Seq. No: 5

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
MSE	Closed	MSE	F2	B3	M

Proposed Corrective Action:
Determine if any changes could be made to the temperature controller or 3SC-5 to prevent 3SC-5 from going to the full bypass position for similar instrument failures (Responsible Group: MSE).

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 01/26/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	TKR7315	GKM7309	MSE	01/27/2000
Approval Assigned To:	RJF2111	RJF2111	MSE	01/27/2000
Approved By:	RJF2111	RJF2111	MSE	01/27/2000

General: Outage: N/A Mode: N/A

Other Tracking Processes
Type Number Text

Problem Investigation Process
Oconee Nuclear Station

Actual Corrective Action:
Actual CAC: Status: Open   Due Date: 07/01/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/01/2000			
Accepted By:	RJF2111	RJF2111	MSE	01/27/2000
Assigned To:	TKR7315	RJF2111	MSE	01/27/2000

CA Seq. No: 6

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	MSE	F2	A3	M

Proposed Corrective Action:

Review and correct operating procedures associated with a stator cooling runback event per the problem evaluation of this PIP. (Responsible Group: OPS)

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 01/26/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	TKR7315	GKM7309	MSE	01/27/2000
Approval Assigned To:	RJF2111	RJF2111	MSE	01/27/2000
Approved By:	RJF2111	RJF2111	MSE	01/27/2000

General: Outage: Mode:

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: Status: Open   Due Date: 07/01/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/01/2000			
Accepted By:	MAP7314	HRL7353	OPS	02/01/2000
Assigned To:	SCP484C	GBJ1009	OPS	02/07/2000

Problem Investigation Process
Oconee Nuclear Station

CA Seq. No: 7

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	MSE	F2	A3	M

Proposed Corrective Action:

Review and correction alarm response guides associated with a stator cooling runback event per the problem evaluation section of this PIP.
(Responsible Group: OPS)

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 01/26/2000

Signature Type	Indiv	Team	Group	Date
Approval Assigned To:	RJF2111	RJF2111	MSE	01/27/2000
Ready For Approval:	TKR7315	GKM7309	MSE	01/27/2000
Approved By:	RJF2111	RJF2111	MSE	01/27/2000

General: Outage: Mode:

ther Tracking Processes

Type Number Text

Actual Corrective Action:

Actual CAC: A3Status: Closed Due Date: 07/01/2000
OP/1/A/6101/003 (Alarm Response Guides 1SA-03), OP/2/A/6102/003 (Alarm Response Guides 2SA-03), OP/3/A/6103/003 (Alarm Response Guides 3SA-03) have been revised to reflect the actual conditions of the Stator Coolant System that result in a stator coolant runback and the correct response of the Turbine Generator to a Stator Coolant Runback. All procedures were approved on 1/26/2000.

Last Updated By: WMB4441: BUCHANAN JR, WILLIAM M Team: GBJ1009 Group: OPS Date: 02/15/2000

Originated By: WMB4441: BUCHANAN JR, WILLIAM M Team: GBJ1009 Group: OPS Date: 02/15/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/01/2000			
Accepted By:	MAP7314	HRL7353	OPS	02/01/2000
Assigned To:	WMB4441	GBJ1009	OPS	02/07/2000
Ready For Approval:	WMB4441	GBJ1009	OPS	02/15/2000
Approval Assigned To:	GBJ1009	DBC7309	OPS	02/15/2000
\pproved By:	GBJ1009	DBC7309	OPS	02/16/2000
Concurrence By:	RWVASSEY	RTB7310	SRG	02/21/2000

Problem Investigation Process
Oconee Nuclear Station

CA Seq. No: 8

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	MSE	F2	C	M

Proposed Corrective Action:

In conjunction with the review and correction of any operating procedures or alarm response guides associated with a stator cooling runback event, provide OPS training as necessary to the appropriate individuals.
(Responsible Group: OPS Training)

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 01/26/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	TKR7315	GKM7309	MSE	01/27/2000
Approval Assigned To:	RJF2111	RJF2111	MSE	01/27/2000
Approved By:	RJF2111	RJF2111	MSE	01/27/2000

General: Outage: Mode:

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: Status: Open Due Date: 07/01/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	07/01/2000			
Accepted By:	MAP7314	HRL7353	OPS	02/01/2000
Assigned To:	SCP484C	GBJ1009	OPS	02/07/2000

Final and Overall PIP Approval

Responsible Group: OPS Status: Screened

Signature Type	Indiv	Team	Group	Date
Assigned To:			OPS	01/04/2000

Problem Investigation Process
Oconee Nuclear Station

Any Supplemental Concurrence Signatures Above Do Not Affect PIP Closure.

Microfilm Roll / Frame: /

Closure Document Type **Closure Document No**

Attachments

Generic Applicability

Responsible Group: OEA Status: Open

GO PIP No:

Assessment Remarks:

OEDB Item 24471 referenced this event. Walt Wylie has been assigned the OEDB item for applicability to other Duke plants.

Determined by Screening Team.

Originated By: DXT7339: TOWER,DEVEREUX Team: JWP7322 Group: OEA Date: 03/01/2000

Signature Type	Indiv	Team	Group	Date
\pproval Assigned To:	WAW0982	JWB7393	OEA	03/09/2000
Assigned To:	WAW0982	JWB7393	OEA	03/09/2000
Due Date:	04/27/2000			

Failure Prevention Investigation

Quality of CA: Quality of Cause: Resp Group: SRG Status: Closed

Special Codes:

N5

Comments

Signature Type	Indiv	Team	Group	Date
Assigned To:			SRG	01/04/2000
Ready For Approval:	RWVASSEY	RTB7310	SRG	03/01/2000
Approval Assigned To:	RTB7310	RTB7310	SRG	03/01/2000
Approved By:	RWVASSEY	RTB7310	SRG	03/01/2000

Remarks

No Remarks for this PIP.

Maintenance Rule

Responsible Group: MSE Status: Closed

Problem Investigation Process
Oconee Nuclear Station

Maintenance Rule SSC

SSC	Description	Risk Significant	Primary System
SC	Stator Coolant System	No	Yes

Equipment Group: E01
Applicable Unit: Unit 3
Functional Failure: Yes MPFF: Yes Repetitive MPFF: No

Functional Failure Comments:
The failure of the Unit 3 Stator Cooling temperature controller caused a stator coolant runback to 55%. The control room crew tripped the turbine at this power level. Since this failure exceeded the plant level performance criteria of "unplanned load changes > 10%/unit/fuel cycle", a functional failure occurred.

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 03/07/2000

MPFF Comments:
The failure of the temperature controller was due to a break in the controller's capillary tube. The capillary tube was not supported :cording to manufacturer's recommendations and capillary bends exceeded specified 3" radius. In addition, it was possible that improper handling of the capillary tube through the years could have contributed to the problem. It is concluded that this failure is a MPFF.

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 03/07/2000

Repetitive MPFF Comments:
Using the search function in this PIP for repetitive MPFFs, a list of 23 selected PIPS were reviewed. The results of this review did not uncover any repeat failures of this type that caused the stator coolant runback described in this PIP. Therefore, there is no Repetitive MPFF.

Originated By: TKR7315: ROYAL, TEDDY K Team: GKM7309 Group: MSE Date: 03/07/2000

Reactor Trip: No Safety System Actuation: No Loss of Heat Decay Removal: No
Force Outage Rate or Plant Transient: Yes Loss Of Spent Fuel: No

Comments:

Signature Type	Indiv	Team	Group	Date
Assigned To:	TKR7315	RJF2111	MSE	01/21/2000
Due Date:	03/31/2000			
Ready For Approval:	TKR7315	GKM7309	MSE	03/23/2000
Approval Assigned To:	RJF2111	RJF2111	MSE	03/23/2000
Approved By:	RJF2111	RJF2111	MSE	03/28/2000

End of the Document for PIP No: O-0-10
The status of this PIP is: Screened
The duration of this PIP was: 3 days

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM NRC-002/SIM
RC-66 (PZR PORV) Stoke Test
Alternate Path

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:
Perform RC-66 PORV Stroke Test

Alternate Path:

When 1RC-66 (PZR PORV) fails to manually close after stroke testing 1RC-4 (PZR PORV Block) is closed to isolate a stuck open PORV.

Facility JPM #:

NEW

K/A Rating(s):

010A4.03 [4.0/3.8]

Task Standard:

Accomplish the stoke test for 1RC-66 (PORV) per PT/201/004.
Close 1RC-4 to isolate a stuck open PORV

Preferred Evaluation Location: Simulator __X__ In-Plant ______

Preferred Evaluation Method: Perform __X__ Simulate ______

References:
PT/201/004

Validation Time: 10 min. **Time Critical:** NO

==

Candidate: ____________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ____________________________ ________________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall SNAP 104
2. IMPORT NRC-2 Files (PORV fails open after lifting)
3. Set LPSW to both LPI Coolers to ≈ 900 gpm/cooler
4. Place QT in recirc (Open CS 5 and6 then start the Component Drain Pump)
5. Override QT press to 0 psig (prevent increase)

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

PT/201/004

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC
Unit 1 startup is in progress
Establishing a PZR bubble is in progress

INITIATING CUES:

The SRO in the Control Room directs you to complete operations to establish a PZR bubble using the in-progress procedure OP/1103/002 Encl 4.14 starting at step 2.17.

START TIME: _______

STEP 1: Per procedure step 2.17, Perform PT/1/A/0201/004, (1RC-66 Stoke Test) STANDARD: Per PT/1/A/0201/004 verify required unit status: RCS pressure < 45 psig QT lined up for lowering QT temperature per OP/1104/17 ***CUE: Unit status requirements have been met*** COMMENTS:	___ SAT ___ UNSAT
STEP 2: Verify Prerequisite System Conditions: STANDARD: Steam bubble is formed in the PZR RC-4 is operable ***Cue: PORV Outlet thermocouples are available on the OAC.*** COMMENTS:	___ SAT ___ UNSAT
TEP 3: Ensure no personnel are present in the SG cavities of the RB STANDARD: ***CUE: No personnel is in the RB at this time*** COMMENTS:	___ SAT ___ UNSAT
STEP 4: Cycle 1RC-4 (PORV Block) STANDARD: 1RC-4 is positioned to close and verified to be closed by the green light on 1RC-4 is positioned to open and verified to be open by the red light on* ***Reopening 1RC-4 is critical.** COMMENTS:	**CRITICAL STEP*** ___ SAT ___ UNSAT

TEP 5:	Record data required on Enclosure 13.4 (Information Sheet) prior to opening 1RC-66 (PORV)	___ SAT
STANDARD:	Time, RCS Pressure, QT pressure, QT temperature, QT level, RC-66 Outlet temperature, PZR level. This information is recorded on Encl 13.4	___ UNSAT
COMMENTS:		
		CRITICAL STEP
STEP 6:	Open 1RC-66 (PORV)	___ SAT
STANDARD:	RC-66 switch positioned to OPEN and the OPEN PERMIT depressed	
COMMENTS:		___ UNSAT
		CRITICAL STEP
STEP 7:	Monitor the following to verify 1RC-66 is open, then close 1RC-66 after positive indications are verified	___ SAT
TANDARD:	Parameters/indications are verified to ensure 1RC-66 is open • PRZ Relief valve flow monitor • 1RC-66 position indication (pilot valve)	___ UNSAT
COMMENTS:		
		CRITICAL STEP
STEP 8:	Close 1RC-66 (PORV)	___ SAT
STANDARD:	Select LOW on the 1RC-66 switch When 1RC-66 is attempted to be closed diagnose 1RC-66 failed to close and close 1RC-4 (1RC-66 Block)	___ UNSAT
COMMENTS:		
	END TASK	

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
4	Reopening 1RC-4 establishes a flowpath from the PZR to QT.
6	This step opens 1RC-66.
7	Verifying PORV is open by alternate means. PORV pilot valve light indication is not used.
8	After 1RC-66 failure to close is diagnosed 1RC-4 is closed to stop flow.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the unit 1 OATC
Unit 1 startup is in progress
Establishing a PZR bubble is in progress

INITIATING CUES:

The SRO in the Control Room directs you to complete operations to establish a PZR bubble using the in-progress procedure OP/1103/002 Encl 4.13 starting at step 2.2
Unit 1 startup is in progress
Establishing a PZR bubble is in progress

INITIATING CUES:

The SRO in the Control Room directs you to complete operations to establish a PZR bubble using the in-progress procedure OP/1103/002 Encl 4.13 starting at step 2.2

(R06-97) TNI/WHC/RAQ
SIM, SR(13)

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No PT/0/A/0201/004
Revision No 004

OTC MASTER FILE

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title PORV OPERABILITY TEST

(4) Prepared By Michael D. Eberhart Date 10/20/97

(5) Requires 10CFR50.59 evaluation?
☐ Yes (New procedure or revision with major changes)
☑ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By RC Gamberg (QR) Date 10/23/97
Cross-Disciplinary Review By [signature] (QR)NA Date 10/23/97
Reactivity Mgmt. Review By (QR)NA [initials] Date 10/23/97

(7) Additional Reviews
Reviewed By Date
Reviewed By Date

(8) Temporary Approval *(if necessary)*
By (SRO/QR) Date
By (QR) Date

(9) Approved By [signature] Curtis Date 10/26/97

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy Date
Compared with Control Copy Date
Compared with Control Copy Date

(11) Date(s) Performed
Work Order Number (WO#)

COMPLETION

(12) Procedure Completion Verification
☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By Date

(13) Procedure Completion Approved Date

(14) Remarks *(Attach additional pages, if necessary)*

24 25 26 27 28 29 30 31 1 2 3 4 5 6 7 8 ... 17 18 15
1997
SIMULATOR
CONTROL COPY

Duke Power Company Oconee Nuclear Station	Procedure No. **PT/0/A/0201/004**
PORV Operability Test	Revision No. 004
Continuous Use	Electronic Reference No. OX002VWW

PORV Operability Test

1. Purpose

1.1 To verify operability of (1) (2) (3) RC-66 (PORV) prior to exceeding 200°F (RC average temperature).

2. References

2.1 OFD-100A

3. Time Required

3.1 30 minutes

4. Prerequisite Test

None

5. Test Equipment

None

6. Limits and Precautions

6.1 Ensure that no personnel are in the Reactor Building SG cavities while (1)(2)(3) RC-66 (PORV) is being tested.

6.2 Do not perform test at greater than 45 psig in RCS.

6.3 Do not exceed 200°F (RC average temperature) until (1) (2) (3) RC-66 (PORV) is verified operable.

7. Required Unit Status

_____ 7.1 Plant in the startup mode with RC pressure ≤ 45 psig.

_____ 7.2 Quench Tank lined up for lowering Quench Tank temperature per OP/1,2,3/A/1104/17 (Quench Tank Operation).

8. Prerequisite System Conditions

_____ 8.1 Verify (1) (2) (3) RC-66 (PORV) outlet thermocouple temperature available on computer.

_____ 8.2 Steam bubble established in Pressurizer.

_____ 8.3 (1) (2) (3) RC-4 (POWER RELIEF BLOCK) operable.

9. Test Method

9.1 With steam bubble in Pressurizer and RCS pressure < 45 psig, (1) (2) (3) RC-66 (PORV) will be opened with (1)(2)(3) RC-4 (POWER RELIEF BLOCK) open and then (1)(2)(3) RC-66 (PORV) will be closed. Operator will verify opening of valve by observing a change in: Valve outlet temperature, Pressurizer level, Quench Tank pressure, level, temperature and PORV Flow Monitor.

10. Data Required

10.1 Quench Tank pressure and temperature

10.2 PORV outlet temperature

10.3 Pressurizer level

10.4 RCS pressure

11. Acceptance Criteria

11.1 Test results are acceptable upon positive indication that (1) (2) (3) RC-66 (PORV) opens and closes.

12. Procedure

12.1 Verify operability of 1RC-66 (PORV) using Enclosure 13.1 (Unit 1 PORV Test).

12.2 Verify operability of 2RC-66 (PORV) using Enclosure 13.2 (Unit 2 PORV Test).

12.3 Verify operability of 3RC-66 (PORV) using Enclosure 13.3 (Unit 3 PORV Test).

12.4 Record data required on Enclosure 13.4 (Information Sheet).

13. Enclosures

13.1 Unit 1 PORV Test

13.2 Unit 2 PORV Test

13.3 Unit 3 PORV Test

13.4 Information Sheet

1. Procedure:

_____ 1.1 Verify that no personnel are present in the SG cavities of the Reactor Building.

1.2 Prior to cycling 1RC-66 (PORV), verify operability of 1RC-4 (POWER RELIEF BLOCK) as follows:

_____ 1.2.1 Close 1RC-4 (POWER RELIEF BLOCK).

_____ _____ 1.2.2 Open 1RC-4 (POWER RELIEF BLOCK).

_____ 1.3 Record data required on Enclosure 13.4 (Information Sheet) prior to opening 1RC-66 (PORV).

1.4 Open 1RC-66 (PORV) as follows:

_____ 1.4.1 Position the 1RC-66 SETPOINT SELECTOR to "OPEN".

NOTE:	1RC-66 remains open when the "OPEN PERMIT" Pushbutton is released until 1RC-66 SETPOINT SELECTOR position is changed to "LOW".

_____ _____ 1.4.2 Depress the 1RC-66 "OPEN PERMIT" Pushbutton.

_____ 1.5 Monitor the following to verify 1RC-66 (PORV) is open:

- Quench Tank pressure increasing
- Quench Tank level increasing
- 1RC-66 (PORV) indicates open on Control Board
- PORV Flow Monitor indicates flow
- Pressurizer Relief Valve Flow Statalarm if 5 or more lights are lit.

_____ _____ 1.6 When there is positive indication that 1RC-66 (PORV) is open, select "Low" on 1RC-66 Setpoint Selector.

_____ 1.6.1 Verify 1RC-66 (PORV) is closed.

_____ _____ 1.6.2 If 1RC-66 (PORV) fails to close, close 1RC-4 (POWER RELIEF BLOCK) immediately.

_____ 1.7 Record data required on Enclosure 13.4 (Information Sheet) when 1RC-66 (PORV) is closed.

1. Procedure:

_____ 1.1 Verify that no personnel are present in the SG cavities of the Reactor Building.

1.2 Prior to cycling 2RC-66 (PORV), verify operability of 2RC-4 (POWER RELIEF BLOCK) as follows:

_____ 1.2.1 Close 2RC-4 (POWER RELIEF BLOCK).

_____ _____ 1.2.2 Open 2RC-4 (POWER RELIEF BLOCK).

_____ 1.3 Record data required on Enclosure 13.4 (Information Sheet) prior to opening 2RC-66 (PORV).

1.4 Open 2RC-66 (PORV) as follows:

_____ 1.4.1 Position the 2RC-66 SETPOINT SELECTOR to "OPEN".

NOTE:	2RC-66 remains open when the "OPEN PERMIT" Pushbutton is released until 2RC-66 SETPOINT SELECTOR position is changed to "LOW".

_____ _____ 1.4.2 Depress the 2RC-66 "OPEN PERMIT" Pushbutton.

_____ 1.5 Monitor the following to verify 2RC-66 (PORV) is open:

- Quench Tank pressure increasing
- Quench Tank level increasing
- 2RC-66 (PORV) indicates open on Control Board
- PORV Flow Monitor indicates flow
- Pressurizer Relief Valve Flow Statalarm if 5 or more lights are lit.

_____ _____ 1.6 When there is positive indication that 2RC-66 (PORV) is open, select "Low" on 2RC-66 Setpoint Selector.

_____ 1.6.1 Verify 2RC-66 (PORV) is closed.

_____ _____ 1.6.2 If 2RC-66 (PORV) fails to close, close 2RC-4 (POWER RELIEF BLOCK) immediately.

_____ 1.7 Record data required on Enclosure 13.4 (Information Sheet) when 2RC-66 (PORV) is closed.

1. Procedure:

_____ 1.1 Verify that no personnel are present in the SG cavities of the Reactor Building.

1.2 Prior to cycling 3RC-66 (PORV), verify operability of 3RC-4 (POWER RELIEF BLOCK) as follows:

_____ 1.2.1 Close 3RC-4 (POWER RELIEF BLOCK).

_____ _____ 1.2.2 Open 3RC-4 (POWER RELIEF BLOCK)

_____ 1.3 Record data required on Enclosure 13.4 (Information Sheet) prior to opening 3RC-66 (PORV).

1.4 Open 3RC-66 (PORV) as follows::

_____ 1.4.1 Position the 3RC-66 SETPOINT SELECTOR to "OPEN"

NOTE:	3RC-66 remains open when the "OPEN PERMIT" Pushbutton is released until 3RC-66 SETPOINT SELECTOR position is changed to "LOW".

_____ _____ 1.4.2 Depress the 3RC-66 "OPEN PERMIT" Pushbutton.

_____ 1.5 Monitor the following to verify 3RC-66 (PORV) is open

- Quench Tank pressure increasing
- Quench Tank level increasing
- 3RC-66 (PORV) indicates open on Control Board
- PORV Flow Monitor indicates flow
- Pressurizer Relief Valve Flow Statalarm if 5 or more lights are lit.

_____ _____ 1.6 When there is positive indication that 3RC-66 (PORV) is open, select "Low" on 3RC-66 Setpoint Selector.

_____ 1.6.1 Verify 3RC-66 (PORV) is closed.

_____ _____ 1.6.2 If 3RC-66 (PORV) fails to close, close 3RC-4 (POWER RELIEF BLOCK) immediately.

_____ 1.7 Record data required on Enclosure 13.4 (Information Sheet) when 3RC-66 (PORV) is closed.

Enclosure 13.4

Information Sheet

UNIT ____________

Time	RCS Press ≤ 45 PSIG	Quench Tank Press	Quench Tank Level	Quench Tank Temp.	RC-66 Outlet Temp	PZR Level

Enclosure 13.4

Information Sheet

UNIT ____________

Time	RCS Press ≤ 45 PSIG	Quench Tank Press	Quench Tank Level	Quench Tank Temp.	RC-66 Outlet Temp	PZR Level
1337	38.0	.1	84	123	48	90

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

CRO-013/SIM
ALIGN MD EFDWP SUCTION TO THE HOTWELL AND FEED THE STEAM GENERATORS

CANDIDATE ____________________

EXAMINER ____________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

ALIGN MDEFDWP SUCTION TO THE HOTWELL AND FEED THE STEAM GENERATORS

Alternate Path:

N/A

Facility JPM #:

CRO-13

K/A Rating(s):

010A4.02 [3.6/3.4]

Task Standard:

The MDEFDWPs are aligned to the Hotwell and providing flow to the SGs within limits prior to reaching a level of 0" in the Hotwell. Step 8.0 of Section 503 of AP/1/A/1700/19 is properly completed.

Preferred Evaluation Location:

Simulator ___X__ In-Plant ______

Preferred Evaluation Method:

Perform ___X__ Simulate ______

References:

AP/1/A/1700/19

Validation Time: 20 min. **Time Critical:** NO

==

Candidate: ____________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ____________________________ ________________________________/________
NAME SIGNATURE DATE

==

COMMENTS

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall SNAP 209
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

AP/1/A/1700/19

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

The unit has experienced a Loss of Power. The TD EFDWP is unavailable. Actions of the EOP have been completed and power has been restored. AP/1/A/1700/19, Loss of Main Feedwater, Section 503 has been completed up to step 8.0. Main FDW is not expected back for several hours.

UST makeup flow capability has been lost.

INITIATING CUE:

The Control Room Supervisor directs you to continue with AP/1/A/1700/19, Loss of Main Feedwater.

START TIME: _______

STEP	SAT/UNSAT
STEP 1: WHEN the UST < 4 feet THEN dispatch two operators to perform EP/1/A/1800/01, Enclosure 7.7, "Operation of the Atmospheric Dump Valves". STANDARD: Determines UST is < 4 feet by monitoring: • OAC analog points • UST B LEVEL meter on 1AB-1 • UST A LEVEL meter on 1AB-3 • UST LEVEL chart recorder on 1VB-1 • Statalarm 1SA-6/A-11, Upper Surge Tank Level Low Dispatches two operators to Atmospheric Dump Valves. COMMENTS:	___ SAT ___ UNSAT

STEP 2: WHEN the UST level < 3 feet THEN align the Emergency Feedwater Pumps Suction to the Hotwell as follows: Stop all CBPs Stop all HWP's STANDARD: Determines UST is < 3 feet by monitoring: • OAC analog points • UST B LEVEL meter on 1AB-1 • UST A LEVEL meter on 1AB-3 • UST LEVEL chart recorder on 1VB-1 • Statalarm 1SA-6/A-11, Upper Surge Tank Level Low Places all CBP control switches in OFF Places all HWP control switches in OFF COMMENTS:	___ SAT ___ UNSAT

TEP 3: Control SG pressure with ADV's as necessary. STANDARD: ***CUE: Another RO is coordinating with an NLO to maintain SG pressure via ADVs.*** COMMENTS:	___ SAT ___ UNSAT
STEP 4: **IF** power is available **THEN** perform the following Open 1V-186 (Vacuum Breaker) STANDARD: Opens 1V-186 COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: Dispatch an operator with a safety harness to 1C-573 to standby until further notice. STANDARD: Dispatches an operator with a safety harness to 1C-573 ***CUE: Inform student that an operator has been dispatched to 1C-573.*** COMMENTS:	___ SAT ___ UNSAT
STEP 6: Close the following 1MS-47, MS to CSAE's 1AS-40, CSAE Aux Steam Supply STANDARD: Closes 1MS-47 Closes 1AS-40 COMMENTS:	___ SAT ___ UNSAT

TEP 7: Monitor and adhere to the following flow limits: MD EFDW Pump flow rates <440 gpm / pump (0.22 x 10^6 lbm / hr) STANDARD: Monitors MD EFDW Pump flow rates and throttles 1FDW-315 and 1FDW-316 as necessary to maintain < 440 gpm/pump. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 8: **IF AT ANY TIME** UST level ≤ 1 foot, **ND** 1C-573 (MD EFDWP Suction from UST) open. **THEN** secure all Emergency FDWPS. STANDARD: Monitors UST level and secures all EFDW pumps if level is ≤ 1 foot. COMMENTS:	**CRITICAL STEP**

STEP 9: **WHEN** vacuum is broken, **THEN** locally close 1C-573 (MD EFDWP Suction from UST). STANDARD: Monitors Vacuum and then closes 1C-573. COMMENTS:	**CRITICAL STEP**
STEP 10: Dispatch and operator to 1C-157 (TD EFDWP Suction from UST) to standby until further notice. STANDARD: Dispatch an operator to 1C-157. ***CUE: Inform student an operator has been dispatched to 1C-157.*** COMMENTS:	

TEP 11: Open 1C-391 (TD EFDWP Suction from Hotwell) STANDARD: Open 1C-391 COMMENTS:	**CRITICAL STEP**
STEP 12: Close 1C-157 (TD EFDWP Suction from UST) STANDARD: Instruct operator to close 1C-157. ***CUE: Inform student that 1C-157 is closed.*** COMMENTS: **END TASK**	

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
4	Condenser vacuum must be broken thus increasing the NPSH to the EFDWPs. This prevents EFDWP damage due to not meeting suction head requirements when Hotwell level is < 2 feet.
7	MD EFDWP flow is throttled via FDW-315 and 316 to limit flow < 440 gpm to prevent pump run-out damage.
8	1C-573 is closed to prevent air from entering the suction of the MD EFDWPs.
9	1C-573 is closed to prevent air from entering the suction of the MD EFDWPs.
11	1C-391 aligns a suction flow path to the TD EFDW pump from the Hotwell.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

The unit has experienced a Loss of Power. The TD EFDWP is unavailable. Actions of the EOP have been completed and power has been restored. AP/1/A/1700/19, Loss of Main Feedwater, Section 503 has been completed up to step 8.0. Main FDW is not expected back for several hours.

UST makeup flow capability has been lost.

INITIATING CUE:

The Control Room Supervisor directs you to continue with AP/1/A/1700/19, Loss of Main Feedwater.

Extended EFDW Operation

_____ 1. Monitor Emergency FDW parameters.

- **REFER TO** OAC Computer Group Display - GD AP19.

CAUTION 2:	Pump damage may occur if a MDEFDW Pump is operated with discharge pressure > 1420 psig and discharge flow < 110 gpm.

_____ 2. **IF** **AT ANY TIME** either MDEFDW Pump discharge pressure indicates > 1420 psig,

AND indicated flow from that MDEFDW Pump on the "MDEFWP DISCH FLOW" meter is < 110 gpm,

THEN perform the following:

_____ 2.1 Immediately stop the affected MDEFDW Pump.

_____ 2.2 Start the TDEFDW Pump.

_____ 2.3 **IF** NO Emergency FDW Pumps are available,

THEN **GO TO** Section 501, Establishing Emergency Feedwater.

_____ 3. **IF** both MDEFDW Pumps are running,

AND Emergency FDW flow in each header is < 600 gpm,

THEN perform the following:

_____ 3.1 Place the TDEFDW Pump switch in "PULL TO LOCK".

_____ 3.2 **GO TO** Step 5.

_____ 4. **IF** TDEFDW Pump is supplying feedwater to <u>any</u> SG,

THEN perform the following:

_____ 4.1 Place the TDEFDW Pump switch in "RUN".

_____ 4.2 Ensure 1LPSW-137 (LPSW TO TDEFDWP COOLING JACKET) is open.

_____ 4.3 Ensure the Emer FDWPT Brng Oil Cooling Pump is on.

5. Maintain UST level > 7 ft by performing the following as required:

_____ Make up to UST with Demin water

_____ Verify CST pumps are in "AUTO".

_____ 5.1 **IF** UST level **CANNOT** be maintained > 7 ft,

THEN dispatch an operator to close 1C-186 (Hotwell Emergency Makeup #1 Control Inlet). (T-1, West of E-24)

_____ 6. Establish Condensate System recirc.

- **REFER TO** Enclosure 6.2, "Aligning For Recirculation Of The Condensate System".

_____ 7. **IF** **AT ANY TIME** conditions allow Main FDW Pump restoration,

THEN **REFER TO** Section 504, <u>Establishing Main Feedwater</u>.

Extended EFDW Operation

_____ 8. **WHEN** UST level < 4 feet,

THEN dispatch two operators to perform EP/1/A/1800/001 (Emergency Operating Procedure) Enclosure 7.7, "Operation Of The Atmospheric Dump Valves".

CAUTION 9:	With vacuum broken, pump damage may occur if Emergency FDW Pump suction is NOT aligned to the Hotwell prior to reaching 1 foot in the UST.

_____ 9. **WHEN** UST level < 3 feet, {8}

THEN align Emergency FDW Pumps suction to the Hotwell as follows:

_____ 9.1 Stop all CBPs.

_____ 9.2 Stop all HWPs.

_____ 9.3 Control SG pressure with ADVs, as necessary.

_____ 9.4 **IF** power is available,

THEN perform the following:

_____ 9.4.1 Open 1V-186 (VACUUM BREAKER).

_____ 9.4.2 Dispatch an operator with a safety harness to standby until further notice near 1C-573 (MD EFDWPS Suction from UST). (T-1, E-24 South West, 8' above floor)

9.4.3 Close the following valves:

_____ 1MS-47 (MS TO CSAE)

_____ 1AS-40 (AS TO CSAE).

_____ 9.4.4 Monitor <u>and</u> adhere to the following flow limits:

- MDEFDW Pump flow rates < 440 gpm/pump (0.22 x 10^6 lbm/hr)
- TDEFDW Pump flow rate < 1000 gpm total (0.5 x 10^6 lbm/hr).

_____ 9.4.5 **<u>IF</u>** **<u>AT ANY TIME</u>** UST level ≤ 1 foot,

<u>AND</u> 1C-573 (MD EFDWPS Suction from UST) open,

<u>THEN</u> perform the following:

_____ 9.4.5.1 Stop <u>all</u> Emergency FDWPs.

_____ 9.4.5.2 **<u>IF</u>** vacuum **<u>CANNOT</u>** be broken,

<u>THEN</u> **GO TO** Step 10.

_____ 9.4.6 **<u>WHEN</u>** vacuum is broken,

<u>THEN</u> locally close 1C-573 (MD EFDWPS Suction from UST).

_____ 9.4.7 **<u>IF</u>** MDEFDW Pumps stopped in Step 9.4.5,

<u>THEN</u> restart MDEFDW Pumps.

NOTE 9.4.8:	The TDEFDW Pump is stopped to prevent air binding during the transfer from the UST to the Hotwell.

_____ 9.4.8 Ensure the TDEFDW Pump is NOT operating.

_____ 9.4.9 Dispatch an operator to 1C-157 (TD EFDWP Suction from UST) to standby until further notice. (T-1/C-20)

NOTE 9.4.10:	During the time 1C-391 (TD EFDWP SUCT FROM HOTWELL) and 1C-157 (TDEFDWP Suction from UST) are both open, water will drain from the UST to the Hotwell.

_____ 9.4.10 Open 1C-391 (TDEFDWP SUCT FROM HOTWELL).

_____ 9.4.11 Locally close 1C-157 (TD EFDWP Suction from UST). (T-1/C-20)

_____ 9.5 **IF** power is NOT available,

THEN perform the following:

_____ 9.5.1 Dispatch an operator with a safety harness to open 1V-186 (VACUUM BREAKER). (T-3, catwalk at 1C2 waterbox)

_____ 9.5.2 Dispatch an operator with a safety harness to standby until further notice near 1C-573 (MD EFDWPS Suction from UST). (T-1/E-24 South West, 8' above floor)

Extended EFDW Operation

_____ 9.5.3 Monitor and adhere to the following flow limits:

- MDEFDW Pump flow rates < 440 gpm/pump (0.22 x 10^6 lbm/hr)
- TDEFDW Pump flow rate < 1000 gpm total (0.5 x 10^6 lbm/hr).

_____ 9.5.4 **IF** **AT ANY TIME** UST level ≤ 1 foot,

AND 1C-573 (MD EFDWPS Suction from UST) is open,

THEN perform the following:

_____ 9.5.4.1 Stop all Emergency FDW Pumps.

_____ 9.5.4.2 **IF** vacuum **CANNOT** be broken,

THEN **GO TO** Step 10.

_____ 9.5.5 **WHEN** vacuum is broken,

THEN locally close 1C-573 (MD EFDWPS Suction from UST).

_____ 9.5.6 **IF** MDEFDW Pumps stopped in Step 9.5.4,

THEN restart MDEFDW Pumps.

NOTE 9.5.7:	The TDEFDW Pump is stopped to prevent air binding during the transfer from the UST to the Hotwell.

_____ 9.5.7 Ensure the TDEFDW Pump is **NOT** operating.

Extended EFDW Operation

_____ 9.5.8 Dispatch an operator to 1C-391 (TD EFDWP SUCT FROM HOTWELL) and 1C-157 (TD EFDWP Suction From UST). (T-1/C-20)

NOTE 9.5.9:	During the time 1C-391 (TD EFDWP SUCT FROM HOTWELL) and 1C-157 (TD EFDWP Suction from UST) are both open, water will drain from the UST to the Hotwell.

_____ 9.5.9 Locally open 1C-391 (TD EFDWP SUCT FROM HOTWELL). (T-1/C-20)

_____ 9.5.10 Locally close 1C-157 (TD EFDWP Suction from UST). (T-1/C-20)

9.5.11 Dispatch an operator to close the following valves:

_____ 1MS-49 (1A CSAE Steam Supply) (T-3/F-26)

_____ 1MS-58 (1B CSAE Steam Supply) (T-3/G-26)

_____ 1MS-67 (1C CSAE Steam Supply). (T-3/H-26)

Extended EFDW Operation

_____ 9.6 **IF** TDEFDW Pump operation is desired,

THEN perform the following:

_____ 9.6.1 **WHEN** suction alignment to the Hotwell is complete,

THEN start the TDEFDW Pump.

_____ 9.7 Monitor Emergency FDW Pump parameters:

- MDEFDW Pump flow rates < 440 gpm/pump (0.22 x 10^6 lbm/hr)
- TDEFDW Pump flow rate < 1000 gpm total (0.5 x 10^6 lbm/hr)
- Emergency FDW Pump discharge pressure > SG pressure.

_____ 9.8 Dispatch an operator to open 1C-188 (Hotwell Emerg Makeup #1 Control Bypass). (T-1, West of E-24) {3}

_____ 9.9 Notify the TSC to evaluate methods to maintain secondary inventory.

Extended EFDW Operation

_____ 10. **IF** **AT ANY TIME** the Hotwell is **NOT** available,

- Hotwell level ≤ 1 inch
- vacuum **CANNOT** be broken,

THEN perform the following:

_____ 10.1 Trip all Emergency FDWPs.

_____ 10.2 Perform EP/1/A/1800/001 (Emergency Operating Procedure) Rule #4, "Loss Of FDW".

_____ 10.3 **GO TO** Section 501, Establishing Emergency Feedwater.

••• **END** •••

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM CRO-95/SIM

Restore RBCUs to normal after an inadvertent ES Channel 5 actuation.

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

Restore RBCUs to normal after an inadvertent ES Channel 5 actuation.

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

022 A4.01 (3.6 / 3.6)

Task Standard:

The 1A and 1B RBCU are stopped. The 1A RBCU is restarted in HIGH speed.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

AP/1/A/1104/015, Reactor Building Cooling

Validation Time: 10 min. **Time Critical:** NO

==

Candidate: ________________________________
NAME

Time Start : _________
Time Finish: _________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: _________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall SNAP # ________
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/1/A/1104/015

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 BOP

An inadvertent actuation of ES Channel 5 has occurred on Unit 1.

1CC-7, 1LPSW-6, 1LPSW-15 and the "A" PR Ventilation fan have been return to their normal condition.

INITIATING CUES:

The SRO in the Control Room directs you to return the RBCUs to their normal alignment.

START TIME: _______

STEP 1: Obtain a copy of the appropriate procedure OP/1/A/1104/015, RBC System. STANDARD: Operator obtains a copy of OP/1/A/1104/015, RBC System. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Equipment on ES Channel 5 is taken to MANUAL. STANDARD: The MANUAL push button is depressed for the following equipment on ES Channel 5. • 1LPSW-565 (not critical) • 1LPSW-566 (not critical) • 1A RBCU • 1B RBCU • 1LPSW-18 (not critical) • 1LPSW-21 (not critical) COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: The 1A and 1B RBCUs are stopped to prevent RBCUs from running in mixed speed. ***CUE: Inform candidate that 30 minutes has passed.*** STANDARD: The 1A RBCU switch is place in the OFF position and then the ES reset push button is depressed. The 1A RBCU is verified off. The 1B RBCU switch is verified in the OFF position and then the ES reset push button is depressed. The 1B RBCU is verified off. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 4: The 1A RBCU is started in HIGH speed. STANDARD: The candidate rotates the 1A RBCU switch to the HIGH position and verifies that the RBCU goes to high speed. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5: Open 1 LPSW 565 STANDARD: Locates the switch and goes to open. Verifies that 1LPSW-565 opens. COMMENTS:	___ SAT ___ UNSAT
STEP 6: Throttles 1LPSW18 and 1LPSW-21 STANDARD: 1LPSW-18 and 1LPSW-21 are throttled so that ≈ 1400 gpm of LPSW is achieved to both the 1A and 1B RBCUs. COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANTIONS:

STEP #	Explanation
2	This step is required, because components must be placed in MANUAL to be able to reposition them.
3	This step is required, because the RBCUs should not be run in a mixed speed configuration.
4	This step is required, because HIGH speed is the normal alignment for the RBCUs.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the unit 1 BOP

An inadvertent actuation of ES Channel 5 has occurred has occurred on Unit 1.

1CC-7, 1LPSW-6, 1LPSW-15 and the "A" PR Ventilation fan have been return to their normal condition.

INITIATING CUES:

The SRO in the Control Room directs you return the RBCUs to their normal alignment.

(R06-97)

SR
Sum
NRC
106
115
JPP/JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No OP/1/A/1104/015

Revision No 020

REPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Reactor Building Cooling System

(4) Prepared By Dennis L. Masteller (Signature) Dennis L. Masteller Date 03/07/00

(5) Requires 10CFR50.59 evaluation?
☐ Yes (New procedure or revision with major changes)
X No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By George Ridgeway (QR) Date 3/9/00

Cross-Disciplinary Review By ______ (QR)NA GAR Date ______

Reactivity Mgmt. Review By ______ (QR)NA GAR Date ______

(7) Additional Reviews

Reviewed By [signature] Date 3/9/00

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By [signature] Date 3/13/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification:

☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Reactor Building Cooling System**	Procedure No. OP/1/A/1104/015
	Revision No. 020
Multiple Use	Electronic Reference No. OX002VM0

Reactor Building Cooling System

1. Purpose

To describe the proper method for operating Reactor Building Cooling System.

2. Limits And Precautions

2.1 RBCUs should **NOT** be started, stopped, or speed changed to equalize fan run times.

NOTE:
- A step of "OFF" to "LOW" (0-600 rpm) is considered one start.
- A step of "OFF" to "HIGH" (0-600-1200 rpm) is considered one start.

2.2 RBCU motor shall be off for 30 minutes, or allowed to operate for 30 minutes, prior to starting or changing speed.

2.3 30 minute speed change time interval may be waived in emergencies.

2.4 Manual speed changes should be minimized where possible.

2.5 During non-emergency operation, maximum RBCU motor bearing temperature: 220°F. (computer point: RBV CLR FAN IB/OB BRG TEMP).

2.6 1B RBCU may be operated while LPSW is diverted to Aux Fan Coolers.

2.7 Do **NOT** operate RBCUs in mixed speed combinations. Excess back pressure is placed on low speed fans.

2.8 Proper damper operation is **NOT** required for RBCU operability per Improved Technical Specifications (ITS).

- If dampers are **NOT** operating properly, high vibration and temperature problems may be encountered. {1}

2.9 When Reactor Building Cooling System Operability is required (TS 3.6.5, MODES 1, 2, 3, and 4), LPSW flow to all RBCUs must be ≥ 550 gpm. {2}

- If LPSW to an RBCU is < 550 gpm, LCO 3.0.3 applies. {2}

3. Enclosures

3.1 RBCU System Startup

3.2 RBCU Operation

3.3 Reduction Of Cooling Capacity

3.4 Valve Alignment For Temporary RB Chiller During Unit Outage

* Appendix *

Information Use

1. Initial Conditions

1.1 All RBCUs shutdown.

1.2 Review Limits and Precautions.

2. Procedure

2.1 Start desired number of RBCUs by selecting "HIGH" or "LOW":

- 1A RBCU
- 1B RBCU
- 1C RBCU

NOTE:
- 1LPSW-21 should remain open if RB Aux Fans require LPSW.
- Do **NOT** throttle RBCU LPSW flow < 550 gpm. {2}

2.2 Position RBCU valves as required for RB cooling:

- 1LPSW-18 (1A RBCU OUTLET)
- 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)
- 1LPSW-565 (RB AUX FANS COOLERS INLET)
- 1LPSW-24 (1C RBCU OUTLET)
- 1LPSW-566 (1B RBCU ISOLATION)

2.3 Verify RBCU Damper on shutdown fan(s) closed.

2.4 Verify RBCU Dampers on operating fan(s) open.

NOTE:
- If 1B RBCU paired with 1A or 1C RBCU, light closest to paired fans should be lit.
- For all other combinations both 1A and 1C lights should be off.

2.5 Verify 1B RBCU Dampers positioned properly.

Information Use

1. Initial Conditions

1.1 RBCU(s) operating.

1.2 Review Limits and Precautions.

2. Procedure

2.1 To stop RBCU(s), place desired switch to "OFF":

- 1A RBCU
- 1B RBCU
- 1C RBCU

2.2 To start RBCU(s), place desired switch to "HIGH" or "LOW":

- 1A RBCU
- 1B RBCU
- 1C RBCU

NOTE:
- 1LPSW-21 should remain open if RB Aux Fans require LPSW.
- Do **NOT** throttle RBCU LPSW flow < 550 gpm. {2}

2.3 Position RBCU valves as required for RB cooling:

- 1LPSW-18 (1A RBCU OUTLET)
- 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)
- 1LPSW-565 (RB AUX FANS COOLERS INLET)
- 1LPSW-24 (1C RBCU OUTLET)
- 1LPSW-566 (1B RBCU ISOLATION)

2.4 Verify RBCU Damper on shutdown fan(s) closed.

2.5 Verify RBCU Dampers on operating fan(s) open.

NOTE:	• If 1B RBCU paired with 1A or 1C RBCU, light closest to paired fans should be lit. • For all other combinations both 1A and 1C lights should be off.

2.6 Verify 1B RBCU Dampers positioned properly.

Information Use

1. Initial Conditions

1.1 Reduction of cooling to RxV and SG cavities desired.

1.2 Review Limits and Precautions.

2. Procedure

NOTE:	• 1LPSW-21 should remain open if RB Aux Fans require LPSW. • Do **NOT** throttle RBCU LPSW flow < 550 gpm. {2}

2.1 Place control switch of operating fan(s) in "OFF" or "LOW" as required:

- 1A RBCU
- 1B RBCU
- 1C RBCU

Continuous Use

1. Initial Conditions

NOTE: 1B RBCU and RB Aux Fans will be available for Loss of Decay Heat Removal heat sink.

_____ _____ 1.1 Unit is in MODE 5 or below. {2}

SRO

_____ 1.2 Temporary RB Chiller installed per Maintenance procedure.

_____ 1.3 Review Limits and Precautions.

2. Procedure

_____ 2.1 Establish communications with Maintenance personnel at Chiller.

NOTE: 1B RBCU and RB Aux Fans supplied by non-safety related cooling water source.

_____ 2.2 Unlock and White Tag closed 1LPSW-82 (RBCU 1B Outlet Block). (A-4-E Pen Rm)

_____ 2.3 IF desired to place Chilled Water through Aux Fans, ensure Open 1LPSW-565 (RB AUX FAN COOLER INLET).

_____ 2.4 IF desired to place Chilled Water through 1B RBCU, ensure Open 1LPSW-566 (1B RBCU ISOLATION).

2.5 Isolate 1B RBCU:

2.5.1 Unlock and White Tag closed the following valves: (A-4-E Pen Rm)

- _____ 1LPSW-79 (RBCU 1A & 1B Supply Tie)
- _____ 1LPSW-80 (RBCU 1B & 1C Supply Tie)

2.5.2 Close the following valves:

- _____ 1LPSW-19 (1B & AUX FAN COOLER RBCU INLET)
- _____ 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)

NOTE:	1LPSW-878 opened per maintenance procedure.

2.6 Ensure open and White Tag open the following valves: (A-4-E Pen Rm)

_____ • 1LPSW-878 (RBCU Chilled Water Supply)

_____ • 1LPSW-879 (RBCU Chilled Water Return)

_____ 2.7 Notify Maintenance to start Temporary RB Chiller.

Maintenance ______________________________

2.8 Run 1B RBCU Fan as required.

2.9 **WHEN** changing chilled water flow paths, notify Maintenance to monitor Temporary RB Chiller operation.

2.10 **WHEN** notified by Maintenance, perform the following: (A-4 E Pen Rm)

_____ 2.10.1 Verify Temporary RB Chiller shutdown.

_____ 2.10.2 Remove White Tag and Close 1LPSW-878 (RBCU Chilled Water Supply).

_____ 2.10.3 Remove White Tag and Close 1LPSW-879 (RBCU Chilled Water Return).

2.11 Remove White Tags from the following: (A-4-E Pen Rm)

_____ • 1LPSW-79 (RBCU 1A & 1B Supply Tie)

_____ • 1LPSW-80 (RBCU 1B & 1C Supply Tie)

_____ • 1LPSW-82 (RBCU 1B Outlet Block)

2.12 Prior to entering MODE 4, return 1B RBCU to service as follows:

2.12.1 Open and lock open the following: (A-4-E Pen Rm)

_____ • 1LPSW-79 (RBCU 1A & 1B Supply Tie)

_____ • 1LPSW-80 (RBCU 1B & 1C Supply Tie)

_____ • 1LPSW-82 (RBCU 1B Outlet Block)

_____ 2.12.2 Open 1LPSW-19 (1B RBCU & AUX FAN COOLER INLET).

_____ 2.12.3 Position 1LPSW-566 (1B RBCU ISOLATION) as required.

_____ 2.12.4 Position 1LPSW-565 (RB AUX FAN COOLER INLET) as required.

_____ 2.12.5 Position 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET) as required.

Appendix

1. Limit and Precaution added in response to PIP 2-O90-0080. Isolation dampers, located below the fusible patches, serve no safety function. Since ductwork below coils/patches is assumed to be "crimped" after a LOCA blowdown (which closes the duct), misalignment of dampers has no effect on RBCU's expected post-accident response, which is to shift to low speed, open discharge valve, and drop patches. Therefore, an alignment problem with isolation dampers **CANNOT** make RBCUs inoperable from a nuclear safety standpoint.

2. Added in response to PIP 2-O98-3629. RBCU LPSW flow ≥ 550 gpm is required when RCS ≥ 200°F and ≥ 300 psig.

OTC CONTROL COPY

JOB PERFORMANCE MEASURE
CRO-095

Prepared By:	Bobby Ayers	Date:	3-29-99
Reviewed By:	[illegible]	Date:	4-21-99
Approved By:	[illegible]	Date:	4-23-99

TASK TITLE: **SWAP RBCUs**

TASK NUMBER: OO2630501

K/A REFERENCE: System: SF5-022 Containment Cooling System (CCS)
K/A: A 4.01
Rating: 3.6/3.6

CLASSIFICATION: **SRO** X **CRO** X **NLO** _

EVALUATION METHOD: **PERFORM** X **SIMULATE** _

EVALUATION LOCATION: **PLANT** _ **SIMULATOR** X **CONTROL ROOM** _

TASK STANDARDS: RBCU 1C is stopped and re-started per OP/0/A/1104/15

APPROXIMATE COMPLETION TIME: 5 minutes

REQUIRED TOOLS OR MATERIALS: OP/0/A/1104/15

REFERENCES: OP/0/A/1104/15

READ TO STUDENT

When I tell you to begin, you are to **STOP/RE-START REACTOR BUILDING COOLING UNIT 1C**. Before you start, I will describe the general plant conditions, state the initiating cues, and answer any questions. Perform procedure steps and make notifications as if you were actually performing the task.

INITIAL CONDITIONS:

Unit 1 is at 100% power.
1A, 1B, and 1C RBCUs are operating in high speed
Maintenance requests to perform PMs on 1C RBCU breaker

INITIATING CUE:

SRO in Control Room, instructs you, the BOP operator, to stop RBCUs 1C and following maintenance activities re-start 1C RBCU per OP/0/A/1104/15, RB Cooling.

ARE THERE ANY QUESTIONS?

JPM INSTRUCTION SHEET

DIRECTIONS TO STUDENT:

When I tell you to begin, you are to **STOP/RE-START REACTOR BUILDING COOLING UNIT 1C**. Before you start, I will describe the general plant conditions, state the initiating cues, and answer any questions. Perform procedure steps and make notifications as if you were actually performing the task.

INITIAL CONDITIONS:

Unit 1 is at 100% power.
1A, 1B, and 1C RBCUs are operating in high speed
Maintenance requests to perform PMs on 1C RBCU breaker

INITIATING CUE:

SRO in Control Room, instructs you, the BOP operator, to stop RBCU 1C and following maintenance activities re-start 1C RBCU per OP/0/A/1104/15, RB Cooling.

***Denotes Critical Step**
#Denotes Sequential Step

#	STEP	STANDARD	S/U
1	Review Limits and Precautions.	OP/0/A/1104/15, Enclosure 3.2 Limits and Precautions are obtained and reviewed prior to operation of RBCUs.	
*2	Select OFF on 1C RBCU. **CUE: After 1C RBCU is stopped inform operator that PMs are complete and 1C RBCU is ready to be returned to service.**	1C RBCU controls are located on 1AB3 and switch is rotated to the OFF position. Red HIGH indicating light is OFF, Green OFF light is ON, Green damper CLOSED light ON. Verify 1C RBCU Discharge Damper closed **NOTE: Sequence of the critical steps is <u>not</u> critical. Damper reposition not critical**	
*3	Start 1C RBCU.	1C RBCU controls are located on 1AB3 and switch is rotated to the LOW or HIGH position. Amber LOW speed light ON, ammeter responds upscale, then fan shifts to HIGH speed. Verify Red HIGH indicating light is ON, Green OFF and Amber LOW lights OFF. 1C RBCU Discharge Damper opens **CUE: If requested, inform student to leave LPSW lined up to Aux Fan Coolers.**	
4	Position the RBCU valves as required for RB Cooling.	Verify valves OPEN: 1LPSW-18 (1A RBCU OUTLET) 1LPSW-21 (1B RBCU OUTLET) 1LPSW-565 (RB AUX FANS INLET) 1LPSW-566 (1B RBCU ISOLATION) 1LPSW-24 (1C RBCU OUTLET) Verify Red OPEN indicating light is ON, Green CLOSE light ON.	
6	Verify 1C RBCU Damper OPEN.	1C RBCU Damper indication is located on 1AB3, and Red OPEN indicating light is ON, Green CLOSE light is OFF.	

END TASK

CRITICAL STEP EXPLANATION:

STEP # - EXPLANATION

2 – Stop 1C RBCU

3 – Start 1C RBCU

#	QUESTION	CORRECT ANSWER
1	Explain why "B" RBCU has several damper positions.	"B" RBCU has the ability to feed into either or both of the cavity duct works. By angling its dampers it may direct the air discharge as needed depending on the RBCU combination running.

Student Response	S/U

K/A #	A4.03	Rating:	3.2/3.2

#	QUESTION	CORRECT ANSWER
2	Briefly describe the actions required to regain control of RBCU operation after ES actuation has occurred.	Place RZ module in Manual, or have initiating signal cleared. Depress "Push to Reset" Pushbutton on 1AB3 associated with each RBCU. Then switch will respond.

Student Response	S/U

K/A #	A4.01	Rating:	3.6/3.6

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM CRO-009/SIM

FOLLOWING KEOWEE EMERGENCY START, TRANSFER MFB POWER FROM CT-4 TO CT-5

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

FOLLOWING KEOWEE EMERGENCY START, TRANSFER MFB POWER FROM CT-4 TO CT-5

Alternate Path:

Facility JPM #:

N/A

K/A Rating(s):

062-A4.01 3.3/3.1

Task Standard:

Auxiliary power is swapped from CT-4 to CT-5.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

OP/0/A/1106/19 Encl. 3.12

Validation Time: 10 min. **Time Critical:** NO

===

Candidate: ____________________________________
NAME

Time Start: ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: __________________________ ______________________________/________
NAME SIGNATURE DATE

===

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC # SNAP ______
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/0/A/1106/19 Encl. 3.12

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

CT-1 is out of service for repairs. A switchyard isolation has resulted in a reactor trip and Unit 1's Main Feeder Busses are being supplied from CT-4 via the Standby Busses. Keowee personnel have requested that the Keowee units be shutdown. CT-5 has been energized from a Lee Gas Turbine and the dedicated path, bypassing the Central switchyard, has been established. OP/0/A/1106/19, Keowee Hydro at Oconee, Enclosure 3.12 has been completed up to step 2.1.4.

INITIATING CUE:

The Control Room SRO directs you to utilize Enclosure 3.12 of OP/0/A/1106/19, Keowee Hydro at Oconee, to transfer MFB power from CT-4 to CT-5 beginning at step 2.1.4.

START TIME: _______

STEP 1: 2.1.4 Place the following transfer switches in MANUAL: • CT-4 BUS 1 AUTO/MAN • CT-4 BUS 2 AUTO/MAN • CT-5 BUS 1 AUTO/MAN • CT-5 BUS 2 AUTO/MAN STANDARD: The following transfer switches are placed in the MANUAL position: • CT-4 BUS 1 AUTO/MAN • CT-4 BUS 2 AUTO/MAN • CT-5 BUS 1 AUTO/MAN **Not Critical** • CT-5 BUS 2 AUTO/MAN **Not Critical** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 2: 2.1.5 Open SK 1 (CT-4 Stby Bus 1 Feeder). STANDARD: SK 1 (CT-4 Stby Bus 1 Feeder) is OPENED. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 3: 2.1.6 Energize the STBY BUSES from CT-5. STANDARD: The following breakers are operated in the listed sequence: SK 2 (CT-4 Stby Bus 2 Fdr) is OPENED. SL 1 (CT-5 Stby Bus 1 Fdr) is CLOSED. SL 2 (CT-5 Stby Bus 2 Fdr) is CLOSED. **NOTE: The time period between opening SK2 and closing SL1 should be > 3 seconds and < 20 seconds.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: 2.1.7 Return the following transfer switches to AUTO: • CT-4 BUS 1 AUTO/MAN • CT-4 BUS 2 AUTO/MAN • CT-5 BUS 1 AUTO/MAN • CT-5 BUS 2 AUTO/MAN STANDARD: The following transfer switches are placed in the AUTO position: • CT-4 BUS 1 AUTO/MAN • CT-4 BUS 2 AUTO/MAN • CT-5 BUS 1 AUTO/MAN • CT-5 BUS 2 AUTO/MAN COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	Step is required, because transfer switches have to be in MANUAL for the breakers to be operated.
2	Step is required, because the SK breaker must be open for the SL breaker to be closed.
3	Step is required, because this is the proper sequence to provide power from CT-5.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

CT-1 is out of service for repairs. A switchyard isolation has resulted in a reactor trip and Unit 1's Main Feeder Busses are being supplied from CT-4 via the Standby Busses. Keowee personnel have requested that the Keowee units be shutdown. CT-5 has been energized from a Lee Gas Turbine and the dedicated path, bypassing the Central switchyard, has been established. OP/0/A/1106/19, Keowee Hydro at Oconee, Enclosure 3.12 has been completed up to step 2.1.4.

INITIATING CUE:

The Control Room SRO directs you to utilize Enclosure 3.12 of OP/0/A/1106/19, Keowee Hydro at Oconee, to transfer MFB power from CT-4 to CT-5 beginning at step 2.1.4.

1. Initial Conditions

_____ 1.1 Keowee Units have been started by emergency actuation and it is desired to shut down the Keowee Units.

_____ 1.2 It is desired to supply power from CT 5.

_____ 1.3 Review Limits and Precautions.

2. Procedure

2.1 Perform a Dead Bus transfer to CT5 from CT4 while CT4 is <u>supplying</u> **Unit 1, 2, <u>OR</u> 3** MFB by:

_____ _____ 2.1.1 Verify CT 5 is energized and ready to power auxiliary loads.

2.1.2 Prior to performing Dead Bus transfer, notify the following:

_____ • Security Force

_____ • Chemistry Department

_____ • Group Heads

_____ • Keowee Operator

_____ 2.1.3 Verify reset **<u>OR</u>** reset MFB Monitor Panel for any Oconee Units receiving power from the STBY Buses.

2.1.4 Place the following transfer switches in "MANUAL":

_____ _____ • CT 4 BUS 1 AUTO/MAN

_____ _____ • CT 4 BUS 2 AUTO/MAN

_____ _____ • CT 5 BUS 1 AUTO/MAN

_____ _____ • CT 5 BUS 2 AUTO/MAN

_____ _____ 2.1.5 Open SK 1 CT 4 STANDBY BUS 1 FEEDER.

CAUTION:	Transfer should be made in > 3 but < 20 seconds to prevent picking up MFB Monitor Panel actuation which will cause a Load Shed, Keowee Emergency start and possible EPSL actuation. Undervoltage relays will cause a loss of most non-safety loads.

2.1.6 Energize STBY BUSES from CT 5 by performing the following:

_____ _____ A. Open SK 2 CT 4 STBY BUS 2 FEEDER.

_____ _____ B. Close SL-1 CT 5 STBY BUS 1 FEEDER.

_____ _____ C. Close SL-2 CT 5 STBY BUS 2 FEEDER.

2.1.7 Return the following Transfer Switches to "AUTO":

_____ _____ • CT4 BUS 1 AUTO/MAN

_____ _____ • CT4 BUS 2 AUTO/MAN

_____ _____ • CT5 BUS 1 AUTO/MAN

_____ _____ • CT5 BUS 2 AUTO/MAN

_____ 2.1.8 Recover any loads lost in transfer.

NOTE:	<u>IF</u> Keowee Unit(s) are generating with Overhead ACB closed prior to an Emergency Start Actuation, that Keowee Unit(s) will shutdown when ES Channel has been reset unless ACB is currently closed.

2.2 When all three units no longer require an energized Underground Power Path and a Normal Lockout does <u>NOT</u> exist on either Keowee Unit supplying power to an Oconee Unit, completely shut down the Keowee Unit tied to the Underground by:

2.2.1 <u>IF</u> ES 1 <u>OR</u> 2 has actuated, either reset ES 1 and 2 channels <u>OR</u> press "MANUAL" on the following ES 1 and 2 modules:

_____ _____ • Keowee Emer Start Ch A

_____ _____ • Keowee Emer Start Ch B

_____ _____ • Load Shed and STBY Bkr 1

_____ _____ • Load Shed and STBY Bkr 2

2.2.2 <u>IF</u> a manual Keowee Emergency start has been performed from any Oconee Unit, return both Keowee Emergency Start Channel switches on the affected unit to "OFF" position.

_____ _____ • Keowee Emergency Start Channel A

_____ _____ • Keowee Emergency Start Channel B

_____ _____ 2.2.3 Reset <u>OR</u> verify reset Main Feeder Bus Monitor Panels.

2.2.4 Reset External Grid Trouble Protection System by depressing the following buttons. (**Unit 1/2**):

_____ _____ • GRID TROUBLE PROTECTIVE SYSTEM U.V. CHANNEL 1 RESET

_____ _____ • GRID TROUBLE PROTECTIVE SYSTEM U.V. CHANNEL 2 RESET

_____ _____ • GRID TROUBLE PROTECTIVE SYSTEM U.F. CHANNEL 1 RESET

_____ _____ • GRID TROUBLE PROTECTIVE SYSTEM U.F. CHANNEL 2 RESET

2.2.5 Verify External Grid Trouble Protection has been reset. **(Unit 1/2)**:

_____ • SA-15, A-2 Channel #1 Underfrequency

_____ • SA-15, A-4 Channel #2 Underfrequency

_____ • SA-15, C-1 Channel #1 Undervoltage

_____ • SA-15, C-3 Channel #2 Undervoltage

NOTE:	External Grid Trouble Protection System actuates Keowee Emergency Start from Oconee **Unit 1** circuitry.

2.2.6 Depress Keowee "PUSH TO RET TO NORMAL AFT ES RESET" pushbutton on ALL Oconee Units which have generated a Keowee Emergency Start signal:

A. **Unit 1**

_____ _____ • KEOWEE LOGIC RESET CHANNEL 1

_____ _____ • KEOWEE LOGIC RESET CHANNEL 2

B. **Unit 2**

_____ _____ • KEOWEE LOGIC RESET CHANNEL 1

_____ _____ • KEOWEE LOGIC RESET CHANNEL 2

C. **Unit 3**

_____ _____ • KEOWEE ES CHANNEL A

_____ _____ • KEOWEE ES CHANNEL B

_____ 2.2.7 Notify Keowee Operator to shutdown the Keowee Hydro Unit(s) per OP/0/A/2000/041 (Keowee - Modes of Operation).

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM NRC-1998
(ALTERNATE PATH)

RIA-57 Operability Check (RIA-57 fails to meet acceptance criteria following maintenance)

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:
RIA-57 Surveillance
Operation of RIA 57

Alternate Path:
RIA-57 Area Monitor Fault alarm received when performing RIA-57 operability check per PT/230/01

Facility JPM #:
JPM NRC-1998
Used during the 1998 NRC exam as a portion of the Admin exam
Not an assigned to the Oconee JPM bank

K/A Rating(s):
RO/SRO - 072A2.02 [2.8/2.9], RIA Detector Failure and 2.2.12 [3.0/3.4], RIA-57 Surveillance
SRO - 2.1.12 [2.9/4.0], Ability to apply TS for a system

Task Standard:
1. RIA-57 declared inoperable and work request initiated to have I&E correct the problem.
2. R.P. notified that RIA-57 is inoperable.
3. SRO ONLY - T.S. 3.3.8 is referred to for LCO guidelines.

Preferred Evaluation Location:
Simulator __X___ In-Plant __X___

Preferred Evaluation Method:
Perform _____ Simulate __X___

References:
PT/0/A/0230/001

Validation Time: __10-15 minutes_____ **Time Critical:** NO

===

Candidate: ______________________________ Time Start:: _______
NAME Time Finish: _______

Performance Rating: SAT _____ UNSAT _____ Question Grade ____ Performance Time _____

Examiner: ______________________ ______________________________ / ________
NAME SIGNATURE DATE

===

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. RECALL ______
2. IMPORT Files for CRO-SCM+RIA
3. T1 = Area Monitor Fault S/A
4. T2 = Area Monitor Fault clear
5. At the Examiner cue actuate Timer #1 and 2

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:
PT/0/A/0230/001, and Technical Specifications

READ TO OPERATOR

DIRECTION TO TRAINEE:

When I tell you to begin, you are to check the operability of RIA-57 at the individual monitor. Before you start, I will describe the general plant conditions, state the initiating cues, and answer any questions. Perform procedure steps and make notifications as if you were actually performing the task.

INITIAL CONDITIONS:

Unit is at 100% power

RIA-57 has just been returned to service following detector replacement.

INITIATING CUES:

The Control Room Supervisor directs you to check the operation of RIA-57 from the individual monitor per PT/0/A/0230/001.

START TIME:________

<table>
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STEP 1:

VERIFY alarm setpoints.

STANDARD:
- Locate_RIA-57 Individual Monitor on VB2/3

From the key pad:
- Depress "Clear" **CUE: Clear depressed**
- "Enter" the number 009 **CUE: 009 depressed**
- Depress "Item." **CUE: This will display the High alarm setpoint of :**

5.9 E+4 RAD/hr.

- **Depress "+" CUE: + depressed and 010 indicated in the window**
- Depress "item" **CUE: this will display the Alert alarm setpoint of**

5.9 E+3 RAD/hr

COMMENTS:

</td><td>

CRITICAL STEP

____ SAT

____ UNSAT

</td></tr>
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STEP 2:

Return to normal.

STANDARD:
- Select "Clear" **CUE: Clear depressed**
- Depress R/hr button **CUE: R/hr depressed. (Display returns to actual indication)**

COMMENTS:

</td><td>

____ SAT

____ UNSAT

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<u>STEP 3*:</u>

Perform Check source.

Depress “C/S” button. CUE: C/S depressed
Check readings between (Units 1 and 3, 5 E-1 and 1.0 E+0; Unit 2, 1 E+0 and 1.5) E+0 CUE: Reading indicates 1.5 E+1
Check “Area Monitor Fault” alarm NOT in If in Control Room –

CUE: Area Monitor Fault” alarms in the simulator or if in the control room, Tell the candidate the alarm has alarmed (SA-8, A-10, AREA MONITOR FAULT.

Refers to Operability Criteria for RIA-57 Enclosure 13.3

<u>STANDARD:</u>

Depress “C/S” button.
Understand that the RIA reading at 1.5E+1 is not normal
Understand the “Area Monitor Fault” alarm is not a normal response and refer to the Alarm Response Guide (ARG) for SA-8, A10…Issue WR for I&E
Refers to Operability Criteria for RIA-57 Enclosure 13.3 and determines that RIA-57 is <u>NOT</u> operable.

<u>COMMENTS:</u>
</td>
<td>
CRITICAL STEP

____ SAT

____ UNSAT
</td>
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</table>

STEP 4*: INITIATE Enclosure 13.3 corrective actions. 1. Initiate work request to have I&E correct the problem. ***CUE: Work request has been initiated.*** 2. Notify R.P RIA-57 is inoperable. ***CUE: R.P. has been notified.*** 3. List RIA-57 on the Control Room Shift Turnover sheet in the T.S. section. ***CUE: RIA-57 listed on TS Turnover Sheet*** 4. Refer to T.S. 3.3.8 for LCO guidelines ***CUE: If RO candidate, advise him that another operator will refer to the TS. IF SRO CANDIDATE, he should refer to the ITS for proper LCO determination.*** STANDARD: 1. Initiate work request to have I&E correct the problem. 2. Notify R.P that RIA-57 is inoperable. 3. List RIA-57 on the Control Room Shift Turnover sheet in the T.S. section. 4. **SRO ONLY -** Refer to T.S. 3.3.8 for LCO guidelines. Table 3.3.8-1 Accident Monitoring Inst. Item # 9, 2 of 2 channels required => 1 OOS, Condition A - restore w/in 30 days. COMMENTS:	____ SAT ____ UNSAT **CRITICAL STEP** **CRITICAL STEP SRO ONLY**

TIME STOPPED: ____________________

CRITICAL STEP EXPLANATIONS

STEP #	Explanation
1	This step is critical to ensure the monitor will alarm at preset accident values.
3	This step is critical to ensure detector operability.
4 (3)	Step is required to ensure that other operators on the unit are aware of the status with RIA-57.
4 (4)	SRO ONLY, Step is required to determine TS requirements.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

RO and SRO

INITIAL CONDITIONS:

RIA-57 has just been returned to service following detector replacement.

INITIATING CUES:

The Control Room Supervisor directs you to check the operation of RIA-57 from the individual monitor per PT/0/A/0230/001.

Operation Of RIA-57 And 58

1.0 From Individual Monitors

1.1 Verify alarm setpoints by this sequence:

- Depress "Clear"
- "Enter" the number 009
- Depress "Item." This will display the High alarm setpoint.
- Depress "+" (This will display the numbers 010)
- Depress "Item." This will display the Alert alarm setpoint.

1.2 Select "Clear." To return to normal, depress R/hr button.

1.3 To perform Checksource for 1, 3RIA-57, and 1, 2, 3RIA-58, depress "C/S" button. Readings should remain between 5.0 E-1 and 1.0 E+0 and **NOT** give an "Area Monitor Fault" alarm. **IF** alarm is received, refer to Enclosure "Operability Criteria For RIA-57 and RIA-58".

To perform Checksource for 2RIA-57, depress "C/S" button. Readings should remain between 1 E+0 and 1.5 E+0 and **NOT** give an "Area Monitor Fault" alarm. **IF** alarm is received, refer to Enclosure "Operability Criteria For RIA-57 and RIA-58".

Operability Criteria For RIA-57 And RIA-58

Purpose: The radiation indication alarm system is operable when it is capable of performing its intended function within the required range.

Criteria: The RIA system is considered to have this capability when:

1. Periodic calibration has been completed.
2. No Area Monitor Fault alarms exists due to RIA-57 and RIA-58.
3. 1,3RIA-57 and 1,2,3RIA-58 indicate a reading of between 5.0 E-1 to 1.0 E+0 R/hr on either the CRT or individual monitors.
4. 2RIA-57 indicates a reading of between 1.0 E+0 to 1.5 E+0 R/hr on either the CRT or individual monitors.

NOTE:	RIA-57 and 58 individual Display Modules have backup power available from the station battery system. The RIA-57 and 58 Monitors are powered from non-loadshed power panels SKJ and SKK.

IF the above conditions cannot be met:

1. Operations shall consider the RIA inoperable and initiate a work request to have I&E correct the problem.
2. Operations shall notify R.P. of the inoperable RIA and list it on the appropriate log sheets.
3. Operations shall refer to TS 3.3.8 for guidelines.

1,2,3 RIA-56

Function: Monitor gaseous effluent from station vent, (High Range Post-Accident).

Setpoint : Alert - Set at 10 RAD/hr

High - Set at 20 RAD/hr

Bases: Reg Guide 1.109

ODCM

NUMARC EALs

A setpoint of 10 RAD/hr is based on the Emergency Action Level from NUMARC EALs that would indicate a Site Area Emergency (500 mRAD/hr to the Whole Body).

A setpoint of 20 RAD/hr is based on the Emergency Action Level from NUMARC EALs that would indicate a General Emergency (1 RAD/hr to the Whole Body).

1,2,3 RIA 57 and 58

Function: Post Accident Containment High Range Monitors

Setpoint:	Alert -	RIA-57 5.9 E+3 RAD/hr	RIA-58 2.6 E+3 RAD/hr
	High -	RIA-57 5.9 E+4 RAD/hr	RIA-58 2.6 E+4 RAD/hr

Bases: Emergency Action Levels, Oconee Nuclear Site

The "Alert" setpoint is based on the dose rate necessary inside the containment building to reach one tenth of the protective action guide levels over a one hour period at the site boundary (1 mile). One tenth of the protective action guide level is 500 mrem thyriod dose (thyroid dose is the limiting pathway). This is also the reading at which a Site Area Emergency should be declared. Setpoint was calculated using annual average meteorology, design basis leak rate, and is for the time period up to 1/2 hour after Unit trip.

The "High" setpoint is based on the dose rate necessary inside the containment building to reach the protective action guide levels over a one hour period at the site boundary (1 mile). The protective action guide level is 5 REM thyroid dose (thyroid dose is the limiting pathway). This is also the reading at which a General Emergency should be declared. Setpoint was calculated using annual average meteorology, design basis leak rate, and is for the time period up to 1/2 hour after Unit trip.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM IA.1-2

ADMIN. TOPICS:
A.1 Shift Turnover (Short-term Information)
A.2 (Surveillance Testing)

(ALTERNATE PATH)

RO and SRO JPM

CANDIDATE ______________________________

EXAMINER ______________________________

Bank
1998 exam

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:
_RIA-57 Surveillance
Manage Short Term Information (Operation of RIA 57). Documentation of OOS RIA on turnover sheet and Tech. Spec log.

Alternate Path:
_RIA-57 Area Monitor Fault alarm received when performing_RIA-57 operability check

Facility JPM #:
NEW

K/A Rating(s):
2.2.12 [3.0/3.4] -_RIA-57 Surveillance
2.1.3 [3.0/3.4] - Manage Short Term Information; Turnover sheet and Tech. Spec log

Task Standard:
1. _RIA-57 declared inoperable and work request initiated to have I&E correct the problem.
2. R.P. notified that_RIA-57 is inoperable and_RIA-57 is listed on the Control Room Shift Turnover sheet in the T.S. section.
3. T.S. 3.5.6 is referred to for LCO guidelines.
4. Documentation on T.S. log

Preferred Evaluation Location:
Simulator _____ In-Plant __X___

Preferred Evaluation Method:
Perform _____ Simulate __X___

References:
PT/0/A/0230/001
OP/_/A/1102/20, ENCLOSURE 5.6, Shift Turnover Sheet, completed.
Tech Spec Log

Validation Time: __________________ **Time Critical:** NO

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Candidate: ________________________________ Time Start:: _______
NAME Time Finish: _______

Performance Rating: SAT _____ UNSAT _____ Question Grade ____ Performance Time _____

Examiner: ________________________ ________________________________ / ________
NAME SIGNATURE DATE

===

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

N/A

SIMULATOR OPERATOR INSTRUCTIONS:

N/A

Tools/Equipment/Procedures Needed:
PT/_/A/600/01, PT/0/A/0230/001, OP/_/A/1102/20, ENCLOSURE 5.6, Shift Turnover Sheet, and Technical Specifications

READ TO OPERATOR

DIRECTION TO TRAINEE:

When I tell you to begin, you are to check the operability of_RIA-57 at the individual monitor. Before you start, I will describe the general plant conditions, state the initiating cues, and answer any questions. Perform procedure steps and make notifications as if you were actually performing the task.

INITIAL CONDITIONS:

Unit_ is at 100% power

_RIA-57 has just been returned to service following detector replacement.

INITIATING CUES:

The Control Room Supervisor directs you to check the operation of_RIA-57 from the individual monitor per PT/0/A/230/01 Enclosure 13.14, Operation of RIA-57, Step 2.

START TIME:________

STEP 1: VERIFY alarm setpoints. STANDARD: • Locate_RIA-57 Individual Monitor on_VB2/3 From the key pad: • Depress "Clear" **CUE: Clear depressed** • "Enter" the number 009 **CUE: 009 depressed** • Depress "Item." **CUE: This will display the High alarm setpoint of 1.20 E+3 RAD/hr.** • **Depress "+" CUE: + depressed and 010 indicated in the window** • Depress "item" **CUE: this will display the Alert alarm setpoint of 286 E+2 RAD/hr** COMMENTS:	**Critical** ____ SAT ____ UNSAT
STEP 2: Return to normal. STANDARD: • Select "Clear" **CUE: Clear depressed** • Depress R/hr button **CUE: R/hr depressed** COMMENTS:	____ SAT ____ UNSAT
STEP 3*: Perform Check source. • Depress "C/S" button. **CUE: C/S depressed** • Check readings between (Units 1 and 3, 5 E-1 and 1.0 E+0; Unit 2, 1 E+0 and 1.5) E+0 ***CUE: Reading indicates 1.5 E+1*** • Check "Area Monitor Fault" alarm NOT in ***CUE: Area Monitor Fault" alarm has alarmed (_SA-8,A-10, AREA MONITOR FAULT. Alarm located above RIAs)*** • Refers to Operability of_RIA-57 Enclosure 13.5 STANDARD: • Depress "C/S" button. • Understand that the RIA reading at 1.5E+1 is not normal • Understand the "Area Monitor Fault" alarm is not a normal response and refer to the Alarm Response Guide (ARG) for _SA-8, A10…Issue WR for I&E • Refers to Operability of_RIA-57 Enclosure 13.5 and determines that_RIA-57 is NOT operable. COMMENTS:	**Critical** ____ SAT ____ UNSAT

STEP 4*: INITIATE Enclosure 13.5 corrective actions. 1. Initiate work request to have I&E correct the problem. ***CUE: Work request has been inititated.*** 2. Notify R.P_RIA-57 is inoperable. ***CUE: R.P. has been notified.*** 3. List_RIA-57 on the Control Room Shift Turnover sheet in the T.S. section. → 4. Refer to T.S. 3.5.6 for LCO guidelines and list on the Normal TS log. → STANDARD: 1. Initiate work request to have I&E correct the problem. 2. Notify R.P that_RIA-57 is inoperable. 3. List_RIA-57 on the Control Room Shift Turnover sheet in the T.S. section. 4. Refer to T.S. 3.5.6 for LCO guidelines. Table 3.5.6-1 Accident Monitoring Inst. Item # 3. 2 of 2 channels required => 1 OOS, Action #2(>HSD) restore w/in 30 days or HSD in next 12 hours. Document RIA OOS on the normal TS log. COMMENTS:	____ SAT ____ UNSAT **CRITICAL** **CRITICAL**

TIME STOPPED: ____________________

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

RO and SRO

INITIAL CONDITIONS:

_RIA-57 has just been returned to service following detector replacement.

INITIATING CUES:

The Control Room Supervisor directs you to check the operation of_RIA-57 from the individual monitor per PT/0/A/230/01 Enclosure 13.14, Operation of _RIA-57, Step 2.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NLO-017/PLANT

ALIGN COOLING WATER TO HIGH PRESSURE INJECTION PUMP MOTOR COOLERS FROM AUX. SERVICE WATER PUMP

CANDIDATE ____

EXAMINER ____

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

ALIGN COOLING WATER TO HIGH PRESSURE INJECTION PUMP MOTOR COOLERS FROM AUX. SERVICE WATER PUMP.

Alternate Path:

N/A

Facility JPM #:

NLO-017

K/A Rating(s):

076 A2.01 3.5/3.7

Task Standard:

Preferred Evaluation Location:

Simulator _____ In-Plant __X____

Preferred Evaluation Method:

Perform _____ Simulate ___X___

References:

AP/1/A/1700/07

Validation Time: 16 min. **Time Critical:** NO

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Candidate: ______________________________ Time Start: ________
NAME Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________ ____________________________/________
NAME SIGNATURE DATE

===

COMMENTS

___

___

___

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

Ensure enough copies of AP/1/A/1700/07 are available in the Simulator file cabinet, since Operators will obtain their own copy of the procedure.

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

A Station Blackout has occurred. The SSF Diesel Generator will not start, rendering the SSF RCMU Pumps inoperable. Standby Bus #1 has subsequently been energized from CT-5. I&E personnel have aligned the 'A' HPIP to the ASWP Switchgear. AP/1,2,3/A/1700/11, (Loss of Power), Enclosure 6.3 of has been completed up to step 2.0.

INITIATING CUES:

AP/1,2,3/A/1700/11, Loss Of Power, directs the operator to align cooling water to the High Pressure Injection Pumps.

The Control Room Supervisor directs you to align cooling water to the High Pressure Injection Pumps on Unit___ (specify unit) per Enclosure 6.3 of AP/1,2,3/A/1700/11, Loss of Power.

START TIME: _______

STEP 1: Ensure closed " AUX. SER. WTR. SWGR 4160 VOLT FDR B1T - UNIT 10" breaker. STANDARD: "AUX. SER. WTR. SWGR. 4160-Volt FDR B1T-UNIT 10" breaker indicates closed on the ASW SWGR 600V LOAD CENTER. Two red CLOSED lights are on. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Ensure closed the "AUX. SER. WTR. SWGR TRANSFORMER" breaker STANDARD: Student verifies the "AUX. SER. WTR SWGR TRANSFORMER" breaker is Closed. **Location: ASW SWGR 600V LC Unit 5** ***CUE: Inform the student that the red CLOSED light is lit and that the green OPEN light is off at the control switch for Aux. Ser. Wtr. Swgr. Xfrmr. Bkr.*** COMMENTS:	___ SAT ___ UNSAT
STEP 3: Rack in "AUXILIARY SERVICE WATER PUMP" breaker at the ASW SWGR 600V LOAD CENTER Unit 6. **STANDARD**: Student opens shutter, inserts 600v breaker rackout tool, and turns tool clockwise to rack breaker in. ***CUE: After breaker is racked in, inform student that the AUX SERVICE WATER PUMP MTOR breaker green OPEN indicating light is ON.*** COMMENTS: Student is expected to follow (simulate) all applicable safe electrical work practices.	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 4: CLOSE CCW-309 (ASWP Disch Drain). STANDARD: CCW-309 (ASWP Disch Drain) is manually CLOSED. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5 OPEN CCW-99 (ASWP Suction). STANDARD: CCW-99 (ASWP Suction) is manually opened. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6 OPEN CCW-101 (ASWP Disch). STANDARD: CCW-101 (ASWP Disch) is manually opened. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7 OPEN CCW-247 (ASWP Recirc). STANDARD: CCW-247 (ASWP Recirc) is manually OPENED. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8 Vent the Aux. Service Water Pump using CCW-308 (ASWP Vent). STANDARD: CCW-308 ASWP vent is throttled open until water issues and then is then closed. COMMENTS:	___ SAT ___ UNSAT
STEP 9 START the Aux. Service Water Pump Motor. STANDARD: Student locates AUX SERVICE WATER PUMP MOTOR control switch and rotates switch to the CLOSE position. **CUE: After switch is rotated, inform student that the AUX SERVICE WATER PUMP MOTOR breaker red CLOSED indicating light is ON.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 10: VERIFY adequate HPIP motor cooler flow indication locally (>1 gpm) by local flow indication. Location: Aux. Bldg. 1st – HPI Pump Room STANDARD: Student verifies flow located AB-1st HPI pump Room. ***Cue: A picture of the "A" HPI Pump Motor Cooler Flow indication may be given the student to be used in explaining how the flow would be verified.*** COMMENTS: **For ALARA and time considerations, do not allow the student to enter the HPI Pump Room. Stop him/her at the plan view of the HPI Room and have him/her indicate where the flow would be verified.** **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANTIONS:

STEP #	Explanation
3	Supplies power to the Auxiliary Service Water Pump.
4	Ensures that water is not introduced to the Aux. Bldg. when the ASWP is started.
5	Ensures that a suction supply of water is available to the ASWP.
6	Ensures that water is supplied to the discharge header.
7	Prevents pump damage due to the possibility that low flow conditions may exist.
9	Supplies the HPI Pump Motor Coolers with water.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

A Station Blackout has occurred. The SSF Diesel Generator will not start, rendering the SSF RCMU Pumps inoperable. Standby Bus #1 has subsequently been energized from CT-5. I&E personnel have aligned the 'A' HPIP to the ASWP Switchgear. AP/1,2,3/A/1700/11, (Loss of Power), Enclosure 6.3 of has been completed up to step 2.0.

INITIATING CUES:

AP/1,2,3/A/1700/11, Loss Of Power, directs the operator to align cooling water to the High Pressure Injection Pumps.

The Control Room Supervisor directs you to align cooling water to the High Pressure Injection Pumps on Unit___ (specify unit) per Enclosure 6.3 of AP/1,2,3/A/1700/11, Loss of Power.

Aux Service Water To HPI Pump Motor Coolers

1. Energize Standby Bus #1:

_____ 1.1 **IF** either Keowee Unit is available,

THEN Emergency Start available Keowee Units:

_____ "Keowee Emer Start Channel A"

_____ "Keowee Emer Start Channel B".

_____ 1.1.1 **WHEN** available Keowee Units are running,

THEN perform the following:

_____ 1.1.1.1 Ensure closed ACB3 **OR** ACB 4.

_____ 1.1.1.2 Place "CT4 BUS 1 AUTO/MANUAL" transfer switch in "MANUAL".

_____ 1.1.1.3 Place "STBY BUS 1 SYNCHRONIZING" switch to "ON".

_____ 1.1.1.4 Close "SK1 CT4 STBY BUS 1 FEEDER".

_____ 1.1.1.5 Verify "STANDBY BUS 1" voltmeter indicates ≈ 4160v.

_____ 1.1.1.6 Place "STBY BUS 1 SYNCHRONIZING" switch to "OFF".

Aux Service Water To HPI Pump Motor Coolers

_____ 1.2 **IF** **NO** Keowee Unit is available,

AND CT-5 voltmeter indicates ≈ 4160v,

THEN perform the following:

1.2.1 Place the following "AUTO/MAN" transfer switches in "MANUAL":

_____ "CT4 BUS 1 AUTO/MAN"

_____ "CT4 BUS 2 AUTO/MAN"

_____ "CT5 BUS 1 AUTO/MAN"

_____ "CT5 BUS 2 AUTO/MAN".

1.2.2 Ensure open the following breakers:

_____ "SK1 CT 4 STBY BUS 1 FEEDER"

_____ "$S1_1$ STBY BUS 1 TO MFB 1"

_____ "$S2_1$ STBY BUS 2 TO MFB 2".

CAUTION 1.2.3:	If statalarm "TRANSFORMER CT-5 UNDERVOLTAGE" (SA-16/ C-4) is received, additional loading of CT-5 (if powered from Central Switchyard) may result in an undervoltage trip of breakers SL1 and SL2 if Standby Bus voltage reaches 3890 volts.

_____ 1.2.3 Close "SL1 CT5 STBY BUS 1 FDR".

_____ 1.2.4 Place "CT5 BUS 1 AUTO/MAN" switch in "AUTO".

Aux Service Water To
HPI Pump Motor Coolers

_____ 2. **WHEN** Standby Bus 1 is energized,

THEN perform the following:

_____ 2.1 Ensure closed breaker "AUX. SER. WTR. SWGR. 4160 VOLT FDR. B1T - UNIT 10". (Control switch at ASW SWGR 600V LOAD CENTER Unit 5)

NOTE:	Breaker indication will **NOT** be on unless the Standby Bus is energized.

_____ 2.2 Ensure closed breaker "AUX. SERV. WTR. SWGR. TRANSFORMER". (located at ASW SWGR 600V LOAD CENTER Unit 5)

_____ 2.3 Rack in breaker "AUX. SERVICE WATER PUMP" at the ASW SWGR 600V LOAD CENTER Unit 6.

_____ 2.4 Close CCW-309 (ASWP Disch Drn).

_____ 2.5 Open CCW-99 (ASWP Suct).

_____ 2.6 Open CCW-101 (ASWP Disch).

_____ 2.7 Open CCW-247 (ASWP Recirc).

_____ 2.8 Vent the Aux Service Water Pump using CCW-308 (ASWP VENT).

_____ 3. Start the Aux. Service Water Pump Motor.

_____ 4. Locally ensure adequate HPIP motor cooler flow (> 1 gpm) by local flow indication.

END

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NLO-004

Manually Bypassing the KI/KU Inverter

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:
MANUALLY BYPASS THE KI/KU INVERTER
TASK NUMBER: OO1345501

Alternate Path:

N/A

Facility JPM #:

NLO-004 (MODIFIED)

K/A Rating(s):
System: APE-057 Loss of Vital AC Instrument Bus
K/A: EA1.01
Rating: 3.7/3.7

Task Standard:
KI/KU Inverter is located and bypassed correctly

Preferred Evaluation Location:

Simulator _____ In-Plant __X___

Preferred Evaluation Method:

Perform _____ Simulate __X__

References:

Validation Time: 10 min. **Time Critical:** NO

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Candidate: ________________________________ Time Start : ________
NAME Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ____________________________ ____________________________/________
NAME SIGNATURE DATE

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COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

AP/1/A/1700/23, AP/2/A/1700/23, AP/3/A/1700/23 (Enclosure 6.1 Bypass of the KI and KU Inverters)

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit _____ (specify unit) was operating at 100% power when it experienced a loss of ICS Power followed by a unit trip. The subsequent actions of the EOP and the Abnormal Procedure for Loss of ICS Power have been completed up to the point of regaining ICS Power.

INITIATING CUES:

The Control Room Supervisor directs you to manually bypass the KI and KU Inverters on Unit _____ (specify unit) per AP/1,2,3/A/1700/23, Loss of ICS Power, Enclosure 6.1.

START TIME: ______

STEP 1: OPEN KI Inverter Bypass Switch cabinet door. STANDARD: Student locates the KI Inverter Bypass Switch cabinet and opens cabinet door. (Located in Equipment Room) COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 2: MANUALLY BYPASS KI Inverter STANDARD: Inverter is MANUALLY BYPASSED by positioning the following switches in sequence: Sw #1 (left switch) And Sw #3 (right switch) are OPENED Then Sw #2 (center switch) is CLOSED COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 3: OPEN KU Inverter Bypass Switch cabinet door. STANDARD: Student locates the KU Inverter Bypass Switch cabinet and opens cabinet door. (Located in Equipment Room) COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: MANUALLY BYPASS KU Inverter STANDARD: Inverter is MANUALLY BYPASSED by positioning the following switches in sequence: Sw #1 (left switch) And Sw #3 (right switch) are OPENED Then Sw #2 (center switch) is CLOSED STANDARD: COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: Call the Control Room to determine if ICS AUTO and Hand Power have been restored. ***CUE: ICS AUTO has been restored but, Hand Power has NOT been restored*** STANDARD: Locate phone and simulate call to the unit's control room COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6: If ICS HAND power has not been restored. STANDARD: In the unit's Cable Room locate __ KRA power panel, ***CUE: Breaker #13 (175A 2P, Power Panelboard 1KU)) is "loose" and in the "mid" position.*** Reset breaker KRA #13 (175A 2P, Power Panelboard 1KU) ***CUE: Breaker #13 (175A 2P, Power Panelboard 1KU)) is RESET.*** In the unit's Cable Room locate __ KU power panel, ***CUE: Breaker #21 (30A 1P, ICS/NNI Hand Power) is "loose" and in the "mid" position.*** Reset breaker #21 (30A 1P, ICS/NNI Hand Power) ***CUE: Breaker #21 (30A 1P, ICS/NNI Hand Power) is RESET.*** Notify the Control Room ***CUE: BOTH ICS AUTO and HAND power has been restored.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	KI cabinet must be located and the door opened to reach the bypass switches
2	Switch 1, 2, and 3 properly operated to bypass the inverter
3	KU cabinet must be located and the door opened to reach the bypass switches
4	Switch 1, 2, and 3 properly operated to bypass the inverter
5	Communicate with the control to determine that the HAND power has not been restored.
6	Properly reset KU power supplies breakers to restore ICS HAND power

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit _____ (specify unit) was operating at 100% power when it experienced a loss of ICS Power followed by a unit trip. The subsequent actions of the EOP and the Abnormal Procedure for Loss of ICS Power have been completed up to the point of regaining ICS Power.

INITIATING CUES:

The Control Room Supervisor directs you to manually bypass the KI and KU Inverters on Unit _____ (specify unit) per AP/1,2,3/A/1700/23, Loss of ICS Power, Enclosure 6.1.

Bypass Of The 1KI And 1KU Inverters

1. Perform the following in the Unit 1 Equipment Room:

1.1 Bypass 1KI Inverter by performing the following:

_____ 1.1.1 Open "SW#1" (Left Switch).

_____ 1.1.2 Open "SW#3" (Right Switch).

_____ 1.1.3 Close "SW#2" (Center Switch).

1.2 Bypass 1KU Inverter by performing the following:

_____ 1.2.1 Open "SW#1" (Left Switch).

_____ 1.2.2 Open "SW#3" (Right Switch).

_____ 1.2.3 Close "SW#2" (Center Switch).

_____ 1.3 Call Unit 1 Control Room (ext. 2261, 2159, or 2335) to determine if ICS AUTO and HAND Power have been restored:

- The following statalarms off:
  - "ICS AUTO POWER FAILURE" (1SA-02/B-11)
  - "ICS HAND POWER FAILURE" (1SA-02/B-12).

Bypass Of The 1KI And 1KU Inverters

_____ 2. **IF** ICS AUTO Power has **NOT** been restored,

THEN perform the following in the Unit 1 Cable Room:

2.1 Reset and close the following:

_____ "1KRA breaker #1" (100A 1P, POWER PANELBOARD 1KI)

_____ "1KI breaker #1" (30A 1P, AUTO POWER (ICS)).

_____ 3. **IF** ICS HAND Power has **NOT** been restored,

THEN perform the following in the Unit 1 Cable Room:

3.1 Reset and close the following:

_____ "1KRA breaker #13" (175A 2P, POWER PANELBOARD 1KU)

_____ "1KU breaker #21" (30A 1P, ICS/NNI HAND POWER).

_____ 4. Immediately notify the Unit 1 Control Room (ext. 2261, 2159, or 2335) that all applicable steps of this Enclosure have been completed.

••• **END** •••

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NLO-041

RESTART THE PRIMARY IA COMPRESSOR FOLLOWING A COMPRESSOR TRIP
(Alternate Path)

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

RESTART THE PRIMARY IA COMPRESSOR FOLLOWING A COMPRESSOR TRIP
TASK NUMBER: OO1333002

Alternate Path:

YES

Facility JPM #:

NLO-041

K/A Rating(s):
System: SF8-078 Instrument Air System
K/A: 2.1.30
Rating: 3.9/3.4

Task Standard:

The Primary IA Compressor is restarted by procedure

Preferred Evaluation Location:

Simulator _____ In-Plant __X___

Preferred Evaluation Method:

Perform _____ Simulate __X__

References:

Enclosure 4.11 of OP/0/A/1106/27

Validation Time: 10 minutes

Time Critical: NO

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Candidate: ______________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

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COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:
Enclosure 4.11 of OP/0/A/1106/27

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit 1 is at 100% power.
The Transmission Dept. was performing PM checks on B3T switchgear when the incoming feeder breaker tripped open.
When B3T de-energized, the automatic transfer to the backup source (B4T) did not occur and the Primary IA Compressor tripped.
The RO entered "Loss of IA" AP/1/A/1700/22 as IA header pressure decreased to ≈ 85 psig and has reached step 5.10, which refers the operator to OP/0/A/1106/27 to restore operable IA compressors.

INITIATING CUES:

The SRO in the control room instructs you utilize Enclosure 4.11 restart of the Primary IA Compressor following a Trip of OP/0/A/1106/27 (Compressed Air System) to **RESTART** the Primary IA Compressor. Initial Conditions of the enclosure have been completed.

START TIME: _______

STEP 1: Position the following valves: Close IA-2730 (Primary IA Desiccant Air Filter "A" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2730 (Primary IA "A" Desiccant Filter Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT
STEP 2: Position the following valves: Close IA-2731 (Primary IA Desiccant Air Filter "B" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2731 (Primary IA "B" Desiccant Filter Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT

STEP 3: At the Primary IA Dryer A Control Panel, position the (ON/OFF) switch to OFF. STANDARD: The student REMOVES the Primary IA Dryers from service by rotating the following switches, located on the A Dryer control panels, to the "OFF" position: Primary IA Dryer "A" On/Off selector. COMMENTS:	___ SAT ___ UNSAT
STEP 4: At the Primary IA Dryer B Control Panel, position the (ON/OFF) switch to OFF. STANDARD: The student REMOVES the Primary IA Dryer from service by rotating the following switches, located on the B Dryer control panels, to the "OFF" position: Primary IA Dryer "B" On/Off selector. COMMENTS:	___ SAT ___ UNSAT

STEP 5: Position the following valves: Close IA-2735 (Primary Air Filter "A" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2735 (Primary Air Filter "A" Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT
STEP 6: Position the following valves: Close IA-2736 (Primary Air Filter "B" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2736 (Primary Air Filter "B" Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT

STEP 7: Open HPSW-771 (Primary IA Comp. Disc. Block) (TB5 M-39) STANDARD: The student LOCATES and OPENS HPSW-771 (Primary IA Compressor Cooling Discharge Block) by rotating the switch to the "Open" position. **NOTE: HPSW-771 control switch and the cooling water inlet pressure gauges are located north of the compressor next to the west Turbine floor wall.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 8: Verify adequate cooling water flow as follows: IF OHPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure) does NOT read between 61 and 67 psig. Backwash HPSW-764 (Primary IA Comp. Disch. Control) (TB5 M-39) per Backwash of Primary IA Compressor HPSW Pressure Regulator enclosure. Verify HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) in Locked Open Position. STANDARD: The student VERIFIES adequate cooling water flow by monitoring the following gauges: - OHPS-PG-0823 (Primary IA Comp. Cooling Water Inlet Pressure). **CUE: Using a pointing device, indicate to the student the following readings:** **- OHPS-PG-0823 = 64 psig.** HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) is verified in the Locked Open Position. COMMENTS:	___ SAT ___ UNSAT

STEP 9: Depress (RESET/LAMP TEST) pushbutton. • Verify all alarm indicators light. • Release (RESET/LAMP TEST) pushbutton and verify all alarm indicator lamps extinguish. 2.7.1 Any alarm condition indicated on the Primary IA Compressor control panel must be resolved before starting the compressor. STANDARD: The student RESETS the Primary IA Compressor by depressing the black "Reset" pushbutton on the compressor control panel located on the north side of the compressor housing. **CUE: While RESET/LAMP TEST pushbutton is depressed, inform student that all alarm indicators are lit. When RESET/LAMP TEST pushbutton is released, inform student that all alarm indicator lamps extinguish.** COMMENTS:	___ SAT ___ UNSAT

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STEP 10:

- Depress (START) pushbutton.
- Verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is with in the range specified (in procedure).

STANDARD:

- The student STARTS the Primary Air Compressor by depressing the "Start" pushbutton on the control panel located on the north side housing of the compressor.

CUE: Inform the student that the green "Machine Run" light has illuminated.

- The student VERIFIES adequate cooling water flow by monitoring the following gauges:

- OHPS-PG-0823 (Primary IA Comp. Cooling Water Inlet Pressure).
- OHPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure).

CUE: Using a pointing device, indicate to the student the following readings:

- **OHPS-PG-0823 = 64 psig.**
- **OHPS-PG-0824 = 9 psig.**

Student should simulate throttling HPSW-767 (Pri. IA Comp. Disch. Cont.) to achieve the proper flow/outlet pressure range.

When HPSW-767 is throttled closed, indicate with the pointing device that flow is 98 gpm and outlet pressure is 18 psig

COMMENTS:

</td>
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CRITICAL STEP

___ SAT

___ UNSAT

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<u>STEP 11</u>: VERIFY selected Enclosure fan is running and all door panels are installed on compressor enclosure. Verify all door panels are installed on the Primary IA Compressor Enclosure. <u>STANDARD</u>: The student determines that selected enclosure fan is operating and all door panels located on the compressor enclosure are installed. **CUE: Inform the student that the selected Enclosure Fan is running properly. Inform student that all door panels are installed on enclosure.** <u>COMMENTS</u>:	___ SAT ___ UNSAT
<u>STEP 12</u>: Throttle open IA-2735 (Primary Air Filter "A" Outlet) or IA-2736 (Primary Air Filter "B" Outlet) (TB5 L-39) to SLOWLY pressurize the Dryer tanks to system pressure (100-110 psig). <u>STANDARD</u>: The student throttles open one of the following valves to SLOWLY PRESSURIZE the Desiccant Dryers: IA-2735 (Primary Air Filter "A" Outlet) **<u>OR</u>** IA-2736 (Primary Air Filter "B" Outlet) **CUE: Once the student has demonstrated his/her ability to properly throttle the valve, indicate to the student with a pointing device that the Desiccant Dryers have reached 104 psig.** <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 13: At the Primary IA Dryer A panel, position the (ON/OFF) switch to ON. STANDARD: The student PLACES the Primary IA Dryers in service by positioning the following switches to the "ON" position: Primary IA Dryer "A" On/Off Selector COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 14: At the Primary IA Dryer B panel, position the (ON/OFF) switch to ON. STANDARD: The student PLACES the Primary IA Dryers in service by positioning the following switches to the "ON" position: Primary IA Dryer "B" On/Off Selector COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 15: CONNECT the Primary IA Compressor to the IA Header. Open IA-2735 (Primary Air Filter A Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2735 (Primary Air Filter "A" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

<u>STEP 16</u>: CONNECT the Primary IA Compressor to the IA Header. Open IA-2736 (Primary Air Filter B Outlet). (TB5 L-39) <u>STANDARD</u>: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2736 (Primary Air Filter "B" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT
<u>STEP 17</u>: CONNECT the Primary IA Compressor to the IA Header. Slowly open IA 2730 (Primary Desiccant Air Filter A Outlet). (TB5 L-39) <u>STANDARD</u>: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2730 (Primary Desiccant Air Filter "A" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 18: CONNECT the Primary IA Compressor to the IA Header. Slowly open IA 2731 (Primary Desiccant Air Filter B Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2731 (Primary Desiccant Air Filter "B" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 19: As system pressure increases check for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters. **NOTE:** Enclosure "Startup Of The Primary IA Compressor" contains a detailed list of expected Primary Air Compressor normal operating parameters. STANDARD: The student checks for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters as system pressure increases. **CUE: No air leaks are found.** Primary Air Compressor monitored for normal operation. COMMENTS:	___ SAT ___ UNSAT

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
7	Open HPSW-771 (Primary IA Comp. Disc. Block) aligns cooling water to the compressor
10	Depress (START) pushbutton starts the compressor, verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is with in the range specified , and establish proper cooling water flow to the compressor
12	Pressurizes and places in service the primary air filter
13	Places the "A" Air Dryer in service
14	Places the "B" Air Dryer in service
15-18	Establishes an air flow path from the compressor to the IA Header

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit 1 is at 100% power.
The Transmission Dept. was performing PM checks on B3T switchgear when the incoming feeder breaker tripped open.
When B3T de-energized, the automatic transfer to the backup source (B4T) did not occur and the Primary IA Compressor tripped.
The RO entered "Loss of IA" AP/1/A/1700/22 as IA header pressure decreased to ≈ 85 psig and has reached step 5.10, which refers the operator to OP/0/A/1106/27 to restore operable IA compressors.

INITIATING CUES:

The SRO in the control room instructs you utilize Enclosure 4.11 restart of the Primary IA Compressor following a Trip of OP/0/A/1106/27 (Compressed Air System) to **RESTART** the Primary IA Compressor.
Initial Conditions of the enclosure have been completed.

Loss of Instrument Air

______ 5.10 Restore Instrument Air header pressure:

- Isolate IA header leakage.
- Start operable IA compressors.
  - **REFER TO** OP/0/A/1106/027, (Compressed Air System).

______ 5.11 **IF** efforts to restore IA are **NOT** successful,

AND both Main Feedwater Pumps have tripped,

THEN establish Condensate Recirc:

______ 5.11.1 Trip all CBPs.

______ 5.11.2 Trip all but one HWP.

______ 5.11.3 Open 1C-124 (Condensate Recirc to UST).

______ 5.11.4 Send an Operator to throttle 1C-129 (Condensate Recirc Control Bypass). (TB5/M-22NE)

______ 5.11.5 Establish ≈ 2300 gpm Condensate Recirc flow by computer point O1A0156 (CBP DISCH HDR FLOW).

Restart Of The Primary IA Compressor Following A Trip

1. Initial Conditions

_____ 1.1 Worthington IA compressors may or may not be supplying IA header.

_____ 1.2 Reason for loss of Primary IA compressor has been corrected.

_____ 1.3 System alignment unchanged from time of compressor trip.

2. Procedure

2.1 Position the following valves:

_____ 2.1.1 Close IA-2730 (Primary Desiccant Air Filter 'A' Outlet). (TB5 L-39)

_____ 2.1.2 Close IA-2731 (Primary Desiccant Air Filter 'B' Outlet). (TB5 L-39)

_____ 2.2 At the Primary IA Dryer A Control Panel, position the (ON/OFF) switch to "OFF."

_____ 2.3 At the Primary IA Dryer B Control Panel, position the (ON/OFF) switch to "OFF."

2.4 Position the following valves:

_____ 2.4.1 Close IA-2735 (Primary Air Filter 'A' Outlet). (TB5 L-39)

_____ 2.4.2 Close IA-2736 (Primary Air Filter 'B' Outlet). (TB5 L-39)

_____ 2.5 Open HPSW-771 (Primary IA Comp. Disch. Block) (TB5 M-39).

2.6 Verify adequate cooling water flow as follows:

_____ 2.6.1 IF OHPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure) does NOT read between 61 and 67 psig. backwash HPSW-764 (Primary IA Comp. Disch. Control) (TB5 M-39) per Backwash of Primary IA Compressor HPSW Pressure Regulator enclosure.

_____ 2.6.2 Verify HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) in Locked Open position.

_____ 2.7 Depress (RESET/LAMP TEST) pushbutton.

- Verify all alarm indicators light.
- Release (RESET/LAMP TEST) pushbutton and verify all alarm indicator lamps extinguish.

_____ 2.7.1 Any alarm conditions indicated on the Primary IA Compressor control panel must be resolved before starting the compressor.

_____ 2.8 Depress (START) pushbutton.

_____ 2.8.1 Verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is within the range specified below for the value of 0HPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure).

Cooling Water Inlet Pressure (psig)	Acceptable Range for Cooling Water Outlet Pressure (psig).
61	18 - 10
62	19 - 11
63	20 - 12
64	21 - 12
65	22 - 13
66	23 - 14
67	24 - 14

_____ • **IF** OHPS-PG-0824 is **NOT** within the Acceptable Range For Cooling Water Outlet Pressure for the value of OHPS-PG-0823, throttle HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) to obtain the required cooling water outlet pressure.

_____ • **IF** OHPS-PG-0824 is **NOT** within the Acceptable Range For Cooling Water Outlet Pressure for the value of OHPS-PG-0823, **THEN** closely monitor Primary IA Compressor Discharge Temperature and Injection Temperature until acceptable Cooling Water pressures can be obtained.

_____ 2.9 **IF** Compressor fails to start, notify WCC SRO for aid in resolving problem.

_____ 2.10 Verify selected Enclosure fan is running.

_____ 2.11 Verify all door panels are installed on the Primary IA Compressor Enclosure.

CAUTION: The Outlet Filter Valves must be opened in increments very slowly to prevent disintegration of the Dryer desiccant.

_____ 2.12 Throttle open IA-2735 (Primary Air Filter 'A' Outlet) (TB5 L-39) **OR** IA-2736 (Primary Air Filter 'B' Outlet) (TB5 L-39) to SLOWLY pressurize the Dryer Tanks to system pressure (100-110 psig).

_____ 2.13 At the Primary IA Dryer A panel, position the (ON/OFF) switch to "ON."

_____ 2.14 At the Primary IA Dryer B Panel, position the (ON/OFF) switch to "ON."

_____ 2.15 Open IA-2735 (Primary Air Filter A Outlet). (TB5 L-39)

_____ 2.16 Open IA-2736 (Primary Air Filter B Outlet). (TB5 L-39)

CAUTION: Rapid opening of the Primary Desiccant Air Filter Outlet Valves can cause collapse of the Desiccant Filters.

_____ 2.17 SLOWLY open IA-2730 (Primary Desiccant Air Filter A Outlet). (TB5 L-39)

_____ 2.18 SLOWLY open IA-2731 (Primary Desiccant Air Filter B Outlet). (TB5 L-39)

_____ 2.19 As system pressure increases check for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters.

NOTE: Enclosure "Startup Of The Primary IA Compressor" contains a detailed list of expected Primary Air Compressor normal operating parameters.

_____ 2.20 Monitor Primary Air Compressor for normal operation.

NOTE: If Primary Air Compressor is operating normally, Backup IA Compressors will be running unloaded.

2.21 Return the Backup IA Compressors to their normal lineup:

_____ 2.21.1 Place the C Backup IA Compressor control switch in "STBY 1."

_____ 2.21.2 Place either the A or B Backup IA Compressor control switch in "STBY 1."

_____ 2.21.3 Place the remaining Backup IA Compressor control switch in "STBY 2."

2.22 Close or verify closed the following valves:

_____ • RCW-25 (Backup IA Compressor A Temp. Control Bypass) (TB1 L-31)

_____ • RCW-31 (Backup IA Compressor B Temp. Control Bypass) (TB1 L-32)

_____ • RCW-37 (Backup IA Compressor C Temp. Control Bypass) (TB1 L-32)

INITIAL SUBMITTAL

OCONEE EXAM 2000-301
50-269, 270, AND 287/2000-301

JULY 10 - 14, 18,19, AND 20, 2000

INITIAL SUBMITTAL

JPMS -- SRO

JPM's - SRO

Oconee
2000

NRC Copy

Every JPM should:

1. BA be supported by facility licensee's job task analysis.
2. BA be operationally important (meets NRC K/A Catalog threshold criterion of 2.5 (3 for requalification exams) or as determined by the facility and agreed to by the NRC).
3. BA be designed as either SRO only, RO/SRO or AO/RO/SRO.
4. include the following, as applicable:
   a. BA initial conditions
   b. BA initiating cues
   c. BA references and tools, including associated procedures
   d. BA validated time limits (average time allowed for completion) and specific designation of those JPMs that are deemed to be time-critical by the facility operations department
   e. BA specific performance criteria that include:
      (1) BA expected actions with exact control and indication nomenclature and criteria (switch position, meter reading), even if these criteria are not specified in the procedural step
      (2) BA system response and other cues that are complete and correct so that the examiner can properly cue the examinee, if asked
      (3) BA statements describing important observations that should be made by the examinee
      (4) BA criteria for successful completion of the task
      (5) BA identification of those steps that are considered critical
      (6) BA restrictions on the sequence of steps

H- Bobby Ayers

Facility: Oconee
Exam Level: **SRO-I**

Date of Examination: 7-10/17-00
Operating Test No.: 1

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
a. CRO-12A, Recover a Dropped Control Rod; (20 min.) AP/1/A/1700/15, OP/0/A/1105/09, Encl 4.10 [KA: 005AA2.03 (3.4/4.4)] Note: This JPM conducted with Admin A.1 CRO-40A, (Calculate SDM) (SRO ONLY) (5 min.)	D, S, A	1
b. CRO-096, Align ECCS Suction from Emergency Sump (LP-20, Emergency Sump Suction failed closed) (9 min.) EP/1/A/1800/01, CP-601; AP/1/A/1700/07 [KA: 002A2.04 (4.3/4.6)] Note: SRO required ECCS (SRO ONLY) (PRA)	N, S, L	2
c. NRC-002, PORV Stroke Test (10 min.) PT/0/A/0201/004 [KA: 010A4.03 (4.0/3.8)] Note: Performed following completion of NRC-001, (Establish Steam Bubble)	N, S, L, A	3
d. CRO-013, Swap TD EFDWP Suction to Hotwell; (10 min.) AP/1/A/1700/19 [KA: APE054AA1.01 (4.5/4.4)]	D, C/S, L	4S
e. CRO-095, Swap RBCU's (Inadvertent Ch. 5 ES actuation); (10 min.) OP/0/A/1104/15 [KA: 022A4.01 (3.6/3.6)]	M, C/S	5
f. CRO-009, Following Keowee Emergency Start, transfer from CT-4 to CT-5; (10 min.) OP/0/A/1106/19, Encl. 3.13 [KA: 062A4.01 (3.3-3.1)]	D, C/S, L	6
g. JPM NRC-1998 – RIA-57 Operability Check; (RIA-57 fails to meet acceptance criteria) (10 min.) PT/0/1/0230/001, Encl. 13.9 and Encl. 13.3 &4 [KA: 072A2.02 (2.8/2.9)]	D, C/S, A	7
B.2 Facility Walk-Through		
a. NLO-017, Align Cooling Water to HPIP's from Station ASWP; (16 min.) AP/1,2,3/A/1700/11, Encl 6.3; OP/0/A/1102/06, Encl. 3.3 [KA: 076A2.01 (3.5/3.7)]	D, R, L	4S
b. NLO-004, Manually Bypass the KI Inverter; (5 min.) AP/1,2,3/A/1700/23 Encl 6.1 [KA: 063K4.01 (2.7/3.0)]	D, L	6
c. NLO-041, Restart the Primary Instrument Air Compressor following a Compressor trip; (10 min.) OP/0/1106/27, Encl. 4.11; APO/1/A/1700/22 [KA: 078G2.1.30 (3.9/3.4)]	D, A	8

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM CRO-012A/SIM

RECOVERY OF A DROPPED CONTROL ROD

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

RECOVERY OF A DROPPED CONTROL ROD

Alternate Path:

Unit is tripped upon receipt of second dropped CR

Facility JPM #:

CRO-12A

K/A Rating(s):

005-AA2.03 3.5/4.4

Task Standard:

Control Rod recovery
Unit is tripped upon receipt of second dropped CR

Preferred Evaluation Location:

Simulator ___X__ In-Plant ______

Preferred Evaluation Method:

Perform ___X__ Simulate ______

References:

AP/1/A/1700/15, Dropped Control Rods

Validation Time: 20 min. **Time Critical:** NO

==

Candidate: ______________________________________ Time Start : ________
NAME Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________________ ___________________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC # _______
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/0/A/1105/009, Control Rod Drive System

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.6.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

START TIME: _______

STEP 1: Take manual control of rods at the Diamond Control Station by performing the following: Place the Diamond Station in MANUAL STANDARD: The AUTO/MANUAL pushbutton on the Diamond Control Panel is depressed, the MANUAL half of the Push Button is backlighted. Location 1UB1 ***Cue: Inform candidate time compression has taken place and the Control Rod has been repaired and should be withdrawn.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 2: Obtain copy of Enclosure 4.10, Recovery of Dropped/Misaligned Regulating Control Rod, of OP/0/A/1105/009, Control Rod Drive System. STANDARD: Obtain a copy of OP/0/A/1105/009, Control Rod Drive System and determine that Enclosure 4.10, Recovery of Dropped/Misaligned Regulating Control Rod, is the proper enclosure for this condition and obtain a copy from the procedure file. COMMENTS:	___ SAT ___ UNSAT

<u>STEP 3</u>: Take manual control of rods at the Diamond Control Station by performing the following: Place the SG Master in HAND Place the Diamond Station in MANUAL (**Diamond is already in HAND per step 1**) <u>STANDARD</u>: The manual pushbutton for the SG Master hand/auto station is depressed, The White Hand light comes ON and the Red Auto light Goes OFF. Location 1UB1 <u>COMMENTS</u>:	___ SAT ___ UNSAT
<u>STEP 4</u>: SELECT group with dropped/misaligned rod on the Group Select Switch <u>STANDARD</u>: GROUP SELECT SWITCH on 1UB1 is located by the student and rotated to Group 6. <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: Press selector for SEQ OVERRIDE. STANDARD: The SEQ/SEQ OR pushbutton is located on the Diamond Control panel on 1UB1 and depressed. "SEQ OR" is backlighted. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6: Select JOG on the Speed Selector STANDARD: The SPEED Selector is located by the student on the Diamond Control panel on 1UB1 and rotated to the JOG position. COMMENTS:	___ SAT ___ UNSAT

STEP 7:	CRITICAL STEP
Press selector for LATCH switch and insert group for approximately 15 seconds or until the group OUT LIMIT lamp on the Diamond Panel goes off. Release LATCH switch.	___ SAT
STANDARD:	___ UNSAT
The IN LIMIT (LATCH) BYPASS pushbutton is located by the student and depressed and held while the INSERT/WITHDRAW joystick is used to insert Group 6 until the Group 6 Out limit lamp, located on the Diamond Control Panel on 1UB1, extinguishes.	
The LATCH pushbutton is then released and the INSERT/WITHDRAW joystick returned to neutral.	
COMMENTS:	

STEP 8: TRANSFER the dropped/misaligned rod to the auxiliary power supply. Select dropped/misaligned rod on the Single Select Switch Press selector for SEQ OVERRIDE Press selector for AUXILIARY Press selector for CLAMP Press selector for MANUAL TRANSFER switch until TRANSFER CONFIRM lamp and the CONTROL ON lamp on the PI panel light Press selector for CLAMP RELEASE STANDARD: On the CRD Panel on 1UB1: SELECT dropped/misaligned rod on the SINGLE SELECT SWITCH. VERIFY SEQ OR is backlit **(Not Critical).** Depresses GROUP/AUXIL pushbutton to make transfer to AUXIL. Verifies SYNC is backlit on MAN TRANS/SY/TR CF pushbutton **(Not Critical)** Depresses CLAMP/CLAMP REL pushbutton to make transfer to CLAMP. CLAMP will be backlit. Depresses MAN TRANS/SY/TR CF pushbutton. TR CF will become backlit. White CONTROL ON lights will illuminate for the Dropped Rod on the Position Indication panel. Depresses CLAMP/CLAMP REL pushbutton and verifies CLAMP REL is backlit. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 9: Perform PI alignment on the dropped/misaligned rod as follows: Press selector for the LATCH switch and insert rod for 15 seconds. Release LATCH switch. Compare absolute and relative readings on the PI panel. Adjust RPI to equal API with POSITION RESET RAISE/LOWER switch. STANDARD: Absolute and relative indications on the PI panel, on 1UB1, are compared using toggle switch to make comparison. RPI is selected with the select toggle switch. The POSITION RESET RAISE/LOWER toggle switch is then placed in the lower position and RPI indication is matched to API position. When matched the RAISE/LOWER toggle is released to neutral. The select toggle switch is returned to the API position. COMMENTS:	___ SAT ___ UNSAT

STEP 10: SELECT RUN on the Speed Selector. STANDARD: SPEED SELECTOR is located by the student on 1UB1 and rotated to the run position. ***CUE: Rod has been misaligned for less than 24 hours.*** COMMENTS:	___ SAT ___ UNSAT
STEP 11: Withdraw dropped/misaligned rod until power begins to increase and then stop withdrawal. STANDARD: Rod is withdrawn while monitoring reactor power for an increase. ***NOTE: When rod is 50% withdrawn, booth operator drop second rod.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 12: Manually trip the reactor STANDARD: The student recognizes the second control rod inserting and manually trips the reactor by depressing the Reactor Trip pushbutton and performs IMAs. COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	The Diamond is taken to manual before repairs on the CR begins.
4	Instructs the rod logic as to which group the rod is in that the operator wants to recover.
5	Allows the operator to withdraw the dropped rod.
7	The latching of the group to clear the out limit is necessary so that the individual rod can be withdrawn.
8	Places the dropped rod on the auxiliary power supply for withdrawal while leaving the group on the group power supply
11	Necessary to withdraw dropped CR.
12	The second dropped rod places the unit in an unanalyzed condition and this is a direction, which is given by OMP 1-18, Operator memory Items.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.6.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods

(R06-97) WHC/JMB/JPP

Duke Power Company TRNS. (1) ID No AP/1/A/1700/015

PROCEDURE PROCESS RECORD

Revision No 4

LAN Location: SAROS

EPARATION

Station OCONEE NUCLEAR STATION

(3) Procedure Title Dropped Control Rods

(4) Prepared By Dennis Gordon. Date 2/17/99

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)

☐ No (Revision with minor changes)

☐ No (To incorporate previously approved changes)

(6) Reviewed By Walter M. Boulware (QR) Date 2/25/99

Cross-Disciplinary Review By ______ (QR)NA WB Date ______

Reactivity Mgmt. Review By Walter M. Boulware (QR)NA ______ Date 2/25/99

(7) Additional Reviews

Reviewed By ______ Date ______

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By [signature] Date 3/8/99

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?

☐ Yes ☐ NA Listed enclosures attached?

☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?

☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?

☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Dropped Control Rods** **Continuous Use** **Reactivity Management Related**	Procedure No. AP/1/A/1700/015 Revision No. 004 Electronic Reference No. OX002RGS

DROPPED CONTROL RODS
Reactivity Management Related

TABLE OF CONTENTS

OCONEE NUCLEAR STATION

Dropped Control Rods

1. Purpose

This procedure provides the actions necessary to maintain the plant in a safe condition following a dropped control rod(s).

2. Symptoms

- "CRD POSITION ERROR" statalarm (1SA-2, B-10)
- "CRD SAFETY RODS NOT AT UPPER LIMIT" statalarm (1SA-2, C-10)
- Control Rod "IN-LIMIT" light on (Green light on Position Indication Panel)
- Control Rod "ASYMM. RODS" light on (Yellow light on Diamond).

3. Automatic Systems Actions

3.1 **IF** ICS is in Auto,

AND an "ASYMM. RODS" (Yellow Light on Diamond) occurs,

THEN an "OUT" inhibit at 60% power is established and the Reactor will runback to 55% power.

3.1.1 **IF** the "ASYMM. RODS" (Yellow Light on Diamond) clears,

THEN runback may stop before reaching 55% power.

4. Immediate Manual Actions

____ 4.1 **IF** more than one Control Rod has dropped,

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

____ 4.2 **IF** more than one Control Rod is misaligned > 9" (6%),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

____ 4.3 **IF** due to a malfunction, a Control Rod Group is misaligned > 9" (6%) from the acceptable region of PT/1/A/0600/001, (Periodic Instrument Surveillance),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

____ 4.4 **IF** a Control Rod has dropped on an approach to criticality,

OR a dropped Control Rod results in a return to subcriticality from a critical condition,

THEN manually insert all Control Rods to Group 1 at 50% WD.

5. Subsequent Actions

_____ 5.1 **IF** the Reactor has tripped,

THEN **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 5.2 Verify Reactor runback <60% Full Power is in progress.:

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

NOTE 5.2.1:	If a control rod drops while the ICS Diamond Panel is in MANUAL, an "Auto Inhibit" (safety rod) or "Sequence Enable" (regulating rod) alarm will prevent placing the ICS in automatic.

_____ 5.2.1 **IF** the Reactor has **NOT** runback,

THEN commence manual Reactor Power reduction to < 60% of the allowable thermal power for the RCP combination.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.3 **IF** operating with only three (3)RCPs,

THEN commence manual Reactor Power reduction to < 45% Full Power.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.4 Notify I&E to begin investigation for the cause of the Dropped Control Rod.

_____ 5.5 Initiate actions to meet the Required Action of ITS 3.1.4 by performing the following:

_____ 5.5.1 Within one hour verify > 1% SDM
with allowance for the inoperable control rod(s):

- Perform PT/1/A/1103/15, (Reactivity Balance Calculation).

_____ 5.5.2 Within two hours reduce Reactor Power < 60%
of the allowable thermal power for the RCP combination.

NOTE 5.5.3:	Notification to I&E should be made as soon as possible due to the complexity of resetting the RPS trip setpoints and the short ITS allowable time limits.

_____ 5.5.3 Notify I&E to reduce RPS Flux/Flow-Imbalance trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.5.4 Notify I&E to reduce RPS High Flux trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.6 **WHEN** Reactor Power is < 60%
of the allowable thermal power for the RCP combination,

THEN notify I&E to begin repair of the Dropped Control Rod.

_____ 5.7 **WHEN** I&E is ready to begin repairs on the Dropped Control Rod,

THEN Place the ICS Diamond control station in MANUAL,

AND permit I&E to repair Dropped Control Rod.

CAUTION:5.8	The Duty Reactor Engineer must evaluate the effects of local power distribution and the necessity for special maneuvering limits prior to the recovery of a dropped or an asymmetric Control Rod.

_____ 5.8 **WHEN** I&E has repaired the Dropped Control Rod,

THEN recover the Dropped Control Rod per OP/0/A/1105/009, (Control Rod Drive System).

END

Dropped Control Rods

Appendix

1. PIP # 0-O98-2734

END

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

CRO-096

ALIGN ECCS SUCTION FROM EMERGENCY SUMP
(Alternate Path)

CANDIDATE ____

EXAMINER ____

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

ALIGN ECCS SUCTION FROM EMERGENCY SUMP

TASK NUMBER: OO2650301

Alternate Path:

YES

Facility JPM #:

CRO-096

K/A Rating(s):
System: EPE 011 LARGE BREAK LOCA
K/A: EA1.11
Rating: 4.2/4.2

Task Standard:

The EOP Enclosure 7.11 is properly completed to align ECCS suction from the Emergency sump.

Preferred Evaluation Location: Simulator __X__ In-Plant _____

Preferred Evaluation Method: Perform __X__ Simulate ____

References:
EP/1/A/1800/0, Enclosure 7.11

Validation Time: 15 minutes

Time Critical: NO

==

Candidate: __________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: __________________________ _______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC or SNAP # ________
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:

EP/1/A/1800/01, Enclosure 7.11, ECCS Suction Swap to the RBES With Both LPI Header Flows > 1000 gpm.

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

A large break LOCA has occurred which is depleting the BWST.
LPI flow is ≥ 1000 gpm per header
EOP is in progress, currently on step 2.0 of CP-601 Cooldown Following Large LOCA.

INITIATING CUES:

BWST level is approaching 13 feet

START TIME: _______

STEP	CRITICAL STEP
STEP 1: IF AT ANY TIME BWST level reaches 13 feet, AND RB Level is increasing THEN transfer LPI and RBS suction to the RBES per Enclosure 7.11, ECCS Suction Swap to the RBES With Both LPI Header Flows > 1000 gpm. STANDARD: The student locates the BWST level gauges on 1UB2. The student determines level to be ≤13 feet. or The student may obtain BWST level from the OAC (Operator Aid Computer), at 1UB1, 1UB2, or STA monitor. or ICCM monitors on 1UB1. The student locates the RB level Train 1 and Train 2 gauges on 1UB1. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 2: Refer to Enclosure 7.11, "ECCS Suction Swap to RBES With Both LPI Header Flows ≥ 1000 gpm". STANDARD: Candidate refers to Enclosure 7.11, "ECCS Suction Swap to RBES With Both LPI Header Flows ≥ 1000 gpm". COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: SECURE all HPI pumps. _____ 1A HPI pump _____ 1B HPI pump _____ 1C HPI pump STANDARD: Student verifies all HPI Pumps are secured by verifying the "RED" on lights are not ON. COMMENTS:	___ SAT ___ UNSAT

STEP 4: Throttle RBS Flow in all headers with an operating pump to 900 - 1000 gpm per header: _____ 1BS-1 (1A HDR RB ISOLATION) _____ 1BS-2 (1B HDR RB ISOLATION) STANDARD: RBS Flow in 1A header throttled to 900 - 1000 gpm RBS Flow in 1B header throttled to 900 - 1000 gpm COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5: WHEN BWST level reaches 9 feet, AND RB level is increasing THEN perform the following to swap LPI suction to RBES: NOTE: RB level of ≥ 2 feet is expected STANDARD: Candidate determines BWST Level is ≤ 9 feet (decreasing) **IF BWST level is not <9 feet then,** **CUE: BWST is < 9 feet.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 6: Simultaneously open the following valves ______ 1LP-19 ('1A' RX. BLDG. SUCTION) ______ 1LP-20 ('1B' RX. BLDG. SUCTION) STANDARD: The student locates the control switch for 1LP-19 and 1LP20 ('1A' and '1B' RX. BLDG. SUCTION) on 1UB2 and rotates the switches in the OPEN direction. Verify Green CLOSED light ON, Red OPEN light OFF. **CUE: 1LP-20 will NOT respond. Student may attempt to dispatch NLOs to either manually open 1LP-20 or RESET the breaker.** **Inform student all attempts to open 1LP-20 are unsuccessful.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: IF 1LP-19 (1A RX BLDG SUCTION) fails to open **NOTE: Go to step 3.3** STANDARD: Candidate determines that 1LP-19 is OPEN COMMENTS:	___ SAT ___ UNSAT

STEP 8: IF 1LP-20 (1B RX BLDG SUCTION fails to open **THEN** perform the following: STANDARD: Candidate determines that 1LP-20 remains CLOSED COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 9: IF BWST level is continuing to decrease, THEN wait until BWST level is ≤ 6 feet before proceeding STANDARD: Candidate observes that the BWST level is ≤ 6 feet before proceeding. **If BWST is not a t 6 feet then,** **CUE: BWST level is 6 feet** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 10: IF LP-20 fails to open IF BWST level is ≤ 6 feet, THEN immediately align the following valves: _____ Close 1LP-21 (1A LPI BWST SUCTION) _____ Open 1LP-9 (1C LIP DISCH TO 1A LPI HDR) _____ Open 1LP-10 (1C LIP DISCH TO 1B LPI HDR)\ **NOTE: 1LP-20 (1A LPI BWST SUCTION) is closed** STANDARD: 1LP-21 (1A LPI BWST SUCTION) is closed 1LP-9 (1C LPI DISCH TO 1A LPI HDR) is opened 1LP-10 (1C LPI DISCH TO 1B LPI HDR) is opened COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 11: Stop the following pumps _____ 1B LPI pump _____ 1B RBS Pump STANDARD: 1B LPI pump is stopped 1B RBS Pump is stopped COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

<u>STEP 12</u>: Throttle total LPI flow per the following: A. <u>IF</u> 1LP-14 (1B LPI Cooler Outlet) has been locally throttled, <u>THEN</u> throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize "A" LPI header flow ≤1100 gpm. B. <u>IF</u> 1LP-14 (1B LPI Cooler Outlet has NOT been locally throttled, <u>THEN</u> throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize in each LPI header flow ≤1100 gpm. <u>STANDARD</u>: Candidate determines LPI Cooler out flow has NOT been locally throttled <u>COMMENTS</u>:	___ SAT ___ UNSAT
<u>STEP 13</u>: GO TO step 7 <u>STANDARD</u>: Transitions to step 7 <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT
<u>STEP 14</u>: Throttle RBS flow in all headers with and operating pump to 900 - 1000 gpm per header _____ 1BS-1 (1A HDE RB ISOLATION) _____ 1BS-2 (1B HDR RB ISOLATION) <u>STANDARD</u>: Verification of ≈1000 gpm flow is indicated in the 1A RB Spray header **NOTE: "A" and "B" RBS flow was throttled in Step 2. And "B" RBS Pump has been secured in Step 3.3.2** <u>COMMENTS</u>:	___ SAT ___ UNSAT

<u>STEP 15</u>: Notify Chemistry to perform the following ____ Commence caustic addition ____ Periodically sample the LPI discharge to determine RBES boron concentration. <u>STANDARD</u>: Chemistry is notified. <u>COMMENTS</u>:	___ SAT ___ UNSAT
<u>STEP 16</u>: <u>IF AT ANY TIME</u> BWST Level is ≤ 6 feet, <u>THEN</u> dispatch and operator to close 1LP-28 (BWST Outlet). (East of Unit 1 BWST) <u>STANDARD</u>: NLO is dispatched to close 1LP-28 (BWST Outlet) <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT
<u>STEP 17</u>: Perform the following to align LPSW to LPI Coolers: Close 1LPSW-139 (Unit 1 NONESSENTIAL HEADER ISOLATION). <u>IF</u> Unit 2 Turbine is tripped,... **CUE: Unit 2 Turbine is operating** <u>STANDARD</u>: Close 1LPSW-139 (Unit 1 NONESSENTIAL HEADER ISOLATION) <u>COMMENTS</u>:	___ SAT ___ UNSAT

STEP 18: Place the following switches in the FAIL OPEN position. _____ 1LPSW-251 FAIL SWITCH _____ 1LPSW-252 FAIL SWITCH STANDARD: 1LPSW-251 FAIL SWITCH in FAIL position 1LPSW-252 FAIL SWITCH in FAIL position COMMENTS:	___ SAT ___ UNSAT
STEP 19: If either of the following conditions exist: Three LPSW pumps are operating Two LPSW pumps are operating and only two LPSW pumps are required to be operable by TS, THEN perform the following:... STANDARD: Candidate determines LPSW pump operating status. **CUE: Two LPSW pumps are operating and only two LPSW pumps are required to be operable by TS** COMMENTS:	___ SAT ___ UNSAT

STEP 20: Open the following valves _____ 1LPSW-4 (1ALPI CLR SHELL OUTLET) _____ 1LPSW-5 (1ALPI CLR SHELL OUTLET) STANDARD: LPSW-4 (1ALPI CLR SHELL OUTLET) is opened 1LPSW-5 (1ALPI CLR SHELL OUTLET) is opened COMMENTS:	___ SAT ___ UNSAT
STEP 21: GO TO step 10.8 STANDARD: Transitions to step 10.8 COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 22: IF only one LPI cooler is available STANDARD: Determines availability of LPI coolers. Both are available. COMMENTS:	___ SAT ___ UNSAT

STEP 23: WHEN 1LP-28 (BWST Outlet is closed THEN perform the following . . . ***CUE: 1LP-28 is closed.*** STANDARD: Candidate determines that steps 11.1 and 11.2 are N/A. COMMENTS:	___ SAT ___ UNSAT
STEP 24: IF Two LPI Pumps are operating, THEN perform the following: STANDARD: Candidate determines that only 1 LPI pump is operating COMMENTS:	___ SAT ___ UNSAT
STEP 25: Initiate makeup to the BWST with boron concentration> COLR limit to provide a backup to ECCS suction source. STANDARD: **CUE: Another operator will initiate makeup to the BWST.** COMMENTS:	___ SAT ___ UNSAT

STOP TASK___________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1, 2	Transfer to Enclosure 7.11 to provide guidance in swapping suction to the RBES
4	Decreases RBS flow to prevent pump runout when suction is swapped to the RBES
5	Monitors BWST for 9 feet to actually perform suction swap to the RBES
6	Open RBES suction valves (LP-20 does not open)
8	Determines LP-20 will not open
9	Determine BWST level is < 6 feet
10	Isolates the "A" Suction line from the BWST and cross-connects the LPIP discharge header.
11	Secures the "B" train pumps to prevent air from the BWST entering the suction source
16	Manually isolates the BWST suction to prevent air in the suction
21	Proper transfer in Enclosure 7.11

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

A large break LOCA has occurred which is depleting the BWST.
LPI flow is ≥ 1000 gpm per header
EOP is in progress, currently on step 2.0 of CP-601 Cooldown Following Large LOCA.

INITIATING CUES:

BWST level is approaching 13 feet

Cooldown Following Large LOCA

_____ 1. **IF** increasing RB level **CANNOT** be verified,

THEN ensure RB Sump isolation valves are closed:

_____ 1LWD-1 (RB NORMAL SUMP ISOLATION)

_____ 1LWD-2 (RB NORMAL SUMP ISOLATION).

_____ 2. **IF** **AT ANY TIME** BWST level reaches 13 feet,

AND RB level is increasing,

THEN transfer LPI and RBS suction to the RBES.

- **REFER TO** Enclosure 7.11, "ECCS Suction Swap To RBES With Both LPI Header Flows ≥ 1000 gpm". [3]

_____ 3. **IF** **AT ANY TIME** LPI flow is required,

AND NONE of the following LPI Pumps can be operated:

- 1A LPI Pump
- 1B LPI Pump,

THEN **REFER TO** AP/1/A/1700/007, (Loss Of LPI System).

_____ 4. **IF** **AT ANY TIME** all the following exist:

- LPI flow is required
- ECCS Pump suction is aligned to the RBES
- Either operating LPI Pump fails,

THEN **REFER TO** AP/1/A/1700/007, (Loss Of LPI System).

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

1. Secure all HPI Pumps:

_____ 1A HPI Pump

_____ 1B HPI Pump

_____ 1C HPI Pump.

2. Throttle RBS flow in all headers with an operating pump to 900 - 1000 gpm per header:

_____ 1BS-1 (1A HDR RB ISOLATION)

_____ 1BS-2 (1B HDR RB ISOLATION).

_____ 2.1 **IF** RBS flow **CANNOT** be throttled ≤ 1000 gpm in either header,

THEN secure the associated RBS Pump:

_____ 1A RBS Pump

_____ 1B RBS Pump.

NOTE 3: A RB level of ≥ 2 feet is expected when BWST level reaches 9 feet.

_____ 3. **WHEN** BWST level reaches 9 feet,

AND RB level is rising,

THEN perform the following to swap LPI suction to RBES: {3}

3.1 Simultaneously open the following valves:

_____ 1LP-19 (1A RX BLDG SUCTION)

_____ 1LP-20 (1B RX BLDG SUCTION).

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

_____ 3.2 **IF** 1LP-19 (1A RX BLDG SUCTION) fails to open,

THEN perform the following:

_____ 3.2.1 **IF** BWST level is continuing to decrease,

THEN wait until BWST level is ≤ 6 feet before proceeding.

_____ 3.2.2 **IF** AT ANY TIME BWST level is ≤ 6 feet,

THEN immediately align the following valves:

_____ Close 1LP-22 (1B LPI BWST SUCTION)

_____ Open 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)

_____ Open 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

3.2.2.1 Stop the following pumps:

_____ 1A LPI Pump

_____ 1A RBS Pump.

3.2.2.2 Throttle total LPI flow per the following:

_____ A. **IF** 1LP-12 (1A LPI COOLER OUTLET) has been locally throttled,

THEN throttle 1LP-14 (1B LPI COOLER OUTLET) to maximize 'B' LPI Header flow ≤ 1100 gpm.

_____ B. **IF** 1LP-12 (1A LPI COOLER OUTLET) has **NOT** been locally throttled,

THEN maximize flow in each LPI Header ≤ 1100 gpm.

_____ 3.2.2.3 **GO TO** Step 7.

ECCS Suction Swap To RBES With
<u>Both</u> LPI Header Flows ≥ 1000 gpm

_____ 3.3 **<u>IF</u>** 1LP-20 (1B RX BLDG SUCTION) fails to open,

<u>THEN</u> perform the following:

_____ 3.3.1 **<u>IF</u>** BWST level is continuing to decrease,

<u>THEN</u> wait until BWST level is ≤ 6 feet before proceeding.

_____ 3.3.2 **<u>IF</u>** <u>AT ANY TIME</u> BWST level is ≤ 6 feet,

<u>THEN</u> <u>immediately</u> align the following valves:

_____ Close 1LP-21 (1A LPI BWST SUCTION)

_____ Open 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)

_____ Open 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

3.3.2.1 Stop the following pumps:

_____ 1B LPI Pump

_____ 1B RBS Pump.

3.3.2.2 Throttle <u>total</u> LPI flow per the following:

_____ A. **<u>IF</u>** 1LP-14 (1B LPI COOLER OUTLET) has been locally throttled,

<u>THEN</u> throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize 'A' LPI Header flow ≤ 1100 gpm.

_____ B. **<u>IF</u>** 1LP-14 (1B LPI COOLER OUTLET) has <u>NOT</u> been locally throttled,

<u>THEN</u> maximize flow in <u>each</u> LPI Header ≤ 1100 gpm.

_____ 3.3.2.3 **GO TO** Step 7.

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

_____ 4. **IF** BWST level is continuing to decrease,

THEN wait until BWST level is ≤ 6 feet before proceeding.

_____ 5. **IF** AT ANY TIME BWST level is ≤ 6 feet,

THEN immediately perform the following:

5.1 Simultaneously close the following valves:

_____ 1LP-21 (1A LPI BWST SUCTION)

_____ 1LP-22 (1B LPI BWST SUCTION).

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

_____ 5.2 **IF** 1LP-21 (1A LPI BWST SUCTION) fails to close,

THEN perform the following:

5.2.1 Simultaneously open the following valves:

_____ 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)

_____ 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

5.2.2 Stop the following pumps:

_____ 1A LPI Pump

_____ 1A RBS Pump.

5.2.3 Throttle total LPI flow per the following:

_____ 5.2.3.1 **IF** 1LP-12 (1A LPI COOLER OUTLET) has been locally throttled,

THEN throttle 1LP-14 (1B LPI COOLER OUTLET) to maximize 'B' LPI Header flow ≤ 1100 gpm.

_____ 5.2.3.2 **IF** 1LP-12 (1A LPI COOLER OUTLET) has NOT been locally throttled,

THEN maximize flow in each LPI Header ≤ 1100 gpm.

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

_____ 5.3 **IF** 1LP-22 (1B LPI BWST SUCTION) fails to close,

THEN perform the following:

5.3.1 Simultaneously open the following valves:

_____ 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)

_____ 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

5.3.2 Stop the following pumps:

_____ 1B LPI Pump

_____ 1B RBS Pump.

5.3.3 Throttle total LPI flow per the following:

_____ 5.3.3.1 **IF** 1LP-14 (1B LPI COOLER OUTLET) has been locally throttled,

THEN throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize 'A' LPI Header flow ≤ 1100 gpm.

_____ 5.3.3.2 **IF** 1LP-14 (1B LPI COOLER OUTLET) has NOT been locally throttled,

THEN maximize flow in each LPI Header ≤ 1100 gpm.

6. Throttle total LPI flow per the following:

____ 6.1 **IF** only one LPI Pump is operating,

THEN limit LPI flow per the following:

____ 6.1.1 **IF** 1LP-12 (1A LPI COOLER OUTLET) has been locally throttled,

THEN throttle 1LP-14 (1B LPI COOLER OUTLET) to maximize 'B' LPI Header flow ≤ 1100 gpm.

____ 6.1.2 **IF** 1LP-14 (1B LPI COOLER OUTLET) has been locally throttled,

THEN throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize 'A' LPI Header flow ≤ 1100 gpm.

____ 6.1.3 **IF** NEITHER of the following valves have been locally throttled:

- 1LP-12 (1A LPI COOLER OUTLET)
- 1LP-14 (1B LPI COOLER OUTLET),

THEN maximize LPI flow in each header ≤ 1100 gpm.

ECCS Suction Swap To RBES With
<u>Both</u> LPI Header Flows ≥ 1000 gpm

_____ 6.2 **<u>IF</u>** <u>two</u> LPI Pumps are operating,

<u>THEN</u> limit LPI flow per the following:

_____ 6.2.1 **<u>IF</u>** 1LP-12 (1A LPI COOLER OUTLET) has been locally throttled,

<u>THEN</u> throttle 1LP-14 (1B LPI COOLER OUTLET) to maximize 'B' LPI Header flow ≤ 3000 gpm.

_____ 6.2.2 **<u>IF</u>** 1LP-14 (1B LPI COOLER OUTLET) has been locally throttled,

<u>THEN</u> throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize 'A' LPI Header flow ≤ 3000 gpm.

_____ 6.2.3 **<u>IF</u>** <u>NEITHER</u> of the following valves have been locally throttled:

- 1LP-12 (1A LPI COOLER OUTLET)
- 1LP-14 (1B LPI COOLER OUTLET),

<u>THEN</u> maximize LPI flow in <u>each</u> header ≤ 3000 gpm.

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

7. Throttle RBS flow in all headers with an operating pump to 900 - 1000 gpm/header:

_____ 1BS-1 (1A HDR RB ISOLATION)

_____ 1BS-2 (1B HDR RB ISOLATION).

8. Notify Chemistry to perform the following:

_____ Commence caustic addition

_____ Periodically sample the LPI discharge to determine RBES boron concentration.

_____ 9. **IF** AT ANY TIME BWST level is ≤ 6 feet,

THEN dispatch an operator to close 1LP-28 (BWST Outlet).
(East of Unit 1 BWST)

10. Perform the following to align LPSW to LPI Coolers:

_____ 10.1 Close 1LPSW-139 (UNIT 1 NONESSENTIAL HEADER ISOLATION).

_____ 10.2 **IF** Unit 2 Turbine is tripped,

THEN close 2LPSW-139 (U-2 NON-ESSENTIAL HDR ISOLATION VALVE).

10.3 Place the following switches in the "FAIL OPEN" position:

_____ "1LPSW-251 FAIL SWITCH"

_____ "1LPSW-252 FAIL SWITCH".

_____ 10.4 **IF** either of the following conditions exists:

- Three LPSW Pumps are operating
- Two LPSW Pumps are operating **AND** only two LPSW Pumps are required to be operable by TS,

THEN perform the following:

10.4.1 Open the following valves:

_____ 1LPSW-4 (1A LPI CLR SHELL OUTLET)

_____ 1LPSW-5 (1B LPI CLR SHELL OUTLET).

_____ 10.4.2 **GO TO** Step 10.8.

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

_____ 10.5 **IF** either LPI Cooler LPSW Flow DIXSON indicator is blank (de-energized),

THEN consider the associated LPI Cooler inoperable.

_____ 10.5.1 **GO TO** Step 10.8.

NOTE:	The DIXSON LPSW flow indicators must be used when determining post accident flow readings.

_____ 10.6 Throttle open 1LPSW-4 (1A LPI CLR SHELL OUTLET) to establish 3000-3300 gpm LPSW flow to the 1A LPI Cooler.

_____ 10.7 Throttle open 1LPSW-5 (1B LPI CLR SHELL OUTLET) to establish 3000-3300 gpm LPSW flow to the 1B LPI Cooler.

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

NOTE 10.8:	If only two LPSW pumps are available and LPI is in operation on Unit 2, LPSW loads may need to be reduced to attain required flow through the Unit 1 LPI cooler.

_____ 10.8 **IF** only one LPI cooler is available,

THEN perform the following to establish proper LPSW flow:

_____ 10.8.1 **IF** 1A cooler is available,

THEN perform the following:

_____ 10.8.1.1 Close 1LPSW-5 (1B LPI CLR SHELL OUTLET).

_____ 10.8.1.2 Throttle 1LPSW-4 (1A LPI CLR SHELL OUTLET) to establish ≈ 5500 gpm in the 1A LPI Cooler.

_____ 10.8.2 **IF** 1B cooler is available,

THEN perform the following:

_____ 10.8.2.1 Close 1LPSW-4 (1A LPI CLR SHELL OUTLET).

_____ 10.8.2.2 Throttle 1LPSW-5 (1B LPI CLR SHELL OUTLET) to establish ≈ 5500 gpm in the 1B LPI Cooler.

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

_____ 11. **WHEN** 1LP-28 (BWST Outlet) is closed,

THEN perform the following:

_____ 11.1 **IF** 1LP-21 (1A LPI BWST SUCTION) failed to close,

AND 1LP-19 (1A RX BLDG SUCTION) is open,

THEN restart 1A LPI Pump.

_____ 11.2 **IF** 1LP-22 (1B LPI BWST SUCTION) failed to close,

AND 1LP-20 (1B RX BLDG SUCTION) is open,

THEN restart 1B LPI Pump.

ECCS Suction Swap To RBES With
Both LPI Header Flows ≥ 1000 gpm

_____ 12. **IF** two LPI Pumps are operating,

THEN perform the following:

12.1 Close the following valves:

_____ 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)

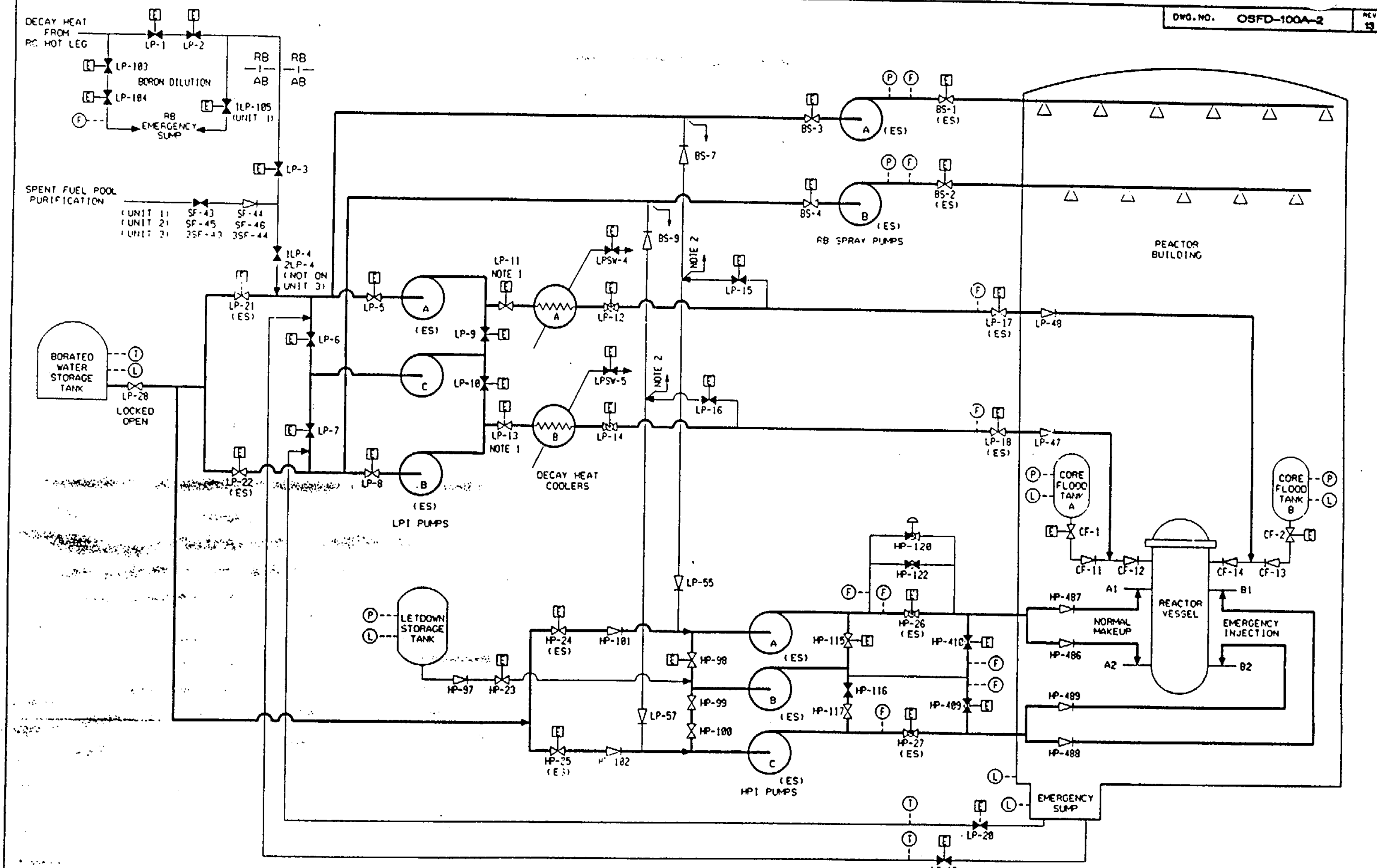
_____ 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

_____ 12.2 Maximize LPI flow in each header ≤ 3000 gpm.

_____ 13. Initiate makeup to the BWST with boron concentration > COLR limit to provide a backup ECCS suction source.

- **REFER TO** OP/1/A/1104/004A (BWST Operation).

••• END •••



NOTES:

1. LP-11 & LP-13 ARE MANUAL ON UNIT #3.
2. THE R.B. SPRAY TIE-IN CONNECTION SHOWN IS FOR UNIT 1 & 3 ONLY.

ERN: 0X001I7W

LEGEND

- SHUTOFF VALVE
- FLOW CONTROL VALVE
- CHECK VALVE (ALL TYPES)
- RELIEF VALVE
- NORMALLY OPEN
- NORMALLY CLOSED
- NORMALLY THROTTLED
- F-FLOW, L-LEVEL, P-PRESSURE, T-TEMPERATURE, R-RADIATION MONITOR
- E-ELECTRIC, H-HYDRAULIC, P-PISTON (PNEUMATIC), S-SOLENOID
- DIAPHRAGM (PNEUMATIC)
- (ES)-RECEIVES ENGINEERED SAFEGUARD SIGNAL

THIS DRAWING IS A SUMMARY FLOW DIAGRAM. FOR COMPLETE SYSTEM DESIGN INFORMATION REFER TO FLOW DIAGRAMS LISTED BELOW.

OFD-101A-1.2, -2.2, -3.2	LETDOWN STORAGE TANK
OFD-101A-1.3, -2.3, -3.3	HPI PUMPS
OFD-101A-1.4, -2.4, -3.4	HPI TO RC SYSTEM
OFD-102A-1.1, -2.1, -3.1	BWST & EMERGENCY SUMP
OFD-102A-1.2, -2.2, -3.2	LPI PUMPS & COOLERS
OFD-102A-3.1, -2.3, -3.3	CORE FLOOD TANKS
OFD-103A-1.1, -2.1, -3.1	RB SPRAY SYSTEM
OFD-104A-1.2, -2.2, -3.2	SPENT FUEL POOL PURIF.

TYPICAL FOR UNITS 1, 2 & 3

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

SUMMARY FLOW DIAGRAM OF EMERGENCY CORE COOLING AND RB SPRAY SYSTEMS

NO.	REVISIONS
13	REV. PER OE-[illegible]
12	REV. PER [illegible]
11	REV. PER NSM ON-[illegible]
-	NSM ON-[illegible], NSM ON-[illegible]
[illegible]	ORIGINAL DRAWING RETIRED

DWG. NO. OSFD-100A-2 REV. 13

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM NRC-002/SIM
RC-66 (PZR PORV) Stoke Test
Alternate Path

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:
Perform RC-66 PORV Stroke Test

Alternate Path:

When 1RC-66 (PZR PORV) fails to manually close after stroke testing 1RC-4 (PZR PORV Block) is closed to isolate a stuck open PORV.

Facility JPM #:

NEW

K/A Rating(s):

010A4.03 [4.0/3.8]

Task Standard:

Accomplish the stoke test for 1RC-66 (PORV) per PT/201/004.
Close 1RC-4 to isolate a stuck open PORV

Preferred Evaluation Location: Simulator __X__ In-Plant ______

Preferred Evaluation Method: Perform __X__ Simulate ______

References:
PT/201/004

Validation Time: 10 min. **Time Critical:** NO

===

Candidate: ______________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________________ __________________________________/________
NAME SIGNATURE DATE

===

COMMENTS

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall SNAP 104
2. IMPORT NRC-2 Files (PORV fails open after lifting)
3. Set LPSW to both LPI Coolers to ≈ 900 gpm/cooler
4. Place QT in recirc (Open CS 5 and6 then start the Component Drain Pump)
5. Override QT press to 0 psig (prevent increase)

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

PT/201/004

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC
Unit 1 startup is in progress
Establishing a PZR bubble is in progress

INITIATING CUES:

The SRO in the Control Room directs you to complete operations to establish a PZR bubble using the in-progress procedure OP/1103/002 Encl 4.14 starting at step 2.17.

START TIME: _______

STEP 1: Per procedure step 2.17, Perform PT/1/A/0201/004, (1RC-66 Stoke Test) STANDARD: Per PT/1/A/0201/004 verify required unit status: RCS pressure < 45 psig QT lined up for lowering QT temperature per OP/1104/17 ***CUE: Unit status requirements have been met*** COMMENTS:	___ SAT ___ UNSAT
STEP 2: Verify Prerequisite System Conditions: STANDARD: Steam bubble is formed in the PZR RC-4 is operable ***Cue: PORV Outlet thermocouples are available on the OAC.*** COMMENTS:	___ SAT ___ UNSAT
.EP 3: Ensure no personnel are present in the SG cavities of the RB STANDARD: ***CUE: No personnel is in the RB at this time*** COMMENTS:	___ SAT ___ UNSAT
STEP 4: Cycle 1RC-4 (PORV Block) STANDARD: 1RC-4 is positioned to close and verified to be closed by the green light on 1RC-4 is positioned to open and verified to be open by the red light on* ***Reopening 1RC-4 is critical.** COMMENTS:	**CRITICAL STEP*** ___ SAT ___ UNSAT

STEP 5: Record data required on Enclosure 13.4 (Information Sheet) prior to opening 1RC-66 (PORV) STANDARD: Time, RCS Pressure, QT pressure, QT temperature, QT level, RC-66 Outlet temperature, PZR level. This information is recorded on Encl 13.4 COMMENTS:	___ SAT ___ UNSAT
STEP 6: Open 1RC-66 (PORV) STANDARD: RC-66 switch positioned to OPEN and the OPEN PERMIT depressed COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: Monitor the following to verify 1RC-66 is open, then close 1RC-66 after positive indications are verified STANDARD: Parameters/indications are verified to ensure 1RC-66 is open • PRZ Relief valve flow monitor • 1RC-66 position indication (pilot valve) COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 8: Close 1RC-66 (PORV) STANDARD: Select LOW on the 1RC-66 switch When 1RC-66 is attempted to be closed diagnose 1RC-66 failed to close and close 1RC-4 (1RC-66 Block) COMMENTS: END TASK	**CRITICAL STEP** ___ SAT ___ UNSAT

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
4	Reopening 1RC-4 establishes a flowpath from the PZR to QT.
6	This step opens 1RC-66.
7	Verifying PORV is open by alternate means. PORV pilot valve light indication is not used.
8	After 1RC-66 failure to close is diagnosed 1RC-4 is closed to stop flow.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the unit 1 OATC
Unit 1 startup is in progress
Establishing a PZR bubble is in progress

INITIATING CUES:

The SRO in the Control Room directs you to complete operations to establish a PZR bubble using the in-progress procedure OP/1103/002 Encl 4.13 starting at step 2.2
Unit 1 startup is in progress
Establishing a PZR bubble is in progress

INITIATING CUES:

The SRO in the Control Room directs you to complete operations to establish a PZR bubble using the in-progress procedure OP/1103/002 Encl 4.13 starting at step 2.2

(R06-97) TNH/WHC/RAQ
SIM, SR(13)

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No PT/0/A/0201/004
Revision No 004

OTC MASTER FILE

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title PORV OPERABILITY TEST

(4) Prepared By Michele D. Eberhart Date 10/20/97

(5) Requires 10CFR50.59 evaluation?
☐ Yes (New procedure or revision with major changes)
☑ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By RC Gamberg (QR) Date 10/23/97
Cross-Disciplinary Review By [signature] Jr (QR)NA Date 10/23/97
Reactivity Mgmt. Review By (QR)NA [initials] Date 10/23/97

(7) Additional Reviews
Reviewed By Date
Reviewed By Date

(8) Temporary Approval *(if necessary)*
By (SRO/QR) Date
By (QR) Date

(9) Approved By [signature] Curtis Date 10/26/97

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy Date
Compared with Control Copy Date
Compared with Control Copy Date

(11) Date(s) Performed
Work Order Number (WO#)

COMPLETION

(12) Procedure Completion Verification
☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By Date

(13) Procedure Completion Approved Date

(14) Remarks *(Attach additional pages, if necessary)*

SIMULATOR
CONTROL COPY
1997

Duke Power Company Oconee Nuclear Station **PORV Operability Test** **Continuous Use**	Procedure No. PT/0/A/0201/004 Revision No. 004 Electronic Reference No. OX002VWW

PORV Operability Test

1. Purpose

1.1 To verify operability of (1) (2) (3) RC-66 (PORV) prior to exceeding 200°F (RC average temperature).

2. References

2.1 OFD-100A

3. Time Required

3.1 30 minutes

4. Prerequisite Test

None

5. Test Equipment

None

6. Limits and Precautions

6.1 Ensure that no personnel are in the Reactor Building SG cavities while (1)(2)(3) RC-66 (PORV) is being tested.

6.2 Do not perform test at greater than 45 psig in RCS.

6.3 Do not exceed 200°F (RC average temperature) until (1) (2) (3) RC-66 (PORV) is verified operable.

7. Required Unit Status

_____ 7.1 Plant in the startup mode with RC pressure ≤ 45 psig.

_____ 7.2 Quench Tank lined up for lowering Quench Tank temperature per OP/1,2,3/A/1104/17 (Quench Tank Operation).

8. Prerequisite System Conditions

_____ 8.1 Verify (1) (2) (3) RC-66 (PORV) outlet thermocouple temperature available on computer.

_____ 8.2 Steam bubble established in Pressurizer.

_____ 8.3 (1) (2) (3) RC-4 (POWER RELIEF BLOCK) operable.

9. Test Method

9.1 With steam bubble in Pressurizer and RCS pressure < 45 psig, (1) (2) (3) RC-66 (PORV) will be opened with (1)(2)(3) RC-4 (POWER RELIEF BLOCK) open and then (1)(2)(3) RC-66 (PORV) will be closed. Operator will verify opening of valve by observing a change in: Valve outlet temperature, Pressurizer level, Quench Tank pressure, level, temperature and PORV Flow Monitor.

10. Data Required

10.1 Quench Tank pressure and temperature

10.2 PORV outlet temperature

10.3 Pressurizer level

10.4 RCS pressure

11. Acceptance Criteria

11.1 Test results are acceptable upon positive indication that (1) (2) (3) RC-66 (PORV) opens and closes.

12. Procedure

12.1 Verify operability of 1RC-66 (PORV) using Enclosure 13.1 (Unit 1 PORV Test).

12.2 Verify operability of 2RC-66 (PORV) using Enclosure 13.2 (Unit 2 PORV Test).

12.3 Verify operability of 3RC-66 (PORV) using Enclosure 13.3 (Unit 3 PORV Test).

12.4 Record data required on Enclosure 13.4 (Information Sheet).

13. Enclosures

13.1 Unit 1 PORV Test

13.2 Unit 2 PORV Test

13.3 Unit 3 PORV Test

13.4 Information Sheet

1. Procedure:

_____ 1.1 Verify that no personnel are present in the SG cavities of the Reactor Building.

1.2 Prior to cycling 1RC-66 (PORV), verify operability of 1RC-4 (POWER RELIEF BLOCK) as follows:

_____ 1.2.1 Close 1RC-4 (POWER RELIEF BLOCK).

_____ _____ 1.2.2 Open 1RC-4 (POWER RELIEF BLOCK).

_____ 1.3 Record data required on Enclosure 13.4 (Information Sheet) prior to opening 1RC-66 (PORV).

1.4 Open 1RC-66 (PORV) as follows:

_____ 1.4.1 Position the 1RC-66 SETPOINT SELECTOR to "OPEN".

NOTE:	1RC-66 remains open when the "OPEN PERMIT" Pushbutton is released until 1RC-66 SETPOINT SELECTOR position is changed to "LOW".

_____ _____ 1.4.2 Depress the 1RC-66 "OPEN PERMIT" Pushbutton.

_____ 1.5 Monitor the following to verify 1RC-66 (PORV) is open:

- Quench Tank pressure increasing
- Quench Tank level increasing
- 1RC-66 (PORV) indicates open on Control Board
- PORV Flow Monitor indicates flow
- Pressurizer Relief Valve Flow Statalarm if 5 or more lights are lit.

_____ _____ 1.6 When there is positive indication that 1RC-66 (PORV) is open, select “Low” on 1RC-66 Setpoint Selector.

_____ 1.6.1 Verify 1RC-66 (PORV) is closed.

_____ _____ 1.6.2 If 1RC-66 (PORV) fails to close, close 1RC-4 (POWER RELIEF BLOCK) immediately.

_____ 1.7 Record data required on Enclosure 13.4 (Information Sheet) when 1RC-66 (PORV) is closed.

1. Procedure:

_____ 1.1 Verify that no personnel are present in the SG cavities of the Reactor Building.

1.2 Prior to cycling 2RC-66 (PORV), verify operability of 2RC-4 (POWER RELIEF BLOCK) as follows:

_____ 1.2.1 Close 2RC-4 (POWER RELIEF BLOCK).

_____ _____ 1.2.2 Open 2RC-4 (POWER RELIEF BLOCK).

_____ 1.3 Record data required on Enclosure 13.4 (Information Sheet) prior to opening 2RC-66 (PORV).

1.4 Open 2RC-66 (PORV) as follows:

_____ 1.4.1 Position the 2RC-66 SETPOINT SELECTOR to "OPEN".

NOTE:	2RC-66 remains open when the "OPEN PERMIT" Pushbutton is released until 2RC-66 SETPOINT SELECTOR position is changed to "LOW".

_____ _____ 1.4.2 Depress the 2RC-66 "OPEN PERMIT" Pushbutton.

_____ 1.5 Monitor the following to verify 2RC-66 (PORV) is open:

- Quench Tank pressure increasing
- Quench Tank level increasing
- 2RC-66 (PORV) indicates open on Control Board
- PORV Flow Monitor indicates flow
- Pressurizer Relief Valve Flow Statalarm if 5 or more lights are lit.

_____ _____ 1.6 When there is positive indication that 2RC-66 (PORV) is open, select “Low” on 2RC-66 Setpoint Selector.

_____ 1.6.1 Verify 2RC-66 (PORV) is closed.

_____ _____ 1.6.2 If 2RC-66 (PORV) fails to close, close 2RC-4 (POWER RELIEF BLOCK) immediately.

_____ 1.7 Record data required on Enclosure 13.4 (Information Sheet) when 2RC-66 (PORV) is closed.

1. Procedure:

_____ 1.1 Verify that no personnel are present in the SG cavities of the Reactor Building.

1.2 Prior to cycling 3RC-66 (PORV), verify operability of 3RC-4 (POWER RELIEF BLOCK) as follows:

_____ 1.2.1 Close 3RC-4 (POWER RELIEF BLOCK).

_____ _____ 1.2.2 Open 3RC-4 (POWER RELIEF BLOCK)

_____ 1.3 Record data required on Enclosure 13.4 (Information Sheet) prior to opening 3RC-66 (PORV).

1.4 Open 3RC-66 (PORV) as follows::

_____ 1.4.1 Position the 3RC-66 SETPOINT SELECTOR to "OPEN"

NOTE:	3RC-66 remains open when the "OPEN PERMIT" Pushbutton is released until 3RC-66 SETPOINT SELECTOR position is changed to "LOW".

_____ _____ 1.4.2 Depress the 3RC-66 "OPEN PERMIT" Pushbutton.

_____ 1.5 Monitor the following to verify 3RC-66 (PORV) is open

- Quench Tank pressure increasing
- Quench Tank level increasing
- 3RC-66 (PORV) indicates open on Control Board
- PORV Flow Monitor indicates flow
- Pressurizer Relief Valve Flow Statalarm if 5 or more lights are lit.

_____ _____ 1.6 When there is positive indication that 3RC-66 (PORV) is open, select “Low” on 3RC-66 Setpoint Selector.

_____ 1.6.1 Verify 3RC-66 (PORV) is closed.

_____ _____ 1.6.2 If 3RC-66 (PORV) fails to close, close 3RC-4 (POWER RELIEF BLOCK) immediately.

_____ 1.7 Record data required on Enclosure 13.4 (Information Sheet) when 3RC-66 (PORV) is closed.

UNIT ____________

Time	RCS Press $\leq$ 45 PSIG	Quench Tank Press	Quench Tank Level	Quench Tank Temp.	RC-66 Outlet Temp	PZR Level

Enclosure 13.4

Information Sheet

UNIT ____________

Time	RCS Press ≤ 45 PSIG	Quench Tank Press	Quench Tank Level	Quench Tank Temp.	RC-66 Outlet Temp	PZR Level
[illegible]	38.0	.1	84	123	48	90

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

CRO-013/SIM
ALIGN MD EFDWP SUCTION TO THE HOTWELL AND FEED THE STEAM GENERATORS

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

ALIGN MDEFDWP SUCTION TO THE HOTWELL AND FEED THE STEAM GENERATORS

Alternate Path:

N/A

Facility JPM #:

CRO-13

K/A Rating(s):

010A4.02 [3.6/3.4]

Task Standard:

The MDEFDWPs are aligned to the Hotwell and providing flow to the SGs within limits prior to reaching a level of 0" in the Hotwell. Step 8.0 of Section 503 of AP/1/A/1700/19 is properly completed.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

AP/1/A/1700/19

Validation Time: 20 min. **Time Critical:** NO

==

Candidate: ____________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ____________________________ __________________________________ / ________
NAME SIGNATURE DATE

==

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall SNAP 209
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

AP/1/A/1700/19

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

The unit has experienced a Loss of Power. The TD EFDWP is unavailable. Actions of the EOP have been completed and power has been restored. AP/1/A/1700/19, Loss of Main Feedwater, Section 503 has been completed up to step 8.0. Main FDW is not expected back for several hours.

UST makeup flow capability has been lost.

INITIATING CUE:

The Control Room Supervisor directs you to continue with AP/1/A/1700/19, Loss of Main Feedwater.

START TIME: _______

STEP 1: WHEN the UST < 4 feet THEN dispatch two operators to perform EP/1/A/1800/01, Enclosure 7.7, "Operation of the Atmospheric Dump Valves". STANDARD: Determines UST is < 4 feet by monitoring: • OAC analog points • UST B LEVEL meter on 1AB-1 • UST A LEVEL meter on 1AB-3 • UST LEVEL chart recorder on 1VB-1 • Statalarm 1SA-6/A-11, Upper Surge Tank Level Low Dispatches two operators to Atmospheric Dump Valves. COMMENTS:	___ SAT ___ UNSAT

STEP 2: WHEN the UST level < 3 feet THEN align the Emergency Feedwater Pumps Suction to the Hotwell as follows: Stop all CBPs Stop all HWP's STANDARD: Determines UST is < 3 feet by monitoring: • OAC analog points • UST B LEVEL meter on 1AB-1 • UST A LEVEL meter on 1AB-3 • UST LEVEL chart recorder on 1VB-1 • Statalarm 1SA-6/A-11, Upper Surge Tank Level Low Places all CBP control switches in OFF Places all HWP control switches in OFF COMMENTS:	___ SAT ___ UNSAT

STEP 3: Control SG pressure with ADV's as necessary. STANDARD: ***CUE: Another RO is coordinating with an NLO to maintain SG pressure via ADVs.*** COMMENTS:	___ SAT ___ UNSAT
STEP 4: **IF** power is available **THEN** perform the following Open 1V-186 (Vacuum Breaker) STANDARD: Opens 1V-186 COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

EP 5: Dispatch an operator with a safety harness to 1C-573 to standby until further notice. STANDARD: Dispatches an operator with a safety harness to 1C-573 ***CUE: Inform student that an operator has been dispatched to 1C-573.*** COMMENTS:	___ SAT ___ UNSAT
STEP 6: Close the following 1MS-47, MS to CSAE's 1AS-40, CSAE Aux Steam Supply STANDARD: Closes 1MS-47 Closes 1AS-40 COMMENTS:	___ SAT ___ UNSAT

STEP 7:

Monitor and adhere to the following flow limits:

MD EFDW Pump flow rates <440 gpm / pump (0.22 x 10^6 lbm / hr)

STANDARD:

Monitors MD EFDW Pump flow rates and throttles 1FDW-315 and 1FDW-316 as necessary to maintain < 440 gpm/pump.

COMMENTS:

CRITICAL STEP

___ SAT

___ UNSAT

STEP 8:

IF AT ANY TIME UST level ≤ 1 foot,

AND 1C-573 (MD EFDWP Suction from UST) open.

THEN secure all Emergency FDWPS.

STANDARD:

Monitors UST level and secures all EFDW pumps if level is ≤ 1 foot.

COMMENTS:

CRITICAL STEP

STEP 9: **WHEN** vacuum is broken, **THEN** locally close 1C-573 (MD EFDWP Suction from UST). STANDARD: Monitors Vacuum and then closes 1C-573. COMMENTS:	**CRITICAL STEP**
STEP 10: Dispatch and operator to 1C-157 (TD EFDWP Suction from UST) to standby until further notice. STANDARD: Dispatch an operator to 1C-157. ***CUE: Inform student an operator has been dispatched to 1C-157.*** COMMENTS:	

STEP 11: Open 1C-391 (TD EFDWP Suction from Hotwell) STANDARD: Open 1C-391 COMMENTS:	**CRITICAL STEP**
STEP 12: Close 1C-157 (TD EFDWP Suction from UST) STANDARD: Instruct operator to close 1C-157. ***CUE: Inform student that 1C-157 is closed.*** COMMENTS: **END TASK**	

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
4	Condenser vacuum must be broken thus increasing the NPSH to the EFDWPs. This prevents EFDWP damage due to not meeting suction head requirements when Hotwell level is < 2 feet.
7	MD EFDWP flow is throttled via FDW-315 and 316 to limit flow < 440 gpm to prevent pump run-out damage.
8	1C-573 is closed to prevent air from entering the suction of the MD EFDWPs.
9	1C-573 is closed to prevent air from entering the suction of the MD EFDWPs.
11	1C-391 aligns a suction flow path to the TD EFDW pump from the Hotwell.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

The unit has experienced a Loss of Power. The TD EFDWP is unavailable. Actions of the EOP have been completed and power has been restored. AP/1/A/1700/19, Loss of Main Feedwater, Section 503 has been completed up to step 8.0. Main FDW is not expected back for several hours.

UST makeup flow capability has been lost.

INITIATING CUE:

The Control Room Supervisor directs you to continue with AP/1/A/1700/19, Loss of Main Feedwater.

Extended EFDW Operation

_____ 1. Monitor Emergency FDW parameters.

- **REFER TO** OAC Computer Group Display - GD AP19.

CAUTION 2:	Pump damage may occur if a MDEFDW Pump is operated with discharge pressure > 1420 psig and discharge flow < 110 gpm.

_____ 2. **IF** **AT ANY TIME** either MDEFDW Pump discharge pressure indicates > 1420 psig,

AND indicated flow from that MDEFDW Pump on the "MDEFWP DISCH FLOW" meter is < 110 gpm,

THEN perform the following:

_____ 2.1 Immediately stop the affected MDEFDW Pump.

_____ 2.2 Start the TDEFDW Pump.

_____ 2.3 **IF** **NO** Emergency FDW Pumps are available,

THEN **GO TO** Section 501, Establishing Emergency Feedwater.

_____ 3. **IF** both MDEFDW Pumps are running,

AND Emergency FDW flow in each header is < 600 gpm,

THEN perform the following:

_____ 3.1 Place the TDEFDW Pump switch in "PULL TO LOCK".

_____ 3.2 **GO TO** Step 5.

_____ 4. **IF** TDEFDW Pump is supplying feedwater to any SG,

THEN perform the following:

_____ 4.1 Place the TDEFDW Pump switch in "RUN".

_____ 4.2 Ensure 1LPSW-137 (LPSW TO TDEFDWP COOLING JACKET) is open.

_____ 4.3 Ensure the Emer FDWPT Brng Oil Cooling Pump is on.

5. Maintain UST level > 7 ft by performing the following as required:

_____ Make up to UST with Demin water

_____ Verify CST pumps are in "AUTO".

_____ 5.1 **IF** UST level **CANNOT** be maintained > 7 ft,

THEN dispatch an operator to close 1C-186 (Hotwell Emergency Makeup #1 Control Inlet). (T-1, West of E-24)

_____ 6. Establish Condensate System recirc.

- **REFER TO** Enclosure 6.2, "Aligning For Recirculation Of The Condensate System".

_____ 7. **IF** AT ANY TIME conditions allow Main FDW Pump restoration,

THEN **REFER TO** Section 504, Establishing Main Feedwater.

Extended EFDW Operation

_____ 8. **WHEN** UST level < 4 feet,

THEN dispatch two operators to perform EP/1/A/1800/001 (Emergency Operating Procedure) Enclosure 7.7, "Operation Of The Atmospheric Dump Valves".

CAUTION 9:	With vacuum broken, pump damage may occur if Emergency FDW Pump suction is NOT aligned to the Hotwell prior to reaching 1 foot in the UST.

_____ 9. **WHEN** UST level < 3 feet, {8}

THEN align Emergency FDW Pumps suction to the Hotwell as follows:

_____ 9.1 Stop all CBPs.

_____ 9.2 Stop all HWPs.

_____ 9.3 Control SG pressure with ADVs, as necessary.

_____ 9.4 **IF** power is available,

THEN perform the following:

_____ 9.4.1 Open 1V-186 (VACUUM BREAKER).

_____ 9.4.2 Dispatch an operator with a safety harness to standby until further notice near 1C-573 (MD EFDWPS Suction from UST). (T-1, E-24 South West, 8' above floor)

Extended EFDW Operation

9.4.3 Close the following valves:

_____ 1MS-47 (MS TO CSAE)

_____ 1AS-40 (AS TO CSAE).

_____ 9.4.4 Monitor <u>and</u> adhere to the following flow limits:

- MDEFDW Pump flow rates < 440 gpm/pump (0.22 x 10^6 lbm/hr)
- TDEFDW Pump flow rate < 1000 gpm total (0.5 x 10^6 lbm/hr).

_____ 9.4.5 **<u>IF</u>** **<u>AT ANY TIME</u>** UST level ≤ 1 foot,

<u>AND</u> 1C-573 (MD EFDWPS Suction from UST) open,

<u>THEN</u> perform the following:

_____ 9.4.5.1 Stop <u>all</u> Emergency FDWPs.

_____ 9.4.5.2 **<u>IF</u>** vacuum **<u>CANNOT</u>** be broken,

<u>THEN</u> **GO TO** Step 10.

_____ 9.4.6 **<u>WHEN</u>** vacuum is broken,

<u>THEN</u> locally close 1C-573 (MD EFDWPS Suction from UST).

_____ 9.4.7 **<u>IF</u>** MDEFDW Pumps stopped in Step 9.4.5,

<u>THEN</u> restart MDEFDW Pumps.

Extended EFDW Operation

NOTE 9.4.8:	The TDEFDW Pump is stopped to prevent air binding during the transfer from the UST to the Hotwell.

_____ 9.4.8 Ensure the TDEFDW Pump is <u>NOT</u> operating.

_____ 9.4.9 Dispatch an operator to 1C-157 (TD EFDWP Suction from UST) to standby until further notice. (T-1/C-20)

NOTE 9.4.10:	During the time 1C-391 (TD EFDWP SUCT FROM HOTWELL) and 1C-157 (TDEFDWP Suction from UST) are <u>both</u> open, water will drain from the UST to the Hotwell.

_____ 9.4.10 Open 1C-391 (TDEFDWP SUCT FROM HOTWELL).

_____ 9.4.11 Locally close 1C-157 (TD EFDWP Suction from UST). (T-1/C-20)

_____ 9.5 **<u>IF</u>** power is <u>NOT</u> available,

<u>THEN</u> perform the following:

_____ 9.5.1 Dispatch an operator <u>with a safety harness</u> to open 1V-186 (VACUUM BREAKER). (T-3, catwalk at 1C2 waterbox)

_____ 9.5.2 Dispatch an operator <u>with a safety harness</u> to standby until further notice near 1C-573 (MD EFDWPS Suction from UST). (T-1/E-24 South West, 8' above floor)

Extended EFDW Operation

_____ 9.5.3 Monitor and adhere to the following flow limits:

- MDEFDW Pump flow rates < 440 gpm/pump (0.22 x 10^6 lbm/hr)
- TDEFDW Pump flow rate < 1000 gpm total (0.5 x 10^6 lbm/hr).

_____ 9.5.4 **IF** **AT ANY TIME** UST level ≤ 1 foot,

AND 1C-573 (MD EFDWPS Suction from UST) is open,

THEN perform the following:

_____ 9.5.4.1 Stop all Emergency FDW Pumps.

_____ 9.5.4.2 **IF** vacuum **CANNOT** be broken,

THEN **GO TO** Step 10.

_____ 9.5.5 **WHEN** vacuum is broken,

THEN locally close 1C-573 (MD EFDWPS Suction from UST).

_____ 9.5.6 **IF** MDEFDW Pumps stopped in Step 9.5.4,

THEN restart MDEFDW Pumps.

NOTE 9.5.7:	The TDEFDW Pump is stopped to prevent air binding during the transfer from the UST to the Hotwell.

_____ 9.5.7 Ensure the TDEFDW Pump is NOT operating.

Extended EFDW Operation

_____ 9.5.8 Dispatch an operator to 1C-391 (TD EFDWP SUCT FROM HOTWELL) and 1C-157 (TD EFDWP Suction From UST). (T-1/C-20)

NOTE 9.5.9:	During the time 1C-391 (TD EFDWP SUCT FROM HOTWELL) and 1C-157 (TD EFDWP Suction from UST) are both open, water will drain from the UST to the Hotwell.

_____ 9.5.9 Locally open 1C-391 (TD EFDWP SUCT FROM HOTWELL). (T-1/C-20)

_____ 9.5.10 Locally close 1C-157 (TD EFDWP Suction from UST). (T-1/C-20)

9.5.11 Dispatch an operator to close the following valves:

_____ 1MS-49 (1A CSAE Steam Supply) (T-3/F-26)

_____ 1MS-58 (1B CSAE Steam Supply) (T-3/G-26)

_____ 1MS-67 (1C CSAE Steam Supply). (T-3/H-26)

Extended EFDW Operation

_____ 9.6 **IF** TDEFDW Pump operation is desired,

THEN perform the following:

_____ 9.6.1 **WHEN** suction alignment to the Hotwell is complete,

THEN start the TDEFDW Pump.

_____ 9.7 Monitor Emergency FDW Pump parameters:

- MDEFDW Pump flow rates < 440 gpm/pump (0.22 x 10^6 lbm/hr)
- TDEFDW Pump flow rate < 1000 gpm total (0.5 x 10^6 lbm/hr)
- Emergency FDW Pump discharge pressure > SG pressure.

_____ 9.8 Dispatch an operator to open 1C-188 (Hotwell Emerg Makeup #1 Control Bypass). (T-1, West of E-24) {3}

_____ 9.9 Notify the TSC to evaluate methods to maintain secondary inventory.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM CRO-95/SIM

Restore RBCUs to normal after an inadvertent ES Channel 5 actuation.

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

Restore RBCUs to normal after an inadvertent ES Channel 5 actuation.

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

022 A4.01 (3.6 / 3.6)

Task Standard:

The 1A and 1B RBCU are stopped. The 1A RBCU is restarted in HIGH speed.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

AP/1/A/1104/015, Reactor Building Cooling

Validation Time: 10 min. **Time Critical:** NO

===

Candidate: ____________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ____________________________ __________________________________ / ________
NAME SIGNATURE DATE

===

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall SNAP # ________
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/1/A/1104/015

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 BOP

An inadvertent actuation of ES Channel 5 has occurred on Unit 1.

1CC-7, 1LPSW-6, 1LPSW-15 and the "A" PR Ventilation fan have been return to their normal condition.

INITIATING CUES:

The SRO in the Control Room directs you to return the RBCUs to their normal alignment.

START TIME: _______

STEP 1: Obtain a copy of the appropriate procedure OP/1/A/1104/015, RBC System. STANDARD: Operator obtains a copy of OP/1/A/1104/015, RBC System. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Equipment on ES Channel 5 is taken to MANUAL. STANDARD: The MANUAL push button is depressed for the following equipment on ES Channel 5. • 1LPSW-565 (not critical) • 1LPSW-566 (not critical) • 1A RBCU • 1B RBCU • 1LPSW-18 (not critical) • 1LPSW-21 (not critical) COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: The 1A and 1B RBCUs are stopped to prevent RBCUs from running in mixed speed. ***CUE: Inform candidate that 30 minutes has passed.*** STANDARD: The 1A RBCU switch is place in the OFF position and then the ES reset push button is depressed. The 1A RBCU is verified off. The 1B RBCU switch is verified in the OFF position and then the ES reset push button is depressed. The 1B RBCU is verified off. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 4: The 1A RBCU is started in HIGH speed. STANDARD: The candidate rotates the 1A RBCU switch to the HIGH position and verifies that the RBCU goes to high speed. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5: Open 1 LPSW 565 STANDARD: Locates the switch and goes to open. Verifies that 1LPSW-565 opens. COMMENTS:	___ SAT ___ UNSAT
STEP 6: Throttles 1LPSW18 and 1LPSW-21 STANDARD: 1LPSW-18 and 1LPSW-21 are throttled so that ≈ 1400 gpm of LPSW is achieved to both the 1A and 1B RBCUs. COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANTIONS:

STEP #	Explanation
2	This step is required, because components must be placed in MANUAL to be able to reposition them.
3	This step is required, because the RBCUs should not be run in a mixed speed configuration.
4	This step is required, because HIGH speed is the normal alignment for the RBCUs.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the unit 1 BOP

An inadvertent actuation of ES Channel 5 has occurred has occurred on Unit 1.

1CC-7, 1LPSW-6, 1LPSW-15 and the "A" PR Ventilation fan have been return to their normal condition.

INITIATING CUES:

The SRO in the Control Room directs you return the RBCUs to their normal alignment.

(R06-97)

SR
SUM
NRC
106
115
JPP/JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No OP/1/A/1104/015
Revision No 020

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Reactor Building Cooling System

(4) Prepared By Dennis L. Masteller (Signature) Dennis L. Masteller Date 03/07/00

(5) Requires 10CFR50.59 evaluation?
☐ Yes (New procedure or revision with major changes)
X No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By George Ridgeway (QR) Date 3/9/00
Cross-Disciplinary Review By ______ (QR)NA GAR Date ______
Reactivity Mgmt. Review By ______ (QR)NA GAR Date ______

(7) Additional Reviews
Reviewed By [signature] Date 3/9/00
Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*
By ______ (SRO/QR) Date ______
By ______ (QR) Date ______

(9) Approved By DBCoyle Date 3/13/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______
Compared with Control Copy ______ Date ______
Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______
Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification:
☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?
☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Reactor Building Cooling System** **Multiple Use**	Procedure No. OP/1/A/1104/015 Revision No. 020 Electronic Reference No. OX002VM0

Reactor Building Cooling System

1. Purpose

To describe the proper method for operating Reactor Building Cooling System.

2. Limits And Precautions

2.1 RBCUs should NOT be started, stopped, or speed changed to equalize fan run times.

NOTE:	
	• A step of "OFF" to "LOW" (0-600 rpm) is considered one start.
	• A step of "OFF" to "HIGH" (0-600-1200 rpm) is considered one start.

2.2 RBCU motor shall be off for 30 minutes, or allowed to operate for 30 minutes, prior to starting or changing speed.

2.3 30 minute speed change time interval may be waived in emergencies.

2.4 Manual speed changes should be minimized where possible.

2.5 During non-emergency operation, maximum RBCU motor bearing temperature: 220°F. (computer point: RBV CLR FAN IB/OB BRG TEMP).

2.6 1B RBCU may be operated while LPSW is diverted to Aux Fan Coolers.

2.7 Do NOT operate RBCUs in mixed speed combinations. Excess back pressure is placed on low speed fans.

2.8 Proper damper operation is NOT required for RBCU operability per Improved Technical Specifications (ITS).

- If dampers are NOT operating properly, high vibration and temperature problems may be encountered. {1}

2.9 When Reactor Building Cooling System Operability is required (TS 3.6.5, MODES 1, 2, 3, and 4), LPSW flow to all RBCUs must be ≥ 550 gpm. {2}

- If LPSW to an RBCU is < 550 gpm, LCO 3.0.3 applies. {2}

3. Enclosures

3.1 RBCU System Startup

3.2 RBCU Operation

3.3 Reduction Of Cooling Capacity

3.4 Valve Alignment For Temporary RB Chiller During Unit Outage

* Appendix *

Information Use

1. Initial Conditions

1.1 All RBCUs shutdown.

1.2 Review Limits and Precautions.

2. Procedure

2.1 Start desired number of RBCUs by selecting "HIGH" or "LOW":

- 1A RBCU
- 1B RBCU
- 1C RBCU

NOTE:	• 1LPSW-21 should remain open if RB Aux Fans require LPSW. • Do **NOT** throttle RBCU LPSW flow < 550 gpm. {2}

2.2 Position RBCU valves as required for RB cooling:

- 1LPSW-18 (1A RBCU OUTLET)
- 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)
- 1LPSW-565 (RB AUX FANS COOLERS INLET)
- 1LPSW-24 (1C RBCU OUTLET)
- 1LPSW-566 (1B RBCU ISOLATION)

2.3 Verify RBCU Damper on shutdown fan(s) closed.

2.4 Verify RBCU Dampers on operating fan(s) open.

NOTE:	• If 1B RBCU paired with 1A or 1C RBCU, light closest to paired fans should be lit. • For all other combinations both 1A and 1C lights should be off.

2.5 Verify 1B RBCU Dampers positioned properly.

Information Use

1. Initial Conditions

1.1 RBCU(s) operating.

1.2 Review Limits and Precautions.

2. Procedure

2.1 To stop RBCU(s), place desired switch to "OFF":

- 1A RBCU
- 1B RBCU
- 1C RBCU

2.2 To start RBCU(s), place desired switch to "HIGH" or "LOW":

- 1A RBCU
- 1B RBCU
- 1C RBCU

NOTE:	
	• 1LPSW-21 should remain open if RB Aux Fans require LPSW.
	• Do **NOT** throttle RBCU LPSW flow < 550 gpm. {2}

2.3 Position RBCU valves as required for RB cooling:

- 1LPSW-18 (1A RBCU OUTLET)
- 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)
- 1LPSW-565 (RB AUX FANS COOLERS INLET)
- 1LPSW-24 (1C RBCU OUTLET)
- 1LPSW-566 (1B RBCU ISOLATION)

2.4 Verify RBCU Damper on shutdown fan(s) closed.

2.5 Verify RBCU Dampers on operating fan(s) open.

NOTE:	• If 1B RBCU paired with 1A or 1C RBCU, light closest to paired fans should be lit. • For all other combinations both 1A and 1C lights should be off.

2.6 Verify 1B RBCU Dampers positioned properly.

Information Use

1. Initial Conditions

1.1 Reduction of cooling to RxV and SG cavities desired.

1.2 Review Limits and Precautions.

2. Procedure

NOTE:	• 1LPSW-21 should remain open if RB Aux Fans require LPSW.
	• Do **NOT** throttle RBCU LPSW flow < 550 gpm. {2}

2.1 Place control switch of operating fan(s) in "OFF" or "LOW" as required:

- 1A RBCU
- 1B RBCU
- 1C RBCU

Continuous Use

1. Initial Conditions

NOTE:	1B RBCU and RB Aux Fans will be available for Loss of Decay Heat Removal heat sink.

_____ SRO _____ 1.1 Unit is in MODE 5 or below. {2}

_____ 1.2 Temporary RB Chiller installed per Maintenance procedure.

_____ 1.3 Review Limits and Precautions.

2. Procedure

_____ 2.1 Establish communications with Maintenance personnel at Chiller.

NOTE:	1B RBCU and RB Aux Fans supplied by non-safety related cooling water source.

_____ 2.2 Unlock and White Tag closed 1LPSW-82 (RBCU 1B Outlet Block). (A-4-E Pen Rm)

_____ 2.3 IF desired to place Chilled Water through Aux Fans, ensure Open 1LPSW-565 (RB AUX FAN COOLER INLET).

_____ 2.4 IF desired to place Chilled Water through 1B RBCU, ensure Open 1LPSW-566 (1B RBCU ISOLATION).

2.5 Isolate 1B RBCU:

2.5.1 Unlock and White Tag closed the following valves: (A-4-E Pen Rm)

- _____ 1LPSW-79 (RBCU 1A & 1B Supply Tie)
- _____ 1LPSW-80 (RBCU 1B & 1C Supply Tie)

2.5.2 Close the following valves:

- _____ 1LPSW-19 (1B & AUX FAN COOLER RBCU INLET)
- _____ 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)

NOTE: 1LPSW-878 opened per maintenance procedure.

2.6 Ensure open and White Tag open the following valves: (A-4-E Pen Rm)

_____ • 1LPSW-878 (RBCU Chilled Water Supply)

_____ • 1LPSW-879 (RBCU Chilled Water Return)

_____ 2.7 Notify Maintenance to start Temporary RB Chiller.

Maintenance ________________________________

2.8 Run 1B RBCU Fan as required.

2.9 WHEN changing chilled water flow paths, notify Maintenance to monitor Temporary RB Chiller operation.

2.10 WHEN notified by Maintenance, perform the following: (A-4 E Pen Rm)

_____ 2.10.1 Verify Temporary RB Chiller shutdown.

_____ 2.10.2 Remove White Tag and Close 1LPSW-878 (RBCU Chilled Water Supply).

_____ 2.10.3 Remove White Tag and Close 1LPSW-879 (RBCU Chilled Water Return).

2.11 Remove White Tags from the following: (A-4-E Pen Rm)

_____ • 1LPSW-79 (RBCU 1A & 1B Supply Tie)

_____ • 1LPSW-80 (RBCU 1B & 1C Supply Tie)

_____ • 1LPSW-82 (RBCU 1B Outlet Block)

2.12 Prior to entering MODE 4, return 1B RBCU to service as follows:

2.12.1 Open and lock open the following: (A-4-E Pen Rm)

_____ • 1LPSW-79 (RBCU 1A & 1B Supply Tie)

_____ • 1LPSW-80 (RBCU 1B & 1C Supply Tie)

_____ • 1LPSW-82 (RBCU 1B Outlet Block)

_____ 2.12.2 Open 1LPSW-19 (1B RBCU & AUX FAN COOLER INLET).

_____ 2.12.3 Position 1LPSW-566 (1B RBCU ISOLATION) as required.

_____ 2.12.4 Position 1LPSW-565 (RB AUX FAN COOLER INLET) as required.

_____ 2.12.5 Position 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET) as required.

Appendix

1. Limit and Precaution added in response to PIP 2-O90-0080. Isolation dampers, located below the fusible patches, serve no safety function. Since ductwork below coils/patches is assumed to be "crimped" after a LOCA blowdown (which closes the duct), misalignment of dampers has no effect on RBCU's expected post-accident response, which is to shift to low speed, open discharge valve, and drop patches. Therefore, an alignment problem with isolation dampers **CANNOT** make RBCUs inoperable from a nuclear safety standpoint.

2. Added in response to PIP 2-O98-3629. RBCU LPSW flow ≥ 550 gpm is required when RCS ≥ 200°F and ≥ 300 psig.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM CRO-009/SIM

FOLLOWING KEOWEE EMERGENCY START, TRANSFER MFB POWER FROM CT-4 TO CT-5

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

FOLLOWING KEOWEE EMERGENCY START, TRANSFER MFB POWER FROM CT-4 TO CT-5

Alternate Path:

Facility JPM #:

N/A

K/A Rating(s):

062-A4.01 3.3/3.1

Task Standard:

Auxiliary power is swapped from CT-4 to CT-5.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

OP/0/A/1106/19 Encl. 3.12

Validation Time: 10 min. **Time Critical:** NO

==

Candidate: ______________________________ Time Start: ________
NAME Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ________________________ ____________________________ / ________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC # SNAP ______
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/0/A/1106/19 Encl. 3.12

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

CT-1 is out of service for repairs. A switchyard isolation has resulted in a reactor trip and Unit 1's Main Feeder Busses are being supplied from CT-4 via the Standby Busses. Keowee personnel have requested that the Keowee units be shutdown. CT-5 has been energized from a Lee Gas Turbine and the dedicated path, bypassing the Central switchyard, has been established. OP/0/A/1106/19, Keowee Hydro at Oconee, Enclosure 3.12 has been completed up to step 2.1.4.

INITIATING CUE:

The Control Room SRO directs you to utilize Enclosure 3.12 of OP/0/A/1106/19, Keowee Hydro at Oconee, to transfer MFB power from CT-4 to CT-5 beginning at step 2.1.4.

START TIME: _______

STEP 1: 2.1.4 Place the following transfer switches in MANUAL: • CT-4 BUS 1 AUTO/MAN • CT-4 BUS 2 AUTO/MAN • CT-5 BUS 1 AUTO/MAN • CT-5 BUS 2 AUTO/MAN STANDARD: The following transfer switches are placed in the MANUAL position: • CT-4 BUS 1 AUTO/MAN • CT-4 BUS 2 AUTO/MAN • CT-5 BUS 1 AUTO/MAN **Not Critical** • CT-5 BUS 2 AUTO/MAN **Not Critical** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 2: 2.1.5 Open SK 1 (CT-4 Stby Bus 1 Feeder). STANDARD: SK 1 (CT-4 Stby Bus 1 Feeder) is OPENED. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 3: 2.1.6 Energize the STBY BUSES from CT-5. STANDARD: The following breakers are operated in the listed sequence: SK 2 (CT-4 Stby Bus 2 Fdr) is OPENED. SL 1 (CT-5 Stby Bus 1 Fdr) is CLOSED. SL 2 (CT-5 Stby Bus 2 Fdr) is CLOSED. **NOTE: The time period between opening SK2 and closing SL1 should be > 3 seconds and < 20 seconds.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: 2.1.7 Return the following transfer switches to AUTO: • CT-4 BUS 1 AUTO/MAN • CT-4 BUS 2 AUTO/MAN • CT-5 BUS 1 AUTO/MAN • CT-5 BUS 2 AUTO/MAN STANDARD: The following transfer switches are placed in the AUTO position: • CT-4 BUS 1 AUTO/MAN • CT-4 BUS 2 AUTO/MAN • CT-5 BUS 1 AUTO/MAN • CT-5 BUS 2 AUTO/MAN COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	Step is required, because transfer switches have to be in MANUAL for the breakers to be operated.
2	Step is required, because the SK breaker must be open for the SL breaker to be closed.
3	Step is required, because this is the proper sequence to provide power from CT-5.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

CT-1 is out of service for repairs. A switchyard isolation has resulted in a reactor trip and Unit 1's Main Feeder Busses are being supplied from CT-4 via the Standby Busses. Keowee personnel have requested that the Keowee units be shutdown. CT-5 has been energized from a Lee Gas Turbine and the dedicated path, bypassing the Central switchyard, has been established. OP/0/A/1106/19, Keowee Hydro at Oconee, Enclosure 3.12 has been completed up to step 2.1.4.

INITIATING CUE:

The Control Room SRO directs you to utilize Enclosure 3.12 of OP/0/A/1106/19, Keowee Hydro at Oconee, to transfer MFB power from CT-4 to CT-5 beginning at step 2.1.4.

1. Initial Conditions

_____ 1.1 Keowee Units have been started by emergency actuation and it is desired to shut down the Keowee Units.

_____ 1.2 It is desired to supply power from CT 5.

_____ 1.3 Review Limits and Precautions.

2. Procedure

2.1 Perform a Dead Bus transfer to CT5 from CT4 while CT4 is <u>supplying</u> **Unit 1, 2,** <u>**OR**</u> **3** MFB by:

_____ _____ 2.1.1 Verify CT 5 is energized and ready to power auxiliary loads.

2.1.2 Prior to performing Dead Bus transfer, notify the following:

- _____ Security Force
- _____ Chemistry Department
- _____ Group Heads
- _____ Keowee Operator

_____ 2.1.3 Verify reset <u>**OR**</u> reset MFB Monitor Panel for any Oconee Units receiving power from the STBY Buses.

2.1.4 Place the following transfer switches in "MANUAL":

- _____ _____ CT 4 BUS 1 AUTO/MAN
- _____ _____ CT 4 BUS 2 AUTO/MAN
- _____ _____ CT 5 BUS 1 AUTO/MAN
- _____ _____ CT 5 BUS 2 AUTO/MAN

_____ _____ 2.1.5 Open SK 1 CT 4 STANDBY BUS 1 FEEDER.

CAUTION: Transfer should be made in > 3 but < 20 seconds to prevent picking up MFB Monitor Panel actuation which will cause a Load Shed, Keowee Emergency start and possible EPSL actuation. Undervoltage relays will cause a loss of most non-safety loads.

2.1.6 Energize STBY BUSES from CT 5 by performing the following:

_____ _____ A. Open SK 2 CT 4 STBY BUS 2 FEEDER.

_____ _____ B. Close SL-1 CT 5 STBY BUS 1 FEEDER.

_____ _____ C. Close SL-2 CT 5 STBY BUS 2 FEEDER.

2.1.7 Return the following Transfer Switches to "AUTO":

_____ _____ • CT4 BUS 1 AUTO/MAN

_____ _____ • CT4 BUS 2 AUTO/MAN

_____ _____ • CT5 BUS 1 AUTO/MAN

_____ _____ • CT5 BUS 2 AUTO/MAN

_____ 2.1.8 Recover any loads lost in transfer.

> **NOTE:** __IF__ Keowee Unit(s) are generating with Overhead ACB closed prior to an Emergency Start Actuation, that Keowee Unit(s) will shutdown when ES Channel has been reset unless ACB is currently closed.

2.2 When all three units no longer require an energized Underground Power Path and a Normal Lockout does __NOT__ exist on either Keowee Unit supplying power to an Oconee Unit, completely shut down the Keowee Unit tied to the Underground by:

2.2.1 __IF__ ES 1 __OR__ 2 has actuated, either reset ES 1 and 2 channels __OR__ press "MANUAL" on the following ES 1 and 2 modules:

_____ _____ • Keowee Emer Start Ch A

_____ _____ • Keowee Emer Start Ch B

_____ _____ • Load Shed and STBY Bkr 1

_____ _____ • Load Shed and STBY Bkr 2

2.2.2 __IF__ a manual Keowee Emergency start has been performed from any Oconee Unit, return both Keowee Emergency Start Channel switches on the affected unit to "OFF" position.

_____ _____ • Keowee Emergency Start Channel A

_____ _____ • Keowee Emergency Start Channel B

_____ _____ 2.2.3 Reset __OR__ verify reset Main Feeder Bus Monitor Panels.

2.2.4 Reset External Grid Trouble Protection System by depressing the following buttons. **(Unit 1/2)**:

_____ _____ • GRID TROUBLE PROTECTIVE SYSTEM U.V. CHANNEL 1 RESET

_____ _____ • GRID TROUBLE PROTECTIVE SYSTEM U.V. CHANNEL 2 RESET

_____ _____ • GRID TROUBLE PROTECTIVE SYSTEM U.F. CHANNEL 1 RESET

_____ _____ • GRID TROUBLE PROTECTIVE SYSTEM U.F. CHANNEL 2 RESET

2.2.5 Verify External Grid Trouble Protection has been reset. **(Unit 1/2)**:

_____ • SA-15, A-2 Channel #1 Underfrequency

_____ • SA-15, A-4 Channel #2 Underfrequency

_____ • SA-15, C-1 Channel #1 Undervoltage

_____ • SA-15, C-3 Channel #2 Undervoltage

NOTE:	External Grid Trouble Protection System actuates Keowee Emergency Start from Oconee Unit 1 circuitry.

2.2.6 Depress Keowee "PUSH TO RET TO NORMAL AFT ES RESET" pushbutton on ALL Oconee Units which have generated a Keowee Emergency Start signal:

A. **Unit 1**

_____ _____ • KEOWEE LOGIC RESET CHANNEL 1

_____ _____ • KEOWEE LOGIC RESET CHANNEL 2

B. **Unit 2**

_____ _____ • KEOWEE LOGIC RESET CHANNEL 1

_____ _____ • KEOWEE LOGIC RESET CHANNEL 2

C. **Unit 3**

_____ _____ • KEOWEE ES CHANNEL A

_____ _____ • KEOWEE ES CHANNEL B

_____ 2.2.7 Notify Keowee Operator to shutdown the Keowee Hydro Unit(s) per OP/0/A/2000/041 (Keowee - Modes of Operation).

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM NRC-1998
(ALTERNATE PATH)

RIA-57 Operability Check (RIA-57 fails to meet acceptance criteria following maintenance)

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:
RIA-57 Surveillance
Operation of RIA 57

Alternate Path:
RIA-57 Area Monitor Fault alarm received when performing RIA-57 operability check per PT/230/01

Facility JPM #:
JPM NRC-1998
Used during the 1998 NRC exam as a portion of the Admin exam
Not an assigned to the Oconee JPM bank

K/A Rating(s):
RO/SRO - 072A2.02 [2.8/2.9], RIA Detector Failure and 2.2.12 [3.0/3.4], RIA-57 Surveillance
SRO - 2.1.12 [2.9/4.0], Ability to apply TS for a system

Task Standard:
1. RIA-57 declared inoperable and work request initiated to have I&E correct the problem.
2. R.P. notified that RIA-57 is inoperable.
3. SRO ONLY - T.S. 3.3.8 is referred to for LCO guidelines.

Preferred Evaluation Location:
Simulator __X___ In-Plant __X___

Preferred Evaluation Method:
Perform _____ Simulate __X___

References:
PT/0/A/0230/001

Validation Time: __10-15 minutes_____ **Time Critical:** NO

==

Candidate: ______________________________ Time Start:: _______
NAME Time Finish: _______

Performance Rating: SAT _____ UNSAT _____ Question Grade ____ Performance Time _____

Examiner: ______________________ ______________________________ / _______
NAME SIGNATURE DATE

==

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. RECALL ______
2. IMPORT Files for CRO-SCM+RIA
3. T1 = Area Monitor Fault S/A
4. T2 = Area Monitor Fault clear
5. At the Examiner cue actuate Timer #1 and 2

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:
PT/0/A/0230/001, and Technical Specifications

READ TO OPERATOR

DIRECTION TO TRAINEE:

When I tell you to begin, you are to check the operability of RIA-57 at the individual monitor. Before you start, I will describe the general plant conditions, state the initiating cues, and answer any questions. Perform procedure steps and make notifications as if you were actually performing the task.

INITIAL CONDITIONS:

Unit is at 100% power

RIA-57 has just been returned to service following detector replacement.

INITIATING CUES:

The Control Room Supervisor directs you to check the operation of RIA-57 from the individual monitor per PT/0/A/0230/001.

START TIME:________

STEP	CRITICAL STEP
STEP 1: VERIFY alarm setpoints. STANDARD: • Locate_RIA-57 Individual Monitor on VB2/3 From the key pad: • Depress "Clear" **CUE: Clear depressed** • "Enter" the number 009 **CUE: 009 depressed** • Depress "Item." **CUE: This will display the High alarm setpoint of :** **5.9 E+4 RAD/hr.** • **Depress "+" CUE: + depressed and 010 indicated in the window** • Depress "item" **CUE: this will display the Alert alarm setpoint of** **5.9 E+3 RAD/hr** COMMENTS:	**CRITICAL STEP** ____ SAT ____ UNSAT
STEP 2: Return to normal. STANDARD: • Select "Clear" **CUE: Clear depressed** • Depress R/hr button **CUE: R/hr depressed. (Display returns to actual indication)** COMMENTS:	____ SAT ____ UNSAT

STEP 3*: Perform Check source. • Depress "C/S" button. **CUE: C/S depressed** • Check readings between (Units 1 and 3, 5 E-1 and 1.0 E+0; Unit 2, 1 E+0 and 1.5) E+0 ***CUE: Reading indicates 1.5 E+1*** • Check "Area Monitor Fault" alarm NOT in If in Control Room – ***CUE: Area Monitor Fault" alarms in the simulator or if in the control room, Tell the candidate the alarm has alarmed (SA-8, A-10, AREA MONITOR FAULT.*** • Refers to Operability Criteria for RIA-57 Enclosure 13.3 STANDARD: • Depress "C/S" button. • Understand that the RIA reading at 1.5E+1 is not normal • Understand the "Area Monitor Fault" alarm is not a normal response and refer to the Alarm Response Guide (ARG) for SA-8, A10...Issue WR for I&E • Refers to Operability Criteria for RIA-57 Enclosure 13.3 and determines that RIA-57 is NOT operable. COMMENTS:	**CRITICAL STEP** ____ SAT ____ UNSAT

STEP 4*: INITIATE Enclosure 13.3 corrective actions. 1. Initiate work request to have I&E correct the problem. ***CUE: Work request has been initiated.*** 2. Notify R.P RIA-57 is inoperable. ***CUE: R.P. has been notified.*** 3. List RIA-57 on the Control Room Shift Turnover sheet in the T.S. section. ***CUE: RIA-57 listed on TS Turnover Sheet*** 4. Refer to T.S. 3.3.8 for LCO guidelines ***CUE: If RO candidate, advise him that another operator will refer to the TS. IF SRO CANDIDATE, he should refer to the ITS for proper LCO determination.*** STANDARD: 1. Initiate work request to have I&E correct the problem. 2. Notify R.P that RIA-57 is inoperable. 3. List RIA-57 on the Control Room Shift Turnover sheet in the T.S. section. 4. **SRO ONLY -** Refer to T.S. 3.3.8 for LCO guidelines. Table 3.3.8-1 Accident Monitoring Inst. Item # 9, 2 of 2 channels required => 1 OOS, Condition A - restore w/in 30 days. COMMENTS:	____ SAT ____ UNSAT **CRITICAL STEP** **CRITICAL STEP SRO ONLY**

TIME STOPPED: ______________________

CRITICAL STEP EXPLANATIONS

STEP #	Explanation
1	This step is critical to ensure the monitor will alarm at preset accident values.
3	This step is critical to ensure detector operability.
4 (3)	Step is required to ensure that other operators on the unit are aware of the status with RIA-57.
4 (4)	SRO ONLY, Step is required to determine TS requirements.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

RO and SRO

INITIAL CONDITIONS:

RIA-57 has just been returned to service following detector replacement.

INITIATING CUES:

The Control Room Supervisor directs you to check the operation of RIA-57 from the individual monitor per PT/0/A/0230/001.

1.0 From Individual Monitors

1.1 Verify alarm setpoints by this sequence:

- Depress "Clear"
- "Enter" the number 009
- Depress "Item." This will display the High alarm setpoint.
- Depress "+" (This will display the numbers 010)
- Depress "Item." This will display the Alert alarm setpoint.

1.2 Select "Clear." To return to normal, depress R/hr button.

1.3 To perform Checksource for 1, 3RIA-57, and 1, 2, 3RIA-58, depress "C/S" button. Readings should remain between 5.0 E-1 and 1.0 E+0 and **NOT** give an "Area Monitor Fault" alarm. **IF** alarm is received, refer to Enclosure "Operability Criteria For RIA-57 and RIA-58".

To perform Checksource for 2RIA-57, depress "C/S" button. Readings should remain between 1 E+0 and 1.5 E+0 and **NOT** give an "Area Monitor Fault" alarm. **IF** alarm is received, refer to Enclosure "Operability Criteria For RIA-57 and RIA-58".

Enclosure 13.3

Operability Criteria For RIA-57 And RIA-58

Purpose: The radiation indication alarm system is operable when it is capable of performing its intended function within the required range.

Criteria: The RIA system is considered to have this capability when:

1. Periodic calibration has been completed.
2. No Area Monitor Fault alarms exists due to RIA-57 and RIA-58.
3. 1,3RIA-57 and 1,2,3RIA-58 indicate a reading of between 5.0 E-1 to 1.0 E+0 R/hr on either the CRT or individual monitors.
4. 2RIA-57 indicates a reading of between 1.0 E+0 to 1.5 E+0 R/hr on either the CRT or individual monitors.

NOTE:	RIA-57 and 58 individual Display Modules have backup power available from the station battery system. The RIA-57 and 58 Monitors are powered from non-loadshed power panels SKJ and SKK.

IF the above conditions cannot be met:

1. Operations shall consider the RIA inoperable and initiate a work request to have I&E correct the problem.
2. Operations shall notify R.P. of the inoperable RIA and list it on the appropriate log sheets.
3. Operations shall refer to TS 3.3.8 for guidelines.

1,2,3 RIA-56

Function: Monitor gaseous effluent from station vent, (High Range Post-Accident).

Setpoint : Alert - Set at 10 RAD/hr

High - Set at 20 RAD/hr

Bases: Reg Guide 1.109

ODCM

NUMARC EALs

A setpoint of 10 RAD/hr is based on the Emergency Action Level from NUMARC EALs that would indicate a Site Area Emergency (500 mRAD/hr to the Whole Body).

A setpoint of 20 RAD/hr is based on the Emergency Action Level from NUMARC EALs that would indicate a General Emergency (1 RAD/hr to the Whole Body).

1,2,3 RIA 57 and 58

Function: Post Accident Containment High Range Monitors

Setpoint: Alert - RIA-57 5.9 E+3 RAD/hr RIA-58 2.6 E+3 RAD/hr

High - RIA-57 5.9 E+4 RAD/hr RIA-58 2.6 E+4 RAD/hr

Bases: Emergency Action Levels, Oconee Nuclear Site

The "Alert" setpoint is based on the dose rate necessary inside the containment building to reach one tenth of the protective action guide levels over a one hour period at the site boundary (1 mile). One tenth of the protective action guide level is 500 mrem thyriod dose (thyroid dose is the limiting pathway). This is also the reading at which a Site Area Emergency should be declared. Setpoint was calculated using annual average meteorology, design basis leak rate, and is for the time period up to 1/2 hour after Unit trip.

The "High" setpoint is based on the dose rate necessary inside the containment building to reach the protective action guide levels over a one hour period at the site boundary (1 mile). The protective action guide level is 5 REM thyroid dose (thyroid dose is the limiting pathway). This is also the reading at which a General Emergency should be declared. Setpoint was calculated using annual average meteorology, design basis leak rate, and is for the time period up to 1/2 hour after Unit trip.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NLO-017/PLANT

ALIGN COOLING WATER TO HIGH PRESSURE INJECTION PUMP MOTOR COOLERS FROM AUX. SERVICE WATER PUMP

CANDIDATE ____

EXAMINER ____

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

ALIGN COOLING WATER TO HIGH PRESSURE INJECTION PUMP MOTOR COOLERS FROM AUX. SERVICE WATER PUMP.

Alternate Path:

N/A

Facility JPM #:

NLO-017

K/A Rating(s):

076 A2.01 3.5/3.7

Task Standard:

Preferred Evaluation Location:

Simulator ____ In-Plant __X__

Preferred Evaluation Method:

Perform ____ Simulate __X__

References:

AP/1/A/1700/07

Validation Time: 16 min. **Time Critical:** NO

===

Candidate: ____________________________
NAME

Time Start: ________
Time Finish: ________

Performance Rating: SAT _____ UNSAT _____ Question Grade _____ Performance Time _____

Examiner: ______________________ ______________________/________
NAME SIGNATURE DATE

===

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

Ensure enough copies of AP/1/A/1700/07 are available in the Simulator file cabinet, since Operators will obtain their own copy of the procedure.

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

A Station Blackout has occurred. The SSF Diesel Generator will not start, rendering the SSF RCMU Pumps inoperable. Standby Bus #1 has subsequently been energized from CT-5. I&E personnel have aligned the 'A' HPIP to the ASWP Switchgear. AP/1,2,3/A/1700/11, (Loss of Power), Enclosure 6.3 of has been completed up to step 2.0.

INITIATING CUES:

AP/1,2,3/A/1700/11, Loss Of Power, directs the operator to align cooling water to the High Pressure Injection Pumps.

The Control Room Supervisor directs you to align cooling water to the High Pressure Injection Pumps on Unit___ (specify unit) per Enclosure 6.3 of AP/1,2,3/A/1700/11, Loss of Power.

START TIME: _______

STEP 1: Ensure closed " AUX. SER. WTR. SWGR 4160 VOLT FDR B1T - UNIT 10" breaker. STANDARD: "AUX. SER. WTR. SWGR. 4160-Volt FDR B1T-UNIT 10" breaker indicates closed on the ASW SWGR 600V LOAD CENTER. Two red CLOSED lights are on. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Ensure closed the "AUX. SER. WTR. SWGR TRANSFORMER" breaker STANDARD: Student verifies the "AUX. SER. WTR SWGR TRANSFORMER" breaker is Closed. **Location: ASW SWGR 600V LC Unit 5** ***CUE: Inform the student that the red CLOSED light is lit and that the green OPEN light is off at the control switch for Aux. Ser. Wtr. Swgr. Xfrmr. Bkr.*** COMMENTS:	___ SAT ___ UNSAT
STEP 3: Rack in "AUXILIARY SERVICE WATER PUMP" breaker at the ASW SWGR 600V LOAD CENTER Unit 6. STANDARD: Student opens shutter, inserts 600v breaker rackout tool, and turns tool clockwise to rack breaker in. ***CUE: After breaker is racked in, inform student that the AUX SERVICE WATER PUMP MTOR breaker green OPEN indicating light is ON.*** COMMENTS: Student is expected to follow (simulate) all applicable safe electrical work practices.	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 4: CLOSE CCW-309 (ASWP Disch Drain). STANDARD: CCW-309 (ASWP Disch Drain) is manually CLOSED. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5 OPEN CCW-99 (ASWP Suction). STANDARD: CCW-99 (ASWP Suction) is manually opened. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6 OPEN CCW-101 (ASWP Disch). STANDARD: CCW-101 (ASWP Disch) is manually opened. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7 OPEN CCW-247 (ASWP Recirc). STANDARD: CCW-247 (ASWP Recirc) is manually OPENED. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8 Vent the Aux. Service Water Pump using CCW-308 (ASWP Vent). STANDARD: CCW-308 ASWP vent is throttled open until water issues and then is then closed. COMMENTS:	___ SAT ___ UNSAT
STEP 9 START the Aux. Service Water Pump Motor. STANDARD: Student locates AUX SERVICE WATER PUMP MOTOR control switch and rotates switch to the CLOSE position. **CUE: After switch is rotated, inform student that the AUX SERVICE WATER PUMP MOTOR breaker red CLOSED indicating light is ON.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 10: VERIFY adequate HPIP motor cooler flow indication locally (>1 gpm) by local flow indication. Location: Aux. Bldg. 1st – HPI Pump Room STANDARD: Student verifies flow located AB-1st HPI pump Room. *Cue: A picture of the "A" HPI Pump Motor Cooler Flow indication may be given the student to be used in explaining how the flow would be verified.* COMMENTS: **For ALARA and time considerations, do not allow the student to enter the HPI Pump Room. Stop him/her at the plan view of the HPI Room and have him/her indicate where the flow would be verified.** **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANTIONS:

STEP #	Explanation
3	Supplies power to the Auxiliary Service Water Pump.
4	Ensures that water is not introduced to the Aux. Bldg. when the ASWP is started.
5	Ensures that a suction supply of water is available to the ASWP.
6	Ensures that water is supplied to the discharge header.
7	Prevents pump damage due to the possibility that low flow conditions may exist.
9	Supplies the HPI Pump Motor Coolers with water.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

A Station Blackout has occurred. The SSF Diesel Generator will not start, rendering the SSF RCMU Pumps inoperable. Standby Bus #1 has subsequently been energized from CT-5. I&E personnel have aligned the 'A' HPIP to the ASWP Switchgear. AP/1,2,3/A/1700/11, (Loss of Power), Enclosure 6.3 of has been completed up to step 2.0.

INITIATING CUES:

AP/1,2,3/A/1700/11, Loss Of Power, directs the operator to align cooling water to the High Pressure Injection Pumps.

The Control Room Supervisor directs you to align cooling water to the High Pressure Injection Pumps on Unit___ (specify unit) per Enclosure 6.3 of AP/1,2,3/A/1700/11, Loss of Power.

Aux Service Water To HPI Pump Motor Coolers

1. Energize Standby Bus #1:

_____ 1.1 **IF** either Keowee Unit is available,

THEN Emergency Start available Keowee Units:

_____ "Keowee Emer Start Channel A"

_____ "Keowee Emer Start Channel B".

_____ 1.1.1 **WHEN** available Keowee Units are running,

THEN perform the following:

_____ 1.1.1.1 Ensure closed ACB3 OR ACB 4.

_____ 1.1.1.2 Place "CT4 BUS 1 AUTO/MANUAL" transfer switch in "MANUAL".

_____ 1.1.1.3 Place "STBY BUS 1 SYNCHRONIZING" switch to "ON".

_____ 1.1.1.4 Close "SK1 CT4 STBY BUS 1 FEEDER".

_____ 1.1.1.5 Verify "STANDBY BUS 1" voltmeter indicates ≈ 4160v.

_____ 1.1.1.6 Place "STBY BUS 1 SYNCHRONIZING" switch to "OFF".

Aux Service Water To
HPI Pump Motor Coolers

_____ 1.2 **IF** NO Keowee Unit is available,

AND CT-5 voltmeter indicates ≈ 4160v,

THEN perform the following:

1.2.1 Place the following "AUTO/MAN" transfer switches in "MANUAL":

_____ "CT4 BUS 1 AUTO/MAN"

_____ "CT4 BUS 2 AUTO/MAN"

_____ "CT5 BUS 1 AUTO/MAN"

_____ "CT5 BUS 2 AUTO/MAN".

1.2.2 Ensure open the following breakers:

_____ "SK1 CT 4 STBY BUS 1 FEEDER"

_____ "$S1_1$ STBY BUS 1 TO MFB 1"

_____ "$S2_1$ STBY BUS 2 TO MFB 2".

CAUTION 1.2.3:	If statalarm "TRANSFORMER CT-5 UNDERVOLTAGE" (SA-16/ C-4) is received, additional loading of CT-5 (if powered from Central Switchyard) may result in an undervoltage trip of breakers SL1 and SL2 if Standby Bus voltage reaches 3890 volts.

_____ 1.2.3 Close "SL1 CT5 STBY BUS 1 FDR".

_____ 1.2.4 Place "CT5 BUS 1 AUTO/MAN" switch in "AUTO".

Aux Service Water To
HPI Pump Motor Coolers

_____ 2. **WHEN** Standby Bus 1 is energized,

THEN perform the following:

_____ 2.1 Ensure closed breaker "AUX. SER. WTR. SWGR. 4160 VOLT FDR. B1T - UNIT 10". (Control switch at ASW SWGR 600V LOAD CENTER Unit 5)

NOTE:	Breaker indication will **NOT** be on unless the Standby Bus is energized.

_____ 2.2 Ensure closed breaker "AUX. SERV. WTR. SWGR. TRANSFORMER". (located at ASW SWGR 600V LOAD CENTER Unit 5)

_____ 2.3 Rack in breaker "AUX. SERVICE WATER PUMP" at the ASW SWGR 600V LOAD CENTER Unit 6.

_____ 2.4 Close CCW-309 (ASWP Disch Drn).

_____ 2.5 Open CCW-99 (ASWP Suct).

_____ 2.6 Open CCW-101 (ASWP Disch).

_____ 2.7 Open CCW-247 (ASWP Recirc).

_____ 2.8 Vent the Aux Service Water Pump using CCW-308 (ASWP VENT).

_____ 3. Start the Aux. Service Water Pump Motor.

_____ 4. Locally ensure adequate HPIP motor cooler flow (> 1 gpm) by local flow indication.

END

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NLO-004

Manually Bypassing the KI/KU Inverter

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:
MANUALLY BYPASS THE KI/KU INVERTER
TASK NUMBER: OO1345501

Alternate Path:

N/A

Facility JPM #:

NLO-004 (MODIFIED)

K/A Rating(s):
System: APE-057 Loss of Vital AC Instrument Bus
K/A: EA1.01
Rating: 3.7/3.7

Task Standard:
KI/KU Inverter is located and bypassed correctly

Preferred Evaluation Location:

Simulator _____ In-Plant __X___

Preferred Evaluation Method:

Perform _____ Simulate __X__

References:

Validation Time: 10 min. **Time Critical:** NO
==

Candidate: ______________________________________ Time Start : _________
NAME Time Finish: _________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________________ ________________________________/________
NAME SIGNATURE DATE
==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

AP/1/A/1700/23, AP/2/A/1700/23, AP/3/A/1700/23 (Enclosure 6.1 Bypass of the KI and KU Inverters)

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit _____ (specify unit) was operating at 100% power when it experienced a loss of ICS Power followed by a unit trip. The subsequent actions of the EOP and the Abnormal Procedure for Loss of ICS Power have been completed up to the point of regaining ICS Power.

INITIATING CUES:

The Control Room Supervisor directs you to manually bypass the KI and KU Inverters on Unit _____ (specify unit) per AP/1,2,3/A/1700/23, Loss of ICS Power, Enclosure 6.1.

START TIME: _______

STEP 1: OPEN KI Inverter Bypass Switch cabinet door. STANDARD: Student locates the KI Inverter Bypass Switch cabinet and opens cabinet door. (Located in Equipment Room) COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 2: MANUALLY BYPASS KI Inverter STANDARD: Inverter is MANUALLY BYPASSED by positioning the following switches in sequence: Sw #1 (left switch) And Sw #3 (right switch) are OPENED Then Sw #2 (center switch) is CLOSED COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 3: OPEN KU Inverter Bypass Switch cabinet door. STANDARD: Student locates the KU Inverter Bypass Switch cabinet and opens cabinet door. (Located in Equipment Room) COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: MANUALLY BYPASS KU Inverter STANDARD: Inverter is MANUALLY BYPASSED by positioning the following switches in sequence: Sw #1 (left switch) And Sw #3 (right switch) are OPENED Then Sw #2 (center switch) is CLOSED STANDARD: COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

<u>STEP 5</u>: Call the Control Room to determine if ICS AUTO and Hand Power have been restored. ***CUE: ICS AUTO <u>has</u> been restored but, Hand Power has <u>NOT</u> been restored*** <u>STANDARD</u>: Locate phone and simulate call to the unit's control room <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT
<u>STEP 6</u>: If ICS HAND power has not been restored. <u>STANDARD</u>: In the unit's Cable Room locate __ KRA power panel, ***CUE: Breaker #13 (175A 2P, Power Panelboard 1KU)) is "loose" and in the "mid" position.*** Reset breaker KRA #13 (175A 2P, Power Panelboard 1KU) ***CUE: Breaker #13 (175A 2P, Power Panelboard 1KU)) is RESET.*** In the unit's Cable Room locate __ KU power panel, ***CUE: Breaker #21 (30A 1P, ICS/NNI Hand Power) is "loose" and in the "mid" position.*** Reset breaker #21 (30A 1P, ICS/NNI Hand Power) ***CUE: Breaker #21 (30A 1P, ICS/NNI Hand Power) is RESET.*** Notify the Control Room ***CUE: BOTH ICS AUTO and HAND power <u>has</u> been restored.*** <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	KI cabinet must be located and the door opened to reach the bypass switches
2	Switch 1, 2, and 3 properly operated to bypass the inverter
3	KU cabinet must be located and the door opened to reach the bypass switches
4	Switch 1, 2, and 3 properly operated to bypass the inverter
5	Communicate with the control to determine that the HAND power has not been restored.
6	Properly reset KU power supplies breakers to restore ICS HAND power

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit _____ (specify unit) was operating at 100% power when it experienced a loss of ICS Power followed by a unit trip. The subsequent actions of the EOP and the Abnormal Procedure for Loss of ICS Power have been completed up to the point of regaining ICS Power.

INITIATING CUES:

The Control Room Supervisor directs you to manually bypass the KI and KU Inverters on Unit _____ (specify unit) per AP/1,2,3/A/1700/23, Loss of ICS Power, Enclosure 6.1.

Enclosure 6.1

Bypass Of The 1KI And 1KU Inverters

1. Perform the following in the Unit 1 Equipment Room:

1.1 Bypass 1KI Inverter by performing the following:

_____ 1.1.1 Open "SW#1" (Left Switch).

_____ 1.1.2 Open "SW#3" (Right Switch).

_____ 1.1.3 Close "SW#2" (Center Switch).

1.2 Bypass 1KU Inverter by performing the following:

_____ 1.2.1 Open "SW#1" (Left Switch).

_____ 1.2.2 Open "SW#3" (Right Switch).

_____ 1.2.3 Close "SW#2" (Center Switch).

_____ 1.3 Call Unit 1 Control Room (ext. 2261, 2159, or 2335) to determine if ICS AUTO and HAND Power have been restored:

- The following statalarms off:
  - "ICS AUTO POWER FAILURE" (1SA-02/B-11)
  - "ICS HAND POWER FAILURE" (1SA-02/B-12).

_____ 2. **IF** ICS AUTO Power has **NOT** been restored,

THEN perform the following in the Unit 1 Cable Room:

2.1 Reset and close the following:

_____ "1KRA breaker #1" (100A 1P, POWER PANELBOARD 1KI)

_____ "1KI breaker #1" (30A 1P, AUTO POWER (ICS)).

_____ 3. **IF** ICS HAND Power has **NOT** been restored,

THEN perform the following in the Unit 1 Cable Room:

3.1 Reset and close the following:

_____ "1KRA breaker #13" (175A 2P, POWER PANELBOARD 1KU)

_____ "1KU breaker #21" (30A 1P, ICS/NNI HAND POWER).

_____ 4. Immediately notify the Unit 1 Control Room (ext. 2261, 2159, or 2335) that all applicable steps of this Enclosure have been completed.

•••END•••

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NLO-041

RESTART THE PRIMARY IA COMPRESSOR FOLLOWING A COMPRESSOR TRIP
(Alternate Path)

CANDIDATE ______________________________

EXAMINER ______________________________

**REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE**

Task:

RESTART THE PRIMARY IA COMPRESSOR FOLLOWING A COMPRESSOR TRIP
TASK NUMBER: OO1333002

Alternate Path:

YES

Facility JPM #:

NLO-041

K/A Rating(s):
System: SF8-078 Instrument Air System
K/A: 2.1.30
Rating: 3.9/3.4

Task Standard:

The Primary IA Compressor is restarted by procedure

Preferred Evaluation Location:

Simulator _____ In-Plant __X__

Preferred Evaluation Method:

Perform _____ Simulate __X__

References:

Enclosure 4.11 of OP/0/A/1106/27

Validation Time: 10 minutes

Time Critical: NO

==

Candidate: ______________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________________ ____________________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:
Enclosure 4.11 of OP/0/A/1106/27

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit 1 is at 100% power.
The Transmission Dept. was performing PM checks on B3T switchgear when the incoming feeder breaker tripped open.
When B3T de-energized, the automatic transfer to the backup source (B4T) did not occur and the Primary IA Compressor tripped.
The RO entered "Loss of IA" AP/1/A/1700/22 as IA header pressure decreased to ≈ 85 psig and has reached step 5.10, which refers the operator to OP/0/A/1106/27 to restore operable IA compressors.

INITIATING CUES:

The SRO in the control room instructs you utilize Enclosure 4.11 restart of the Primary IA Compressor following a Trip of OP/0/A/1106/27 (Compressed Air System) to **RESTART** the Primary IA Compressor. Initial Conditions of the enclosure have been completed.

START TIME: _______

STEP 1: Position the following valves: Close IA-2730 (Primary IA Desiccant Air Filter "A" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2730 (Primary IA "A" Desiccant Filter Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT
STEP 2: Position the following valves: Close IA-2731 (Primary IA Desiccant Air Filter "B" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2731 (Primary IA "B" Desiccant Filter Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT

STEP 3: At the Primary IA Dryer A Control Panel, position the (ON/OFF) switch to OFF. STANDARD: The student REMOVES the Primary IA Dryers from service by rotating the following switches, located on the A Dryer control panels, to the "OFF" position: Primary IA Dryer "A" On/Off selector. COMMENTS:	___ SAT ___ UNSAT
STEP 4: At the Primary IA Dryer B Control Panel, position the (ON/OFF) switch to OFF. STANDARD: The student REMOVES the Primary IA Dryer from service by rotating the following switches, located on the B Dryer control panels, to the "OFF" position: Primary IA Dryer "B" On/Off selector. COMMENTS:	___ SAT ___ UNSAT

<u>STEP 5</u>: Position the following valves: Close IA-2735 (Primary Air Filter "A" Outlet). (TB5 L-39) <u>STANDARD</u>: The student LOCATES and CLOSES IA-2735 (Primary Air Filter "A" Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** <u>COMMENTS</u>:	___ SAT ___ UNSAT
<u>STEP 6</u>: Position the following valves: Close IA-2736 (Primary Air Filter "B" Outlet). (TB5 L-39) <u>STANDARD</u>: The student LOCATES and CLOSES IA-2736 (Primary Air Filter "B" Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** <u>COMMENTS</u>:	___ SAT ___ UNSAT

STEP 7: Open HPSW-771 (Primary IA Comp. Disc. Block) (TB5 M-39) STANDARD: The student LOCATES and OPENS HPSW-771 (Primary IA Compressor Cooling Discharge Block) by rotating the switch to the "Open" position. **NOTE: HPSW-771 control switch and the cooling water inlet pressure gauges are located north of the compressor next to the west Turbine floor wall.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 8: Verify adequate cooling water flow as follows: IF OHPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure) does NOT read between 61 and 67 psig. Backwash HPSW-764 (Primary IA Comp. Disch. Control) (TB5 M-39) per Backwash of Primary IA Compressor HPSW Pressure Regulator enclosure. Verify HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) in Locked Open Position. STANDARD: The student VERIFIES adequate cooling water flow by monitoring the following gauges: - OHPS-PG-0823 (Primary IA Comp. Cooling Water Inlet Pressure). **CUE: Using a pointing device, indicate to the student the following readings:** - **OHPS-PG-0823 = 64 psig.** HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) is verified in the Locked Open Position. COMMENTS:	___ SAT ___ UNSAT

STEP 9: Depress (RESET/LAMP TEST) pushbutton. • Verify all alarm indicators light. • Release (RESET/LAMP TEST) pushbutton and verify all alarm indicator lamps extinguish. 2.7.1 Any alarm condition indicated on the Primary IA Compressor control panel must be resolved before starting the compressor. STANDARD: The student RESETS the Primary IA Compressor by depressing the black "Reset" pushbutton on the compressor control panel located on the north side of the compressor housing. **CUE: While RESET/LAMP TEST pushbutton is depressed, inform student that all alarm indicators are lit. When RESET/LAMP TEST pushbutton is released, inform student that all alarm indicator lamps extinguish.** COMMENTS:	___ SAT ___ UNSAT

<table>
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<u>STEP 10</u>:

- Depress (START) pushbutton.
- Verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is with in the range specified (in procedure).

<u>STANDARD</u>:

- The student STARTS the Primary Air Compressor by depressing the "Start" pushbutton on the control panel located on the north side housing of the compressor.

CUE: Inform the student that the green "Machine Run" light has illuminated.

- The student VERIFIES adequate cooling water flow by monitoring the following gauges:

- OHPS-PG-0823 (Primary IA Comp. Cooling Water Inlet Pressure).
- OHPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure).

CUE: Using a pointing device, indicate to the student the following readings:

- **OHPS-PG-0823 = 64 psig.**
- **OHPS-PG-0824 = 9 psig.**

Student should simulate throttling HPSW-767 (Pri. IA Comp. Disch. Cont.) to achieve the proper flow/outlet pressure range.

When HPSW-767 is throttled closed, indicate with the pointing device that flow is 98 gpm and outlet pressure is 18 psig

<u>COMMENTS</u>:

</td>
<td>

CRITICAL STEP

___ SAT

___ UNSAT

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STEP 11: VERIFY selected Enclosure fan is running and all door panels are installed on compressor enclosure. Verify all door panels are installed on the Primary IA Compressor Enclosure. STANDARD: The student determines that selected enclosure fan is operating and all door panels located on the compressor enclosure are installed. **CUE: Inform the student that the selected Enclosure Fan is running properly. Inform student that all door panels are installed on enclosure.** COMMENTS:	___ SAT ___ UNSAT
STEP 12: Throttle open IA-2735 (Primary Air Filter "A" Outlet) or IA-2736 (Primary Air Filter "B" Outlet) (TB5 L-39) to SLOWLY pressurize the Dryer tanks to system pressure (100-110 psig). STANDARD: The student throttles open one of the following valves to SLOWLY PRESSURIZE the Desiccant Dryers: IA-2735 (Primary Air Filter "A" Outlet) **OR** IA-2736 (Primary Air Filter "B" Outlet) **CUE: Once the student has demonstrated his/her ability to properly throttle the valve, indicate to the student with a pointing device that the Desiccant Dryers have reached 104 psig.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 13: At the Primary IA Dryer A panel, position the (ON/OFF) switch to ON. STANDARD: The student PLACES the Primary IA Dryers in service by positioning the following switches to the "ON" position: Primary IA Dryer "A" On/Off Selector COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 14: At the Primary IA Dryer B panel, position the (ON/OFF) switch to ON. STANDARD: The student PLACES the Primary IA Dryers in service by positioning the following switches to the "ON" position: Primary IA Dryer "B" On/Off Selector COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 15: CONNECT the Primary IA Compressor to the IA Header. Open IA-2735 (Primary Air Filter A Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2735 (Primary Air Filter "A" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 16: CONNECT the Primary IA Compressor to the IA Header. Open IA-2736 (Primary Air Filter B Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2736 (Primary Air Filter "B" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 17: CONNECT the Primary IA Compressor to the IA Header. Slowly open IA 2730 (Primary Desiccant Air Filter A Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2730 (Primary Desiccant Air Filter "A" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 18: CONNECT the Primary IA Compressor to the IA Header. Slowly open IA 2731 (Primary Desiccant Air Filter B Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2731 (Primary Desiccant Air Filter "B" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 19: As system pressure increases check for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters. **NOTE:** Enclosure "Startup Of The Primary IA Compressor" contains a detailed list of expected Primary Air Compressor normal operating parameters. STANDARD: The student checks for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters as system pressure increases. **CUE: No air leaks are found.** Primary Air Compressor monitored for normal operation. COMMENTS:	___ SAT ___ UNSAT

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
7	Open HPSW-771 (Primary IA Comp. Disc. Block) aligns cooling water to the compressor
10	Depress (START) pushbutton starts the compressor, verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is with in the range specified , and establish proper cooling water flow to the compressor
12	Pressurizes and places in service the primary air filter
13	Places the "A" Air Dryer in service
14	Places the "B" Air Dryer in service
15-18	Establishes an air flow path from the compressor to the IA Header

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit 1 is at 100% power.
The Transmission Dept. was performing PM checks on B3T switchgear when the incoming feeder breaker tripped open.
When B3T de-energized, the automatic transfer to the backup source (B4T) did not occur and the Primary IA Compressor tripped.
The RO entered "Loss of IA" AP/1/A/1700/22 as IA header pressure decreased to ≈ 85 psig and has reached step 5.10, which refers the operator to OP/0/A/1106/27 to restore operable IA compressors.

INITIATING CUES:

The SRO in the control room instructs you utilize Enclosure 4.11 restart of the Primary IA Compressor following a Trip of OP/0/A/1106/27 (Compressed Air System) to **RESTART** the Primary IA Compressor.
Initial Conditions of the enclosure have been completed.

Loss of Instrument Air

_____ 5.10 Restore Instrument Air header pressure:

- Isolate IA header leakage.
- Start operable IA compressors.
  - **REFER TO** OP/0/A/1106/027, (Compressed Air System).

_____ 5.11 **IF** efforts to restore IA are **NOT** successful,

AND both Main Feedwater Pumps have tripped,

THEN establish Condensate Recirc:

_____ 5.11.1 Trip all CBPs.

_____ 5.11.2 Trip all but one HWP.

_____ 5.11.3 Open 1C-124 (Condensate Recirc to UST).

_____ 5.11.4 Send an Operator to throttle 1C-129 (Condensate Recirc Control Bypass). (TB5/M-22NE)

_____ 5.11.5 Establish ≈ 2300 gpm Condensate Recirc flow by computer point O1A0156 (CBP DISCH HDR FLOW).

Restart Of The Primary IA Compressor
Following A Trip

1. Initial Conditions

____ 1.1 Worthington IA compressors may or may not be supplying IA header.

____ 1.2 Reason for loss of Primary IA compressor has been corrected.

____ 1.3 System alignment unchanged from time of compressor trip.

2. Procedure

2.1 Position the following valves:

____ 2.1.1 Close IA-2730 (Primary Desiccant Air Filter 'A' Outlet). (TB5 L-39)

____ 2.1.2 Close IA-2731 (Primary Desiccant Air Filter 'B' Outlet). (TB5 L-39)

____ 2.2 At the Primary IA Dryer A Control Panel, position the (ON/OFF) switch to "OFF."

____ 2.3 At the Primary IA Dryer B Control Panel, position the (ON/OFF) switch to "OFF."

2.4 Position the following valves:

____ 2.4.1 Close IA-2735 (Primary Air Filter 'A' Outlet). (TB5 L-39)

____ 2.4.2 Close IA-2736 (Primary Air Filter 'B' Outlet). (TB5 L-39)

____ 2.5 Open HPSW-771 (Primary IA Comp. Disch. Block) (TB5 M-39).

2.6 Verify adequate cooling water flow as follows:

____ 2.6.1 **IF** OHPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure) does **NOT** read between 61 and 67 psig. backwash HPSW-764 (Primary IA Comp. Disch. Control) (TB5 M-39) per Backwash of Primary IA Compressor HPSW Pressure Regulator enclosure.

____ 2.6.2 Verify HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) in Locked Open position.

____ 2.7 Depress (RESET/LAMP TEST) pushbutton.

- Verify all alarm indicators light.
- Release (RESET/LAMP TEST) pushbutton and verify all alarm indicator lamps extinguish.

____ 2.7.1 Any alarm conditions indicated on the Primary IA Compressor control panel must be resolved before starting the compressor.

____ 2.8 Depress (START) pushbutton.

____ 2.8.1 Verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is within the range specified below for the value of 0HPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure).

Cooling Water Inlet Pressure (psig)	Acceptable Range for Cooling Water Outlet Pressure (psig).
61	18 - 10
62	19 - 11
63	20 - 12
64	21 - 12
65	22 - 13
66	23 - 14
67	24 - 14

____ • IF OHPS-PG-0824 is **NOT** within the Acceptable Range For Cooling Water Outlet Pressure for the value of OHPS-PG-0823, throttle HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) to obtain the required cooling water outlet pressure.

____ • IF OHPS-PG-0824 is **NOT** within the Acceptable Range For Cooling Water Outlet Pressure for the value of OHPS-PG-0823, THEN closely monitor Primary IA Compressor Discharge Temperature and Injection Temperature until acceptable Cooling Water pressures can be obtained.

____ 2.9 IF Compressor fails to start, notify WCC SRO for aid in resolving problem.

____ 2.10 Verify selected Enclosure fan is running.

____ 2.11 Verify all door panels are installed on the Primary IA Compressor Enclosure.

CAUTION: The Outlet Filter Valves must be opened in increments very slowly to prevent disintegration of the Dryer desiccant.

____ 2.12 Throttle open IA-2735 (Primary Air Filter 'A' Outlet) (TB5 L-39) OR IA-2736 (Primary Air Filter 'B' Outlet) (TB5 L-39) to SLOWLY pressurize the Dryer Tanks to system pressure (100-110 psig).

____ 2.13 At the Primary IA Dryer A panel, position the (ON/OFF) switch to "ON."

_____ 2.14 At the Primary IA Dryer B Panel, position the (ON/OFF) switch to "ON."

_____ 2.15 Open IA-2735 (Primary Air Filter A Outlet). (TB5 L-39)

_____ 2.16 Open IA-2736 (Primary Air Filter B Outlet). (TB5 L-39)

CAUTION: Rapid opening of the Primary Desiccant Air Filter Outlet Valves can cause collapse of the Desiccant Filters.

_____ 2.17 SLOWLY open IA-2730 (Primary Desiccant Air Filter A Outlet). (TB5 L-39)

_____ 2.18 SLOWLY open IA-2731 (Primary Desiccant Air Filter B Outlet). (TB5 L-39)

_____ 2.19 As system pressure increases check for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters.

NOTE: Enclosure "Startup Of The Primary IA Compressor" contains a detailed list of expected Primary Air Compressor normal operating parameters.

_____ 2.20 Monitor Primary Air Compressor for normal operation.

NOTE: If Primary Air Compressor is operating normally, Backup IA Compressors will be running unloaded.

2.21 Return the Backup IA Compressors to their normal lineup:

_____ 2.21.1 Place the C Backup IA Compressor control switch in "STBY 1."

_____ 2.21.2 Place either the A or B Backup IA Compressor control switch in "STBY 1."

_____ 2.21.3 Place the remaining Backup IA Compressor control switch in "STBY 2."

2.22 Close or verify closed the following valves:

_____ • RCW-25 (Backup IA Compressor A Temp. Control Bypass) (TB1 L-31)

_____ • RCW-31 (Backup IA Compressor B Temp. Control Bypass) (TB1 L-32)

_____ • RCW-37 (Backup IA Compressor C Temp. Control Bypass) (TB1 L-32)

Facility: Oconee
Exam Level: **SRO(U)**

Date of Examination: 7-10/17-00
Operating Test No.: 1

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
a. CRO-12A, Recover a Dropped Control Rod; (20 min.) AP/1/A/1700/15, OP/0/A/1105/09, Encl 4.10 [KA: 005AA2.03 (3.4/4.4)] Note: This JPM conducted with Admin A.1 CRO-40A (Calculate SDM) (SRO ONLY) (5 min.)	D, S, A	1
b. CRO-095, Swap RBCU's (Inadvertent ES actuation); (10 min.) OP/0/A/1104/15 [KA: 022A4.01 (3.6/3.6)]	M, C/S	5
c. CRO-096, Align ECCS Suction from Emergency Sump (LP-20, Emergency Sump Suction failed closed) (9 min.) EP/1/A/1800/01, CP-601; AP/1/A/1700/07 [KA: 002A2.04 (4.3/4.6)] Note: SRO required ECCS (SRO ONLY) (PRA)	N, S, L	2

B.2 Facility Walk-Through

a. NLO-017, Align Cooling Water to HPIP's from Station ASWP; AP/1,2,3/A/1700/11, Encl 6.3; OP/0/A/1102/06, Encl. 3.3 (16 min.) [KA: 076A2.01 (3.5/3.7)]	D, R, L	4S
b. NLO-041, Restart the Primary Instrument Air Compressor following a Compressor trip; (10 min.) OP/0/1106/27, Encl. 4.11; APO/1/A/1700/22 [KA: 078G2.1.30 (3.9/3.4)]	D, A	8

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM CRO-012A/SIM

RECOVERY OF A DROPPED CONTROL ROD

CANDIDATE ____________________

EXAMINER ____________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

RECOVERY OF A DROPPED CONTROL ROD

Alternate Path:

Unit is tripped upon receipt of second dropped CR

Facility JPM #:

CRO-12A

K/A Rating(s):

005-AA2.03 3.5/4.4

Task Standard:

Control Rod recovery
Unit is tripped upon receipt of second dropped CR

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

AP/1/A/1700/15, Dropped Control Rods

Validation Time: 20 min. **Time Critical:** NO

==

Candidate: ________________________________
NAME

Time Start : _________
Time Finish: _________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: _________________________ __________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC # _______
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/0/A/1105/009, Control Rod Drive System

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.6.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

START TIME: _______

	CRITICAL STEP
STEP 1: Take manual control of rods at the Diamond Control Station by performing the following: Place the Diamond Station in MANUAL STANDARD: The AUTO/MANUAL pushbutton on the Diamond Control Panel is depressed, the MANUAL half of the Push Button is backlighted. Location 1UB1 ***Cue: Inform candidate time compression has taken place and the Control Rod has been repaired and should be withdrawn.*** COMMENTS:	___ SAT ___ UNSAT
STEP 2: Obtain copy of Enclosure 4.10, Recovery of Dropped/Misaligned Regulating Control Rod, of OP/0/A/1105/009, Control Rod Drive System. STANDARD: Obtain a copy of OP/0/A/1105/009, Control Rod Drive System and determine that Enclosure 4.10, Recovery of Dropped/Misaligned Regulating Control Rod, is the proper enclosure for this condition and obtain a copy from the procedure file. COMMENTS:	___ SAT ___ UNSAT

STEP 3: Take manual control of rods at the Diamond Control Station by performing the following: Place the SG Master in HAND Place the Diamond Station in MANUAL (**Diamond is already in HAND per step 1**) STANDARD: The manual pushbutton for the SG Master hand/auto station is depressed, The White Hand light comes ON and the Red Auto light Goes OFF. Location 1UB1 COMMENTS:	___ SAT ___ UNSAT
STEP 4: SELECT group with dropped/misaligned rod on the Group Select Switch STANDARD: GROUP SELECT SWITCH on 1UB1 is located by the student and rotated to Group 6. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: Press selector for SEQ OVERRIDE. STANDARD: The SEQ/SEQ OR pushbutton is located on the Diamond Control panel on 1UB1 and depressed. "SEQ OR" is backlighted. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6: Select JOG on the Speed Selector STANDARD: The SPEED Selector is located by the student on the Diamond Control panel on 1UB1 and rotated to the JOG position. COMMENTS:	___ SAT ___ UNSAT

<u>STEP 7</u>: Press selector for LATCH switch and insert group for approximately 15 seconds or until the group OUT LIMIT lamp on the Diamond Panel goes off. Release LATCH switch. <u>STANDARD</u>: The IN LIMIT (LATCH) BYPASS pushbutton is located by the student and depressed and held while the INSERT/WITHDRAW joystick is used to insert Group 6 until the Group 6 Out limit lamp, located on the Diamond Control Panel on 1UB1, extinguishes. The LATCH pushbutton is then released and the INSERT/WITHDRAW joystick returned to neutral. <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: TRANSFER the dropped/misaligned rod to the auxiliary power supply. Select dropped/misaligned rod on the Single Select Switch Press selector for SEQ OVERRIDE Press selector for AUXILIARY Press selector for CLAMP Press selector for MANUAL TRANSFER switch until TRANSFER CONFIRM lamp and the CONTROL ON lamp on the PI panel light Press selector for CLAMP RELEASE STANDARD: On the CRD Panel on 1UB1: SELECT dropped/misaligned rod on the SINGLE SELECT SWITCH. VERIFY SEQ OR is backlit **(Not Critical).** Depresses GROUP/AUXIL pushbutton to make transfer to AUXIL. Verifies SYNC is backlit on MAN TRANS/SY/TR CF pushbutton **(Not Critical)** Depresses CLAMP/CLAMP REL pushbutton to make transfer to CLAMP. CLAMP will be backlit. Depresses MAN TRANS/SY/TR CF pushbutton. TR CF will become backlit. White CONTROL ON lights will illuminate for the Dropped Rod on the Position Indication panel. Depresses CLAMP/CLAMP REL pushbutton and verifies CLAMP REL is backlit. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 9: Perform PI alignment on the dropped/misaligned rod as follows: Press selector for the LATCH switch and insert rod for 15 seconds. Release LATCH switch. Compare absolute and relative readings on the PI panel. Adjust RPI to equal API with POSITION RESET RAISE/LOWER switch. STANDARD: Absolute and relative indications on the PI panel, on 1UB1, are compared using toggle switch to make comparison. RPI is selected with the select toggle switch. The POSITION RESET RAISE/LOWER toggle switch is then placed in the lower position and RPI indication is matched to API position. When matched the RAISE/LOWER toggle is released to neutral. The select toggle switch is returned to the API position. COMMENTS:	___ SAT ___ UNSAT

STEP 10: SELECT RUN on the Speed Selector. STANDARD: SPEED SELECTOR is located by the student on 1UB1 and rotated to the run position. ***CUE: Rod has been misaligned for less than 24 hours.*** COMMENTS:	___ SAT ___ UNSAT
STEP 11: Withdraw dropped/misaligned rod until power begins to increase and then stop withdrawal. STANDARD: Rod is withdrawn while monitoring reactor power for an increase. ***NOTE: When rod is 50% withdrawn, booth operator drop second rod.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 12: Manually trip the reactor STANDARD: The student recognizes the second control rod inserting and manually trips the reactor by depressing the Reactor Trip pushbutton and performs IMAs. COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	The Diamond is taken to manual before repairs on the CR begins.
4	Instructs the rod logic as to which group the rod is in that the operator wants to recover.
5	Allows the operator to withdraw the dropped rod.
7	The latching of the group to clear the out limit is necessary so that the individual rod can be withdrawn.
8	Places the dropped rod on the auxiliary power supply for withdrawal while leaving the group on the group power supply
11	Necessary to withdraw dropped CR.
12	The second dropped rod places the unit in an unanalyzed condition and this is a direction, which is given by OMP 1-18, Operator memory Items.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.6.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods

1R(06-97) WHC/JMB/JPP Trns.

Duke Power Company

PROCEDURE PROCESS RECORD

(1) ID No AP/1/A/1700/015

Revision No 4

LAN Location: SAROS

EPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Dropped Control Rods

(4) Prepared By Dennis Jordan Date 2/17/99

(5) Requires 10CFR50.59 evaluation?
- [x] Yes (New procedure or revision with major changes)
- [ ] No (Revision with minor changes)
- [ ] No (To incorporate previously approved changes)

(6) Reviewed By Walter M. Baker (QR) Date 2/25/99

Cross-Disciplinary Review By ________ (QR)NA WB Date ________

Reactivity Mgmt. Review By Walter M. Baker (QR)NA ______ Date 2/25/99

(7) Additional Reviews

Reviewed By ________ Date ________

Reviewed By ________ Date ________

(8) Temporary Approval *(if necessary)*

By ________ (SRO/QR) Date ________

By ________ (QR) Date ________

(9) Approved By Walter Delegat Date 3/8/99

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ________ Date ________

Compared with Control Copy ________ Date ________

Compared with Control Copy ________ Date ________

(11) Date(s) Performed ________

Work Order Number (WO#) ________

COMPLETION

(12) Procedure Completion Verification

- ☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
- ☐ Yes ☐ NA Listed enclosures attached?
- ☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
- ☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
- ☐ Yes ☐ NA Procedure requirements met?

Verified By ________ Date ________

(13) Procedure Completion Approved ________ Date ________

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Dropped Control Rods** **Continuous Use** **Reactivity Management Related**	Procedure No. AP/1/A/1700/015 Revision No. 004 Electronic Reference No. OX002RGS

DROPPED CONTROL RODS
Reactivity Management Related

TABLE OF CONTENTS

OCONEE NUCLEAR STATION

Dropped Control Rods

1. Purpose

This procedure provides the actions necessary to maintain the plant in a safe condition following a dropped control rod(s).

2. Symptoms

- "CRD POSITION ERROR" statalarm (1SA-2, B-10)
- "CRD SAFETY RODS NOT AT UPPER LIMIT" statalarm (1SA-2, C-10)
- Control Rod "IN-LIMIT" light on (Green light on Position Indication Panel)
- Control Rod "ASYMM. RODS" light on (Yellow light on Diamond).

3. Automatic Systems Actions

3.1 **IF** ICS is in Auto,

AND an "ASYMM. RODS" (Yellow Light on Diamond) occurs,

THEN an "OUT" inhibit at 60% power is established and the Reactor will runback to 55% power.

3.1.1 **IF** the "ASYMM. RODS" (Yellow Light on Diamond) clears,

THEN runback may stop before reaching 55% power.

4. Immediate Manual Actions

_____ 4.1 **IF** more than one Control Rod has dropped,

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 4.2 **IF** more than one Control Rod is misaligned > 9" (6%),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.3 **IF** due to a malfunction, a Control Rod Group is misaligned > 9" (6%) from the acceptable region of PT/1/A/0600/001, (Periodic Instrument Surveillance),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.4 **IF** a Control Rod has dropped on an approach to criticality,

OR a dropped Control Rod results in a return to subcriticality from a critical condition,

THEN manually insert all Control Rods to Group 1 at 50% WD.

5. Subsequent Actions

_____ 5.1 **IF** the Reactor has tripped,

THEN **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 5.2 Verify Reactor runback <60% Full Power is in progress.:

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

NOTE 5.2.1:	If a control rod drops while the ICS Diamond Panel is in MANUAL, an "Auto Inhibit" (safety rod) or "Sequence Enable" (regulating rod) alarm will prevent placing the ICS in automatic.

_____ 5.2.1 **IF** the Reactor has **NOT** runback,

THEN commence manual Reactor Power reduction to < 60% of the allowable thermal power for the RCP combination.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.3 **IF** operating with only three (3)RCPs,

THEN commence manual Reactor Power reduction to < 45% Full Power.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.4 Notify I&E to begin investigation for the cause of the Dropped Control Rod.

_____ 5.5 Initiate actions to meet the Required Action of ITS 3.1.4 by performing the following:

_____ 5.5.1 Within one hour verify > 1% SDM
with allowance for the inoperable control rod(s):

- Perform PT/1/A/1103/15, (Reactivity Balance Calculation).

_____ 5.5.2 Within two hours reduce Reactor Power < 60%
of the allowable thermal power for the RCP combination.

NOTE 5.5.3:	Notification to I&E should be made as soon as possible due to the complexity of resetting the RPS trip setpoints and the short ITS allowable time limits.

_____ 5.5.3 Notify I&E to reduce RPS Flux/Flow-Imbalance trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.5.4 Notify I&E to reduce RPS High Flux trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.6 **WHEN** Reactor Power is < 60%
of the allowable thermal power for the RCP combination,

THEN notify I&E to begin repair of the Dropped Control Rod.

_____ 5.7 **WHEN** I&E is ready to begin repairs on the Dropped Control Rod,

THEN Place the ICS Diamond control station in MANUAL,

AND permit I&E to repair Dropped Control Rod.

CAUTION:5.8	The Duty Reactor Engineer must evaluate the effects of local power distribution and the necessity for special maneuvering limits prior to the recovery of a dropped or an asymmetric Control Rod.

_____ 5.8 **WHEN** I&E has repaired the Dropped Control Rod,

THEN recover the Dropped Control Rod per OP/0/A/1105/009, (Control Rod Drive System).

END

Dropped Control Rods

Appendix

1. PIP # 0-O98-2734

END

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM CRO-95/SIM

Restore RBCUs to normal after an inadvertent ES Channel 5 actuation.

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

Restore RBCUs to normal after an inadvertent ES Channel 5 actuation.

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

022 A4.01 (3.6 / 3.6)

Task Standard:

The 1A and 1B RBCU are stopped. The 1A RBCU is restarted in HIGH speed.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

AP/1/A/1104/015, Reactor Building Cooling

Validation Time: 10 min. **Time Critical:** NO

===

Candidate: ____________________________________ Time Start : ________
NAME Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ____________________________ _________________________________/________
NAME SIGNATURE DATE

===

COMMENTS

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall SNAP # ________
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

OP/1/A/1104/015

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 BOP

An inadvertent actuation of ES Channel 5 has occurred on Unit 1.

1CC-7, 1LPSW-6, 1LPSW-15 and the "A" PR Ventilation fan have been return to their normal condition.

INITIATING CUES:

The SRO in the Control Room directs you to return the RBCUs to their normal alignment.

START TIME: _______

STEP 1: Obtain a copy of the appropriate procedure OP/1/A/1104/015, RBC System. STANDARD: Operator obtains a copy of OP/1/A/1104/015, RBC System. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Equipment on ES Channel 5 is taken to MANUAL. STANDARD: The MANUAL push button is depressed for the following equipment on ES Channel 5. • 1LPSW-565 (not critical) • 1LPSW-566 (not critical) • 1A RBCU • 1B RBCU • 1LPSW-18 (not critical) • 1LPSW-21 (not critical) OMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: The 1A and 1B RBCUs are stopped to prevent RBCUs from running in mixed speed. ***CUE: Inform candidate that 30 minutes has passed.*** STANDARD: The 1A RBCU switch is place in the OFF position and then the ES reset push button is depressed. The 1A RBCU is verified off. The 1B RBCU switch is verified in the OFF position and then the ES reset push button is depressed. The 1B RBCU is verified off. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 4: The 1A RBCU is started in HIGH speed. STANDARD: The candidate rotates the 1A RBCU switch to the HIGH position and verifies that the RBCU goes to high speed. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5: Open 1 LPSW 565 STANDARD: Locates the switch and goes to open. Verifies that 1LPSW-565 opens. COMMENTS:	___ SAT ___ UNSAT
STEP 6: Throttles 1LPSW18 and 1LPSW-21 STANDARD: 1LPSW-18 and 1LPSW-21 are throttled so that ≈ 1400 gpm of LPSW is achieved to both the 1A and 1B RBCUs. COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANTIONS:

STEP #	Explanation
2	This step is required, because components must be placed in MANUAL to be able to reposition them.
3	This step is required, because the RBCUs should not be run in a mixed speed configuration.
4	This step is required, because HIGH speed is the normal alignment for the RBCUs.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the unit 1 BOP

An inadvertent actuation of ES Channel 5 has occurred has occurred on Unit 1.

1CC-7, 1LPSW-6, 1LPSW-15 and the "A" PR Ventilation fan have been return to their normal condition.

INITIATING CUES:

The SRO in the Control Room directs you return the RBCUs to their normal alignment.

(R06-97)

SR
SLm
NRC
106
115
JPP/JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No OP/1/A/1104/015

Revision No 020

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Reactor Building Cooling System

(4) Prepared By Dennis L. Masteller (Signature) Dennis L. Masteller Date 03/07/00

(5) Requires 10CFR50.59 evaluation?
☐ Yes (New procedure or revision with major changes)
X No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By George Ridgeway (QR) Date 3/9/00
Cross-Disciplinary Review By ______ (QR)NA GAR Date ______
Reactivity Mgmt. Review By ______ (QR)NA GAR Date ______

(7) Additional Reviews
Reviewed By [signature] Date 3/9/00
Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*
By ______ (SRO/QR) Date ______
By ______ (QR) Date ______

(9) Approved By [signature] Date 3/13/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______
Compared with Control Copy ______ Date ______
Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______
Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification:
☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?
☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Reactor Building Cooling System** **Multiple Use**	Procedure No. OP/1/A/1104/015 Revision No. 020 Electronic Reference No. OX002VM0

Reactor Building Cooling System

1. Purpose

To describe the proper method for operating Reactor Building Cooling System.

2. Limits And Precautions

2.1 RBCUs should NOT be started, stopped, or speed changed to equalize fan run times.

NOTE:
- A step of "OFF" to "LOW" (0-600 rpm) is considered one start.
- A step of "OFF" to "HIGH" (0-600-1200 rpm) is considered one start.

2.2 RBCU motor shall be off for 30 minutes, or allowed to operate for 30 minutes, prior to starting or changing speed.

2.3 30 minute speed change time interval may be waived in emergencies.

2.4 Manual speed changes should be minimized where possible.

2.5 During non-emergency operation, maximum RBCU motor bearing temperature: 220°F. (computer point: RBV CLR FAN IB/OB BRG TEMP).

2.6 1B RBCU may be operated while LPSW is diverted to Aux Fan Coolers.

2.7 Do NOT operate RBCUs in mixed speed combinations. Excess back pressure is placed on low speed fans.

2.8 Proper damper operation is NOT required for RBCU operability per Improved Technical Specifications (ITS).

- If dampers are NOT operating properly, high vibration and temperature problems may be encountered. {1}

2.9 When Reactor Building Cooling System Operability is required (TS 3.6.5, MODES 1, 2, 3, and 4), LPSW flow to all RBCUs must be ≥ 550 gpm. {2}

- If LPSW to an RBCU is < 550 gpm, LCO 3.0.3 applies. {2}

3. Enclosures

3.1 RBCU System Startup

3.2 RBCU Operation

3.3 Reduction Of Cooling Capacity

3.4 Valve Alignment For Temporary RB Chiller During Unit Outage

* Appendix *

Information Use

1. Initial Conditions

1.1 All RBCUs shutdown.

1.2 Review Limits and Precautions.

2. Procedure

2.1 Start desired number of RBCUs by selecting "HIGH" or "LOW":

- 1A RBCU
- 1B RBCU
- 1C RBCU

NOTE:	• 1LPSW-21 should remain open if RB Aux Fans require LPSW. • Do NOT throttle RBCU LPSW flow < 550 gpm. {2}

2.2 Position RBCU valves as required for RB cooling:

- 1LPSW-18 (1A RBCU OUTLET)
- 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)
- 1LPSW-565 (RB AUX FANS COOLERS INLET)
- 1LPSW-24 (1C RBCU OUTLET)
- 1LPSW-566 (1B RBCU ISOLATION)

2.3 Verify RBCU Damper on shutdown fan(s) closed.

2.4 Verify RBCU Dampers on operating fan(s) open.

NOTE:	• If 1B RBCU paired with 1A or 1C RBCU, light closest to paired fans should be lit. • For all other combinations both 1A and 1C lights should be off.

2.5 Verify 1B RBCU Dampers positioned properly.

Information Use

1. Initial Conditions

1.1 RBCU(s) operating.

1.2 Review Limits and Precautions.

2. Procedure

2.1 To stop RBCU(s), place desired switch to "OFF":

- 1A RBCU
- 1B RBCU
- 1C RBCU

2.2 To start RBCU(s), place desired switch to "HIGH" <u>or</u> "LOW":

- 1A RBCU
- 1B RBCU
- 1C RBCU

NOTE:	
	• 1LPSW-21 should remain open if RB Aux Fans require LPSW.
	• Do <u>NOT</u> throttle RBCU LPSW flow < 550 gpm. {2}

2.3 Position RBCU valves as required for RB cooling:

- 1LPSW-18 (1A RBCU OUTLET)
- 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)
- 1LPSW-565 (RB AUX FANS COOLERS INLET)
- 1LPSW-24 (1C RBCU OUTLET)
- 1LPSW-566 (1B RBCU ISOLATION)

2.4 Verify RBCU Damper on shutdown fan(s) closed.

2.5 Verify RBCU Dampers on operating fan(s) open.

NOTE:	• If 1B RBCU paired with 1A or 1C RBCU, light closest to paired fans should be lit. • For all other combinations both 1A and 1C lights should be off.

2.6 Verify 1B RBCU Dampers positioned properly.

Information Use

1. Initial Conditions

1.1 Reduction of cooling to RxV and SG cavities desired.

1.2 Review Limits and Precautions.

2. Procedure

NOTE:	• 1LPSW-21 should remain open if RB Aux Fans require LPSW.
	• Do **NOT** throttle RBCU LPSW flow < 550 gpm. {2}

2.1 Place control switch of operating fan(s) in "OFF" or "LOW" as required:

- 1A RBCU
- 1B RBCU
- 1C RBCU

Valve Alignment For Temporary RB Chiller During Unit Outage

Continuous Use

1. Initial Conditions

NOTE:	1B RBCU and RB Aux Fans will be available for Loss of Decay Heat Removal heat sink.

_____ 1.1 Unit is in MODE 5 or below. {2}

SRO

_____ 1.2 Temporary RB Chiller installed per Maintenance procedure.

_____ 1.3 Review Limits and Precautions.

2. Procedure

_____ 2.1 Establish communications with Maintenance personnel at Chiller.

NOTE:	1B RBCU and RB Aux Fans supplied by non-safety related cooling water source.

_____ 2.2 Unlock and White Tag closed 1LPSW-82 (RBCU 1B Outlet Block). (A-4-E Pen Rm)

_____ 2.3 IF desired to place Chilled Water through Aux Fans, ensure Open 1LPSW-565 (RB AUX FAN COOLER INLET).

_____ 2.4 IF desired to place Chilled Water through 1B RBCU, ensure Open 1LPSW-566 (1B RBCU ISOLATION).

2.5 Isolate 1B RBCU:

2.5.1 Unlock and White Tag closed the following valves: (A-4-E Pen Rm)

- _____ 1LPSW-79 (RBCU 1A & 1B Supply Tie)
- _____ 1LPSW-80 (RBCU 1B & 1C Supply Tie)

2.5.2 Close the following valves:

- _____ 1LPSW-19 (1B & AUX FAN COOLER RBCU INLET)
- _____ 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET)

NOTE:	1LPSW-878 opened per maintenance procedure.

2.6 Ensure open and White Tag open the following valves: (A-4-E Pen Rm)

_____ • 1LPSW-878 (RBCU Chilled Water Supply)

_____ • 1LPSW-879 (RBCU Chilled Water Return)

_____ 2.7 Notify Maintenance to start Temporary RB Chiller.

Maintenance ______________________________

2.8 Run 1B RBCU Fan as required.

2.9 WHEN changing chilled water flow paths, notify Maintenance to monitor Temporary RB Chiller operation.

2.10 WHEN notified by Maintenance, perform the following: (A-4 E Pen Rm)

_____ 2.10.1 Verify Temporary RB Chiller shutdown.

_____ 2.10.2 Remove White Tag and Close 1LPSW-878 (RBCU Chilled Water Supply).

_____ 2.10.3 Remove White Tag and Close 1LPSW-879 (RBCU Chilled Water Return).

2.11 Remove White Tags from the following: (A-4-E Pen Rm)

_____ • 1LPSW-79 (RBCU 1A & 1B Supply Tie)

_____ • 1LPSW-80 (RBCU 1B & 1C Supply Tie)

_____ • 1LPSW-82 (RBCU 1B Outlet Block)

2.12 Prior to entering MODE 4, return 1B RBCU to service as follows:

2.12.1 Open and lock open the following: (A-4-E Pen Rm)

_____ • 1LPSW-79 (RBCU 1A & 1B Supply Tie)

_____ • 1LPSW-80 (RBCU 1B & 1C Supply Tie)

_____ • 1LPSW-82 (RBCU 1B Outlet Block)

_____ 2.12.2 Open 1LPSW-19 (1B RBCU & AUX FAN COOLER INLET).

_____ 2.12.3 Position 1LPSW-566 (1B RBCU ISOLATION) as required.

_____ 2.12.4 Position 1LPSW-565 (RB AUX FAN COOLER INLET) as required.

_____ 2.12.5 Position 1LPSW-21 (1B RBCU & RB AUX FAN CLR OUTLET) as required.

Appendix

1. Limit and Precaution added in response to PIP 2-O90-0080. Isolation dampers, located below the fusible patches, serve no safety function. Since ductwork below coils/patches is assumed to be "crimped" after a LOCA blowdown (which closes the duct), misalignment of dampers has no effect on RBCU's expected post-accident response, which is to shift to low speed, open discharge valve, and drop patches. Therefore, an alignment problem with isolation dampers **CANNOT** make RBCUs inoperable from a nuclear safety standpoint.

2. Added in response to PIP 2-O98-3629. RBCU LPSW flow ≥ 550 gpm is required when RCS ≥ 200°F and ≥ 300 psig.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

CRO-096

ALIGN ECCS SUCTION FROM EMERGENCY SUMP
(Alternate Path)

CANDIDATE ____

EXAMINER ____

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

ALIGN ECCS SUCTION FROM EMERGENCY SUMP

TASK NUMBER: OO2650301

Alternate Path:

YES

Facility JPM #:

CRO-096

K/A Rating(s):
System: EPE 011 LARGE BREAK LOCA
K/A: EA1.11
Rating: 4.2/4.2

Task Standard:

The EOP Enclosure 7.11 is properly completed to align ECCS suction from the Emergency sump.

Preferred Evaluation Location:

Simulator __X__ In-Plant ____

Preferred Evaluation Method:

Perform __X__ Simulate ____

References:
EP/1/A/1800/0, Enclosure 7.11

Validation Time: 15 minutes

Time Critical: NO

==

Candidate: ________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ________________________ __________________________/________
NAME SIGNATURE DATE

==

COMMENTS

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC or SNAP # ________
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:

EP/1/A/1800/01, Enclosure 7.11, ECCS Suction Swap to the RBES With Both LPI Header Flows > 1000 gpm.

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

A large break LOCA has occurred which is depleting the BWST.
LPI flow is ≥ 1000 gpm per header
EOP is in progress, currently on step 2.0 of CP-601 Cooldown Following Large LOCA.

INITIATING CUES:

BWST level is approaching 13 feet

START TIME: ______

STEP 1: IF AT ANY TIME BWST level reaches 13 feet, AND RB Level is increasing THEN transfer LPI and RBS suction to the RBES per Enclosure 7.11, ECCS Suction Swap to the RBES With Both LPI Header Flows > 1000 gpm. STANDARD: The student locates the BWST level gauges on 1UB2. The student determines level to be ≤13 feet. or The student may obtain BWST level from the OAC (Operator Aid Computer), at 1UB1, 1UB2, or STA monitor. or ICCM monitors on 1UB1. The student locates the RB level Train 1 and Train 2 gauges on 1UB1. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 2: Refer to Enclosure 7.11, "ECCS Suction Swap to RBES With Both LPI Header Flows ≥ 1000 gpm". STANDARD: Candidate refers to Enclosure 7.11, "ECCS Suction Swap to RBES With Both LPI Header Flows ≥ 1000 gpm". COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: SECURE all HPI pumps. _____ 1A HPI pump _____ 1B HPI pump _____ 1C HPI pump STANDARD: Student verifies all HPI Pumps are secured by verifying the "RED" on lights are not ON. COMMENTS:	___ SAT ___ UNSAT

STEP 4: Throttle RBS Flow in all headers with an operating pump to 900 - 1000 gpm per header: _____ 1BS-1 (1A HDR RB ISOLATION) _____ 1BS-2 (1B HDR RB ISOLATION) STANDARD: RBS Flow in 1A header throttled to 900 - 1000 gpm RBS Flow in 1B header throttled to 900 - 1000 gpm COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5: WHEN BWST level reaches 9 feet, AND RB level is increasing THEN perform the following to swap LPI suction to RBES: NOTE: RB level of ≥ 2 feet is expected STANDARD: Candidate determines BWST Level is ≤ 9 feet (decreasing) **IF BWST level is not <9 feet then,** **CUE: BWST is < 9 feet.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 6: Simultaneously open the following valves _____ 1LP-19 ('1A' RX. BLDG. SUCTION) _____ 1LP-20 ('1B' RX. BLDG. SUCTION) STANDARD: The student locates the control switch for 1LP-19 and 1LP20 ('1A' and '1B' RX. BLDG. SUCTION) on 1UB2 and rotates the switches in the OPEN direction. Verify Green CLOSED light ON, Red OPEN light OFF. **CUE: 1LP-20 will NOT respond. Student may attempt to dispatch NLOs to either manually open 1LP-20 or RESET the breaker.** **Inform student all attempts to open 1LP-20 are unsuccessful.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: IF 1LP-19 (1A RX BLDG SUCTION) fails to open **NOTE: Go to step 3.3** STANDARD: Candidate determines that 1LP-19 is OPEN COMMENTS:	___ SAT ___ UNSAT

STEP 8: **IF** 1LP-20 (1B RX BLDG SUCTION fails to open **THEN** perform the following: STANDARD: Candidate determines that 1LP-20 remains CLOSED COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 9: IF BWST level is continuing to decrease, THEN wait until BWST level is ≤ 6 feet before proceeding STANDARD: Candidate observes that the BWST level is ≤ 6 feet before proceeding. **If BWST is not a t 6 feet then,** **CUE: BWST level is 6 feet** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 10: IF LP-20 fails to open IF BWST level is ≤ 6 feet, THEN immediately align the following valves: _____ Close 1LP-21 (1A LPI BWST SUCTION) _____ Open 1LP-9 (1C LIP DISCH TO 1A LPI HDR) _____ Open 1LP-10 (1C LIP DISCH TO 1B LPI HDR)\ **NOTE: 1LP-20 (1A LPI BWST SUCTION) is closed** STANDARD: 1LP-21 (1A LPI BWST SUCTION) is closed 1LP-9 (1C LPI DISCH TO 1A LPI HDR) is opened 1LP-10 (1C LPI DISCH TO 1B LPI HDR) is opened COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 11: Stop the following pumps _____ 1B LPI pump _____ 1B RBS Pump STANDARD: 1B LPI pump is stopped 1B RBS Pump is stopped COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

<u>STEP 12</u>: Throttle total LPI flow per the following: A. <u>IF</u> 1LP-14 (1B LPI Cooler Outlet) has been locally throttled, <u>THEN</u> throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize "A" LPI header flow ≤1100 gpm. B. <u>IF</u> 1LP-14 (1B LPI Cooler Outlet has NOT been locally throttled, <u>THEN</u> throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize in each LPI header flow ≤1100 gpm. <u>STANDARD</u>: Candidate determines LPI Cooler out flow has NOT been locally throttled <u>COMMENTS</u>:	___ SAT ___ UNSAT
<u>STEP 13</u>: GO TO step 7 <u>STANDARD</u>: Transitions to step 7 <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT
<u>STEP 14</u>: Throttle RBS flow in all headers with and operating pump to 900 - 1000 gpm per header _____ 1BS-1 (1A HDE RB ISOLATION) _____ 1BS-2 (1B HDR RB ISOLATION) <u>STANDARD</u>: Verification of ≈1000 gpm flow is indicated in the 1A RB Spray header **NOTE: "A" and "B" RBS flow was throttled in Step 2. And "B" RBS Pump has been secured in Step 3.3.2** <u>COMMENTS</u>:	___ SAT ___ UNSAT

STEP 15: Notify Chemistry to perform the following ____ Commence caustic addition ____ Periodically sample the LPI discharge to determine RBES boron concentration. STANDARD: Chemistry is notified. COMMENTS:	___ SAT ___ UNSAT
STEP 16: IF AT ANY TIME BWST Level is ≤ 6 feet, THEN dispatch and operator to close 1LP-28 (BWST Outlet). (East of Unit 1 BWST) STANDARD: NLO is dispatched to close 1LP-28 (BWST Outlet) COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 17: Perform the following to align LPSW to LPI Coolers: Close 1LPSW-139 (Unit 1 NONESSENTIAL HEADER ISOLATION). IF Unit 2 Turbine is tripped,... **CUE: Unit 2 Turbine is operating** STANDARD: Close 1LPSW-139 (Unit 1 NONESSENTIAL HEADER ISOLATION) COMMENTS:	___ SAT ___ UNSAT

STEP 18: Place the following switches in the FAIL OPEN position. _____ 1LPSW-251 FAIL SWITCH _____ 1LPSW-252 FAIL SWITCH STANDARD: 1LPSW-251 FAIL SWITCH in FAIL position 1LPSW-252 FAIL SWITCH in FAIL position COMMENTS:	___ SAT ___ UNSAT
STEP 19: If either of the following conditions exist: Three LPSW pumps are operating Two LPSW pumps are operating and only two LPSW pumps are required to be operable by TS, THEN perform the following:... STANDARD: Candidate determines LPSW pump operating status. **CUE: Two LPSW pumps are operating and only two LPSW pumps are required to be operable by TS** COMMENTS:	___ SAT ___ UNSAT

STEP 20: Open the following valves _____ 1LPSW-4 (1ALPI CLR SHELL OUTLET) _____ 1LPSW-5 (1ALPI CLR SHELL OUTLET) STANDARD: LPSW-4 (1ALPI CLR SHELL OUTLET) is opened 1LPSW-5 (1ALPI CLR SHELL OUTLET) is opened COMMENTS:	___ SAT ___ UNSAT
STEP 21: GO TO step 10.8 STANDARD: Transitions to step 10.8 COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 22: IF only one LPI cooler is available STANDARD: Determines availability of LPI coolers. Both are available. COMMENTS:	___ SAT ___ UNSAT

<u>STEP 23</u>: <u>WHEN</u> 1LP-28 (BWST Outlet is closed <u>THEN</u> perform the following . . . ***CUE: 1LP-28 is closed.*** <u>STANDARD</u>: Candidate determines that steps 11.1 and 11.2 are N/A. <u>COMMENTS</u>:	___ SAT ___ UNSAT
<u>STEP 24</u>: <u>IF</u> Two LPI Pumps are operating, <u>THEN</u> perform the following: <u>STANDARD</u>: Candidate determines that only 1 LPI pump is operating <u>COMMENTS</u>:	___ SAT ___ UNSAT
<u>STEP 25</u>: Initiate makeup to the BWST with boron concentration> COLR limit to provide a backup to ECCS suction source. <u>STANDARD</u>: **CUE: Another operator will initiate makeup to the BWST.** <u>COMMENTS</u>:	___ SAT ___ UNSAT

STOP TASK___________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1, 2	Transfer to Enclosure 7.11 to provide guidance in swapping suction to the RBES
4	Decreases RBS flow to prevent pump runout when suction is swapped to the RBES
5	Monitors BWST for 9 feet to actually perform suction swap to the RBES
6	Open RBES suction valves (LP-20 does not open)
8	Determines LP-20 will not open
9	Determine BWST level is < 6 feet
10	Isolates the "A" Suction line from the BWST and cross-connects the LPIP discharge header.
11	Secures the "B" train pumps to prevent air from the BWST entering the suction source
16	Manually isolates the BWST suction to prevent air in the suction
21	Proper transfer in Enclosure 7.11

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

A large break LOCA has occurred which is depleting the BWST.
LPI flow is ≥ 1000 gpm per header
EOP is in progress, currently on step 2.0 of CP-601 Cooldown Following Large LOCA.

INITIATING CUES:

BWST level is approaching 13 feet

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NLO-017/PLANT

ALIGN COOLING WATER TO HIGH PRESSURE INJECTION PUMP MOTOR COOLERS FROM AUX. SERVICE WATER PUMP

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

ALIGN COOLING WATER TO HIGH PRESSURE INJECTION PUMP MOTOR COOLERS FROM AUX. SERVICE WATER PUMP.

Alternate Path:

N/A

Facility JPM #:

NLO-017

K/A Rating(s):

076 A2.01 3.5/3.7

Task Standard:

Preferred Evaluation Location:

Simulator _____ In-Plant __X____

Preferred Evaluation Method:

Perform _____ Simulate ___X___

References:

AP/1/A/1700/07

Validation Time: 16 min. **Time Critical:** NO

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Candidate: ______________________________ Time Start: _________
NAME Time Finish: _________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________ ______________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

Ensure enough copies of AP/1/A/1700/07 are available in the Simulator file cabinet, since Operators will obtain their own copy of the procedure.

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

A Station Blackout has occurred. The SSF Diesel Generator will not start, rendering the SSF RCMU Pumps inoperable. Standby Bus #1 has subsequently been energized from CT-5. I&E personnel have aligned the 'A' HPIP to the ASWP Switchgear. AP/1,2,3/A/1700/11, (Loss of Power), Enclosure 6.3 of has been completed up to step 2.0.

INITIATING CUES:

AP/1,2,3/A/1700/11, Loss Of Power, directs the operator to align cooling water to the High Pressure Injection Pumps.

The Control Room Supervisor directs you to align cooling water to the High Pressure Injection Pumps on Unit___ (specify unit) per Enclosure 6.3 of AP/1,2,3/A/1700/11, Loss of Power.

START TIME: _______

STEP 1: Ensure closed " AUX. SER. WTR. SWGR 4160 VOLT FDR B1T - UNIT 10" breaker. STANDARD: "AUX. SER. WTR. SWGR. 4160-Volt FDR B1T-UNIT 10" breaker indicates closed on the ASW SWGR 600V LOAD CENTER. Two red CLOSED lights are on. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Ensure closed the "AUX. SER. WTR. SWGR TRANSFORMER" breaker STANDARD: Student verifies the "AUX. SER. WTR SWGR TRANSFORMER" breaker is Closed. **Location: ASW SWGR 600V LC Unit 5** ***CUE: Inform the student that the red CLOSED light is lit and that the green OPEN light is off at the control switch for Aux. Ser. Wtr. Swgr. Xfrmr. Bkr.*** COMMENTS:	___ SAT ___ UNSAT
STEP 3: Rack in "AUXILIARY SERVICE WATER PUMP" breaker at the ASW SWGR 600V LOAD CENTER Unit 6. STANDARD: Student opens shutter, inserts 600v breaker rackout tool, and turns tool clockwise to rack breaker in. ***CUE: After breaker is racked in, inform student that the AUX SERVICE WATER PUMP MTOR breaker green OPEN indicating light is ON.*** COMMENTS: Student is expected to follow (simulate) all applicable safe electrical work practices.	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 4: CLOSE CCW-309 (ASWP Disch Drain). STANDARD: CCW-309 (ASWP Disch Drain) is manually CLOSED. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5 OPEN CCW-99 (ASWP Suction). STANDARD: CCW-99 (ASWP Suction) is manually opened. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
.EP 6 OPEN CCW-101 (ASWP Disch). STANDARD: CCW-101 (ASWP Disch) is manually opened. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7 OPEN CCW-247 (ASWP Recirc). STANDARD: CCW-247 (ASWP Recirc) is manually OPENED. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8 Vent the Aux. Service Water Pump using CCW-308 (ASWP Vent). STANDARD: CCW-308 ASWP vent is throttled open until water issues and then is then closed. COMMENTS:	___ SAT ___ UNSAT
STEP 9 START the Aux. Service Water Pump Motor. STANDARD: Student locates AUX SERVICE WATER PUMP MOTOR control switch and rotates switch to the CLOSE position. **CUE: After switch is rotated, inform student that the AUX SERVICE WATER PUMP MOTOR breaker red CLOSED indicating light is ON.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 10: VERIFY adequate HPIP motor cooler flow indication locally (>1 gpm) by local flow indication. Location: Aux. Bldg. 1st – HPI Pump Room STANDARD: Student verifies flow located AB-1st HPI pump Room. ***Cue: A picture of the "A" HPI Pump Motor Cooler Flow indication may be given the student to be used in explaining how the flow would be verified.*** COMMENTS: **For ALARA and time considerations, do not allow the student to enter the HPI Pump Room. Stop him/her at the plan view of the HPI Room and have him/her indicate where the flow would be verified.** **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANTIONS:

STEP #	Explanation
3	Supplies power to the Auxiliary Service Water Pump.
4	Ensures that water is not introduced to the Aux. Bldg. when the ASWP is started.
5	Ensures that a suction supply of water is available to the ASWP.
6	Ensures that water is supplied to the discharge header.
7	Prevents pump damage due to the possibility that low flow conditions may exist.
9	Supplies the HPI Pump Motor Coolers with water.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

A Station Blackout has occurred. The SSF Diesel Generator will not start, rendering the SSF RCMU Pumps inoperable. Standby Bus #1 has subsequently been energized from CT-5. I&E personnel have aligned the 'A' HPIP to the ASWP Switchgear. AP/1,2,3/A/1700/11, (Loss of Power), Enclosure 6.3 of has been completed up to step 2.0.

INITIATING CUES:

AP/1,2,3/A/1700/11, Loss Of Power, directs the operator to align cooling water to the High Pressure Injection Pumps.

The Control Room Supervisor directs you to align cooling water to the High Pressure Injection Pumps on Unit___ (specify unit) per Enclosure 6.3 of AP/1,2,3/A/1700/11, Loss of Power.

Aux Service Water To HPI Pump Motor Coolers

1. Energize Standby Bus #1:

_____ 1.1 **IF** either Keowee Unit is available,

THEN Emergency Start available Keowee Units:

_____ "Keowee Emer Start Channel A"

_____ "Keowee Emer Start Channel B".

_____ 1.1.1 **WHEN** available Keowee Units are running,

THEN perform the following:

_____ 1.1.1.1 Ensure closed ACB3 OR ACB 4.

_____ 1.1.1.2 Place "CT4 BUS 1 AUTO/MANUAL" transfer switch in "MANUAL".

_____ 1.1.1.3 Place "STBY BUS 1 SYNCHRONIZING" switch to "ON".

_____ 1.1.1.4 Close "SK1 CT4 STBY BUS 1 FEEDER".

_____ 1.1.1.5 Verify "STANDBY BUS 1" voltmeter indicates ≈ 4160v.

_____ 1.1.1.6 Place "STBY BUS 1 SYNCHRONIZING" switch to "OFF".

Aux Service Water To
HPI Pump Motor Coolers

_____ 1.2 **IF** NO Keowee Unit is available,

AND CT-5 voltmeter indicates ≈ 4160v,

THEN perform the following:

1.2.1 Place the following "AUTO/MAN" transfer switches in "MANUAL":

_____ "CT4 BUS 1 AUTO/MAN"

_____ "CT4 BUS 2 AUTO/MAN"

_____ "CT5 BUS 1 AUTO/MAN"

_____ "CT5 BUS 2 AUTO/MAN".

1.2.2 Ensure open the following breakers:

_____ "SK1 CT 4 STBY BUS 1 FEEDER"

_____ "$S1_1$ STBY BUS 1 TO MFB 1"

_____ "$S2_1$ STBY BUS 2 TO MFB 2".

CAUTION 1.2.3:	If statalarm "TRANSFORMER CT-5 UNDERVOLTAGE" (SA-16/ C-4) is received, additional loading of CT-5 (if powered from Central Switchyard) may result in an undervoltage trip of breakers SL1 and SL2 if Standby Bus voltage reaches 3890 volts.

_____ 1.2.3 Close "SL1 CT5 STBY BUS 1 FDR".

_____ 1.2.4 Place "CT5 BUS 1 AUTO/MAN" switch in "AUTO".

**Aux Service Water To
HPI Pump Motor Coolers**

_____ 2. WHEN Standby Bus 1 is energized,

THEN perform the following:

_____ 2.1 Ensure closed breaker "AUX. SER. WTR. SWGR. 4160 VOLT FDR. B1T - UNIT 10". (Control switch at ASW SWGR 600V LOAD CENTER Unit 5)

NOTE:	Breaker indication will **NOT** be on unless the Standby Bus is energized.

_____ 2.2 Ensure closed breaker "AUX. SERV. WTR. SWGR. TRANSFORMER". (located at ASW SWGR 600V LOAD CENTER Unit 5)

_____ 2.3 Rack in breaker "AUX. SERVICE WATER PUMP" at the ASW SWGR 600V LOAD CENTER Unit 6.

_____ 2.4 Close CCW-309 (ASWP Disch Drn).

_____ 2.5 Open CCW-99 (ASWP Suct).

_____ 2.6 Open CCW-101 (ASWP Disch).

_____ 2.7 Open CCW-247 (ASWP Recirc).

_____ 2.8 Vent the Aux Service Water Pump using CCW-308 (ASWP VENT).

_____ 3. Start the Aux. Service Water Pump Motor.

_____ 4. Locally ensure adequate HPIP motor cooler flow (> 1 gpm) by local flow indication.

END

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NLO-041

RESTART THE PRIMARY IA COMPRESSOR FOLLOWING A COMPRESSOR TRIP
(Alternate Path)

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

RESTART THE PRIMARY IA COMPRESSOR FOLLOWING A COMPRESSOR TRIP
TASK NUMBER: OO1333002

Alternate Path:

YES

Facility JPM #:

NLO-041

K/A Rating(s):
System: SF8-078 Instrument Air System
K/A: 2.1.30
Rating: 3.9/3.4

Task Standard:

The Primary IA Compressor is restarted by procedure

Preferred Evaluation Location:

Simulator _____ In-Plant __X___

Preferred Evaluation Method:

Perform _____ Simulate __X__

References:

Enclosure 4.11 of OP/0/A/1106/27

Validation Time: 10 minutes

Time Critical: NO

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Candidate: ________________________________ NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ________________________ NAME ________________________ SIGNATURE /________ DATE

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COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:
Enclosure 4.11 of OP/0/A/1106/27

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit 1 is at 100% power.
The Transmission Dept. was performing PM checks on B3T switchgear when the incoming feeder breaker tripped open.
When B3T de-energized, the automatic transfer to the backup source (B4T) did not occur and the Primary IA Compressor tripped.
The RO entered "Loss of IA" AP/1/A/1700/22 as IA header pressure decreased to ≈ 85 psig and has reached step 5.10, which refers the operator to OP/0/A/1106/27 to restore operable IA compressors.

INITIATING CUES:

The SRO in the control room instructs you utilize Enclosure 4.11 restart of the Primary IA Compressor following a Trip of OP/0/A/1106/27 (Compressed Air System) to **RESTART** the Primary IA Compressor. Initial Conditions of the enclosure have been completed.

START TIME: _______

STEP 1: Position the following valves: Close IA-2730 (Primary IA Desiccant Air Filter "A" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2730 (Primary IA "A" Desiccant Filter Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT
STEP 2: Position the following valves: Close IA-2731 (Primary IA Desiccant Air Filter "B" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2731 (Primary IA "B" Desiccant Filter Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT

STEP 3: At the Primary IA Dryer A Control Panel, position the (ON/OFF) switch to OFF. STANDARD: The student REMOVES the Primary IA Dryers from service by rotating the following switches, located on the A Dryer control panels, to the "OFF" position: Primary IA Dryer "A" On/Off selector. COMMENTS:	___ SAT ___ UNSAT
STEP 4: At the Primary IA Dryer B Control Panel, position the (ON/OFF) switch to OFF. STANDARD: The student REMOVES the Primary IA Dryer from service by rotating the following switches, located on the B Dryer control panels, to the "OFF" position: Primary IA Dryer "B" On/Off selector. COMMENTS:	___ SAT ___ UNSAT

STEP 5: Position the following valves: Close IA-2735 (Primary Air Filter "A" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2735 (Primary Air Filter "A" Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT
STEP 6: Position the following valves: Close IA-2736 (Primary Air Filter "B" Outlet). (TB5 L-39) STANDARD: The student LOCATES and CLOSES IA-2736 (Primary Air Filter "B" Outlet) by rotating the valve operator until the position indicating arrow is perpendicular to the piping. **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.** COMMENTS:	___ SAT ___ UNSAT

STEP 7: Open HPSW-771 (Primary IA Comp. Disc. Block) (TB5 M-39) STANDARD: The student LOCATES and OPENS HPSW-771 (Primary IA Compressor Cooling Discharge Block) by rotating the switch to the "Open" position. **NOTE: HPSW-771 control switch and the cooling water inlet pressure gauges are located north of the compressor next to the west Turbine floor wall.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 8: Verify adequate cooling water flow as follows: IF OHPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure) does NOT read between 61 and 67 psig. Backwash HPSW-764 (Primary IA Comp. Disch. Control) (TB5 M-39) per Backwash of Primary IA Compressor HPSW Pressure Regulator enclosure. Verify HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) in Locked Open Position. STANDARD: The student VERIFIES adequate cooling water flow by monitoring the following gauges: - OHPS-PG-0823 (Primary IA Comp. Cooling Water Inlet Pressure). **CUE: Using a pointing device, indicate to the student the following readings:** - **OHPS-PG-0823 = 64 psig.** HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) is verified in the Locked Open Position. COMMENTS:	___ SAT ___ UNSAT

<u>STEP 9</u>: Depress (RESET/LAMP TEST) pushbutton. • Verify all alarm indicators light. • Release (RESET/LAMP TEST) pushbutton and verify all alarm indicator lamps extinguish. 2.7.1 Any alarm condition indicated on the Primary IA Compressor control panel must be resolved before starting the compressor. <u>STANDARD</u>: The student RESETS the Primary IA Compressor by depressing the black "Reset" pushbutton on the compressor control panel located on the north side of the compressor housing. **CUE: While RESET/LAMP TEST pushbutton is depressed, inform student that all alarm indicators are lit. When RESET/LAMP TEST pushbutton is released, inform student that all alarm indicator lamps extinguish.** <u>COMMENTS</u>:	___ SAT ___ UNSAT

STEP 10: • Depress (START) pushbutton. • Verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is with in the range specified (in procedure). STANDARD: • The student STARTS the Primary Air Compressor by depressing the "Start" pushbutton on the control panel located on the north side housing of the compressor. **CUE: Inform the student that the green "Machine Run" light has illuminated.** • The student VERIFIES adequate cooling water flow by monitoring the following gauges: - OHPS-PG-0823 (Primary IA Comp. Cooling Water Inlet Pressure). - OHPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure). **CUE: Using a pointing device, indicate to the student the following readings:** - **OHPS-PG-0823 = 64 psig.** - **OHPS-PG-0824 = 9 psig.** Student should simulate throttling HPSW-767 (Pri. IA Comp. Disch. Cont.) to achieve the proper flow/outlet pressure range. **When HPSW-767 is throttled closed, indicate with the pointing device that flow is 98 gpm and outlet pressure is 18 psig** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 11: VERIFY selected Enclosure fan is running and all door panels are installed on compressor enclosure. Verify all door panels are installed on the Primary IA Compressor Enclosure. STANDARD: The student determines that selected enclosure fan is operating and all door panels located on the compressor enclosure are installed. **CUE: Inform the student that the selected Enclosure Fan is running properly. Inform student that all door panels are installed on enclosure.** COMMENTS:	___ SAT ___ UNSAT
STEP 12: Throttle open IA-2735 (Primary Air Filter "A" Outlet) or IA-2736 (Primary Air Filter "B" Outlet) (TB5 L-39) to SLOWLY pressurize the Dryer tanks to system pressure (100-110 psig). STANDARD: The student throttles open one of the following valves to SLOWLY PRESSURIZE the Desiccant Dryers: IA-2735 (Primary Air Filter "A" Outlet) **OR** IA-2736 (Primary Air Filter "B" Outlet) **CUE: Once the student has demonstrated his/her ability to properly throttle the valve, indicate to the student with a pointing device that the Desiccant Dryers have reached 104 psig.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 13: At the Primary IA Dryer A panel, position the (ON/OFF) switch to ON. STANDARD: The student PLACES the Primary IA Dryers in service by positioning the following switches to the "ON" position: Primary IA Dryer "A" On/Off Selector COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 14: At the Primary IA Dryer B panel, position the (ON/OFF) switch to ON. STANDARD: The student PLACES the Primary IA Dryers in service by positioning the following switches to the "ON" position: Primary IA Dryer "B" On/Off Selector COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 15: CONNECT the Primary IA Compressor to the IA Header. Open IA-2735 (Primary Air Filter A Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2735 (Primary Air Filter "A" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 16: CONNECT the Primary IA Compressor to the IA Header. Open IA-2736 (Primary Air Filter B Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2736 (Primary Air Filter "B" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 17: CONNECT the Primary IA Compressor to the IA Header. Slowly open IA 2730 (Primary Desiccant Air Filter A Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2730 (Primary Desiccant Air Filter "A" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 18: CONNECT the Primary IA Compressor to the IA Header. Slowly open IA 2731 (Primary Desiccant Air Filter B Outlet). (TB5 L-39) STANDARD: The student CONNECTS the Primary Air compressor to the IA header by slowly opening the following valves: IA-2731 (Primary Desiccant Air Filter "B" Outlet) **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.** CUE: Indicate that the valves are fully open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 19: As system pressure increases check for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters. **NOTE:** Enclosure "Startup Of The Primary IA Compressor" contains a detailed list of expected Primary Air Compressor normal operating parameters. STANDARD: The student checks for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters as system pressure increases. **CUE: No air leaks are found.** Primary Air Compressor monitored for normal operation. COMMENTS:	___ SAT ___ UNSAT

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
7	Open HPSW-771 (Primary IA Comp. Disc. Block) aligns cooling water to the compressor
10	Depress (START) pushbutton starts the compressor, verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is with in the range specified , and establish proper cooling water flow to the compressor
12	Pressurizes and places in service the primary air filter
13	Places the “A” Air Dryer in service
14	Places the “B” Air Dryer in service
15-18	Establishes an air flow path from the compressor to the IA Header

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit 1 is at 100% power.
The Transmission Dept. was performing PM checks on B3T switchgear when the incoming feeder breaker tripped open.
When B3T de-energized, the automatic transfer to the backup source (B4T) did not occur and the Primary IA Compressor tripped.
The RO entered "Loss of IA" AP/1/A/1700/22 as IA header pressure decreased to ≈ 85 psig and has reached step 5.10, which refers the operator to OP/0/A/1106/27 to restore operable IA compressors.

INITIATING CUES:

The SRO in the control room instructs you utilize Enclosure 4.11 restart of the Primary IA Compressor following a Trip of OP/0/A/1106/27 (Compressed Air System) to **RESTART** the Primary IA Compressor.
Initial Conditions of the enclosure have been completed.

Loss of Instrument Air

_____ 5.10 Restore Instrument Air header pressure:

- Isolate IA header leakage.
- Start operable IA compressors.
  - **REFER TO** OP/0/A/1106/027, (Compressed Air System).

_____ 5.11 **IF** efforts to restore IA are **NOT** successful,

AND both Main Feedwater Pumps have tripped,

THEN establish Condensate Recirc:

_____ 5.11.1 Trip all CBPs.

_____ 5.11.2 Trip all but one HWP.

_____ 5.11.3 Open 1C-124 (Condensate Recirc to UST).

_____ 5.11.4 Send an Operator to throttle 1C-129 (Condensate Recirc Control Bypass). (TB5/M-22NE)

_____ 5.11.5 Establish ≈ 2300 gpm Condensate Recirc flow by computer point O1A0156 (CBP DISCH HDR FLOW).

1. Initial Conditions

_____ 1.1 Worthington IA compressors may or may not be supplying IA header.

_____ 1.2 Reason for loss of Primary IA compressor has been corrected.

_____ 1.3 System alignment unchanged from time of compressor trip.

2. Procedure

2.1 Position the following valves:

_____ 2.1.1 Close IA-2730 (Primary Desiccant Air Filter 'A' Outlet). (TB5 L-39)

_____ 2.1.2 Close IA-2731 (Primary Desiccant Air Filter 'B' Outlet). (TB5 L-39)

_____ 2.2 At the Primary IA Dryer A Control Panel, position the (ON/OFF) switch to "OFF."

_____ 2.3 At the Primary IA Dryer B Control Panel, position the (ON/OFF) switch to "OFF."

2.4 Position the following valves:

_____ 2.4.1 Close IA-2735 (Primary Air Filter 'A' Outlet). (TB5 L-39)

_____ 2.4.2 Close IA-2736 (Primary Air Filter 'B' Outlet). (TB5 L-39)

_____ 2.5 Open HPSW-771 (Primary IA Comp. Disch. Block) (TB5 M-39).

2.6 Verify adequate cooling water flow as follows:

_____ 2.6.1 IF OHPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure) does NOT read between 61 and 67 psig. backwash HPSW-764 (Primary IA Comp. Disch. Control) (TB5 M-39) per Backwash of Primary IA Compressor HPSW Pressure Regulator enclosure.

_____ 2.6.2 Verify HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) in Locked Open position.

_____ 2.7 Depress (RESET/LAMP TEST) pushbutton.

- Verify all alarm indicators light.
- Release (RESET/LAMP TEST) pushbutton and verify all alarm indicator lamps extinguish.

_____ 2.7.1 Any alarm conditions indicated on the Primary IA Compressor control panel must be resolved before starting the compressor.

_____ 2.8 Depress (START) pushbutton.

_____ 2.8.1 Verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is within the range specified below for the value of 0HPS-PG-0823 (Primary IA Compressor Cooling Water Inlet Pressure).

Cooling Water Inlet Pressure (psig)	Acceptable Range for Cooling Water Outlet Pressure (psig).
61	18 - 10
62	19 - 11
63	20 - 12
64	21 - 12
65	22 - 13
66	23 - 14
67	24 - 14

_____ • IF OHPS-PG-0824 is NOT within the Acceptable Range For Cooling Water Outlet Pressure for the value of OHPS-PG-0823, throttle HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) to obtain the required cooling water outlet pressure.

_____ • IF OHPS-PG-0824 is NOT within the Acceptable Range For Cooling Water Outlet Pressure for the value of OHPS-PG-0823, THEN closely monitor Primary IA Compressor Discharge Temperature and Injection Temperature until acceptable Cooling Water pressures can be obtained.

_____ 2.9 IF Compressor fails to start, notify WCC SRO for aid in resolving problem.

_____ 2.10 Verify selected Enclosure fan is running.

_____ 2.11 Verify all door panels are installed on the Primary IA Compressor Enclosure.

CAUTION: The Outlet Filter Valves must be opened in increments very slowly to prevent disintegration of the Dryer desiccant.

_____ 2.12 Throttle open IA-2735 (Primary Air Filter 'A' Outlet) (TB5 L-39) OR IA-2736 (Primary Air Filter 'B' Outlet) (TB5 L-39) to SLOWLY pressurize the Dryer Tanks to system pressure (100-110 psig).

_____ 2.13 At the Primary IA Dryer A panel, position the (ON/OFF) switch to "ON."

_____ 2.14 At the Primary IA Dryer B Panel, position the (ON/OFF) switch to "ON."

_____ 2.15 Open IA-2735 (Primary Air Filter A Outlet). (TB5 L-39)

_____ 2.16 Open IA-2736 (Primary Air Filter B Outlet). (TB5 L-39)

CAUTION:	Rapid opening of the Primary Desiccant Air Filter Outlet Valves can cause collapse of the Desiccant Filters.

_____ 2.17 SLOWLY open IA-2730 (Primary Desiccant Air Filter A Outlet). (TB5 L-39)

_____ 2.18 SLOWLY open IA-2731 (Primary Desiccant Air Filter B Outlet). (TB5 L-39)

_____ 2.19 As system pressure increases check for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters.

NOTE:	Enclosure "Startup Of The Primary IA Compressor" contains a detailed list of expected Primary Air Compressor normal operating parameters.

_____ 2.20 Monitor Primary Air Compressor for normal operation.

NOTE:	If Primary Air Compressor is operating normally, Backup IA Compressors will be running unloaded.

2.21 Return the Backup IA Compressors to their normal lineup:

_____ 2.21.1 Place the C Backup IA Compressor control switch in "STBY 1."

_____ 2.21.2 Place either the A or B Backup IA Compressor control switch in "STBY 1."

_____ 2.21.3 Place the remaining Backup IA Compressor control switch in "STBY 2."

2.22 Close or verify closed the following valves:

_____ • RCW-25 (Backup IA Compressor A Temp. Control Bypass) (TB1 L-31)

_____ • RCW-31 (Backup IA Compressor B Temp. Control Bypass) (TB1 L-32)

_____ • RCW-37 (Backup IA Compressor C Temp. Control Bypass) (TB1 L-32)

INITIAL SUBMITTAL

OCONEE EXAM 2000-301
50-269, 270, AND 287/2000-301

JULY 10 - 14, 18, 19, AND 20, 2000

INITIAL SUBMITTAL

ADMINISTRATIVE TOPICS

ADMINISTRATIVE TOPICS

OCONEE
2000

NRC Copy

Facility: Oconee		Date of Examination: 07/10 – 21/2000 .
Examination Level (circle one): **RO**		Operating Test Number: 1 .
Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Plant Parameter Verification	JPM CRO-040 (Bank-modified), Calculate Shutdown Margin with the Computer. KA 2.1.7 [3.7/4.4] (RO ONLY) **Note:** This JPM will be conducted with B.1 Section JPM CRO-012, (Recovery of dropped rod)
	COLR/Tech. Spec Utilization	JPM NRC-005 (New) – Reactor Power Imbalance - Improved Tech Specs/COLR. KA 2.1.11 [3.0/3.8]
A.2	Surveillance Testing	JPM NRC-004 (New) – Perform PT/1/A/0600/001, Enclosure 13.16, ICCM Subcooling Monitor Check. KA 2.2.12 [3.0/3.4]
A.3	Radiation Control	SRO/RO – 2 Questions *Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure [CFR: 43.4/45.10] G2.3.10 (2.9/3.3)* Knowledge of 10 CFR 20 and related facility radiation control requirements. KA 2.3.1 [2.6/3.0] **Note:** These questions to be conducted with B.2 section JPM NLO-040 or 17.
A.4	Emergency Plan Implementation	RO – JPM NRC-003 (New) RP/1000/015A (Offsite Communications - Emergency Communications) (RO ONLY) KA 2.4.43 [2.3/3.5]

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

CRO-40/ADMIN A.1

CALCULATE SDM
WITH A DROPPED CONTROL ROD

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

CALCULATE SDM WITH A DROPPED CONTROL ROD

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

Gen 2.1.7 3.7/4.4

Task Standard:

PT/1/A/1103/15, Reactivity Balance Procedure is used to verify > 1% SDM with one inoperable (dropped) CR within 1 hour.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/1103/15,Reactivity Balance Procedure
AP/1/A/1700/15, Dropped Control Rods

Validation Time: 10 min. **Time Critical:** YES

==

Candidate: __________________________________
NAME

Time Start : _________
Time Finish: _________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ___________________________ ________________________________/_______
NAME SIGNATURE DATE

==

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC # SNAP ________

2. Go to run, acknowledge alarms.

6. Freeze simulator.

10. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:

PT/1/A/1103/015, Reactivity Balance Procedure
OP/0/A/1105/009, Control Rod Drive System

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.5.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

START TIME: _______

STEP 1:	Within one hour verify > 1% SDM with allowance to the inoperable control rod. Perform PT/1/A/1103/15, Reactivity Balance Procedure.	**CRITICAL STEP** ___ SAT ___ UNSAT
STANDARD:	Obtain copy of PT/1/A/1103/15, Reactivity Balance Calculations.	
COMMENTS:		
STEP 2	Determine proper enclosure to use.	**CRITICAL STEP** ___ SAT ___ UNSAT
STANDARD:	Enclosure 13.20, Shutdown Margin at Power, is chosen.	
COMMENTS:		
STEP 3:	Use Enclosure 3.21, Rod Position Limits at Power, 1 Inoperable Rod or 1 Dropped Rod – 4 Pump Flow. Verify available SDM is ≥ 1% ΔK/K by verifying that the control rod position and power level are within the acceptable region on the appropriate curve for the number of RCPs and Inoperable rods in Enclosure 13.21, Rod Position limits at Power.	**CRITICAL STEP** ___ SAT ___ UNSAT
STANDARD:	SDM is determined to be is ≥ 1% ΔK/K.	
COMMENTS:		
	END OF TASK	

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	Step is necessary for the operator to select PT/1/A/1103/15, Reactivity Balance Calculations procedure to obtain correct enclosure to complete step three correctly.
2	Step is necessary for the operator to select to the Enclosure 13.20, Shutdown Margin at Power.
3	Step is necessary, the operator must interpret the 4 RCP curve to ensure adequate SDM.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

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INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

(R06-97) WHC/JMB/JPP

Duke Power Company Trng. (1) ID No AP/1/A/1700/015

PROCEDURE PROCESS RECORD

Revision No 4

LAN Location: SAROS

REPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Dropped Control Rods

(4) Prepared By Dennis Gordon. Date 2/17/99

(5) Requires 10CFR50.59 evaluation?
- ☒ Yes (New procedure or revision with major changes)
- ☐ No (Revision with minor changes)
- ☐ No (To incorporate previously approved changes)

(6) Reviewed By Walter M. Barber (QR) Date 2/25/99

Cross-Disciplinary Review By ______ (QR)NA WB Date ______

Reactivity Mgmt. Review By Walter M. Barber (QR)NA ______ Date 2/25/99

(7) Additional Reviews

Reviewed By ______ Date ______

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By Arthur Delagot Date 3/8/99

PERFORMANCE ***(Compare with control copy every 14 calendar days while work is being performed.)***

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification
- ☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
- ☐ Yes ☐ NA Listed enclosures attached?
- ☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
- ☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
- ☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Dropped Control Rods** **Continuous Use** **Reactivity Management Related**	Procedure No. AP/1/A/1700/015 Revision No. 004 Electronic Reference No. OX002RGS

DROPPED CONTROL RODS
Reactivity Management Related

TABLE OF CONTENTS

Page

OCONEE NUCLEAR STATION

Dropped Control Rods

1. Purpose

This procedure provides the actions necessary to maintain the plant in a safe condition following a dropped control rod(s).

2. Symptoms

- "CRD POSITION ERROR" statalarm (1SA-2, B-10)
- "CRD SAFETY RODS NOT AT UPPER LIMIT" statalarm (1SA-2, C-10)
- Control Rod "IN-LIMIT" light on (Green light on Position Indication Panel)
- Control Rod "ASYMM. RODS" light on (Yellow light on Diamond).

3. Automatic Systems Actions

3.1 **IF** ICS is in Auto,

AND an "ASYMM. RODS" (Yellow Light on Diamond) occurs,

THEN an "OUT" inhibit at 60% power is established and the Reactor will runback to 55% power.

3.1.1 **IF** the "ASYMM. RODS" (Yellow Light on Diamond) clears,

THEN runback may stop before reaching 55% power.

4. Immediate Manual Actions

_____ 4.1 **IF** more than one Control Rod has dropped,

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 4.2 **IF** more than one Control Rod is misaligned > 9" (6%),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.3 **IF** due to a malfunction, a Control Rod Group is misaligned > 9" (6%) from the acceptable region of PT/1/A/0600/001, (Periodic Instrument Surveillance),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.4 **IF** a Control Rod has dropped on an approach to criticality,

OR a dropped Control Rod results in a return to subcriticality from a critical condition,

THEN manually insert all Control Rods to Group 1 at 50% WD.

5. Subsequent Actions

_____ 5.1 **IF** the Reactor has tripped,

THEN **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 5.2 Verify Reactor runback <60% Full Power is in progress.:

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

NOTE 5.2.1:	If a control rod drops while the ICS Diamond Panel is in MANUAL, an "Auto Inhibit" (safety rod) or "Sequence Enable" (regulating rod) alarm will prevent placing the ICS in automatic.

_____ 5.2.1 **IF** the Reactor has **NOT** runback,

THEN commence manual Reactor Power reduction to < 60% of the allowable thermal power for the RCP combination.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.3 **IF** operating with only three (3)RCPs,

THEN commence manual Reactor Power reduction to < 45% Full Power.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.4 Notify I&E to begin investigation for the cause of the Dropped Control Rod.

_____ 5.5 Initiate actions to meet the Required Action of ITS 3.1.4 by performing the following:

_____ 5.5.1 Within one hour verify > 1% SDM with allowance for the inoperable control rod(s):

- Perform PT/1/A/1103/15, (Reactivity Balance Calculation).

_____ 5.5.2 Within two hours reduce Reactor Power < 60% of the allowable thermal power for the RCP combination.

NOTE 5.5.3:	Notification to I&E should be made as soon as possible due to the complexity of resetting the RPS trip setpoints and the short ITS allowable time limits.

_____ 5.5.3 Notify I&E to reduce RPS Flux/Flow-Imbalance trip setpoints to 65.5% of thermal power allowable for the RCP combination.

_____ 5.5.4 Notify I&E to reduce RPS High Flux trip setpoints to 65.5% of thermal power allowable for the RCP combination.

_____ 5.6 **WHEN** Reactor Power is < 60% of the allowable thermal power for the RCP combination,

THEN notify I&E to begin repair of the Dropped Control Rod.

_____ 5.7 **WHEN** I&E is ready to begin repairs on the Dropped Control Rod,

THEN Place the ICS Diamond control station in MANUAL,

AND permit I&E to repair Dropped Control Rod.

Dropped Control Rods

CAUTION:5.8	The Duty Reactor Engineer must evaluate the effects of local power distribution and the necessity for special maneuvering limits prior to the recovery of a dropped or an asymmetric Control Rod.

_____ 5.8 **WHEN** I&E has repaired the Dropped Control Rod,

THEN recover the Dropped Control Rod per OP/0/A/1105/009, (Control Rod Drive System).

END

Dropped Control Rods

Appendix

1. PIP # 0-O98-2734

END

(R06-97)

SR
Slm
NRC
115
JPP
JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. PT/1/A/1103/15
Revision No 51

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Reactivity Balance Procedure (Unit 1)

INFORMATION ONLY

(4) Prepared By Steve Cernuno Date 7/28/99

(5) Requires 10CFR50.59 evaluation?

- ☒ Yes (New procedure or revision with major changes)
- ☐ No (Revision with minor changes)
- ☐ No (To incorporate previously approved changes)

(6) Reviewed By TE Sanders (QR) Date 8/2/99

Cross-Disciplinary Review By Joe Price (QR) NA____ Date 8-16-99

Reactivity Mgmt. Review By TE Sanders (QR) NA____ Date 8/2/99

(7) Additional Reviews

Reviewed By ____________ Date ______

Reviewed By ____________ Date ______

(8) Temporary Approval *(if necessary)*

By ____________ (SRO/QR) Date ______

By ____________ (QR) Date ______

(9) Approved By [signature] Date 2/24/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ____________ Date ______

Compared with Control Copy ____________ Date ______

Compared with Control Copy ____________ Date ______

(11) Date(s) Performed ____________

Work Order Number (WO#) ____________

COMPLETION

(12) Procedure Completion Verification

- ☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
- ☐ Yes ☐ NA Listed enclosures attached?
- ☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
- ☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
- ☐ Yes ☐ NA Procedure requirements met?

Verified By ____________ Date ______

Procedure Completion Approved ____________ Date ______

(13) Remarks *(Attach additional pages, if necessary)*

FEB 2000
Received
Operator
Training

Duke Power Company Oconee Nuclear Station	Procedure No. PT/1/A/1103/15
Reactivity Balance Procedure (Unit 1)	Revision No. 51
*** This procedure has the potential to affect Reactivity Management *** **Continuous Use**	Electronic Reference No.

Reactivity Balance Procedure

1. Purpose

1.1 To calculate the Boron concentration necessary to provide greater than 1% ΔK/K shutdown margin.

1.2 To calculate the actual shutdown margin when the reactor is shutdown.

1.3 To evaluate the available shutdown margin during power operation (e.g., in the event of an inoperable rod.)

1.4 To provide the minimum RCS Boron concentration required to ensure greater than 1% ΔK/K shutdown margin to perform the Control Rod Drive (CRD) patch verification (for initial startup following refueling).

1.5 To estimate the critical rod configuration or the critical Boron concentration prior to startup.

1.6 To provide a method for preventing inadvertent criticality using subcritical multiplication measurement.

1.7 To provide nominal APSR position.

2. References

2.1 Improved Tech Specs:
- 1.1, Definitions - Shutdown Margin
- 3.1.1, Shutdown Margin
- 3.1.4, Control Rod Group Alignment Limits
- 3.1.5, Safety Rod Position Limits
- 3.2.1, Regulating Rod Position Limits
- 3.3.9, Source Range Neutron Flux
- 3.9.1, Boron Concentration

2.2 Selected Licensee Commitments: 16.13.4, Reactivity Anomalies

2.3 Unit 1 - Physics Test Manual (PTM)

2.4 Unit 1 - Core Operating Limits Report (COLR)

2.5 Nuclear Systems Directive 304. Reactivity Management

3. Time Required

Two people - 1 hour for most Enclosures

4. Prerequisite Tests

None

5. Test Equipment

Personal computer (for computerized calculations)

6. Limits and Precautions

6.1 Operations uses the results of this procedure to make important operational decisions, therefore this procedure affects core reactivity.

6.2 Appropriate corrections have been made per this procedure, or actual plant conditions must be the same as the reference conditions stated on the appropriate enclosure(s).

6.3 Independent verification is required for each calculation performed. For hand calculations, this requires that two people separately complete the appropriate enclosures for the desired calculation to verify the results are in agreement. For computerized calculations, this requires that two people separately run the computer code(s) or verify the input.

7. Required Unit Status

None

8. Prerequisite System Conditions

None

9. Test Method

9.1 Shutdown Boron Concentration:

Calculated in Enclosure 13.1 or 13.2.
The shutdown Boron concentration provides a greater than 1.0% $\Delta K/K$ shutdown margin with the worst case stuck rod assumed to be out and with conservatism applied per standard practice for Babcock & Wilcox 177 fuel assembly reactors.

A reference shutdown Boron concentration is obtained based on the cycle burnup, rod positions and RCS temperature. The reactivity worths of Xenon, Samarium, and the inoperable rod penalty (if applicable) are converted into their equivalent Boron concentrations. (Credit is taken only for the minimum Xenon worth occurring in a specified time interval, which should not exceed 12 hours. The Shutdown Boron concentration is valid ONLY during that time interval. Due to inaccuracies in the Xenon models, .8 times the Xenon and Samarium worth are used unless the RCS is below 450°F, in which case .5 times the Xenon and Samarium worths are used. Xenon and Samarium worths may be assumed to be zero for conservatism.) These Boron concentrations are then applied to the reference Boron concentration to provide the <u>required Boron concentration for a greater than 1.0% ΔK/K shutdown margin (i.e., the shutdown Boron concentration).</u>

9.2 Shutdown Margin Calculation while Shutdown:

Calculated in Enclosure 13.1 or 13.2.
The shutdown margin is the amount of reactivity by which the reactor is shutdown. The worst case stuck rod is assumed to be out, and additional conservatisms are applied per standard practice for Babcock and Wilcox 177 fuel assembly reactors. If operating with a known inoperable rod, an additional penalty is applied to account for that rod. This penalty need not be applied when the reactor is shutdown if that rod can be confirmed to be fully inserted by redundant indications. The shutdown Boron concentration must first be found per 9.1. The actual Boron concentration is then subtracted from this concentration and the result converted to % ΔK/K. 1.0% ΔK/K is then subtracted from this value to obtain the shutdown margin, expressed in - % ΔK/K. A separate check for SSF RC Makeup System operability is performed, which takes no credit for Xenon, but does not require the stuck rod penalty. This limit is shown in Enclosure 13.23.

Following a shutdown, Control Rod Position at the time of Shutdown may be used with the Rod Position Limit curves (in Enclosure 13.21) to verify at least 1% ΔK/K shutdown margin for the first 3 hours following shutdown (provided RCS Temperature stays >= 532°F and boron does not decrease). This may be necessary for shutdowns with an inoperable rod, since the more conservative calculation method (in Enclosure 13.1, 13.2) may not show 1% ΔK/K shutdown margin immediately after shutdown. Boration should begin immediately to be able to show 1% ΔK/K shutdown margin using the calculation method.

9.3 Shutdown Margin at Power:

Verified in Enclosure 13.20.
While at power, the available shutdown margin may be verified to be >= 1% ΔK/K by using the Rod Position Limits curves. Operation in the "Acceptable Region" of these

curves ensures that the shutdown margin following a reactor trip will be >= 1% ΔK/K with the worst stuck rod out. There are curves for 3 and 4 RCP operation, and curves for 0 and 1 inoperable rod. A dropped rod is considered inoperable for the purposes of providing shutdown margin while at power.

9.4 Estimated Critical Rod Position:

Calculated in Enclosure 13.3.
The core excess reactivity is obtained based on the cycle burnup. The reactivity worths associated with Boron, Xenon, temperature correction (if RCS temperature not at 532°F) and Samarium are then obtained and summed with the core excess reactivity. The groups 5-7 positions are then determined for which the inserted rod worth when summed with all the above, yields a total core reactivity of 0.0% ΔK/K. The upper and lower rod position limits are then determined and the actual critical rod positions are recorded.

9.5 Estimated Critical Boron Concentration:

Calculated in Enclosure 13.4.
The core excess reactivity is obtained based on the cycle burnup. The reactivity worth associated with Xenon, temperature correction (if RCS temperature not at 532°F), Samarium and the <u>desired</u> critical rod positions are summed with the core excess reactivity. The Boron concentration is then determined for which its reactivity worth, when summed with all the above, yields a total core reactivity of 0.0% ΔK/K.

9.6 Subcritical Multiplication Measurement:

Performed in Enclosure 13.6.
With Group 1 at 50% wd, an initial source range (SR) count rate (C_0) is recorded. During control rod withdrawals, new counts (C) are recorded and used to calculate 1/M, or C_0/C. As criticality is approached, C/C_0 will approach infinity, and 1/M will approach zero. Plotting 1/M versus rod worth provides a rough indication of what rod position will yield a critical condition, and acts as an indication of premature criticality, or criticality more than 0.75% ΔK/K below the Estimated Critical Position calculated in 9.4.

10. Data Required

10.1 For Xenon Worth: cycle burnup and power history to time of last equilibrium xenon.

10.2 For Shutdown Boron Concentration/Shutdown Margin Calculation: Power, cycle burnup, RCS temperature, Group 1 and 8 positions, Xenon worth and the actual boron concentration.

10.3 For Estimated Critical Rod Configuration: RCS temperature, cycle burnup, present boron concentration, Xenon worth, and Samarium worth.

10.4 For Estimated Critical Boron Configuration: RCS temperature, cycle burnup, desired critical rod configuration, Xenon worth, and Samarium worth.

10.5 For Subcritical Multiplication Measurement: Control Rod position and source range (SR) count rate.

11. Acceptance Criteria

Independent/Separate verifications should agree within 10 ppmB (for Shutdown Boron or Estimated Critical Boron) or 5%wd (for Estimated Critical Position). The more conservative Shutdown Boron Concentration calculation shall be used to ensure at least a 1.0% ΔK/K shutdown margin.

12. Procedure

Complete, or refer to, the appropriate enclosure(s):

Shutdown Margin Calculation at power:
Enclosure 13.20

Shutdown Margin Calculation while shutdown:
Enclosure 13.1, "Shutdown Boron Concentration/Shutdown Margin Calculation,"
OR-
Enclosure 13.2 "Computerized Shutdown Margin Calculation"

Refueling Outage Boron Concentrations:
Enclosure 13.14, "Refueling Outage Boron Concentrations"

Estimated Critical Rod Position:
Enclosure 13.3, "Computerized Estimated Critical Rod Position Calculation"

Estimated Critical Boron Concentration:
Enclosure 13.4, "Computerized Estimated Critical Boron Calculation"

Subcritical Multiplication (1/M) Measurement:
Enclosure 13.6, "Subcritical Multiplication (1/M) Measurement"

Instructions for obtaining Xenon Prediction:
Enclosure 13.16, "Instructions for Obtaining Xenon Prediction"

Required Control Rod Group 8 Position:
Enclosure 13.15, "Required Group 8 Position and Designed Cycle Length"

Designed Cycle Length Information:
Enclosure 13.15, "Required Group 8 Position and Designed Cycle Length"

RCS Boron Concentration for SSF Operability:
Enclosure 13.23, "Minimum RCS Boron Concentration to Maintain SSF Operability"

Required Shutdown Margin:
Enclosure 13.18, "Shutdown Margin Requirements"

NOTE:	Only the appropriate completed enclosures need be attached to the procedure cover sheet to be submitted for procedure completion.

13. Enclosures

13.1 Shutdown Boron Concentration/Shutdown Margin Calculation

13.2 Computerized Shutdown Margin Calculation

13.2.1 Computerized Shutdown Margin Calculation Documentation

13.3 Computerized Estimated Critical Rod Position Configuration

13.4 Computerized Estimated Critical Boron Concentration

13.5 Deleted

13.6 Subcritical Multiplication (1/M) Measurement

13.7 Core Excess Reactivity vs. Burnup

13.8 Differential Boron Worth vs. Burnup

13.9 Inserted Control Rod Worth (for 1/M measurement of Groups 1-7)

13.10 Temperature Coefficient vs. RCS Boron Concentration

13.11 Shutdown Boron Concentration vs. Burnup (Group 1 @ 0% wd)

Enclosure 13.20
Shutdown Margin Calculation at Power

Performed By: ____________________

> **NOTE:** A dropped rod is considered inoperable for the purpose of providing shutdown margin while at power.

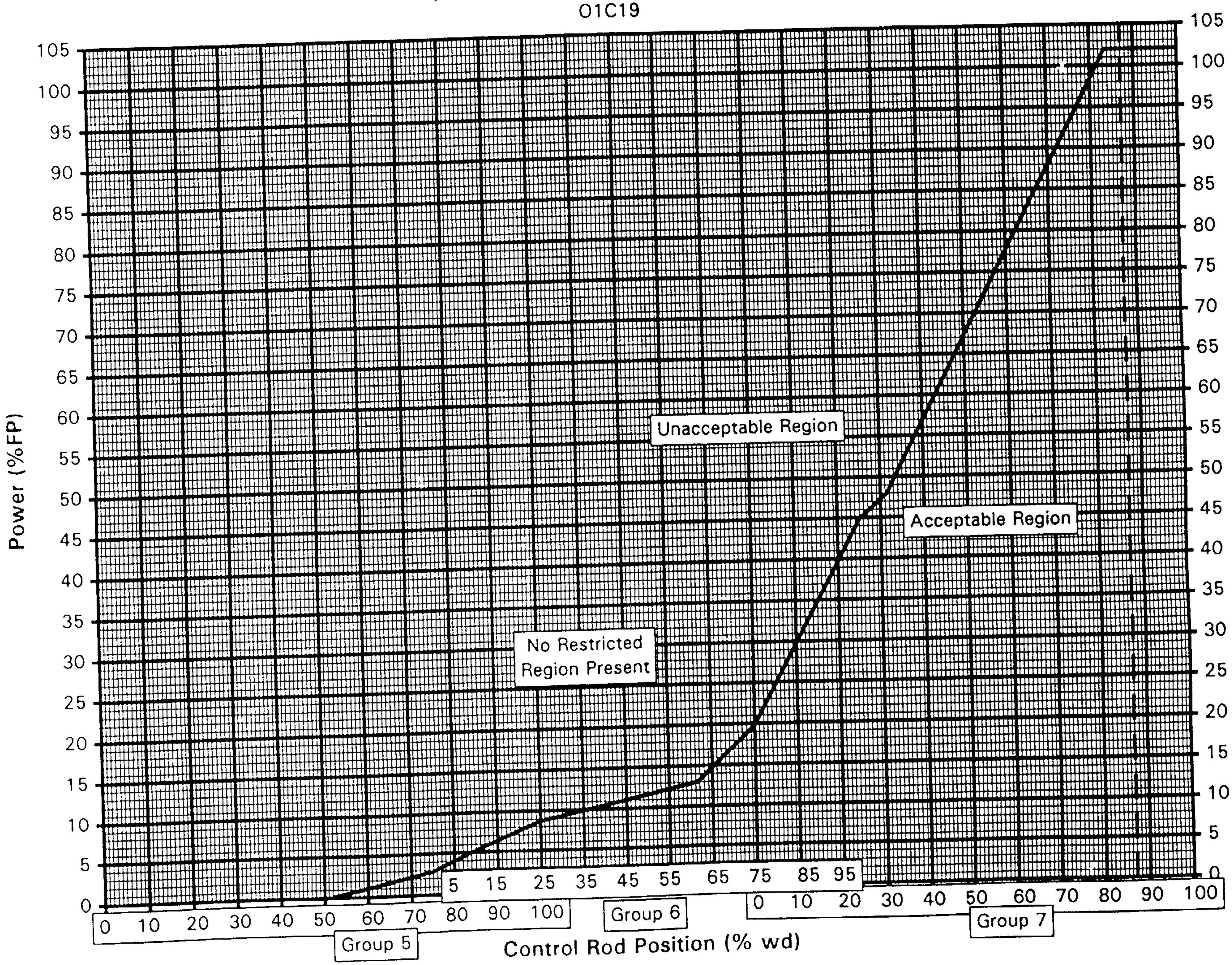
13.20.1 Verify one of the following:

_____ _____ 13.20.1.a Available shutdown margin is ≥1% ΔK/K. This is shown by verifying that the control rod position and power level are within the Acceptable Region or the Restricted Region on the appropriate curve for the number of RC Pumps and Inoperable rods in Enclosure 13.21, Rod Position Limits at Power.
IV

\- OR -

_____ _____ 13.20.1.b Appropriate actions are taken per ITS 3.1.4, 3.1.5 and 3.2.1.
IV

Rod Position Limits at Power
1 Inoperable Rod or 1 Dropped Rod - 4 Pump Flow
01C19



REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NRC-005/ADMIN A.1

REACTOR POWER IMBALANCE
Improved Technical Specifications/COLR

CANDIDATE ____________________

EXAMINER ____________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

Axial Power Imbalance

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

Gen 2.1.11   3.07/3.8

Task Standard:

Perform power imbalance within limits verification.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and Section 12.3
PT/0/A/1103/019, Backup Incore Detector System
Core Operating Limits Report

Validation Time: 25 min.   **Time Critical:** NO

==

Candidate: ________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: _________________________ ______________________________/________
NAME   SIGNATURE   DATE

==

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and Section 12.3
PT/0/A/1103/019, Backup Incore Detector System
Core Operating Limits Report

READ TO OPERATOR

DIRECTIONS TO STUDENT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

The Unit has been operating at 100% power for 2 weeks.
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 has been completed up to page 6, Axial Power Imbalance Operating Limits.

INITIATING CUE:

The SRO directs you to perform the Axial Power Imbalance Operating Limits verification.
The Reactor calculation package is NOT running.

START TIME: ______

STEP 1: Verify Power imbalance within operational alarm limit in COLR when > 40% RTP. **IF** Reactor calculation package is NOT running on computer, refer to Section 12.3. STANDARD: When told Reactor Calculation package not running, refer to Section 12.3. ***CUE: Tell candidate that the Reactor calculation package is not running.*** COMMENTS:	___ SAT ___ UNSAT
STEP 2: Axial Imbalance shall **NOT** exceed appropriate limit curve in COLR. **IF** axial imbalance limit is exceeded, take immediate corrective action to **IF** an acceptable imbalance is **NOT** achieved within 2 hours, reactor power shall be reduced until imbalance limits are met. Refer to TS 3.2.2. STANDARD: Candidate obtains the correct limit curve in COLR. This curve is located on page 12 of 31 (Oconee 1 Cycle 19) (Oconee 2 Cycle 18) (Oconee 3 Cycle 18) **NOTE: Later in JPM when imbalance calculation is made with the Incores a different enclosure from the COLR will be used.** ***CUE: Only Imbalance Surveillance is required for this JPM. Step 12.3.2 is not required.*** COMMENTS:	___ SAT ___ UNSAT

STEP 3: Order of preference of measurement systems to determine axial imbalance and quadrant power tilt is as follows: A. Incore Detectors (Computer Reactor Calculation Package). B. Outcore Detectors (Power Range Outcore Detectors). C. Backup Incore Detectors. Refer to PT/10/A/1103/019 (Backup Incore Detector System). ***CUE: The Backup Incore detectors will be used for this determination.*** STANDARD: Candidate refers to PT/10/A/1103/019 (Backup Incore Detector System). COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 4: Verification of minimum Incore operability. **NOTE: Backup Incore Chart "A" points and information provided to the student.** STANDARD: **NOTE: Give student Backup Incore Chart "A" data sheet.** **CUE: Inform candidate that all points on Backup Incore Chart "A" are operable (no points are off scale or contain a note).** COMMENTS:	___ SAT ___ UNSAT
STEP 5: 12.2.1 Verify the reactor has been at steady state conditions (± 2% FP) for at least 30 minutes. STANDARD: The Candidate determines is reactor power is steady. ***CUE: Reactor power has been at 100% power for the past 2 weeks.*** COMMENTS:	___ SAT ___ UNSAT

STEP 6: Calculate axial imbalance per Enclosure 13.3 using operable recorder points identified on Enclosure 13.1. STANDARD: The candidate refers to and obtains a copy of Enclosures 13.1 and 13.3 The candidate performs calculation per Enclosure 13.3. **NOTE: Refer to completed enclosure 13.3.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: Verify the calculated axial imbalance does not exceed the backup incore limits per 11.1. STANDARD: The candidate verifies the calculated axial imbalance does not exceed the backup incore limits per 11.1, (-18.7 / +18.7) the current Core Operating Limits Report (COLR) on the Backup Incore Setpoint Column of the (Error-Adjusted) "Operational Power Imbalance Setpoints" Table. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: If either step 12.2.4 or 12.2.5 cannot be satisfied, notify the Unit Supervisor and take appropriate actions described in the applicable Technical Specifications as listed below Quadrant Power Tilt - ITS 3.2.3 Axial Power Imbalance - ITS 3.2.2 ***CUE: Inform candidate that for this JPM only imbalance will be checked.*** STANDARD: Candidate determines that step 12.2.4 is satisfied. COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
3	Step is necessary, because reference to the must use Backup Incore System procedure must be used to determine imbalance.
6	Step is necessary, because calculation is needed to determine imbalance.
7	Step is necessary, because imbalance must be compared to COLR to verify within limits.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

The Unit has been operating at 100% power for 2 weeks.
PT/1/A/0600/001, Periodic Instrument Surveillances, Enclosures 13.1 has been completed up to page 6, Axial Power Imbalance Operating Limits.

INITIATING CUE:

The SRO directs you to perform the Axial Power Imbalance Operating Limits verification.
The reactor calculation package is NOT running.

Enclosure 13.1
Required Backup Recorder Points For Calculating Axial Power Imbalance

* UNIT 1 ONLY *

From the following list of allowed point groupings for calculating axial power imbalance, three sets for which the points indicated are all operable and in calibration must be available.

NOTES: "A" recorder is on the Control Board; "B" is on the vertical board.
A set of recorder points for imbalance calculations consist of three operable points grouped together.

Any of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	3	G9	6		B	8	L6	6
A	2	G9	4		B	7	L6	4
A	1	G9	2		B	6	L6	2
A	22	D5	6		B	19	N4	6
A	24	D5	4		B	18	N4	4
A	14	D5	2		B	17	N4	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	6	E9	6		B	4	K5	6
A	5	E9	4		B	5	K5	4
A	4	E9	2		B	10	K5	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	18	F13	6		B	22	O6	6
A	15	F13	4		B	21	O6	4
A	13	F13	2		B	20	O6	2

Recorded By ______________________ Date ______________

BACKUP INCORE CHART "A"		
Point #	%	Location
1	195.1	G09-L2
2	203.0	G09-L4
3	196.1	G09-L6
4	214.1	E09-L2
5	226.6	E09-L4
6	209.6	E09-L6
7	189.4	L06-L4
8	196.8	L06-L6
9	185.7	M09-L2
10	180.7	K05-L2
11	187.5	G11-L2
12	179.7	E07-L2
13	212.3	F13-L2
14	213.2	D04-L2
15	210.4	F13-L4
16	209.5	F03-L6
17	212.5	N04-L2
18	197.6	F13-L6
19	196.7	N04-L6
20	199.3	O06-L2
21	200.2	O06-L4
22	199.4	O06-L6
23	196.8	D05-L6
24	197.7	D05-L4

Enclosure 13.3
Axial Power Imbalance Calculation Sheet

13.3.1 Using Enclosure 13.1 choose a set of backup recorder points for which all points are operable and record the point identifications (recorder, point number, detector location, and level) on the blanks provided below.

13.3.2 Record the backup recorder readings on the blank provided below.

13.3.3 Record current % FP on the blanks provided below using the priority of thermal power indications listed in PT/0/A/0600/001 (usually either power range NIs or from the results of PT/0/A/0205/05, Thermal Power Calculation).

13.3.4 Calculate imbalance for each of the three detector strings using the following formula:

$$\text{IMB} = \frac{\text{R (level 6 or 5) - R (level 2 or 3)}}{\text{R (level 6 or 5) + R (level 4) + R (level 2 or 3)}} \times \text{\% FP}$$

where R = detector reading

13.3.5 The imbalance from the backup recorders is the average for the three detector strings, calculated as indicated below:

	RECORDER ID	POINT #	DETECTOR LOCATION	DETECTOR LEVEL	DETECTOR READING (R)
I	A	3	609-L6	6 or 5	196.1
	A	2	609-L4	4	203.0
	A	1	609-L2	2 or 3	195.1
II	A	6	E9-L6	6 or 5	209.6
	A	5	E9-L4	4	226.6
	A	4	E9-L2	2 or 3	214.1
III	A	18	F13-L6	6 or 5	197.6
	A	15	F13-L4	4	210.4
	A	13	F13-L2	2 or 3	212.3

$\text{IMB}_{I} = \dfrac{(196.1 - 195.1)}{(196.1 + 203.0 + 195.1)} \times 100$ %FP = .17 %IMB

$\text{IMB}_{II} = \dfrac{(209.6 - 214.1)}{(209.6 + 226.6 + 214.1)} \times 100$ %FP = -.69 %IMB

$\text{IMB}_{III} = \dfrac{(197.6 - 212.3)}{(197.6 + 210.4 + 212.3)} \times 100$ %FP = -2.37 %IMB

TOTAL -2.89 %IMB

AVERAGE IMBALANCE = TOTAL/3 = -.96 %IMB

Calculated by ________________ Date/Time ________________ Verified by ________________ Date/Time ________________

BACKUP INCORE CHART "A"		
Point #	%	Location
1	195.1	G09-L2
2	203.0	G09-L4
3	196.1	G09-L6
4	214.1	E09-L2
5	226.6	E09-L4
6	209.6	E09-L6
7	189.4	L06-L4
8	196.8	L06-L6
9	185.7	M09-L2
10	180.7	K05-L2
11	187.5	G11-L2
12	179.7	E07-L2
13	212.3	F13-L2
14	213.2	D04-L2
15	210.4	F13-L4
16	209.5	F03-L6
17	212.5	N04-L2
18	197.6	F13-L6
19	196.7	N04-L6
20	199.3	O06-L2
21	200.2	O06-L4
22	199.4	O06-L6
23	196.8	D05-L6
24	197.7	D05-L4

Enclosure 13.1
Required Backup Recorder Points For Calculating Axial Power Imbalance

* UNIT 1 ONLY *

From the following list of allowed point groupings for calculating axial power imbalance, three sets for which the points indicated are all operable and in calibration must be available.

NOTES: "A" recorder is on the Control Board; "B" is on the vertical board.
A set of recorder points for imbalance calculations consist of three operable points grouped together.

Any of the following strings may be used:

Recorder Number	Point Number	Core Location	Level	Recorder Number	Point Number	Core Location	Level
A	3	G9	6	B	8	L6	6
A	2	G9	4	B	7	L6	4
A	1	G9	2	B	6	L6	2
A	22	D5	6	B	19	N4	6
A	24	D5	4	B	18	N4	4
A	14	D5	2	B	17	N4	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level	Recorder Number	Point Number	Core Location	Level
A	6	E9	6	B	4	K5	6
A	5	E9	4	B	5	K5	4
A	4	E9	2	B	10	K5	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level	Recorder Number	Point Number	Core Location	Level
A	18	F13	6	B	22	O6	6
A	15	F13	4	B	21	O6	4
A	13	F13	2	B	20	O6	2

Recorded By ______________________ Date ______________

Enclosure 13.3
Axial Power Imbalance Calculation Sheet

13.3.1 Using Enclosure 13.1 choose a set of backup recorder points for which all points are operable and record the point identifications (recorder, point number, detector location, and level) on the blanks provided below.

13.3.2 Record the backup recorder readings on the blank provided below.

13.3.3 Record current % FP on the blanks provided below using the priority of thermal power indications listed in PT/0/A/0600/001 (usually either power range NIs or from the results of PT/0/A/0205/05, Thermal Power Calculation).

13.3.4 Calculate imbalance for each of the three detector strings using the following formula:

$$\text{IMB} = \frac{\text{R (level 6 or 5) - R (level 2 or 3)}}{\text{R (level 6 or 5) + R (level 4) + R (level 2 or 3)}} \times \text{\% FP}$$

where R = detector reading

13.3.5 The imbalance from the backup recorders is the average for the three detector strings, calculated as indicated below:

	RECORDER ID	POINT #	DETECTOR LOCATION	DETECTOR LEVEL	DETECTOR READING (R)	
I	______	______	______	6 or 5	______	$\text{IMB}_{\text{I}} = \frac{(\_\_\_\_ - \_\_\_\_)}{(\_\_\_ + \_\_\_ + \_\_\_)} \times$ ____ %FP = ____ %IMB
	______	______	______	4	______	
	______	______	______	2 or 3	______	
II	______	______	______	6 or 5	______	$\text{IMB}_{\text{II}} = \frac{(\_\_\_\_ - \_\_\_\_)}{(\_\_\_ + \_\_\_ + \_\_\_)} \times$ ____ %FP = ____ %IMB
	______	______	______	4	______	
	______	______	______	2 or 3	______	
III	______	______	______	6 or 5	______	$\text{IMB}_{\text{III}} = \frac{(\_\_\_\_ - \_\_\_\_)}{(\_\_\_ + \_\_\_ + \_\_\_)} \times$ ____ %FP = ____ %IMB
	______	______	______	4	______	
	______	______	______	2 or 3	______	

TOTAL ____ %IMB

AVERAGE IMBALANCE = TOTAL/3 = ______ %IMB

Calculated by ____________ Date/Time ______________ Verified by ____________ Date/Time ______________

(R06-97)

SR
Sum
NRC
115
JPP
JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No PT/1/A/0600/001
Revision No 217

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Periodic Instrument Surveillance

(4) Prepared By William M. Buchanan (Signature) [signature] Date 04/11/00

(5) Requires 10CFR50.59 evaluation?
☐ Yes (New procedure or revision with major changes)
☒ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By [signature] (QR) Date 4/12/00
Cross-Disciplinary Review By ______ (QR)NA [initials] Date ______
Reactivity Mgmt. Review By ______ (QR)NA [initials] Date ______

(7) Additional Reviews
Reviewed By (IT) Alan Sweeney (Time Sync Only) Date 4/10/00
Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*
By ______ (SRO/QR) Date ______
By ______ (QR) Date ______

(9) Approved By [signature] Date 4/12/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______
Compared with Control Copy ______ Date ______
Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______
Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification:
☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?
☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

APR 2000
Received
Operator
Training

Duke Power Company Oconee Nuclear Station **Periodic Instrument Surveillance**	Procedure No. PT/1/A/0600/001
	Revision No. 217
Continuous Use	Electronic Reference No. OX002WAS

Periodic Instrument Surveillance

1. Purpose

1.1 To periodically verify proper operation of various instruments and systems.

2. References

2.1 Technical Specifications (TS)

2.2 DPC/Oconee Nuclear Station Core Operating Limits Report (COLR)

2.3 UFSAR Chapter 16 Selected Licensee Commitments (SLC)

3. Time Required

3.1 90 minutes per shift

4. Prerequisite Tests

None

5. Test Equipment

None

6. Limits And Precautions

6.1 This procedure controls activities that have the potential to affect reactivity. Major changes to this procedure shall be reviewed by a Qualified Reviewer to determine if a cross-disciplinary review for Reactivity Management concerns is needed as required by NSD 304 (Reactivity Management).

6.2 Failure to meet the required conditions may be a violation of TS or SLC. If so, it must be reported immediately to OPS Duty Person, Superintendent of Operations, or Station Manager.

6.3 OP/0/A/1103/020 (Loss Of Computer) should be referred to upon a loss of computer or loss of a computer point or function needed for a surveillance.

6.4 When changing Modes, <u>**ALL**</u> TS, SLC, and Surveillance Requirements (SR) prior to changing Modes of operation shall be performed. <u>**ALL**</u> TS, SLC, and Surveillance Requirements (SR) (Semi-Daily, Daily, Weekly, and Monthly) must be initialed prior to changing modes. When the surveillance is completed, initial in the block for the shift you are working, regardless of the time of day, week or month, even if there is not a (N), (D), 1st day of month, etc., indicated in the block.

7. Required Unit Status

7.1 Surveillance of instrumentation per Enclosure "Mode 1 & 2" required:

Prior to entering Mode 2 from Mode 3

AND

During operation in either Mode 1 or Mode 2.

7.2 Surveillance of instrumentation per Enclosure "Mode 3" required:

Prior to entering Mode 3 from Mode 4

AND

During operation in Mode 3.

7.3 Surveillance of instrumentation per Enclosure "Mode 4" required:

Prior to entering Mode 4 from Mode 5

AND

During operation in Mode 4.

7.4 Surveillance of instrumentation per Enclosure "Mode 5" required:

Prior to entering Mode 5 from Mode 6

AND

During operation in Mode 5.

7.5 Surveillance of instrumentation per Enclosure "Mode 6" required:

Prior to entering Mode 6 from No Mode

AND

During operation in Mode 6.

7.6 Surveillance of instrumentation per Enclosure "No Mode" required:

During operation in No Mode.

7.7 Surveillance per Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)" is required when 1LT-5 < 50" and core contains any irradiated fuel.

7.7.1 Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)" is performed in parallel with Enclosure "Mode 5" or "Mode 6".

8. Prerequisite System Conditions

None

9. Test Method

9.1 Component checks will be made according to information given in the following enclosures:

Enclosure "Modes 1 & 2"

Enclosure "Mode 3"

Enclosure "Mode 4"

Enclosure "Mode 5"

Enclosure "Mode 6"

Enclosure "No Mode"

Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)".

10. Data Required

10.1 Data requirements specified in Enclosure "Modes 1 & 2", Enclosure "Mode 3", Enclosure "Mode 4", Enclosure "Mode 5", Enclosure "Mode 6", Enclosure "No Mode", or Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)".

11. Acceptance Criteria

11.1 Systems or components meet TS or SLC requirements applicable to surveillance step.

11.2 Any discrepancy noted during performance of this test shall show corrective action taken.

12. Procedure

12.1 As required, perform component checks according to schedule specified in the following enclosures:

- Enclosure "Modes 1 & 2"
- Enclosure "Mode 3"
- Enclosure "Mode 4"
- Enclosure "Mode 5"
- Enclosure "Mode 6"
- Enclosure "No Mode"
- Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)"

12.1.1 For surveillances required per TS and/or SLC: (D) indicates between 0730-1030 hours; (N) indicates between 1930-2230 hours.

- Example: "N/Tuesday" means signoff shall be completed sometime between Tuesday at 1930 hours and Tuesday at 2230 hours.

12.1.2 For surveillances <u>NOT</u> required per TS and/or SLC: (D) indicates between 0700-1900 hours; (N) indicates between 1900-0700 hours.

- Example: "N/Tuesday" means signoff shall be completed sometime between Tuesday at 1900 hours and Wednesday at 0700 hours.

12.1.3 Instrument operation may be checked either by reading appropriate recorder, gauge, etc., or by selecting computer point ID, where applicable.

12.2 <u>IF</u> any component supplying input to RPS or ES channels fails to meet its Required Condition (i.e., is "out of tolerance"), initiate the following action:

12.2.1 ES Instrument

A. Check other analog channels to see if any other channel is tripped.

NOTE:	For determining appropriate TS condition a tripped channel is considered inoperable.

B. <u>IF</u> no other analog channel is tripped, trip affected analog channel by placing instrument channel for affected parameter (RC pressure or RB pressure) in "TEST-OPERATE". Affected parameter(s) should be left in "TEST-OPERATE" until channel input(s) is repaired.

C. <u>IF</u> any other analog channel is tripped, do **<u>NOT</u>** trip affected channel. Initiate immediate action to have instrument repaired. Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits).

D. Immediate shutdown may be required.
- Refer to TS 3.3.5.
- Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits).

12.2.2 RPS Instrument

A. <u>IF</u> no other RPS channel is in MANUAL BYPASS or no other RPS channel contains a DUMMY BISTABLE, place affected RPS channel in MANUAL BYPASS. Initiate action to have instrument channel repaired.

NOTE:	For determining appropriate TS condition a tripped channel is considered inoperable.

B. <u>IF</u> another RPS channel is in MANUAL BYPASS or contains a DUMMY BISTABLE, trip affected RPS channel by placing any one of its instrument channels in "TEST-OPERATE" (for STAR Modules select "TEST"). Affected parameter(s) should be left in "TEST-OPERATE" (or "TEST") until channel input(s) is repaired. Initiate <u>immediate</u> action to have instrument channel repaired.

C. <u>IF</u> affected RPS channel is already in MANUAL BYPASS, do **<u>NOT</u>** trip affected RPS channel. Initiate action to have instrument channel repaired.

D. <u>IF</u> affected RPS channel contains a DUMMY BISTABLE and no other RPS channel is in MANUAL BYPASS, place affected RPS channel in MANUAL BYPASS. TS allows any one RPS channel to contain more than one DUMMY BISTABLE.

E. <u>IF</u> another RPS channel is tripped, do **<u>NOT</u>** trip affected RPS channel. Initiate <u>immediate</u> action to have instrument channel repaired. Tripping affected RPS channel will cause a reactor trip.

F. Immediate shutdown may be required.
- Refer to TS 3.3.1
- Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits)

12.2.3 Priority of Power Indications to Use for Surveillance.
(A = highest priority, G = lowest priority)

A. OAC Calculated Thermal Power Best.

- O1P2037 (Core Thermal Power Best (60 min avg)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0588 (Core Thermal Power Best (10 min. avg.)) - Transient in last 60 minutes or O1P2037 unavailable.
- O1P0889 (Core Thermal Power Best (snapshot)) - Transient in progress or O1P2037 and O1P0588 unavailable.

B. OAC Calculated Thermal Power Secondary (if above ≈ 25% power).

- O1P0587 (Core Thermal Power Secondary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0888 (Core Thermal Power Secondary (snapshot)) - Transient within the last 60 minutes or O1P0587 unavailable.

C. OAC Calculated Thermal Power Primary (if below ≈ 25% power).

- O1P0576 (Core Thermal Power Primary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0887 (Core Thermal Power Primary (snapshot)) - Transient within the last 60 minutes or O1P0576 unavailable.

D. OAC Calculated Thermal Power ΔT.

- O1P0575 (Core Ther Pwr From Delta Temp (10 min. avg.)) - Steady State Ops. (i.e., no transient in last 10 minutes)
- O1P0326 (Core Thermal Power From Delta T (snapshot)) - Transient within the last 10 minutes or O1P0575 unavailable.

E. Alternate method for determining (%) Reactor Power

$$\frac{NI\text{-}5 + NI\text{-}6 + NI\text{-}7 + NI\text{-}8}{4} = \%\ \text{Rx Power (Avg)}$$

F. Hand-Calculated Thermal Power ΔT using Enclosure "Loop ΔT Vs Reactor Power".

G. Thermal Power from Nuclear Engineering using PT/0/A/0205/002 (Thermal Power Calculation).

12.3 Reactor Power Axial Imbalance and Quadrant Power Tilt

12.3.1 Axial Imbalance shall **NOT** exceed appropriate limit curve in COLR..

A. **IF** axial imbalance limit is exceeded, take immediate corrective action to achieve an acceptable imbalance.

B. **IF** an acceptable imbalance is **NOT** achieved within 2 hours, reactor power shall be reduced until imbalance limits are met. Refer to TS 3.2.2.

12.3.2 Quadrant Power Tilt (QPT) shall **NOT** exceed appropriate positive (+) limit in COLR.

A. **IF** QPT limit is exceeded, take immediate corrective action to achieve an acceptable QPT. Refer to TS 3.2.3. .

B. Alternate method for determining QPT:

$$QPT = 100\left[\frac{\text{power in any quadrant}}{\text{Avg. power of all quadrants}} - 1\right]$$

12.3.3 Order of preference of measurement systems to determine axial imbalance and quadrant power tilt is as follows:

A. Incore Detectors (Computer Reactor Calculation Package).

B. Outcore Detectors (Power Range Outcore Detectors).

C. Backup Incore Detectors. Refer to PT/0/A/1103/019 (Backup Incore Detector System).

12.3.4 **IF** at least one power range outcore detector is **NOT** operable in each quadrant, outcore detectors shall **NOT** be used to measure axial imbalance or quadrant power tilt.

12.3.5 **IF** Outcore Detectors (Power Range Outcore Detectors) are needed for tilt calculations, contact Rx Engineering group to perform PT/0/A/1103/018 (Excore Tilt Calculations).

12.3.6 **IF** Outcore Detectors (Power Range Outcore Detectors) are needed for imbalance calculations, refer to the following alternate method for determining (%) Reactor Power Axial Imbalance:

$$\frac{\text{NI-5* + NI-6* + NI-7* + NI-8*}}{4} = \text{\% Imbalance (Avg.)}$$

* Use Imbalance CR gauges reading for each NI.

12.3.7 **IF** Reactor Calculations package is **NOT** running, verify minimum incore detector operability requirements are met. Refer to PT/0/A/1103/019 (Backup Incore Detector System).

NOTE:	"Steady Conditions" defined as: Operating at a constant power level with no rod motion due to xenon and no plans to change power level in next 24 hours.

12.4 **WHEN** operating at a steady condition above 40% FP:

12.4.1 Control Rods should be positioned at or above dashed vertical lines designating Steady State Operating Bands in COLR.

A. Maneuvering restrictions on Control Rod and APSR movement in OP/1/A/1102/004 (Operation At Power) have priority over 24 hour time limit to resume operation in Steady State Operating Bands.

B. **IF** Control Rod position limits are exceeded, (i.e., operating in restricted region), corrective action shall be taken immediately to achieve an acceptable control rod position. TS 3.2.1 requires an acceptable control rod position be attained within 2 hours.

12.4.2 APSRs should be positioned as required per Enclosure "Required Group 8 Position" of PT/1/A/1103/015 (Reactivity Balance Procedure).

12.4.3 **IF** plant operating conditions or imbalance control requirements prevent steady operation within Control Rod Steady State Operating Bands, contact Systems Engineering/Reactor Group.

12.5 SASS (Smart Automatic Signal Selector) Auto Operation

12.5.1 SASS for Pzr level looks at Pzr level 1, 2, or 3. If level 1 or 2 fails, SASS will AUTO swap to Pzr level 3.

12.5.2 **IF** "AUTO" light is off, "MISMATCH" light is on, and "TRIP 'A'" or "TRIP 'B'" light is on, a SASS trip has occurred.

A. Controlling signal will be signal which does **NOT** have a "TRIP" light illuminated.

NOTE:	Failure to swap switch to valid signal could result in failed signal feeding through if SASS is reset before signal is repaired.

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbuttons for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.5.3 **IF** "AUTO" light is off and "MISMATCH" light is on, a mismatch has occurred.:

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.5.4 Initiate a Work Request to repair faulty signal.

12.5.5 Following repair of faulty signal, reset SASS by pushing "RESET" button. The following should occur:

A. SASS should swap to "AUTO". "AUTO" light should be illuminated, "TRIP 'A' or "TRIP 'B'" light and "MISMATCH" light should be off.

B. Controlling signal should remain unchanged.

12.6 SASS (Smart Automatic Signal Selector) Manual Operation

12.6.1 **IF** "MISMATCH" light is on and "TRIP 'A'" or "TRIP 'B'" light is on, a SASS trip has occurred.

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.6.2 **IF** "MISMATCH" light is on, a mismatch has occurred.:

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.6.3 Initiate a Work Request to repair faulty signal.

12.6.4 Following repair of faulty signal, reset SASS by pushing "RESET" button. The following should occur:

A. SASS should swap to "AUTO". "AUTO" light should be illuminated, "TRIP 'A'" or "TRIP 'B'" light and "MISMATCH" light should be off.

B. Controlling signal should remain unchanged.

12.7 AMSAC/DSS

12.7.1 Refer to the following indications to determine normal status of AMSAC/DSS:

- AMSAC CH 1 and CH 2 **NOT** actuated

  (O1D2928, O1D2929, 1SA-8 D-5/D-8)

- DSS CH 1 and CH 2 **NOT** actuated

  (O1D2930, O1D2931, 1SA-8 C-9/C-10)

- AMSAC/DSS CH 1 and CH 2 **NOT** bypassed

  (O1D2932, O1D2933, Indicating Lights on 1UB1)

- AMSAC/DSS Enabled

  (Indicating Light on 1UB1)

- AMSAC/DSS CH 1 **AND** CH 2 UPS Normal

  (O1D2934, O1D2935)

- "Sy Max" Programmable Controllers

CH 1 AMSAC/DSS	CH 2 AMSAC/DSS
RUN Light (ON)	RUN Light (ON)
HALT Light (OFF)	HALT Light (OFF)

12.7.2 AMSAC/DSS UPS (Uninterruptable Power Supply) has been upgraded with new firmware.

A. UPS will generate an alarm if noise is encountered on its power supply. However, it will automatically re-assess input power supply quality and, if transient has passed, it will reset and clear its alarm.

B. UPS will still generate UPS Trouble alarm. However, it will be more likely to clear automatically without operator intervention.

C. **IF** UPS Trouble alarm does **NOT** automatically reset, issue a Work Request.

12.7.3 IF all of the following conditions are met, AMSAC/DSS may be considered operable:

A. Surveillance requirements of SLC 16.7.2 (Anticipated Transients Without Scram) are satisfied.

B. AMSAC/DSS CH 1 and AMSAC/DSS CH 2 are enabled.

C. AMSAC/DSS CH 1 **AND** AMSAC/DSS CH 2 are capable of generating intended EFDW start signals, control rod drop signals, turbine trip signal, and TBV setpoint shift signal.

- To satisfy these criteria, all AMSAC/DSS circuitry (including input pressure switches/pressure transmitters, electrical isolation devices, logic circuits, programmable controllers, and uninterruptible power supplies) shall be functional and properly calibrated.

D. "Sy Max" Programmable Controllers "RUN" Lights (ON) and "HALT" Lights (OFF) for AMSAC/DSS CH 1 and AMSAC/DSS CH 2.

12.7.4 Inability of EFDW pumps, turbine trip circuit, or control rods to respond to an AMSAC/DSS signal does **NOT** constitute inoperability of AMSAC/DSS system. These malfunctions are governed by applicable TS.

- TS 3.7.5 and TS 3.3.14 for inoperable Emergency Feedwater Pumps or existing Initiation Circuitry.
- TS 3.3.15 for inoperable Turbine Stop Valve closure circuitry.
- TS 3.1.4 for inoperable control rod(s).

12.7.5 IF one or both channels of AMSAC/DSS are inoperable **AND** reactor is critical, refer to SLC 16.7.2. Notify Compliance of inoperabilities extending beyond seven days.

12.7.6 IF any AMSAC/DSS channel is inoperable or generates an invalid trip signal, bypass both AMSAC/DSS channels from control panel in AHU Room located on 6th floor above Units 1 & 2 CR.

A. **IF** reactor is critical, declare AMSAC/DSS system inoperable **AND** refer to SLC 16.7.2. Initiate a Work Request to repair affected channel.

12.7.7 **WHEN** AMSAC/DSS channel has been repaired, return AMSAC/DSS channels to service per Enclosure "Return To Service Of AMSAC/DSS".

12.8 MSLB

12.8.1 __IF__ one or both trains of MSLB do __NOT__ meet Surveillance Requirements:

A. Refer to TS 3.3.11, 3.3.12 and/or 3.3.13 for appropriate TS Condition for inoperability that is indicated.

B. __IF__ entry into condition A of TS 3.3.11 indicated, Immediately Notify I&E to perform IP/0/A/0270/003 (Main Steam Line Break (MSLB) Loss Of An Analog Channel Trip/Restoration) to trip affected channel and prevent entry into condition B of TS 3.3.11.

C. Initiate a Priority Work Request.

D. Initiate a PIP and contact Accountable Systems Engineer.

12.9 Dixon Indicators

12.9.1 Dixons listed on Enclosure "Dixon Meter Information" are on an enhanced surveillance interval.

12.9.2 __IF__ any Dixon listed on Enclosure "Dixon Meter Information" are found to be blinking with a reading of zero, no action is required unless a failure is suspected.

12.9.3 Dixons __NOT__ listed on Enclosure "Dixon Meter Information" are on a surveillance interval. These Dixons have alternate methods of verifying input signal is valid.

12.10 __WHEN__ a computer point needed for a surveillance is __NOT__ available, refer to OP/0/A/1103/020 (Loss Of Computer).

13. Enclosures

13.1 Mode 1 & 2

13.2 Mode 3

13.3 Mode 4

13.4 Mode 5

13.5 Mode 6

13.6 No Mode

13.7 Minimum Temperature For Criticality Surveillance Sheet

13.8 RCS Pressure, Temperature, Heatup And Cooldown Rates Surveillance Sheet

13.9 Pzr Level For LTOP Surveillance Sheet

13.10 RCP Power Supply Verification

13.11 LPI Pump Power Supply Verification

13.12 Loop ΔT Vs Reactor Power

13.13 Gross Load Vs Reactor Power

13.14 Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)

13.15 Return To Service Of AMSAC/DSS

13.16 ICCM Subcooling Monitor Check

13.17 Surveillance Evaluation

13.18 Dixon Meter Information

13.19 Hot Lake Water Surveillance

NOTE:	If Reactor calculations package is <u>NOT</u> running on computer, section 12.3 contains guidance.
NOTE:	If Reactor calculations package is running properly on computer, NAS Loop Counter should differ by ≈ 24 every two hours.
NOTE:	Step 1.1 contains the priority of indications to use for (%) Reactor Power.

TIME	% Reactor Power	NAS Loop Counter O1P5504	INITIALS					
			Step 1	Step 2	Step 3	Step 4	Step 5	RCP Seal Leakoff Flow
2000						N/A		
2200						N/A		
0000								
0200						N/A		
0400						N/A		
0600						N/A		
0800						N/A		
1000						N/A		
1200						N/A		
1400						N/A		
1600						N/A		
1800						N/A		

1. **IF** Thermal Power Best indicates "Good", verify Core Thermal Power Indication (every 2 hours when Rx critical.)

1.1 Priority of Power Indications to Use for Surveillance (A = highest priority, G = lowest priority):

A. OAC Calculated Thermal Power Best.

- O1P2037 (Core Thermal Power Best (60 min avg)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0588 (Core Thermal Power Best (10 min. avg.)) - Transient in last 60 minutes or O1P2037 unavailable.
- O1P0889 (Core Thermal Power Best (snapshot)) - Transient in progress or O1P2037 and O1P0588 unavailable.

B. OAC Calculated Thermal Power Secondary if above ≈ 25% power.

- O1P0587 (Core Thermal Power Secondary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0888 (Core Thermal Power Secondary (snapshot)) - Transient within last 60 minutes or O1P0587 unavailable.

C. OAC Calculated Thermal Power Primary if below ≈ 25% power.

- O1P0576 (Core Thermal Power Primary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0887 (Core Thermal Power Primary (snapshot)) - Transient within last 60 minutes, or O1P0576 unavailable.

D. OAC Calculated Thermal Power Delta T.

- O1P0575 (Core Ther Pwr From Delta Temp (10 min. avg.)) - Steady State Ops. (i.e., no transient in last 10 minutes)
- O1P0326 (Core Thermal Power From Delta T (snapshot)) - Transient within last 10 minutes or O1P0575 unavailable.

E. Alternate method for determining (%) Reactor Power

$$\frac{\text{NI-5} + \text{NI-6} + \text{NI-7} + \text{NI-8}}{4} = \text{\% Rx Power (Avg)}$$

F. Hand-Calculated Thermal Power Δ T using Enclosure "Loop ΔT Vs Reactor Power".

G. Thermal Power from Rx Engineering using PT/0/A/0205/002 (Thermal Power Calculation).

1.2 **IF** Thermal Power Best indicates "Bad", enter "NIS" (**NOT** In Service) and initials in appropriate block (s). Refer to OP/0/A/1103/020 (Loss Of Computer) for other actions.

NOTE:	If either step 1.3 or 1.4 is **NOT** satisfied, Duty Rx Engineer should perform verification of computer calculated TPB indication prior to calibrating NIs.

1.3 Verify Thermal Power Best (TPB) within ± 2.0% Rx Power of percent power from ΔT:

1.3.1 Refer to step 1.1 for priority of indications to determine percent power from ΔT,

OR

1.3.2 Average two RC Loop ΔTs from RC Loop ΔT gauge and using Enclosure "Loop ΔT Vs Reactor Power" determine percent power from ΔT.

1.4 Verify current Gross Load does **NOT** exceed value given by Enclosure "Gross Load Vs Reactor Power" for current Rx power level. Obtain Gross Load from O1P0963 **OR** Watt/Var meter if O1P0963 is **NOT** available.

2. Review Shift Turnover Sheet every 2 Hours.

2.1 Review Enclosure "Shift Turnover Sheet" to verify all turnover items updated and all required testing/surveillance items resulting from a degraded Mode per TS performed.

3. **<u>IF</u>** > 90% RTP and Steady State, **<u>AND</u>** fouling coefficient is less than 1.0, verify every 2 Hours O1P0576 (Core Thermal Power Primary (60 min avg)) does **<u>NOT</u>** exceed O1P0587 (Core Thermal Power Secondary (60 min avg)) by more than 0.2% RTP (i.e., O1P0576 < O1P0587 + 0.2).

   3.1 **<u>IF</u>** fouling coefficient is less than 1.0, **<u>AND</u> <u>IF</u>** O1P0576 (Core Thermal Power Primary (60 min avg)) exceeds O1P0587 (Core Thermal Power Secondary (60 min avg)) by more than 0.2% RTP, contact Duty Rx Engineer.

4. Obtain and call in Daily Dispatcher Readings.

CAUTION: If LDST ≥ 130 °F, HPI System is inoperable.

5. Verify LDST temperature < 120 °F. (CP O1A1240)

6. Initial when verification of steps 1, 2, 3, 4, and 5 completed.

7. Procedure for Periodic Checks:

- Review all in-progress Surveillance Evaluation enclosures in Tech Spec R&R Book:
  - (N) Verify all corrective/compensatory actions still valid. (e.g., WRs, WOs, PIPs open; procedure change(s) **<u>NOT</u>** yet implemented)
- No surveillance or completion time exceeded.
- Update in-progress Surveillance Evaluations by one-lining, initialing, and dating as required (change WR numbers to WO numbers, update resolution times).
  - RO and SRO sign updated line per step 9.1 of Enclosure "Surveillance Evaluation".
- Close out Surveillance Evaluations no longer applicable (e.g., corrective actions completed, TS/SLC no longer applicable).
- Attach completed (closed out) Surveillance Evaluations to this procedure.

- Perform periodic checks as specified.
  - __IF__ check can be performed as specified and is satisfied, initial appropriate block.
  - __IF__ check __CANNOT__ be performed as written or is __NOT__ satisfied, perform the following:
    - __IF__ Surveillance Evaluation is __NOT__ outstanding for check, perform Enclosure "Surveillance Evaluation".
    - Record Surveillance Evaluation in effect in appropriate block for any periodic checks with Surveillance Evaluations issued.
    - Attach a copy of Surveillance Evaluation issued this shift to this procedure.
    - List Surveillance Evaluations in effect in Remarks section of Procedure Process Record.
  - Place Surveillance Evaluations initiated this shift in Tech Spec R&R Book.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.7.6.1 12 Hours SR 16.10.1.1 6 Hours	CST, UST, and HW	______ 1930-2100 ______ 0130-0300	______ 0730-0900 ______ 1330-1500	O1E2250 O1A0152	Verify Combined Inventory for EFW in Acceptable Operation Region of Enclosure "Combined Inventory for EFW" of OP/0/A/1108/001 (Curves And General Information).
SR 3.7.6.1 12 Hours	CST, UST, and HW	______ 1930-2100 ______ 0130-0300	______ 0730-0900 ______ 1330-1500	O1E2250 O1E2295	Verify UST level > 6 ft. (done on 6 hr frequency)
SR 3.4.1.3 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1970	Verify RCS total flow within limits in COLR. Mode 1 only, Steady State Operation
SR 3.2.2.1 12 Hours	Axial Power Imbalance Operating Limits	(N)	(D)	O1P0877	Verify Power imbalance within operational alarm limits in COLR when > 40% RTP. **IF** Reactor calculations package is **NOT** running on computer, refer to Section 12.3. **IF** % Rx Power Imbalance changes > 2% during Steady State Operations, contact Rx Engineering **IF** NI calibration is required under these conditions, contact Rx Engineering.
SR 3.2.3.1 7 Days	QPT	(N)	(D)	O1P0737 O1P0738 O1P0739 O1P0740	Verify QPT within limits in COLR when > 20% RTP. (done on 12 hr frequency) **IF** Reactor calculations package is **NOT** running on computer, refer to Section 12.3.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.3.1.1 12 Hours	RPS Instrumentation NI Power Range NI-5, 6, 7, 8, 9	(N)	(D)	O1A1544 O1A1545 O1A1546 O1A1547 O1A1548	Verify computer readouts agree within 2%. (4% in RPS Cab) **NOTE:** **IF** the channels are off scale, the channel check will only verify that they are off scale in the same direction. (TS Bases SR 3.3.1.1)
SR 3.3.1.2 24 Hours	RPS Instrumentation Heat Balance Check Power Range Amplifiers	(N)	(D)	O1P0889	Verify TPB does **NOT** exceed NI-5, 6, 7, 8 or 9 by more than 2% power. (done on 12 hr frequency) Calibrate NIs when TPB ≥ 2% above any two of the power range NIs. Do **NOT** exceed ≥ 4% in non-conservative direction. **IF** TPB indicates "Bad", contact Duty Rx Engineer to calculate core thermal power per PT/0/A/0205/002 (Thermal Power Calculation). Mode 1 only, **NOT** required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP.
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation RB Pressure Narrow Range	(N)	(D)	O1A1566 O1A1286 O1A1287	Verify computer readouts agree within 0.6 psi (2 psi in ES Cab). **IF** readouts differ by > 0.4 psi, issue a Priority "E" Work Request.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.6.4.1 12 Hours	Containment Pressure NR RB Pressure	(N)	(D)	O1A1566 O1A1286 O1A1287	Verify RB pressure ≥ -2.45 psig but ≤ 1.2 psig. **IF** > + 0.6 psig, depressurize RB prior to > + 0.8 psig per OP/1/A/1102/014 (RB Purge). **IF** ≤ -0.5 psig, notify MCE for operability evaluation. {PIP 98-3976 & OSC-4476}
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Pressure Narrow Range	(N)	(D)	O1A1688 O1A1689 O1A1690 O1A1691	Verify computer readouts agree within 26 psi (48 psi in RPS Cab).
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Temperature T_H	(N)	(D)	O1A1692 O1A1693 O1A1694 O1A1695	Verify computer readouts agree within 3°F (5°F in RPS Cab). **IF** any of CR RCS temperature selectors are changed, notify Rx Engineering to evaluate and update Enclosure "Loop ΔT Vs. Reactor Power" for new selected inputs.
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Flow	(N)	(D)	O1A1549 O1A0877 O1A1420 O1A1712	Verify total flow agrees within 4800 klbm/hr **AND** no computer alarms for high flow present.
SR 3.4.1.2 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1888 O1P1889	Verify RCS loop average temperature: < 580°F on OAC < 579.5°F Dixon indication (OAC unavailable) Mode 1 only, Steady State Operation When 3 RCPs operating, limits applied to loop with lowest loop average temperature for the condition where there is a 0°F ΔTc Setpoint.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.7.11.1 12 Hours	Pressurizer Temperature	(N)	(D)	O1E2298 O1E2299	Verify each temperature channel agrees within 12°F on computer or indicator.
SR 3.4.9.1 12 Hours	Pressurizer Level (Corrected)	(N)	(D)	O1E2275 O1E2276 O1E2277	Verify each level channel agrees within 9" between computer and recorder **OR** between indicator and recorder. Verify Pzr level ≤ 260".
SR 16.7.11.2 24 Hours TS 3.5.2	LDST Level	(N)	(D)	O1A1042 O1A1043	Verify redundant level channels on computer and gauge agrees within 2". (done on 12 hr frequency)
SR 3.3.11.1 12 Hours	MSLB Detection and MFW Isolation Instrumentation	(N)	(D)		Verify redundant outlet pressure channels for 1A and 1B SGs agree within 30 psig: "A" SG / "B" SG O1E2281 / O1E2282 O1E2283 / O1E2284 O1E2111 / O1E2112 **IF** required conditions **NOT** met, refer to step D (MSLB).
SR 3.7.8.3 24 Hours	ECCW		(D)	O1P0761	Verify average CCW inlet temperature ≤ 88°F. **IF** > 88°F, notify MSE for operability evaluation. {PIP 98-3976 & OSC-4476}
SR 3.4.1.1 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1609 O1P1620	Verify RCS loop pressure within limits in COLR. Mode 1 only, Steady State Operation. When 3 RCPs operating, limits applied to loop with highest pressure.
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation RC Pressure Wide Range	(N)	(D)	O1A1416 O1A1417 O1A1418	Verify computer readouts agree within 75 psi (100 psi in ES Cab).
SR 3.1.6.1 12 Hours	APSR Alignment Limits	(N)	(D)	GD60 REG	Verify position of each APSR within 6.5% of group average.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.2.1.1 12 Hours	Regulating Rod Position Limits	(N)	(D)	GD60 REG	Verify regulating rod groups within sequence and overlap limits in COLR.
SR 3.2.1.2 12 Hours	Regulating Rod Position Limits	(N)	(D)	GD60 REG	Verify regulating rod groups within position limits on curve in COLR.
SR 3.1.4.1 12 Hours	Control Rod Group Alignment Limits	(N)	(D)	GD60 REG GD60 SAFETY	Verify all Control Rods in each Group agree within ± 3.5% of group average. **IF** a Control Rod is > ± 3.5% of its Group average, refer to OP/0/A/1105/009 (Control Rod Drive System).
SR 3.1.5.1 12 Hours	Safety Rod Position Limits	(N)	(D)	GD60 SAFETY PI Panel	Verify each safety rod fully withdrawn.
SR 16.7.11.3 31 Days SLC 16.5.13	CBAST Temperature	(N)		O1A0784	Verify computer indication > 125°F. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)
SR 16.5.13.1 7 Days	CBAST	(N)		O1A0797	Verify equivalency of 1100 ft^3 of 11,000 ppm boron per OP/0/A/1108/001 (Curves And General Information). (done on 24 hr frequency)
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation ES Channels 7 & 8 RB 10 psig	(N)	(D)		Verify no trips present. Verify status annunciators operable (lamp test).
SR 3.3.1.1 12 Hours SLC 16.7.9	RPS Instrumentation RP RCP/Flux Trip	(N)	(D)		Verify no Dummy Bistable installed. Verify no trips present. Verify status annunciators operable (lamp test).
SR 3.3.1.1 12 Hours	RPS Instrumentation RB High Press Trip	(N)	(D)		Verify no Dummy Bistable installed. Verify no trips present. Verify status annunciators operable (lamp test).

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.1.7.1 12 Hours	Position Indicator Channels PI Panel	(N)	(D)		Verify all Relative Rod Position indications agree within 5% of Absolute Rod Position indications. **<u>IF NOT</u>**, notify Duty Rx Engineer for evaluation of core parameters and recommended actions.
SR 3.3.9.1 12 Hours	Source Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 SR agree within 1 decade. Mode 2 only
SR 3.3.10.1 12 Hours TS 3.3.8	Wide Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 agree within 3 LED Segments. Mode 2 only
SR 3.4.4.1 12 Hours	RCS Loops	(N)	(D)		Verify required RCPs (3 or 4) in operation with RCS flow indicated.
SR 3.5.4.2 7 Days TS 3.3.8 TS 3.5.4	BWST	(N)			Verify BWST level on ICCM Plasma Displays ≥ 47.0 ft. (done on 24 hr frequency) • A 1LT-BWST 1 • B 1LT-BWST 2 **<u>AND</u>** BWST level ≥ 46.0 ft. on both analog gauges on 1UB2 • BWST Level A • BWST Level B **<u>IF</u>** required conditions **<u>NOT</u>** met, BWST is inoperable.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.5.4.1 24 Hours TS 3.5.4	BWST Temperature	(N)			Verify BWST between ≥ 50°F and ≤ 92.5°F as read on Bailey Indicator. **IF** > 92.5°F, notify MSE for operability evaluation of RBS System. {PIP 98-3976 & OSC-4476} **IF** ≥ 102.5°F or < 50°F, BWST is inoperable.
SR 16.11.3.1 24 Hours	WG Decay Tk Disch Flow Recorder	(N)	(D)		Verify recorder (GWD CR033) indicates flow. Perform anytime during shift hours during GWD Tank releases.
SR 16.11.3.1 24 Hours SR 16.11.3.2 24 Hours	1RIA-35	(N)			Verify 1RIA-35 indicates > zero **AND** no low flow alarm present.
SR 16.11.3.12 24 Hours	1RIA-37		(D)		Perform source check on 1RIA-37.
SR 16.11.3.1 24 Hours	1RIA-38		(D)		Verify 1RIA-38 indicates > zero **AND** no fault alarm present.
SR 16.11.3.2 24 Hours	1RIA-40	(N)			Verify 1RIA-40 indicates > zero **AND** no low flow alarm present.
SR 16.11.3.2 24 Hours	1RIA-43, 44, 45	(N)			Verify the following: 1) 1RIA-43, 44, 45 indicate > zero. 2) Unit Vent Monitor has no low flow alarm. 3) Unit Vent Flow Recorder indicates on scale.
SR 3.4.15.1 12 Hours	RCS Leakage Detection Instrumentation	(N)	(D)		Verify 1RIA-47 indicates > zero **AND** no flow alarm present. **OR** Verify 1RIA-49 indicates > zero **AND** no flow alarm present.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.11.3.1 24 Hours SR 16.11.3.2 24 Hours	1&2RIA-54	(N)			Verify the following: 1) 1&2RIA-54 indicates > zero. 2) No low flow alarm present. 3) "NORMAL/BYPASS" switch in "Normal".
SR 3.5.1.1 12 Hours	CFTs	(N)	(D)		Verify 1CF-1 **AND** 1CF-2 fully open.
SR 16.7.10.1 12 Hours	CFT Instrumentation	(N)	(D)		Verify redundant level channels on each CFT agree within 0.3 ft.
SR 3.5.1.2 12 Hours	CFTs	(N)	(D)		Verify CFT levels between 12.56 ft and 13.44 ft.
SR 16.7.10.1 12 Hours	CFT Instrumentation	(N)	(D)		Verify redundant pressure channels on each CFT agree within 30 psi.
SR 3.5.1.3 12 Hours	CFTs	(N)	(D)		Verify CFT pressures between 575 psig and 625 psig.
SR 16.8.6.1 24 Hours	Lee/Central Alternate Power System		(D)		Verify status of LCTs by contacting Lee Steam Station CR. Operable (✓) ___ 4C ___ 5C ___ 6C **IF** two LCTs are **NOT** operable, refer to Maintenance Rule **AND** contact Switchyard Coordinator.
SR 3.7.8.1 12 Hours	ECCW System	(N)	(D)		Verify two Unit 1 ESV Pumps in operation. **IF** two Unit 1 ESV Pumps **NOT** in operation, Refer To TS 3.7.8 Bases for allowed Pump/Header combinations.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.7.8.2 24 Hours SR 16.9.7.1 12 Hours	ECCW	(N)	(D)		Verify Keowee lake level within limits per SLC 16.9.7. **NOTE:** Instrument error of 1.15 ft. must be added to the absolute lake levels found in SLC 16.9.7 if using a computer point to verify level. Absolute lake level can be determined at the Keowee Hydro Intake structure.
SR 3.4.13.1 72 Hours SLC 16.5.10	RCS Operational Leakage	(N)			<u>Evaluate</u> per PT/1/A/600/010 (Reactor Coolant Leakage) when at steady state for ≥ 12 hours. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)
SR 3.7.16.1 12 Hours SR 16.8.1.1 SR 16.8.1.2 SR 16.8.1.3 24 Hours	Room Temperatures Unit 1 Cable Rm. Unit 1 Equip. Rm. Unit 1&2 Control Rm.	(N) ______ ______ ______	(D) ______ ______ ______		Record and verify room temperatures within respective temperature limits: Unit 1 Cable Rm: ≤ 80°F Unit 1 Equip. Rm: ≤ 85°F Unit 1&2 Control Room: ≤ 80°F **<u>IF</u>** limit is exceeded, refer to OP/0/A/1104/019 (Control Room Ventilation System), notify Unit Coordinator **<u>AND</u>** refer to SLC 16.8.1 and TS 3.7.16.
SR 16.7.11.3 31 Days	BAMT Temperature	(N)			Have NLO verify normal readout at Chemical Addition Panel agrees with local readout within 5°F. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.5.4.3 7 Days	BWST	(N) Wednesday			Verify BWST concentration within limit in COLR. May be performed anytime during shift hours (1900-0700)
SR 16.7.11.3 31 Days	PORV and Safety Valve Flow Monitors	(N) Saturday			Verify power supply lights on **AND** verify flow monitor statalarm actuates from "TEST" switch. (done on 7 Day frequency) May be performed anytime during shift hours (1900-0700)
SR 3.5.1.5 31 Days	CFTs	(N) 1st Day of Month			Verify with NLO 1CF-1 **AND** 1CF-2 breakers open. May be performed anytime during shift hours (1900-0700)
SR 3.5.1.4 31 days	CFTs	(N) 1st Day of Month			Verify each CFT boron concentration within limit in COLR. May be performed anytime during shift hours (1900-0700)
SR 3.6.3.1 31 Days	1PR-1, 2, 3, 4, 5, 6	(N) 1st Day of Month			Verify with NLO 1PR-1 and 1PR-6 breakers open. Verify with I&E links open for 1PR-2, 1PR-3, 1PR-4, and 1PR-5. May be performed anytime during shift hours (1900-0700)

NOTE: Remaining items may be performed anytime during shift hours.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	Pressurizer Level (Uncorrected)	(N)	(D)	O1E2301 O1E2303 O1E2305	Verify redundant level channels agree within 8" on computer.
TS 3.10.1	SFP Temperature	(N)	(D)	O1A0839	Verify SFP temperature ≤ 143°F. **IF** > 143°F, SSF RCMUP is inoperable. Contact Duty MSE Engineer.
TS 3.7.5	UST Temperature	(N)	(D)	O1A0122 O1A0123	Verify UST temperature ≤ 125°F. **IF** > 125°F, notify Unit Coordinator and refer to OP/1/A/1106/006 (Emergency FDW System) for EFDW operability.
SLC 16.7.2	AMSAC/DSS	(N)	(D)	O1D2928 THRU O1D2935	Verify no trips present **AND** status annunciators indicate operable channels. Verify 1SA-8 C-9/C-10/D-5/D-8 **AND** indicating lights on 1UB1. Refer to Section 12.7 for operability determinations.
SLC 16.7.3	SG "A" XSUR Level Redundant Level	(N)	(D)	O1A1213 O1E2052	Verify redundant levels agree within 3" when < 2% RTP. > 3" acceptable during momentary swings (< 30 sec).

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
SLC 16.7.3	SG "A" XSUR Level Minimum Level	(N)	(D)	O1A1213 O1E2052	Verify all XSUR OAC indications ≥ 23" when SG on level control. < 23" are acceptable during momentary swings (< 30 sec)
SLC 16.7.3	SG "B" XSUR Level Redundant Level	(N)	(D)	O1A1215 O1E2053	Verify redundant levels agree within 3" when Rx < 2% RTP. > 3" acceptable during momentary swings (< 30 sec).
SLC 16.7.3	SG "B" XSUR Level Minimum Level	(N)	(D)	O1A1215 O1E2053	Verify all XSUR OAC indications ≥ 23" when SG on level control. < 23" acceptable during momentary swings (< 30 sec).
	1A SG SU Levels	(N)	(D)	O1E2000 O1E2001	Verify redundant levels agree within 2" when Rx < 2% RTP. >2" acceptable during momentary swings (< 30 sec).
	1B SG SU Levels	(N)	(D)	O1E2005 O1E2006	Verify redundant levels agree within 2" when Rx < 2% RTP. >2" acceptable during momentary swings (< 30 sec).
SLC 16.7.5	1A SG OR Levels	(N)			Verify redundant operating range recorder level channels agree within 3%.
SLC 16.7.5	1B SG OR Levels	(N)			Verify redundant operating range recorder level channels agree within 3%.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	"A" SG Shell Temperatures	(N)		O1P1892 O1A0968 O1A0969 O1A0970 O1A0971 O1A0972	Verify O1P1892 agrees with manually calculated average of five "A" SG Shell Temperatures (O1A0968 – O1A0972) within $\pm$ 5°F. **IF** required conditions **NOT** met, issue a Work Request.
	"B" SG Shell Temperatures	(N)		O1P1893 O1A0973 O1A0974 O1A0975 O1A0976 O1A0977	Verify O1P1893 agrees with a manually calculated average of five "B" SG Shell Temperatures (O1A0973 – O1A0977) within $\pm$ 5°F. **IF** required conditions **NOT** met, issue a Work Request.
	Station Condenser ΔT		(D)	O1P1947 or O3P1947	**IF** CCW inlet temperature > 68°F, verify Station Condenser ΔT ≤ 22°F. **IF** Station Condenser ΔT is > 22°F, notify Unit Coordinator **OR** OPS Duty Person. **IF** O1P1947 and O3P1947 are OOS, perform the following: 1) Verify all Units with CCW flow have CP O*P1944 operable **OR** 2) Hourly Inlet/Outlet sheet performed per OP/0/A/1103/020 (Loss Of Computer).
	CCW	(N)	(D)	O1P0761	Verify average CCW inlet temperature ≤ 80°F. **IF** > 80°F, Perform Enclosure 13.19 "Hot Lake Water Surveillance". {OSC-2576}

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	Station CCW Discharge Temperature		(D)	O1P1945 or O3P1945	Verify Station 2 Hour Average CCW discharge temperature ≤ 100°F. **IF** Station 2 Hour Average CCW discharge temperature > 100°F, notify Unit Coordinator **OR** OPS Duty Person. **IF** O1P1945 and O3P1945 are OOS, perform the following: 1) Verify all Units with CCW flow have CP O*P1942 operable **OR** 2) Hourly Inlet/Outlet sheet performed per OP/0/A/1103/020 (Loss Of Computer).
TS 3.5.2	LDST Pressure	(N)	(D)	CR Gage O1A2191	Verify both LDST pressure/level relationships comply with OP/0/A/1108/001 (Curves And General Information).
	Room Temperatures Unit 1 Cable Rm. Unit 1 Equip. Rm. Unit 1&2 Control Rm.	(N) ________ ________ ________	(D) ________ ________ ________		Record and compare room temperatures to those taken on previous shift. A ≥ 3°F temperature increase observed from the previous shift may be an indication of a problem with the WC System. **IF** this is indicated **REFER TO** Enclosure "Control Room, Equipment Room, And Cable Room Temperature Troubleshooting Guide" of OP/0/A/1106/029 (Control Room, Equipment Room, And Cable Room Chillers).
	ES Channels 1 & 2 RC Press	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 3 & 4 RC Press	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 1 & 2 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	ES Channels 3 & 4 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 5 & 6 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP Low Press Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP RCP/Flux/Imb Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Temp Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP Press/Temp Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Press Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Flux Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
TS 3.10.1	RCS Boron Concentration	(N)	(D)		Verify RCS Boron Concentration greater than "Minimum RCS Boron Concentration to Maintain SSF Operability" curve of PT/1/A/1103/015 (Reactivity Balance). **IF** minimum concentration is **NOT** met, SSF RC MU Pump is inoperable. Contact Duty Rx Engineer.
	Control Rod Position	(N)	(D)		Verify limit lamps operable on Diamond and PI Panel.
TS 3.3.10 TS 3.3.8	Wide Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 agree within: 3 LED Segments when < 10% RTP **OR** 2 LED Segments when ≥ 10% RTP. Mode 1 only

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
TS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) **AND** SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). **IF** SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
TS 3.3.8	Subcooling Monitors Core	(N) Thursday			**IF** > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F **AND** (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. **IF** (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	Subcooling Monitors Core	(N) Thursday			**IF** < 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' **AND** Train 'B') within -9 to +11°F. **IF** (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion
TS 3.3.8	ICC Level - Train 'A'		(D)		Verify "MALFUNCT 00" message on diagnostic page of ICCM Plasma Display. Verify "RVLIS/ICCM/RG 1.97 TRAIN A TROUBLE" annunciator (1SA-18/A-3) **NOT** in alarm. **IF** a "MALFUNCTION FF" message **OR** annunciator alarm is present, issue a Priority Work Request **AND** contact I&E to investigate problem to determine operability of ICC channel.
TS 3.3.8	ICC Core Cooling Train 'A'		(D)		Verify from Core Cooling core map ≥ 5 CETCs operable (do **NOT** indicate "FAIL").

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	ICC Level - Train 'B'		(D)		Verify "MALFUNCT 00" message on diagnostic page of ICCM Plasma Display. Verify "RVLIS/ICCM/RG 1.97 TRAIN B TROUBLE" annunciator (1SA-18/A-4) **NOT** in alarm. **IF** a "MALFUNCT FF" message **OR** annunciator alarm is present, issue a Priority Work Request **AND** contact I&E to investigate problem to determine operability of ICC channel.
TS 3.3.8	ICC Core Cooling Train 'B'		(D)		Verify from Core Cooling core map ≥ 5 CETCs operable (do **NOT** indicate "FAIL").
TS 3.3.5 TS 3.5.4	BWST Level Instrument ICCM Plasma Displays	(N)			Verify redundant indicators on ICCM Plasma Displays agree within 2 ft. **IF** required conditions **NOT** met, BWST is inoperable.
TS 3.5.4	BWST Level Instrument Analog Gauges on 1UB2	(N)			Verify redundant indicators on 1UB2 agree within 2 ft. **IF** required conditions **NOT** met, BWST is inoperable.
TS 3.3.14 TS 3.7.5	1A & 1B MD EFDW Pumps "OFF/AUTO/RUN" Lights	(N)	(D)		Verify lights energized. **IF NOT**, MD EFDW Pumps are inoperable and Auto Start capability is lost.
TS 3.3.8	EFDW Total Flow	(N)			Verify Train 'A' **AND** Train 'B' EFDW Hdr Flow to SG indicates < 20 gpm with no EFDWPs operating. (indicators fail high)

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8 TS 3.7.6	UST Level CR Gauges	(N)	(D)		Verify redundant indicators agree within 0.4 ft.
SLC 16.11.3	Sorrento Radiation Monitor Time Check	(N)	(D)		Verify current time on RIA CRT within ± 1 minute of current time on OAC CRT. **<u>IF</u>** > ± 1 minute, Contact IT to reset time.
TS 3.3.8	1RIA-57, 58	(N)			Verify 1RIA-57, 58 indicate between 5.0E-1 and 1.0E0 R/HR. Press "C/S" button. Verify no Area Monitor Fault alarm exists after check source is complete. Press "R/HR" button to return to normal.
TS 3.5.3 TS 3.7.7	LPI Cooler 'A' LPSW Flow Dixon Indicator	(N)	(D)	O1A2124	**<u>IF</u>** dixon indicator zero and blinking, verify computer point **<u>NOT</u>** reading a high negative value.
TS 3.5.3 TS 3.7.7	LPI Cooler 'B' LPSW Flow Dixon Indicator	(N)	(D)	O1A2125	**<u>IF</u>** dixon indicator zero and blinking, verify computer point **<u>NOT</u>** reading a high negative value.
TS 3.6.5	1A, 1B, 1C RBCU LPSW Flow (IN)	(N)	(D)		Verify each RBCU LPSW Flow (IN) ≥ 550 gpm. **<u>IF</u>** Auxiliary Cooling Coils **<u>AND</u>** 1B RBCU **<u>BOTH</u>** have flow established, verify ≥ 1100 gpm Inlet Flow to 1B RBCU. **<u>IF</u>** any RBCU LPSW Flow (IN) < required, enter LCO 3.0.3

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.5.3 TS 3.6.4 TS 3.6.5	RB Dome Temperature	(N)	(D)	O1A0043 Chart Recorder 1RBCCR0007	Verify highest RB Dome temperature ≤ 170°F. Verify lowest RB Dome temperature ≥ 90°F when 100 % RTP. **IF** either limit is exceeded, contact MSE for operability evaluation. (LPI and BS) **IF** > 175°F, contact CEN for operability evaluation. (Reactor Building)
TS 3.3.8	RB Post Accident Water Level Wide Range Indication	(N)		O1A1033 O1A1565	Verify Train 'A' **AND** Train 'B' Meters, Computer, and Recorder (1BS CR0085) agree within 0.5 ft.
TS 3.3.8	RB Post Accident Pressure Wide Range Indication	(N)		O1A1011 O1A1315	Verify Train 'A' **AND** Train 'B' Meters, Computer, and Recorder (1BS CR0085) agree within 6 psi.
TS 3.3.8 TS 3.4.15	RB Normal Sump	(N)	(D)		Verify Train 'A' **AND** Train 'B' Meters and Recorder (1LWDCR0095) agree within 1 ft.
TS 3.4.15	RB Normal Sump	(N)			Verify water level in RBNS on scale.
TS 3.3.8	RB Emerg Sump Narrow Range	(N)		O1A0050	Verify Train 'A' **AND** Train 'B' Meters, Computer and Recorder (1LWDCR0095) agree within 1 ft.
	RB Emergency Sump	(N)			Verify zero water level in RBES.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.10.1	RCP Seal Leakoff Flow	(N)	(D)		Verify RCP seal leakoff flow electronic display on **AND** display present. **IF** seal leakoff flow > 4 gpm on any RCP, increase seal leakoff flow surveillance to every 2 hours and initial on page 1. **IF** the SSF is **NOT** manned and seal leakoff flow > 4.7 gpm for 1A1, 1A2, 1B1 or 1B2 RCP, SSF RCMU Pump is inoperable. **IF** the SSF is manned and seal leakoff flow >6.0 gpm for 1A1 or 1B1 RCP, or > 4.7 gpm for 1A2 RCP, or > 5.5 gpm for 1B2 RCP, SSF RCMU Pump is inoperable.
	Loose Parts Monitor	(N)			Monitor all operable points on LPM. Test alarm circuitry per OP/1/A/1105/011 (Loose Parts Monitoring System).
	Event Recorders	(N)			Verify paper in all Events recorders.
	800 mHz Radio	Sunday (0100-0400) (N)			Test the backup radio communications with the System Operating Center (SOC) **AND** the Transmission Control Center (TCC). SOC code – 96 TCC code – 11 **IF** communications fail notify SPOC.
	Easterline Angus Charts	(N)			Stamp charts.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
SLC 16.8.5	125 VDC Ground Detection System Test	(N)	(D)		Perform Enclosure "125 VDC Ground Detection System Operation" of OP/1/A/1107/010 (Operation Of The Batteries And Battery Chargers). **IF** required conditions **CANNOT** be met, refer to SLC 16.8.5 for required actions
	SASS	(N)	(D)		Verify the following on SASS panels in ICS cabinet #8: 1) All "AUTO" lights on. 2) No "MISMATCH" lights on. 3) All "POWER" lights on.
SLC 16.9.6	Fire Alarm Cabinet	(N)			Verify "Power" **AND** "Run" LEDs are on **AND** no Trouble/Alarm lights present.
TS 3.10.1	SFP Level	(N)	(D)		**IF** all fuel in SFP subcritical ≥ 20 days, verify SFP level > - 2 ft. **IF** any fuel in SFP subcritical < 20 days, verify SFP level greater than Enclosure "Unit 1&2 Spent Fuel Pool Level Vs Temperature Curve (7-19 days))" of OP/0/A/1108/001 (Curves And General Information). **IF** limit exceeded, SSF RCMUP is inoperable.
	RCP Data Sheets	(N)			Complete RCP Data Sheets: • Sunday and Wednesday when at steady-state power. • Daily when changing Rx power **OR** RCS temperature.
SLC 16.11.3	RB Depressurization		(D)		**IF** RB depressurization is in progress, submit a RB Sample Request.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.11	MSLB Digital Channels	(N)	(D)		Verify with NLO all five power supply lamps lit. **IF** required conditions **NOT** met, refer to step D (MSLB).
ISFSI TS 4.1.1 ISFSI C of C 1.3.1 1.3.2	ISFSI Storage Facility		(D)		Verify notified by Security: 1) All Horizontal Storage Modules (HSM) ventilation screens (inlet and outlet) free of debris and no material accumulated between modules to block air flow. 2) All Roof Slab temperatures monitored which contain Spent Fuel < 260°F and < 80°F increase in 24 hrs. **IF** temperature limits are exceeded, issue a Work Request **AND** contact Rx Engineering.
	Unit 1 BWST	(N) Monday			Place in recirc per OP/1&2/A/1104/006 (SF Cooling System).
	LDST Level	(N) Saturday			Verify redundant level channels 1&2 CR Gage and Local Gage (1HPIPG0437) agree within 2".
	LDST Pressure	(N) Saturday			Verify redundant pressure channels 1&2 CR Gage and Local Gage (1HPIPG0438) agree within 1 psig.
	SSF Radio	(N)			Verify communications with CR via SSF radio. Use base station on Channel 2.

Duke Power Company Oconee Nuclear Station **Backup Incore Detector System** *** This procedure has the potential to affect Reactivity Management *** **Continuous Use**	Procedure No. PT/**0**/A/1103/019 Revision No. 4 Electronic Reference No.

Performed By ______________________

Date ______________

Backup Incore Detector System

1. Purpose

1.1 To verify the operable backup recorder points meet the minimum requirements for the incore instrumentation system upon loss of the incore system on the unit computer or loss of the unit computer.

1.2 To provide a method to calculate reactor power axial imbalance and quadrant power tilt using the backup incore detector system when the incore system is not available on the unit computer and one or more of the excore detectors are inoperable.

2. References

2.1 Improved Technical Specifications 3.2.2, Axial Power Imbalance
3.2.3, Quadrant Power Tilt

2.2 OP/0/A/1103/020, Loss of Computer

2.3 PT/1,2,3/A/0600/001, Periodic Instrument Surveillance

2.4 Unit Core Operating Limits Report (COLR)

2.5 NSD 304, Reactivity Management

2.6 Selected Licensee Commitment 16.7.8

3. Time Required

3.1 Verify Backup Incore Recorders operable - 10 minutes - 1 Operator or Reactor Engineer

3.2 Calculate Backup Tilt/Imbalance - 30 minutes - 2 Operators and/or Reactor Engineers

4. Prerequisite Tests

None

5. Test Equipment

Calculator

6. Limits and Precautions

6.1 This procedure has the potential to affect REACTIVITY MANAGEMENT, since the backup incore recorders are used to monitor reactivity.

6.2 If the incore system is not available on the unit computer and the backup recorder points are not operable per this procedure, then the reactor power shall be reduced below 80% of the power allowable for the existing reactor coolant pump combination within eight hours unless:

6.2.1 The incore system is restored on the unit computer.

or

6.2.2 The backup recorder points are restored to meet the minimum requirements for operability. (ref. SLC 16.7.8)

6.3 If the backup incore limits are exceeded then action must be taken per the applicable Technical Specifications as listed below:

6.3.1 Quadrant Power Tilt - ITS 3.2.3

6.3.2 Axial Power Imbalance - ITS 3.2.2

7. Required Plant Status

_____7.1 Quadrant power tilt surveillances are required when the Unit is above 20% full power.

_____7.2 Reactor power imbalance surveillances are required when the Unit is above 40% rated power.

8. Prerequisite System Conditions

Loss of incore system on the unit computer or loss of the unit computer

9. Test Method

9.1 The backup recorder points will be checked to identify which points are a) inoperable as indicated by off-scale readings or b) identified as inoperable or out of calibration during the last functional verification. The remaining operable points will be checked to verify the minimum number of detectors are operable to measure axial imbalance and quadrant power tilt as required.

9.2 Axial Imbalance and quadrant power tilt calculations may be performed using the operable backup recorder points.

10. Data Required

Incore Backup recorder point readings

11. Acceptance Criteria

11.1 The calculated axial imbalance is within the curve for the appropriate pump configuration shown in the current Core Operating Limits Report (COLR) on the Backup Incore Setpoint Column of the (Error-Adjusted) "Operational Power Imbalance Setpoints" Table.

11.2 The calculated quadrant power tilt is less than the value given in the current Core Operating Limits Report (COLR) on the Backup Incore row of the (Error-Adjusted) "Quadrant Power Tilt Setpoints" Table.

12. Procedure

NOTE:	If this procedure is being performed only to satisfy Section 1.1, then perform Section 12.1 and N/A Section 12.2.

_______12.1 Verification of Minimum Incore Detector Operability

_____12.1.1 On Enclosure 13.1 and 13.2, place an "X" next to the backup recorder points which are inoperable as indicated by off-scale readings or notes attached to the recorders.

_____12.1.2 Verify that all three required points on at least three detector strings are operable per instructions on Enclosure 13.1.

_____12.1.3 Verify that all four required points on at least four sets (two sets in each axial core half) are operable per instructions on Enclosure 13.2.

_____12.1.4 If either step 12.1.2 or 12.1.3 cannot be satisfied;

_____12.1.4.1 Notify the Unit Supervisor.

_____12.1.4.2 Take actions described in 6.2.

_______12.2 Calculation of Axial Imbalance and Quadrant Power Tilt

NOTE:	If this portion of the procedure is being performed to fulfill the requirements of PT/1,2,3/A/0600/01, Periodic Instrument Surveillance, then repeat the steps below every twelve hours as required and record the calculated values in that procedure.

_____12.2.1 Verify the reactor has been at steady state conditions (±2% FP) for at least 30 minutes.

_____12.2.2 Calculate axial imbalance per Enclosure 13.3 using operable recorder points identified on Enclosure 13.1.

_____12.2.3 Calculate quadrant power tilt per Enclosure 13.4 using operable recorder points identified on Enclosure 13.2.

_____12.2.4 Verify the calculated axial imbalance does not exceed the backup incore limits per 11.1.

_____12.2.5 Verify the calculated quadrant power tilt does not exceed the backup incore limits per 11.2.

_____12.2.6 If either step 12.2.4 or 12.2.5 cannot be satisfied, notify the Unit Supervisor and take appropriate actions described in the applicable Technical Specification as listed below:

Quadrant Power Tilt - ITS 3.2.3
Axial Power Imbalance - ITS 3.2.2.

13. Enclosures

13.1 Required Backup Recorder Points for Calculating Axial Power Imbalance

13.2 Required Backup Recorder Points for Calculating Quadrant Power Tilt

13.3 Axial Power Imbalance Calculation Sheet

13.4 Quadrant Power Tilt Calculation Sheet

Oconee 1 Cycle 19

Operational Power Imbalance Setpoints

	%FP	Full Incore	Backup Incore	Out of Core
4 Pumps	0	-31.5	-31.0	-31.5
	80	-31.5	-31.0	-31.5
	90	-29.7	-29.3	-29.7
	100	-19.1	-18.7	-19.1
	102	-17.0	-16.5	-17.0
	102	17.0	17.0	17.0
	100	19.1	18.7	19.1
	90	22.4	21.8	22.4
	80	23.1	22.3	23.1
	0	23.1	22.3	23.1
3 Pumps	0.0	-31.5	-31.0	-31.5
	63.30	-31.5	-	-31.5
	63.77	-	-31.0	-
	77.0	-17.0	-16.5	-17.0
	77.0	17.0	17.0	17.0
	71.99	-	22.3	-
	71.24	23.1	-	23.1
	0.0	23.1	22.3	23.1

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NRC-004/ADMIN A.2

ICCM Subcooling Margin Monitor Check

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

SUBCOOLING MONITOR CHECK

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):
System: Conduct of Operations
K/A: G2.2.12
Rating: 3.0/3.8

Task Standard:

Perform Subcooling Monitor Check

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and 13.16

Validation Time: 10 min. **Time Critical:** NO

===

Candidate: ________________________________ Time Start : ________
NAME Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: __________________________ ______________________________/________
NAME SIGNATURE DATE

===

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC or SNAP # ________
2. Go to run, acknowledge alarms.
3. Verify accurate pressure/ temperature values
4. Freeze simulator.
5. Leave simulator in FREEZE to prevent values changing.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and 13.16

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit 1 is at 100% power
Today is Thursday
The time is 2000
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1 has been completed up page 21

INITIATING CUES:

The SRO directs you to complete PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1.

START TIME: ______

STEP 1: Verify Loop A and Loop B RCS WR Pressure and ICCM Plasma Display pressure agree with in 10 psig **"A" WR = 2114, ICCM = 2114** **"B" WR = 2160, ICCM = 2150** STANDARD: Locate and obtain Loop "A" RCS WR Pressure and ICCM Plasma Display pressure and ensure pressures agree within 10 psig Locate and obtain Loop "B" RCS WR Pressure and ICCM Plasma Display pressure and ensure pressures agree within 10 psig COMMENTS:	___ SAT ___ UNSAT
NOTE: Perform this step as the Initial Conditions indicate it is Thursday night shift. STEP 2: Obtain readings from: __(25)_ SCM Loop "A" (OAC) __(24)_ SCM Loop "A" (ICC) STANDARD: Locate and obtain Loop "A" SCM readings. SCM Loop "A" (OAC) - SCM Loop "A" (ICC) = __(+1)__ Verify SCM Loops agree within -6 °F to +9 °F Candidate determines that "A" is within the specified range COMMENTS:	___ SAT ___ UNSAT

STEP 2: Obtain readings from: __(18)__ SCM Loop "B" (OAC) __(27)__ SCM Loop "B" (ICC) STANDARD: Locate and obtain Loop "B" SCM readings. SCM Loop "B" (OAC) - SCM Loop "B" (ICC) = __(-9)__ Verify SCM Loops agree within –6 °F to +9 °F Candidate determines that "B" is NOT within the specified range COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: Candidate determines that "B" is NOT within the specified range. STANDARD: Refers to Enclosure 13.16 (ICCM Subcooling Monitor Check) of PT/1/A/0600/001, Periodic Instrument Surveillance COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: Loop "B" Subcooling Monitor Obtain RCS Loop "A" pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below STANDARD: **(2026)** psig + 14.7 psi = **(2040.7)** psia COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: Using ASME Steam Tables **OR** OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below STANDARD: **(638)** °F COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6: Obtain RCS Loop "B" temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below. STANDARD: **(598)** °F ***CUE: Tell the operator to use 598°F instead of 600 as indicated on the simulator. This will allow the final calculation to within the proper range of +/- 5°F*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below. STANDARD: Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) = Correction. **(22)** °F = (638) °F – (598) °F - 18 °F COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: Verify ICC Train "B" SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4) STANDARD: Determine difference in ICCM and manually calculated SCM agrees within +/- 5°F ICC Train "B" SCM Loop **(27) – 22 = 5** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 9: Calculations performed in step 2.2 require independent verification. **CUE: another operator will perform verification calculations.** STANDARD: Sign the Performed By: ________________________ COMMENTS:	___ SAT ___ UNSAT

TIME END: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
2	Step is necessary, to determine that the "B" SCM is not within the required range and is inoperable. Calculation is -9
3	Step is necessary, Refer to Enclosure 13.16 to perform Manual SCM calculation
4	Step is necessary, calculation of actual RCS pressure in psia to obtain correct saturation temperature
5	Step is necessary, obtain correct saturation temperature based on pressure (psia)
6	Step is necessary, obtain actual RCS Th temperature
7	Step is necessary, obtain actual Loop A SCM
8	Step is necessary, determine that ICCM Loop SCM agrees with the manual calculated SCM within +/- 5. Actual = 5

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit 1 is at 100% power
Today is Thursday
The time is 2000
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1 has been completed up page 21

INITIATING CUES:

The SRO directs you to complete PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit 1 is at 100% power
Today is Thursday
The time is 2000
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1 has been completed up page 21

INITIATING CUES:

The SRO directs you to complete PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) <u>AND</u> SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). <u>IF</u> SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			<u>IF</u> > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F <u>AND</u> (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. <u>IF</u> (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

1. Initial Conditions

_____ 1.1 Manual verification of ICCM Subcooling monitors required.

_____ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

_____ 2.1.2 Using ASME Steam Tables <u>OR</u> OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

__________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

__________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

__________°F = __________°F (step 2.1.2) - __________°F (step 2.1.3) - 18°F

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

_____ 2.2.2 Using ASME Steam Tables OR OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

__________ °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

__________ °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.2.2) (step 2.2.3)

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop ________

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core AND ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core ________

ICC Train 'B' SCM Core ________

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

________________________ ________________________
Performed By IV By

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) AND SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). IF SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			IF > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F AND (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. IF (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

1. Initial Conditions

_____ 1.1 Manual verification of ICCM Subcooling monitors required.

_____ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

_____ 2.1.2 Using ASME Steam Tables <u>OR</u> OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

__________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

__________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.1.2) (step 2.1.3)

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

_____ 2.2.2 Using ASME Steam Tables <u>OR</u> OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

__________ °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

__________ °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.2.2) (step 2.2.3)

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop ________

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core <u>AND</u> ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core ________

ICC Train 'B' SCM Core ________

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

________________________ ________________________
Performed By IV By

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)	Loop A 2114 ICC 2114 Loop B 2160 ICC 2150	Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday		25-24 = 1 18-27 = -9	Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) **AND** SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). **IF** SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			**IF** > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F **AND** (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. **IF** (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

1. Initial Conditions

✓ 1.1 Manual verification of ICCM Subcooling monitors required.

✓ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

_________ psig + 14.7 psi =_________psia

_____ 2.1.2 Using ASME Steam Tables **OR** OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

_________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

_________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

_________°F = _________°F - _________°F - 18°F
(step 2.1.2) (step 2.1.3)

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

2026 psig + 14.7 psi = 2040.7 psia

_____ 2.2.2 Using ASME Steam Tables OR OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

638 °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

CUE → 598 °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

22 °F = 638 °F (step 2.2.2) - 598 °F (step 2.2.3) - 18°F

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop 27

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core AND ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core ________

ICC Train 'B' SCM Core ________

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

Sign
Performed By

____
IV By

QUESTION NO. A.3 SRO/RO-Q1 G2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure [CFR: 43.4/45.10] (2.9/3.3) **REFERENCE ALLOWED**

QUESTION:

You have been asked been asked to verify that 3RC-2 (PZR Spray Bypass) is properly back seated.

- Contamination levels on top of the PZR 500,000 dpm/cm^2
- Radiation levels are 30 mrem/hr β-γ general area

Q1: What RWP will you use to enter the RB for this job?

Q2: What dress requirements are required for this job?

Note: This information can be obtained from the RWPs in the U3 Change Room or from Shift/Unit RP crew.

ANSWER:

A1: RWP 3001, U3 RB Inspections and Valve Operations

A2: Per the RWP Dress Category and Task Description – Dress Category I, Cloth Hood, cloth coverall, cotton gloves, 2 pair of rubber gloves, booties, shoecovers, no personal outer clothing,. Secure gloves and booties (tape, Velcro, straps).

REFERENCE: Reference: Radiation Protection Policy Manual, NSD 507, RP-RPP

COMMENTS:

RADIATION WORK PERMIT # 3001 REV: 9 DATE/TIME: 03/30/00 13:36

OCONEE NUCLEAR STATION ACTIVATION DATE: 04/07/00 00:01

Job Title: U3 RX BLDG INSPECTIONS AND VALVE OPERATIONS

STANDING REQUIREMENTS FOR USE OF THIS RWP

EACH RADIATION WORKER IS RESPONSIBLE FOR:

- KNOWING THEIR WORK AREA DOSE RATES.
- FOLLOWING REQUIREMENTS OF THIS RWP.
- BEING ALARA.
- HOUSEKEEPING.
- WEARING A POCKET OR ELECTRONIC DOSIMETER AND A TLD.
- FOLLOWING POSTED REQUIREMENTS.
- REVIEWING AREA RADIOLOGICAL PLAN VIEW WHEN AVAILABLE PRIOR TO ENTRY.
- NOTIFYING RADIATION PROTECTION PRIOR TO SWEEPING, BRUSHING, GRINDING, WELDING, OR USE OF COMPRESSED AIR IN CONTAMINATED AREAS.
- FOLLOWING POSTED DRESS CATEGORY REQUIREMENTS.
- WEARING MODESTY GARMENTS WHEN NOT WEARING PERSONAL OUTER CLOTHING.
- MONITORING PERSONNEL/TOOL/EQUIPMENT REQUIRED WHEN LEAVING RCA OR CONTAMINATED RCZ.

DRESS CATEGORY AND TASK DESCRIPTION

D 1. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

H 2. WORK IN CONTAMINATED AREA.

I 3. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.

M 4. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

N 5. WORK IN CONTAMINATED AREA WHEN WET CONDITIONS OR POTENTIAL FOR WET CONDITIONS EXIST.

SPECIAL DOSIMETRY	RESPIRATORY

SPECIAL INSTRUCTIONS/PRECAUTIONS

* NOTIFY RP PRIOR TO START OF WORK
* USE HOSE CLAMPS TO SECURE HOSE/TUBING CONNECTIONS

COMMENTS

NOTIFY RP PRIOR TO ENTERING THE REACTOR BUILDING.
NOTIFY RP IF WORK AREA CONDITIONS OR JOB SCOPE CHANGES.
RESPIRATORY REQUIREMENTS WILL BE BASED ON TEDE / ALARA EVALUATIONS.
WORKERS TO INSTALL CATCH CONTAINMENTS / DRAIN RIGS TO PREVENT SPILLS WHILE DRAINING COMPONENTS.
DISPOSABLE (PLASTIC) BOOTIES SHALL BE WORN INSIDE NYLON (WASHABLE) BOOTIES FOR WORK IN WET CONDITIONS.

"EXTRA HIGH RADIATION AREA" DOSE RATES:
5000 MREM/HR HIGH CONTACT ON FLOOR OF DEEP END OF CANAL
UP TO 1000 MREM/HR GENERAL AREA IN DEEP END OF CANAL

ED (MG) SET POINTS
DOSE ALARM - 25 MREM
DOSE RATE ALARM - 100 MREM/HR

APPROVED BY: NRW1552 DATE/TIME: 03/30/00 13:35	TERMINATED BY: DATE/TIME:

5.3.6 PROTECTIVE CLOTHING FOR EACH DRESS CATEGORY

DRESS CATEGORY	PROTECTIVE CLOTHING
A	None.
B	Surgical gloves.
C	Cotton and rubber gloves.
D	Cotton and rubber gloves, booties and shoecovers.
E	Labcoat, cotton and rubber or surgical gloves.
F	Labcoat, cotton and rubber gloves, booties and shoecovers.
G	Cloth hood, disposable coveralls, cotton and rubber gloves, booties and shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
H	Cloth hood, cloth coverall, cotton and rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
I	Cloth hood, cloth coverall, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
J	Cloth hood, cloth coverall, cotton gloves, 2 pair rubber gloves, booties, shoecovers, no personal outer clothing and additional outer booties or shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
K	Cloth hood, cloth coverall, disposable coveralls, cotton gloves, rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
L	Cloth hood, cloth coverall, disposable coveralls, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing and additional outer booties or shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
M	Cloth hood, 2 pair cloth coveralls, cotton gloves, 2 pair rubber gloves, 2 pair booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
N	Cloth hood, cloth coverall, wetsuit, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
O	Cloth hood, cloth coverall, bubble suit, cotton gloves, 2 pair rubber gloves, booties, shoecovers, no personal outer clothing and additional shoe covers or jump boots. Secure gloves and booties (tape, elastic, Velcro, straps).
Z	Special dress as required by Radiation Protection.

RADIATION WORK PERMIT # 3001 REV: 9 DATE/TIME: 03/30/00 13:36

OCONEE NUCLEAR STATION ACTIVATION DATE: 04/07/00 00:01

Job Title: U3 RX BLDG INSPECTIONS AND VALVE OPERATIONS

STANDING REQUIREMENTS FOR USE OF THIS RWP

EACH RADIATION WORKER IS RESPONSIBLE FOR:

- KNOWING THEIR WORK AREA DOSE RATES.
- FOLLOWING REQUIREMENTS OF THIS RWP.
- BEING ALARA.
- HOUSEKEEPING.
- WEARING A POCKET OR ELECTRONIC DOSIMETER AND A TLD.
- FOLLOWING POSTED REQUIREMENTS.
- REVIEWING AREA RADIOLOGICAL PLAN VIEW WHEN AVAILABLE PRIOR TO ENTRY.
- NOTIFYING RADIATION PROTECTION PRIOR TO SWEEPING, BRUSHING, GRINDING, WELDING, OR USE OF COMPRESSED AIR IN CONTAMINATED AREAS.
- FOLLOWING POSTED DRESS CATEGORY REQUIREMENTS.
- WEARING MODESTY GARMENTS WHEN NOT WEARING PERSONAL OUTER CLOTHING.
- MONITORING PERSONNEL/TOOL/EQUIPMENT REQUIRED WHEN LEAVING RCA OR CONTAMINATED RCZ.

DRESS CATEGORY AND TASK DESCRIPTION

D 1. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

H 2. WORK IN CONTAMINATED AREA.

I 3. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.

M 4. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

N 5. WORK IN CONTAMINATED AREA WHEN WET CONDITIONS OR POTENTIAL FOR WET CONDITIONS EXIST.

SPECIAL DOSIMETRY

RESPIRATORY

SPECIAL INSTRUCTIONS/PRECAUTIONS

* NOTIFY RP PRIOR TO START OF WORK
* USE HOSE CLAMPS TO SECURE HOSE/TUBING CONNECTIONS

COMMENTS

NOTIFY RP PRIOR TO ENTERING THE REACTOR BUILDING.
NOTIFY RP IF WORK AREA CONDITIONS OR JOB SCOPE CHANGES.
RESPIRATORY REQUIREMENTS WILL BE BASED ON TEDE / ALARA EVALUATIONS.
RESPIRATORY REQUIREMENTS WILL BE BASED ON TEDE / ALARA EVALUATIONS.
WORKERS TO INSTALL CATCH CONTAINMENTS / DRAIN RIGS TO PREVENT SPILLS WHILE DRAINING COMPONENTS.
DISPOSABLE (PLASTIC) BOOTIES SHALL BE WORN INSIDE NYLON (WASHABLE) BOOTIES FOR WORK IN WET CONDITIONS.

"EXTRA HIGH RADIATION AREA" DOSE RATES:
5000 MREM/HR HIGH CONTACT ON FLOOR OF DEEP END OF CANAL
UP TO 1000 MREM/HR GENERAL AREA IN DEEP END OF CANAL

ED (MG) SET POINTS
DOSE ALARM - 25 MREM
DOSE RATE ALARM - 100 MREM/HR

APPROVED BY: NRH1552
DATE/TIME: 03/30/00 13:35

TERMINATED BY:
DATE/TIME:

Enclosure 5.3

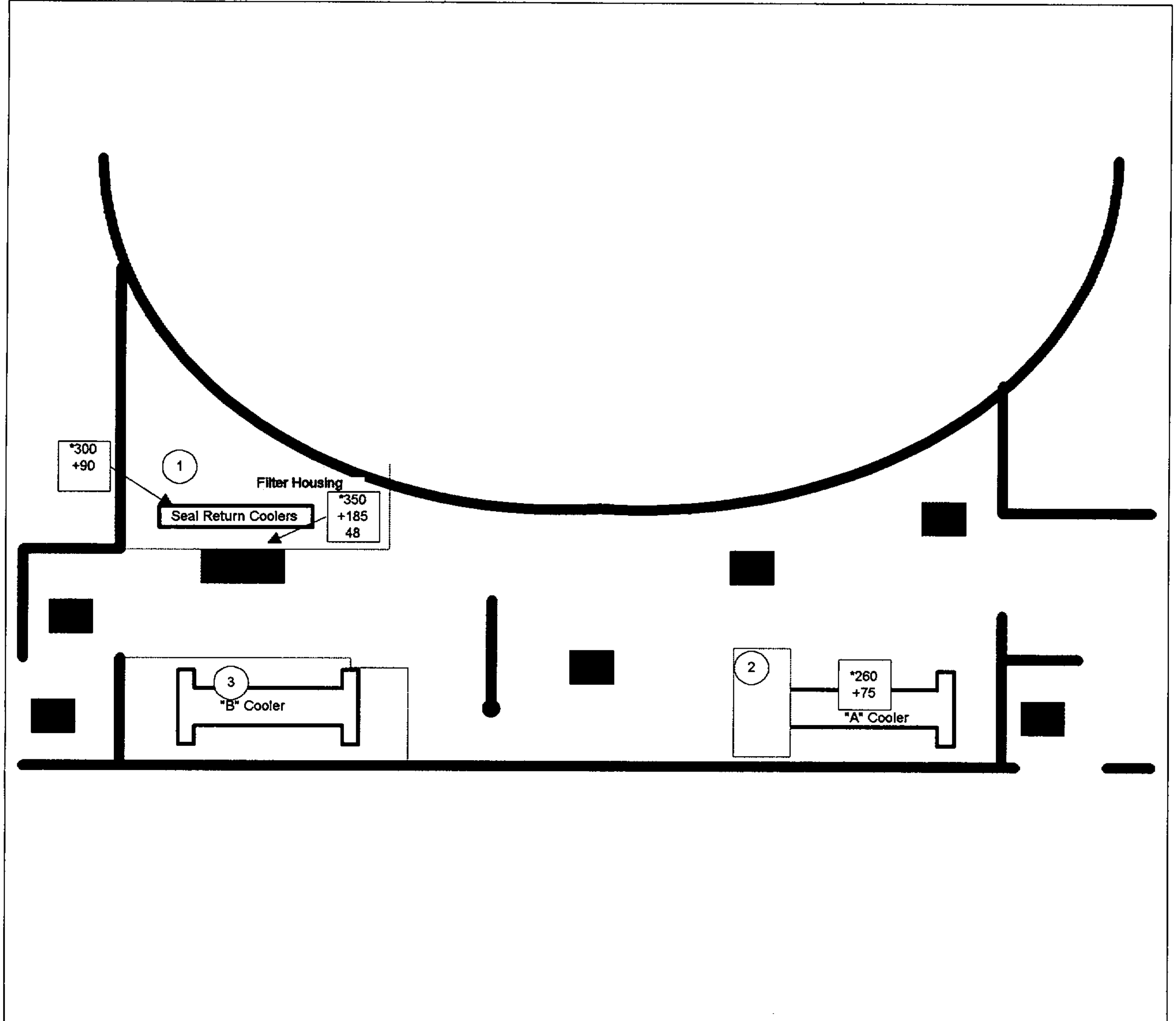
Selection of Protective Clothing

5.3.6 PROTECTIVE CLOTHING FOR EACH DRESS CATEGORY

DRESS CATEGORY	PROTECTIVE CLOTHING
A	None.
B	Surgical gloves.
C	Cotton and rubber gloves.
D	Cotton and rubber gloves, booties and shoecovers.
E	Labcoat, cotton and rubber or surgical gloves.
F	Labcoat, cotton and rubber gloves, booties and shoecovers.
G	Cloth hood, disposable coveralls, cotton and rubber gloves, booties and shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
H	Cloth hood, cloth coverall, cotton and rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
I	Cloth hood, cloth coverall, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
J	Cloth hood, cloth coverall, cotton gloves, 2 pair rubber gloves, booties, shoecovers, no personal outer clothing and additional outer booties or shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
K	Cloth hood, cloth coverall, disposable coveralls, cotton gloves, rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
L	Cloth hood, cloth coverall, disposable coveralls, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing and additional outer booties or shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
M	Cloth hood, 2 pair cloth coveralls, cotton gloves, 2 pair rubber gloves, 2 pair booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
N	Cloth hood, cloth coverall, wetsuit, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
O	Cloth hood, cloth coverall, bubble suit, cotton gloves, 2 pair rubber gloves, booties, shoecovers, no personal outer clothing and additional shoe covers or jump boots. Secure gloves and booties (tape, elastic, Velcro, straps).
Z	Special dress as required by Radiation Protection.

QUESTION NO. A.3 SRO/RO-Q2 G2.3.1 Radiation Exposure Limits [2.6/3.0] **REFERENCE ALLOWED**
QUESTION: Given the attached Oconee Nuclear Station VSDS Survey Report for Room 108: Concerning the area the Room 108 (U-1Decay Heat Removal / Seal Return) plan view 1. Describe the type of posting this area should have to warn radiation worker? Explain your answer. 2. Identify where you should stand while waiting for further direction from the Control Room during a work evaluation. Explain your answer.
ANSWER: 1. High Radiation Area because the dose rate (185 mrem/hour) at 30 centimeters is greater than 100 mrem/hour. This would be marked High radiation area signs. 2. Near the entrance because the general area radiation level is the lowest value at this location.
REFERENCE: NSD 507.8 Exposure and Contamination Control
COMMENTS:

Room 108 Decay Heat Removal / Seal Return	Survey # 050100-27	Date/Time: 07/01/2000 22:48



Comments: 1000/054 ROUTINE SURVEY, PALMVIEW UPDATED, ALL TAKEN IN CLEAN AREA WHERE < 100 CCPM.

Dose Rate
*150 - - Contact Reading
+75 - - 30 cm Reading
20 - - General Area

HS-50 Hot Spot
Posting
Drip Bag

15 Smear 15 Air Sample RM Wipe

Type = Monthly

RWP: 15

Reactor Power = 100%

Unless otherwise Noted, dose rate in mrem/hr.

Surveyor:

Summary of Highest Readings

Smears

2) 3621 DPM/100 CM2 B /γ
1) 1800 DPM/100 CM2 B /γ
3) 1465 DPM/100 CM 2 B /γ

Air Samples & Wipes

Reviewed by:

ENABLING OBJECTIVES (continued)

4. State the approval requirements for an individual at Duke Power Company to exceed the **basic** permissible exposure limit of 2.0 rem. (R4)
5. State the special dose limits established for the general public. (R5)
6. Describe the special dose control measures used to protect the fetus of a "declared" pregnant radiation worker. (R6)
7. Recognize that in "exceptional situations", it is possible to allow an adult radiation worker to receive additional exposure, apart from normal occupational exposure. (R7)
8. Define and describe the specific site area for each of the following terms relating to the control of station areas: (R8)

    8.1 Unrestricted Area

    8.2 Restricted Area

    8.3 Controlled Area

    8.4 Radiation Control Area (RCA)

    8.5 Radiation Control Zone (RCZ)

    8.6 Radiation Area (RA)

    8.7 High Radiation Area (HRA)

    8.8 Extra High Radiation Area (EHRA)

    8.9 Very High Radiation Area (VHRA)

    8.10 Airborne Radioactivity Area

    8.11 Hot Spot

    8.12 Significant Dose Contributor

    8.13 Low Exposure Waiting Area

    8.14 Contaminated Area

ENABLING OBJECTIVES (continued)

18. Describe the method used at Oconee to indicate whether an article is "clean" and can be unconditionally released from the RCA, or is above the contamination limit for unconditional release. (R18)

19. State the maximum contamination limit (in cpm) for personal clothing, body surfaces, and hand-held items for unconditional release from the RCA. (R19)

20. Describe how designated contaminated tools used inside the RCA are identified. (R20)

21. Concerning RWPs/SRWPs: (R21)
    - 21.1 Explain the purpose of RWPs and SRWPs.
    - 21.2 Explain the differences between RWPs and SRWPs.
    - 21.3 List the requirements for re-evaluating SRWPs.
    - 21.4 Understand that indivduals do not have the authority to deviate from RWP or SRWP requirements.
    - 21.5 Identify plant locations of SRWP information.

22. Define ALARA. (R22)

23. Identify the various methods available to aid in maintaining exposures ALARA. (R23)

24. Given a set of conditions, correctly apply the radiation protection practices addressed in this lesson plan. (R24) ✱

✱ P. 1 CHE 11-18-97

507.8 EXPOSURE AND CONTAMINATION CONTROL

DPC adheres to the conservative assumption that there is a risk associated with radiation exposure. Application of the As Low As Reasonably Achievable (ALARA) concept minimizes this risk. Nuclear facility management and all individuals who perform work at the facility share the goal of keeping dose ALARA. Individuals are expected to be knowledgeable in and practice exposure control techniques.

507.8.1 AS LOW AS REASONABLY ACHIEVABLE (ALARA) PROGRAM

The ALARA Program is designed to minimize dose. The ALARA Program is described in the System ALARA Manual, Section III. Some important ALARA Program components are:

- Holding pre-job and post-job briefs
- Pre-planning jobs
- Using training mock-ups
- Using engineering controls
- Removing sources of radiation exposure
- Applying lessons learned from industry events
- Providing job feedback
- Using ALARA Suggestion Forms

A. Planning for Tasks <500 mrem Total Exposure

- Use basic ALARA principles
- RP ALARA Group involvement is not required.

B. ALARA Planning for Tasks Greater Than or Equal to 500 mrem

All Work Order tasks greater than or equal to 500 mrem are planned and tracked using an ALARA package. The package consists of:

- ALARA Planning Worksheet (System ALARA Manual, Section IV)
- ALARA Briefing Checklist (System ALARA Manual, Section IV)
- Execution Team Post-Job ALARA Critique (System ALARA Manual, Section IV)
- RP ALARA Post-Job Critique (System ALARA Manual, Section IV)

C Dose Tracking

- Task supervisor and execution team has primary responsibility for tracking all exposures received.
- RP shall be contacted when received dose exceeds expected values.

D Post-Job Critique

- Provide problem and improvement ideas to RP on Execution Team Post-job Critique (when provided) or use ALARA Suggestion Form.

507.8.2 ACCESS CONTROL

Controls are in place to limit access to site and in-plant areas for security and radiological safety purposes. The majority of radiologically controlled areas are located within the RCA; however, Radiation Control Zones may also be established at locations outside of the RCA.

EMERGENCY NOTIFICATION

1. [A] THIS IS A DRILL [X] ACTUAL EMERGENCY [X] INITIAL [ ] FOLLOW-UP MESSAGE NUMBER 01

2. SITE: Oconee UNIT: one REPORTED BY: Candidate's name

3. TRANSMITTAL TIME/DATE: ______ (Eastern) ___ mm / ___ dd / ___ yy CONFIRMATION PHONE NUMBER: (864) 882-7076

4. AUTHENTICATION (If Required): 60 (Number) Payload (Codeword)

5. EMERGENCY CLASSIFICATION:

[X] NOTIFICATION OF UNUSUAL EVENT [B] ALERT [C] SITE AREA EMERGENCY [D] GENERAL EMERGENCY

6. [X] Emergency Declaration At: [B] Termination At: TIME/DATE: ______ (Eastern) ___ mm / ___ dd / ___ yy (If B, go to item 16.)

7. EMERGENCY DESCRIPTION/REMARKS: Unidentified reactor coolant leakage exists. Current plant conditions DO NOT threaten public safety.

8. PLANT CONDITION: [A] IMPROVING [X] STABLE [C] DEGRADING

9. REACTOR STATUS: [A] SHUTDOWN: TIME/DATE: ______ (Eastern) ___ mm / ___ dd / ___ yy [X] 100 % POWER

10. EMERGENCY RELEASE(S):

[X] NONE (Go to item 14.) [B] POTENTIAL (GO TO ITEM 14.) [C] IS OCCURRING [D] HAS OCCURRED

NOT APPLICABLE

**11. TYPE OF RELEASE: [ ] ELEVATED [ ] GROUND LEVEL

[A] AIRBORNE: Started: ______ Time (Eastern) ___ / ___ Date / ___ Stopped: ______ Time (Eastern) ___ / ___ Date / ___

[B] LIQUID: Started: ______ Time (Eastern) ___ / ___ Date / ___ Stopped: ______ Time (Eastern) ___ / ___ Date / ___

**12. RELEASE MAGNITUDE: [ ] CURIES PER SEC. [ ] CURIES NORMAL OPERATING LIMITS: [ ] BELOW [ ] ABOVE

[A] NOBLE GASES ______ [B] IODINES ______

[C] PARTICULATES ______ [D] OTHER ______

**13. ESTIMATE OF PROJECTED OFFSITE DOSE: [ ] NEW [ ] UNCHANGED PROJECTION TIME: ______ (Eastern)

ESTIMATED DURATION: ______ HRS.

	TEDE mrem	Thyroid CDE mrem
SITE BOUNDARY		
2 MILES		
5 MILES		
10 MILES		

**14. METEOROLOGICAL DATA: [A] WIND DIRECTION (from) ______ ° [B] SPEED (mph) ______

[C] STABILITY CLASS ______ [D] PRECIPITATION (type) ______

15. RECOMMENDED PROTECTIVE ACTIONS:

[X] NO RECOMMENDED PROTECTIVE ACTIONS

[B] EVACUATE ______

[C] SHELTER IN-PLACE ______

[D] OTHER ______

16. APPROVED BY: ______ (Name) Emergency Coordinator (Title) TIME/DATE: ______ (Eastern) ___ mm / ___ dd / ___ yy

* If items 8-14 have not changed, only items 1-7 and 15-16 are required to be completed.

** Information may not be available on initial notifications.

Form 34888 (R1-94)

TRAINING PURPOSES ONLY

GOVERNMENT AGENCIES NOTIFIED

Record the name, date, time and agencies notified:

1. ______________________________
(name)

______________ ______________ OCONEE LAW ENFORCEMENT CENTER
(date) (time) (agency)

2. ______________________________
(name)

______________ ______________ PICKENS LAW ENFORCEMENT CENTER
(date) (time) (agency)

3. ______________________________
(name)

______________ ______________ STATE WARNING POINT (SCHD)
(date) (time) (agency)

4. ______________________________
(name)

______________ ______________ PICKENS EPD
(date) (time) (agency)

5. ______________________________
(name)

______________ ______________ OCONEE EPD
(date) (time) (agency)

6. ______________________________
(name)

______________ ______________ DHEC (BSHWM) Callback only
(date) (time) (agency)

7. ______________________________
(name)

______________ ______________ ______________
(date) (time) (agency)

RCA/RCZs are posted with warning signs that clearly identify the radiological hazard(s) and other pertinent access information. All individuals at a nuclear facility are expected to:

- Comply with RCA/RCZ entrance/exit requirements
- Read and comply with posted warning signs and/or barricades.
- Maintain the integrity of barricades after entering or exiting an RCA/RCZ or when working in close proximity to one.
- Regard RCZ ropes as if they are walls. Do not reach across or move ropes unless authorized by RP.
- Notify RP with questions or problems.

A. **Definitions**

Airborne Radioactivity Area - An area containing airborne radioactivity that is equal to or greater than 25% of 1 Weighted Derived Air Concentration (DAC).

Contaminated Area - An area where loose contamination equal to or greater than 1000 dpm/100cm^2 beta/gamma and/or 20 dpm/100cm^2 alpha exists.

Extra High Radiation Area - An area with a dose rate greater than 1000 mrem/hour at 30 centimeters. These areas are locked or guarded and require continuous RP coverage for entry. In areas that can not be reasonably locked, a flashing yellow light is used as a warning device.

High Radiation Area - An area with a dose rate greater than 100 mrem/hour at 30 centimeters.

Hot Spot - A localized source of radiation that is at least five times the general area dose rates, has a contact dose rate greater than 100 mrem/hour and/or is located where the potential for significant personnel exposure exists.

Low Exposure Waiting Area (LEWA) - An area where the dose rate is less than the general area. Usually, the lowest dose rate location in a room or area.

Protected Area - Area within the double fence around the plant. Access requires security identification.

Radiation Area - An area with a dose rate greater than 5 mrem/hour at 30 centimeters.

Radiation Control Area (RCA) - An area established within the Restricted Area to provide additional access control for radiological safety purposes. Requirements for entry are located in Section 507.5.

Radiation Control Zone (RCZ) - An area where specific radiological hazards exist and which are defined and controlled in accordance with 10CFR20 requirements. Requirements for entry are located in Section 507.5.

Radioactive Material - An area where radioactive materials are stored.

Restricted Area - Any area where access is controlled by the licensee for purposes of protecting individuals from exposure to radiation and radioactive materials. At DPC nuclear facilities, the Restricted Area includes the Reactor Building(s), Auxiliary Building(s), Turbine Building(s), Service Building and fenced area adjacent to the above buildings. At ONS, the Independent Spent Fuel Storage Installation, Radwaste Facility, and some warehouses are also in Restricted Areas.

Significant Dose Contributor - An area normally posted with a Significant Dose Contributor sign and florescent green ribbon to identify highest dose rate area(s) in a Radiation Area, High Radiation Area or Extra High Radiation Area.

Unrestricted Area - Any area where access is neither limited or controlled outside the site boundary fence.

Very High Radiation Area - An area with a dose rate greater than 500 rad/hour at 1 meter. These areas are locked at all times and requires continuous RP coverage for entry.

B. Plan Views

Plan Views are located at various elevation locations and outside of rooms inside the RCA (excluding Reactor Buildings). The Plan Views provide radiological information necessary for individuals to maintain their dose ALARA while performing tasks in a RCA/RCZ. Individuals performing work in an RCA/RCZ are expected to:

- Read Plan View(s) prior to entry.
- Know the general area and contact dose rates in the area where they will be working.
- Know the location of any contaminated areas and the contamination levels in the areas in which they will be working.
- Know the locations of Hot Spots in the work area.
- Know the location of a Low Exposure Waiting Area (LEWA) in the work area.
- Use the information to maintain their dose ALARA and prevent the spread of contamination.

C. Establishing High Radiation Areas (HRA)

Any individual having reason to suspect that an area is (or could be) an HRA and not properly identified as such, should immediately contact Radiation Protection so necessary surveys and corrective actions can be taken. Examples of reasons are:

- Equipment has been moved or rearranged in a room or area
- Dosimeter exposure readings do not correlate with anticipated exposures in the area.

D. HRA Access Requirements

Individuals requesting HRA access are responsible for:

- Ensuring they have not reached 'Alert Status' (80% of exposure limit). Individuals at 'Alert Status' are required to notify RP supervision of their current dose for approval of HRA entry.
- Knowing the radiological conditions in the work area.
- Knowing the location of work to be performed.
- Meeting one of the following requirements:
  1. RP coverage is provided.
  2. Alarming dosimetry is worn or portable area monitor is provided.
  3. Individual uses appropriate portable RP survey instrument and is qualified to instrument by RP standards.

507.8.3 EXPOSURE CONTROL METHODS

A. Time

Minimize time spent in an area by using training mock-ups, pre- job planning, avoiding rework, performing possible work outside of area, etc.

B. Distance

Maintain as great a distance from a radiation source as possible when performing work, using extension-type tools, using LEWAs when not performing work, etc.

C. Shielding

Temporary shielding is used in some situations to reduce dose. Notify RP if use of shielding is being considered for a job.

507.8.4 CONTAMINATION PREVENTION AND CONTROL METHODS

RP radioactive contamination controls are implemented to minimize contamination of personnel, areas and equipment. Surface contamination controls minimize possible inhalation or ingestion of radioactivity, skin dose from small particles of radioactivity and the spread of contamination to the environment. Some methods used to control contamination are:

A. Performing surveys, establishing RCZs and posting warning signs in areas where sources of contamination exist.

B. Limiting eating, drinking, storage of food, chewing and use of tobacco products to authorized areas.

- Persons with medical conditions, such as heart problems, are allowed to carry emergency medication inside RCZs, RP personnel should be contacted in advance for instructions.

C. Planning and performing work to minimize spread of contamination and reduce number of contaminated areas.

D. Decontaminating surfaces whenever practical. See Section 507.8.8 for policy.

E. Bagging and tagging contaminated material, equipment and tools. See Section 507.9.3 for policy.

F. Securing equipment/cords/hoses that cross contaminated RCZ boundaries.

G. Using protective clothing appropriately to prevent becoming contaminated:

- Adhere to protective clothing requirements specified on the SRWP or RWP unless authorized to deviate by RP.
- Consult RP if SRWP or RWP requirements should be changed.
- Use radiological protective clothing only in the RCA or RCZ unless otherwise specified by RP.
- Leave the work area if conditions change and cause specified protective clothing to be inappropriate or inadequate. Notify RP.
- Leave the work area if protective clothing becomes torn, soaked, untaped or unfastened. Notify RP.
- Control protective clothing during removal and place it in appropriate containers.
- Remove all protective clothing (except modesty garments) before exiting the RCZ boundary (unless directed otherwise by RP).

H. Catch Containment Program

Catch containments are used to prevent the spread of contamination that occurs as a result of uncontrolled plant system leaks. Routine inspections/audits of catch containments are performed to ensure containments are functioning properly and in good condition.

I. Radiological Respiratory Protection Program

The primary objective of the Respiratory Program is to minimize inhalation of airborne radioactive materials by individuals. The preferred method of achieving this objective is the use of engineering controls. Engineering controls are built into the nuclear facilities to remove airborne radioactive materials from the work environment. When additional engineering controls, such as local exhaust ventilation, containment or decontamination cannot be used or are not practical, the following methods are used to maintain dose ALARA.

- Increasing monitoring and access control
- Limiting exposure times
- Using respiratory protection equipment

J. Hot Particle Program

Discrete radioactive particles or hot particles are small, loose, highly radioactive particles. These particles produce high dose rates and are highly transportable due to their small size and static charge.

Radiological jobs are evaluated for the potential for hot particles. Controls for hot particles are implemented, as needed, based on historical data, the task performed and plant conditions.

507.8.5 USE OF PROTECTIVE CLOTHING AND RELATED EQUIPMENT

A. Coveralls

- Cloth coveralls are generally worn for normal to heavy work conditions. Outer personal clothing is not worn with cloth coveralls.
- Disposable coveralls may be worn over personal clothing for light work.
- Disposable coveralls may also be worn over cotton coveralls to provide a more impenetrable barrier to contaminants or to provide hot particle control.
- Specialty coveralls may be provided for specific jobs or tasks (for example, welding, fire protection, multiple dosimetry). These coveralls should only be worn for specific applications.

B Head Protection

- Hoods are worn to protect the head, hair and respirator from contamination.

C. Hand Protection

- Cotton liners are provided to facilitate the donning and removal of rubber gloves.

  Note: Cotton liners should not be used alone to protect the hands from contamination.
- Rubber gloves are provided to protect the hands from contamination when handling or working on contaminated components.
- Surgeon's gloves are available and may be used with Radiation Protection approval for work requiring a greater degree of manual dexterity.
- Other types of gloves may be used in addition to rubber gloves for additional safety concerns.

D. Shoe Protection

- Booties are worn over personal shoes and normally under shoe covers. Additional booties may be required by RP for additional protection.
- Rubber shoe covers are normally worn over booties in contaminated areas.
- Specialty boots may be specified for specific applications (such as jump boots, hip waders, knee boots).

E. Protection for Wet Conditions

- Water resistant clothing is usually specified for work in wet areas or for work which could result in the worker getting wet (for example, opening a system which could contain water).
- Water resistant suits may also be used in highly contaminated areas to provide maximum protection from contamination.
- Various types of water resistant suits are available:
  1. A two-piece suit is used for typical wet work.
  2. A one-piece suit is designed for use with air supplied hoods and is provided with air release vents.
  3. Breathable or light weight wet suits are also available when specified by RP.

F. Lab Coats

Lab coats are used for limited purposes such as performing analysis on radioactive samples or laundry operations. Lab coats provide only minimal protection from contamination and should not be used for most contamination control purposes.

G. Modesty Tops/Gym Shorts

These items are provided to be worn under protective clothing to protect personal underclothing and to provide body cover after protective clothing removal.

H. Facial Protection

- Goggles and face shields are provided to protect the eyes and face from contamination when working in close proximity to contaminated components and from the splashing of contaminated liquid. Goggles and face shields also provide protection from beta exposure.
- Facial protection (such as masks, face socks, disposable face shields) provides protection to the skin of the face from contamination.

I. Worker Suiting Up

- Obtain the required protective clothing.
- Remove all personal outer clothing, if required.

  Note: T-shirts are considered outer personal clothing unless worn as a undershirt.
- Put on gym shorts (and modesty top, if worn).
- Put on required coveralls.
- Place dosimetry on a neck strap under the coveralls or place dosimetry inside the coverall pocket with the TLD in front of the ED with the TLD beta window facing away from the body. Secure or tape the pocket opening.
- Put on booties over personal shoes (these may be worn inside or outside of the coveralls).
- Secure outer booties (tape, elastic, velcro, straps).
- Put on rubber shoe covers.
- Put on hood.
- Put on cotton gloves and then rubber gloves or surgical gloves.
- Secure gloves over sleeves of coveralls (tape, elastic, velcro, straps).
- Return unused protective clothing to storage location.

J. Removal of Protective Clothing

Do not throw protective clothing or equipment across the exit area; this can result in the spread of contamination.

- Remove all tape, elastic or velcro, if used (from wrists, ankles, etc.).
- Remove rubber shoe covers.
- Remove rubbers gloves by peeling them off inside out.
- Remove hood, taking care not to contaminate hair or face.
- Remove dosimetry.
- Remove the coveralls, peeling off inside out.
- Remove booties as you transfer to the step-off pad which is clean.
- Remove cotton gloves and proceed to nearest monitor wearing the gym shorts (and modesty top). Hand and foot monitoring is required at minimum before proceeding to change room.
- Monitor whole body to ensure you are not contaminated before dressing in personal clothes.
- In cases where more than one set of overalls are worn or multiple step off pads are used, follow the instructions of RP for dress out and removal.

507.8.6 PERSONNEL CONTAMINATION MONITORING

A. Exit from RCA

- Use Personnel Contamination Monitors (PCMs), when available, at RCA exit to perform whole body frisk prior to exiting.
- Call RP for release requirements when PCMs or hand-held friskers are not available.
- Use equipment and respond to alarms per posted instructions.
- Never attempt to use a hand-held frisker to frisk yourself following an alarm.
- Never attempt to decontaminate yourself following an alarm.

B. Exit from Contaminated RCZ

- Frisk hands and feet using nearest PCM. Hand and Foot Monitor or hand-held frisker prior to changing elevations (unless approved otherwise by RP).
- Proceed to nearest PCM and use a PCM to perform a whole body frisk.
- Use equipment and respond to alarms per posted instructions.
- Never attempt to use a hand held frisker to frisk yourself following an alarm.
- Never attempt to decontaminate yourself following an alarm.

C. Exit Under Emergency Conditions

Emergency conditions (fire, medical, security situations, security drills, etc.) may require personnel to exit without a proper frisk. Proper frisking is required after the emergency. Notify RP if frisking was not performed.

507.8.7 MATERIAL, EQUIPMENT AND TOOL MONITORING

A. Removal from RCA

Green-tagged items or items that were not used on contaminated equipment/system and were not inside a contaminated RCZ:

- Use Small Article Monitors (SAMs), when available, at RCA exit for release when items fit into SAM.
- Use hand-held frisker, when available, to release items that will not fit into SAMs.
- Call RP for release requirements when all SAMs at a location are out of service and a hand-held frisker is not available.
- Use equipment and respond to alarms per posted instructions.

B. Removal from Contaminated RCZ

- Wipe down item as directed by RP.
- Call RP to determine survey requirements.
- Wrap item(s) or place in poly/launderable bag.
- Obtain task qualified person to escort item to desired location if necessary per Section 507.9.3B or 507.9.3C.

507.8.8 DECONTAMINATION

A. Responsibility

Never perform decontamination functions unless authorized by RP.

<u>Large Work areas:</u>

- Immediately notify RP when a clean work area becomes contaminated.
- When the contamination was avoidable, individuals whose work resulted in the contamination should be made available to perform the decontamination work.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NRC-003/ADMIN A.4

OFFSITE COMMUNICATIONS FROM THE CONTROL ROOM

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

Perform Offsite Communications from the Control Room.

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

G2.4.43 2.8/3.5

Task Standard:

Complete the "Emergency Notification" form and make notifications per RP/0/B/1000/15A, Offsite Communications From The Control Room, within 15 minutes.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

RP/0/B/1000/15A, Offsite Communications From The Control Room
Oconee Nuclear Site Emergency Action Level Description Guidelines
Emergency Notification form

Validation Time: 15 min. **Time Critical:** YES

==

Candidate: ________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: _________________________ ______________________________ / ________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall 100% power IC
2. Freeze simulator.
3. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

RP/0/B/1000/15A, Offsite Communications From The Control Room
Emergency Notification form
Oconee Nuclear Site Emergency Action Level Description Guidelines

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

Communications to outside agencies will be simulated for this JPM.

INITIAL CONDITIONS:

- Oconee Unit 1 MODE 1, 100% power
  - 18 gpm unidentified RCS leakage
- Oconee Unit 2 MODE 1, 100% power
- Oconee Unit 3 MODE 1, 100% power

INITIATING CUES:

The OSM/Emergency Coordinator directs you to complete the Emergency Notification form and make the required notifications.

TIME CRITICAL

START TIME: _______

STEP 1:	Obtain a copy of the appropriate procedure.	**CRITICAL STEP**
STANDARD:	Operator obtains a copy RP/0/B/1000/15A, Offsite Communications From The Control Room, from the Emergency Procedures Cart.	___ SAT
COMMENTS:	Emergency Procedures Cart is located in TSC or Unit 3 CR document library in plant. Located in OSM office area in simulator.	___ UNSAT
STEP 2:	Obtain portable phone.	
STANDARD:	Portable phone is picked up.	___ SAT
COMMENTS:	Phone is located on column in Control Room.	___ UNSAT
STEP 3:	Obtain Yellow Folder from Emergency Procedures Cart.	**CRITICAL STEP**
STANDARD:	Yellow Folder is obtained from Emergency Procedures Cart.	___ SAT
COMMENTS:		___ UNSAT
STEP 4:	Obtain Emergency Action Level Guideline from Emergency Procedures Cart.	**CRITICAL STEP**
STANDARD:	Emergency Action Level Guidelines is obtained from Emergency Procedures Cart.	___ SAT
COMMENTS:		___ UNSAT

TEP 5: Review OSM/Emergency Coordinator Log. STANDARD: OSM/Emergency Coordinator Log is reviewed. ***Cue: When asked, give candidate the OSM/Emergency Coordinator Log for review.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6: Complete Line 1. STANDARD: "B" and Initial is marked. Message is numbered "1". COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: Complete Line 2. STANDARD: Site is marked Oconee. Unit affected is "1". Candidate's name is written in the Reported By blank. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: Complete Lines 5-10 STANDARD: Lines completed correctly, refer to completed form. ***Cue: As the OSM/Emergency Coordinator provide the following information to candidate:*** • ***Classification is "Unusual Event" (line 5)*** • ***Emergency Declaration At:: same as determined during Initial Conditions (line 6)*** • ***Emergency Description: Emergency Action Level Description Guidelines 4.2.U.1.a (line 7)*** • ***Plant Condition: "B" STABLE (line 8)*** • ***Reactor Status: U-1, 100% (line 9)*** • ***Emergency Releases: NONE (line 10)*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 9: Complete Lines 11-14 STANDARD: Write "Not Applicable" across Lines 11-14 (line 11-14), refer to completed form. COMMENTS:	___ SAT ___ UNSAT
STEP 10: Complete Lines 15 STANDARD: Recommended Protective Actions: • Mark "A", No Recommended Protective Actions COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 11: Provide completed form to OSM/Emergency Coordinator for approval and completion of Line 16. STANDARD: Give form to OSM/Emergency Coordinator (evaluator) ***Cue: As the OSM/Emergency Coordinator sign date and time on line 16 as shown on completed form.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 12: Copy Emergency Notification Form. STANDARD: Copy Emergency Notification Form using the fax machine. COMMENTS:	___ SAT ___ UNSAT
STEP 13: Fax Emergency Notification form to offsite agencies. STANDARD: Fax Emergency Notification Form using Speed Dial 14. ***Cue: Faxing the form will be simulated. Inform candidate that the fax machine is faxing properly.*** COMMENTS:	___ SAT ___ UNSAT

TEP 14: Notify SC State/County agencies by using Selective Signaling. STANDARD: Notify SC State/County agencies using Selective Signaling by dialing *4. Record Transmittal Time/Date whenever Selective Signaling Group Call number has been dialed and phone begins to ring. Check off the state and County agencies as they answer on back of form. ***Cue: Inform candidate that the following agencies are on the phone:*** ***Oconee County LEC*** ***Pickens County LEC*** ***State Warning Point EPD*** **NOTE: Time critical time stops here.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 15: Message authentication STANDARD: When requested provide authentication by using Authentication Code List. Provide code word "Payload" and record on Line 4. ***Cue: Request authentication. Candidate may request a certain agency to provide code number. Provide code number "60".*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 16: Read information from Emergency Notification Form to offsite agencies. STANDARD: Candidate reads information from Emergency Notification Form to offsite agencies. Read each line slowly and distinctly: Line 1…., Line 2 ….. Record Time/Name of agencies receiving notification (lines 1-3 on back on Emergency Notification Form. COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	Need to obtain correct procedure to complete task
3	Provides authentication code word sheet, which is used, for authentication.
4	proper words to put in the message
5	Needed to determine event that has occurred.
6	Proper filling out of message sheet is the majority of the task.
7	Proper filling out of message sheet is the majority of the task.
8	Proper filling out of message sheet is the majority of the task.
10	Proper filling out of message sheet is the majority of the task.
11	Proper filling out of message sheet is the majority of the task.
14	Have to notify State/County agencies by phone.
15	Needed to verify source of message.
16	Have to notify State/County agencies by phone.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

- Oconee Unit 1 MODE 1, 100% power
  - 18 gpm unidentified RCS leakage
- Oconee Unit 2 MODE 1, 100% power
- Oconee Unit 3 MODE 1, 100% power

INITIATING CUES:

The OSM/Emergency Coordinator directs you to complete the Emergency Notification form and make the required notifications.

TIME CRITICAL

EMERGENCY NOTIFICATION

1. [A] THIS IS A DRILL [B] ACTUAL EMERGENCY [ ] INITIAL [ ] FOLLOW-UP MESSAGE NUMBER ____________

2. SITE: Oconee UNIT: ____________ REPORTED BY: ____________

3. TRANSMITTAL TIME/DATE: ________ (Eastern) ___/___/___ (mm dd yy) CONFIRMATION PHONE NUMBER: (864) 882-7076

4. AUTHENTICATION (If Required): ____________ (Number) ____________ (Codeword)

5. EMERGENCY CLASSIFICATION:
   [A] NOTIFICATION OF UNUSUAL EVENT [B] ALERT [C] SITE AREA EMERGENCY [D] GENERAL EMERGENCY

6. [A] Emergency Declaration At: [B] Termination At: TIME/DATE: ________ (Eastern) ___/___/___ (mm dd yy) (If B, go to item 16.)

7. EMERGENCY DESCRIPTION/REMARKS: ____________

8. PLANT CONDITION: [A] IMPROVING [B] STABLE [C] DEGRADING

9. REACTOR STATUS: [A] SHUTDOWN: TIME/DATE: ________ (Eastern) ___/___/___ (mm dd yy) [B] ______ % POWER

10. EMERGENCY RELEASE(S):
    [A] NONE (Go to item 14.) [B] POTENTIAL (GO TO ITEM 14.) [C] IS OCCURRING [D] HAS OCCURRED

**11. TYPE OF RELEASE: [ ] ELEVATED [ ] GROUND LEVEL
    [A] AIRBORNE: Started: ________ (Time (Eastern)) ___/___/___ (Date) Stopped: ________ (Time (Eastern)) ___/___/___ (Date)
    [B] LIQUID: Started: ________ (Time (Eastern)) ___/___/___ (Date) Stopped: ________ (Time (Eastern)) ___/___/___ (Date)

**12. RELEASE MAGNITUDE: [ ] CURIES PER SEC. [ ] CURIES NORMAL OPERATING LIMITS: [ ] BELOW [ ] ABOVE
    [A] NOBLE GASES ____________ [B] IODINES ____________
    [C] PARTICULATES ____________ [D] OTHER ____________

**13. ESTIMATE OF PROJECTED OFFSITE DOSE: [ ] NEW [ ] UNCHANGED PROJECTION TIME: ________ (Eastern)

ESTIMATED DURATION: ________ HRS.

	TEDE mrem	Thyroid CDE mrem
SITE BOUNDARY		
2 MILES		
5 MILES		
10 MILES		

**14. METEOROLOGICAL DATA: [A] WIND DIRECTION (from) ________ ° [B] SPEED (mph) ________
    [C] STABILITY CLASS ________ [D] PRECIPITATION (type) ________

15. RECOMMENDED PROTECTIVE ACTIONS:
    [A] NO RECOMMENDED PROTECTIVE ACTIONS
    [B] EVACUATE ____________
    [C] SHELTER IN-PLACE ____________
    [D] OTHER ____________

16. APPROVED BY: ____________ (Name) Emergency Coordinator (Title) TIME/DATE: ________ (Eastern) ___/___/___ (mm dd yy)

* If items 8-14 have not changed, only items 1-7 and 15-16 are required to be completed.

** Information may not be available on initial notifications.

TRAINING PURPOSES ONLY

Form 34888 (R1-94)

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

- Oconee Unit 1 MODE 1, 100% power
  - 18 gpm unidentified RCS leakage
- Oconee Unit 2 MODE 1, 100% power
- Oconee Unit 3 MODE 1, 100% power

INITIATING CUES:

The OSM/Emergency Coordinator directs you to complete the Emergency Notification form and make the required notifications.

TIME CRITICAL

OCONEE NUCLEAR SITE
EMERGENCY DRILL/EVENT TIME LOG
TSC EMERGENCY COORDINATOR

DATE ____________ LOGKEEPER NAME ______________________

TIME	EVENT/DECISION DESCRIPTION
	18 gpm unidentified RCS leakage verified on Unit #1.

Authentication Code List
Effective 1/1/1999 – 12/31/2001

1. Explorer
2. Gemini
3. Voyager
4. Viking
5. Fuel
6. Challenger
7. Atlas
8. Apollo
9. Thor
10. Navajo
11. Mercury
12. Nike
13. Galaxy
14. Satellite
15. Agena
16. Centaur
17. Titan
18. Pegasus
19. Jupiter
20. Bomarc
21. Mace
22. Trident
23. Peacekeeper
24. Minuteman
25. Oxydizer
26. Penguin
27. Delta
28. Chevaline
29. Juno
30. Pershing
31. Skybolt
32. Vanguard
33. Malabar
34. Saturn
35. Bumper
36. Lark
37. Sunnyvale
38. Rascal
39. Corporal
40. Polaris
41. Spacecraft
42. Snark
43. Ranger
44. Tiros
45. Echo
46. Vela
47. Surveyor
48. Syncom
49. Mariner
50. Pioneer
51. Launch
52. Orbiter
53. NASA
54. Mariner
55. Westar
56. Skylab
57. Booster
58. Palapa
59. Marisat
60. Payload
61. Columbia
62. Matador
63. Ariane
64. Atlantis
65. Discovery
66. Galileo
67. Telstar
68. Athena
69. Starbird
70. Shuttle
71. Endeavor
72. Antigua
73. Ascension
74. Redstone
75. Andros
76. Sentinel
77. Poseidon
78. Kourou
79. Vandenburg
80. Cape Canaveral
81. Dynasoar
82. Satcom
83. Intelsat
84. Harpoon
85. Hound Dog
86. Tomahawk
87. Lacrosse
88. Spacelab
89. Navstar
90. Magellan
91. Cassini
92. Hubble
93. Skynet
94. Ulysses
95. Rollback
96. Umbilical
97. ARIA
98. Comstar
99. Castor
100. Nimbus
101. Landsat
102. Soyuz
103. Mir
104. Sputnik
105. Astronaut
106. Cosmonaut
107. Aerobee
108. Gantry
109. Blockhouse
110. Telemetry
111. Antenna
112. Aurora
113. Crawler
114. Shroud
115. Dryden
116. White Sands
117. Lockheed
118. Boeing
119. Blue Scout
120. GEMS
121. Star Cast
122. Solar
123. Goddard
124. Bermuda
125. Bahama
126. Analog
127. Digital
128. Honeywell
129. Raytheon
130. Acquisition

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

4.7.S.1 Initiating Condition: Control Room Evacuation And Plant Control Cannot Be Established

Emergency Action Level

4.7.S.1.a *Control Room Evacuation Has Been Initiated AND* ***Control Of The Plant Cannot Be Established*** *From The Aux Shutdown Panel Or The SSF Within 15 Minutes*

An event has occurred requiring evacuation of the control room. Plant operators have not been successful in establishing control of the reactor within 15 minutes from a remote location that is capable of reducing and maintaining reactor coolant temperature less than 250 °F. Current plant conditions *DO NOT* threaten public safety.

4.7.S.2 Initiating Condition: Keowee Hydro Dam Failure

Emergency Action Level

4.7.S.2.a *Imminent/Actual Dam Failure (Includes Any Of The Following):*

- *Keowee Hydro Dam*
- *Little River Dam*
- *Dikes A, B, C, or D*
- *Intake Canal Dike*

NOTE 1: USE THE FOLLOWING EMERGENCY ACTION LEVEL DESCRIPTION FOR NOTIFICATION OF AN IMMINENT FAILURE OF A KEOWEE HYDROELECTRIC PROJECT DAM/DIKE

A 'Condition A - Failure Is Imminent Or Has Occurred' exists for the Keowee Hydroelectric Project due to imminent failure of ______*(state dam/dike)*______. In this situation, current conditions MAY AFFECT public safety. Protective actions for downstream residents are required.

NOTE 2: USE THE FOLLOWING EMERGENCY ACTION LEVEL DESCRIPTION FOR NOTIFICATION OF AN ACTUAL FAILURE OF A KEOWEE HYDROELECTRIC PROJECT DAM/DIKE

A 'Condition A - Failure Is Imminent Or Has Occurred' exists for the Keowee Hydroelectric Project due to failure of ______*(state dam/dike)*______. In this situation, current conditions AFFECT public safety. Protective actions for downstream residents are required.

(R06-97) INFORMATION ONLY

Sim
Sim Cat 8
NRC
JPP
SR

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. RP/0/B/1000/002

Revision No 4

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Control Room Emergency Coordinator Procedure

(4) Prepared By Rodney Brown Date 1/17/2000

(5) Requires 10CFR50.59 evaluation?

■ Yes (New procedure or revision with major changes)

☐ No (Revision with minor changes)

☐ No (To incorporate previously approved changes)

(6) Reviewed By Robert Taylor (QR) Date 1-19-00

Cross-Disciplinary Review By (QR)NA Date 1-17-00

Reactivity Mgmt. Review By Robert Taylor (QR)NA ✓ Date 1-19-00

(7) Additional Reviews

QA Review By ____ Date ____

Reviewed By ____ Date 1/19/2000

Reviewed By ____ Date ____

(8) Temporary Approval *(if necessary)*

By ____ (SRO/QR) Date ____

By ____ (QR) Date ____

(9) Approved By M. R. Thorne Date 2-14-2000

PERFORMANCE ***(Compare with control copy every 14 calendar days while work is being performed.)***

(10) Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

(11) Date(s) Performed ____

Work Order Number (WO#) ____

COMPLETION

(12) Procedure Completion Verification

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?

☐ Yes ☐ NA Listed enclosures attached?

☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?

☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?

☐ Yes ☐ NA Procedure requirements met?

Verified By ____ Date ____

(13) Procedure Completion Approved ____ Date ____

Remarks *(Attach additional pages, if necessary)*

FEB 2000 Received Operator Training

Duke Power Company Oconee Nuclear Site **Control Room Emergency Coordinator Procedure** **Reference Use**	Procedure No. RP/0/B/1000/002
	Revision No. 004
	Electronic Reference No. OX002WOT

Control Room Emergency Coordinator Procedure

NOTE: This procedure is an implementing procedure to the Oconee Nuclear Site Emergency Plan and must be forwarded to Emergency Planning within three (3) working days of approval.

1. Symptoms

1.1 Events are in process or have occurred which require activation of the Oconee Nuclear Site Emergency Plan.

2. Immediate Actions

The Operations Shift Manager/Emergency Coordinator shall use this procedure until relieved by the Station Manager/Alternate in the Technical Support Center.

NOTE: Place Keeping Aids: ☐ at left of steps may be used for procedure place keeping. (☑)

☐ 2.1 **IF** **General Emergency** conditions are met,
THEN **GO TO** Enclosure 4.1 (General Emergency).

☐ 2.2 **IF** **Site Area Emergency** conditions are met,
THEN **GO TO** Enclosure 4.2 (Site Area Emergency).

☐ 2.3 **IF** **Alert** conditions are met,
THEN **GO TO** Enclosure 4.3 (Alert).

☐ 2.4 **IF** **Unusual Event** conditions are met,
THEN **GO TO** Enclosure 4.4 (Unusual Event).

☐ 2.5 **IF** An Emergency Classification does **NOT** exist and ERO Activation is desired,
THEN **GO TO** Step 1.6 of Enclosure 4.4 (Unusual Event).

3. Subsequent Actions

NOTE: Actions are **NOT** required to be followed in any particular sequence.

☐ 3.1 **IF** RIA 46 is on scale,
THEN Use Enclosure 4.3 of RP/0/B/1000/001, (Emergency Classification), to determine if the emergency classification should be upgraded to a Site Area Emergency or General Emergency based on radiation activity.

☐ 3.1.1 Instruct RP to perform an Offsite Dose Calculation and determine any additional Protective Action Recommendations.

☐ 3.2 **IF** RIA 57 or 58 are on scale,
THEN Use Enclosure 4.1 or 4.8 of RP/0/B/1000/001, (Emergency Classification), to determine if the emergency classification should be upgraded to a Site Area Emergency or General Emergency based on radiation activity.

☐ 3.3 **IF** RIA 16 or 17 are in Alert or High Alarm (≥ 2.5 mR/hr),
THEN Instruct RP to perform an Offsite Dose Calculation using the RIA values.

☐ 3.3.1 Use Enclosure 4.3 of RP/0/B/1000/001, (Emergency Classification), and the Offsite Dose Calculation results to determine if the emergency classification should be upgraded to a Site Area Emergency or General Emergency based on dose projection at the site boundary.

☐ 3.3.2 Determine any additional Protective Action Recommendations.

☐ 3.4 **IF** A large scale fire or flood damage has occurred or is occurring,
THEN Use RP/0/B/1000/022, (Procedure For Site Fire Damage Assessment And Repair), to determine additional actions that may be required.

☐ 3.5 **IF** A Security Event is in progress,
THEN Use RP/0/B/1000/007, (Security Event), to determine additional actions that may be required.

☐ 3.6 **IF** A hazardous substance has been released,
THEN Use RP/0/B/1000/017, (Spill Response), to determine additional actions that may be required.

NOTE: Priority should be placed on providing treatment for the most life-threatening event (i.e., medical vs radiation exposure - OSC procedure RP/0/B/1000/011, (Planned Emergency Exposure). The Emergency Coordinator may authorize (either verbal or signature) exposures greater than 25 rem TEDE (Total Effective Dose Equivalent) for lifesaving missions.

☐ 3.7 **IF** A medical response is required,
THEN Use RP/0/1000/016, (Medical Response).

☐ 3.7.1 Document verbal approval of Planned Emergency Exposures required for lifesaving missions in the Control Room Emergency Coordinator Log.

☐ 3.8 **IF** Changing plant conditions require an emergency classification upgrade,
THEN **GO TO** the applicable enclosure, designated in the Immediate Actions section of this procedure, required for the appropriate emergency classification.

☐ 3.9 Announce over the Plant Public address System the following information:

☐ 3.9.1 The current emergency classification level and plant status UE/Alert/SAE/GE

☐ 3.9.2 If appropriate, the status of contamination and how people are to handle themselves:

Plant personnel should assume they are contaminated until surveyed by RP or until they have frisked themselves.

NO eating, drinking, or smoking until the area is cleared by RP

Identify areas of contamination to plant personnel:

____

____

NOTE:
- The Outside Air Booster Fans (Control Room Ventilation System - CRVS) are used to provide positive pressure in the Control Room to prevent smoke, toxic gases, or radioactivity from entering the area as required by NuReg 0737.
- Chlorine Monitor Alarm will either stop the Air Booster Fans or will not allow them to start.

☐ 3.10 **IF** There is an indication that smoke or toxic gases from the Turbine Building may enter the Control Room.
THEN Instruct Control Room personnel to turn on the Outside Air Booster Fans.

Fans On _____ Time: _____

☐ 3.11 **IF** RIA-39 is in **ALARM**,
THEN Follow AP/1/2/3/1700/018, (Abnormal Release Of Radioactivity).

Fans On _____ Time: _____

☐ Secure fans if back-up sample by RP shows RIA-39 is in error.

☐ Isolate source of airborne contamination to the Control Room if sample from RP shows RIA alarm is valid.

☐ Secure fans if dose levels in CR/TSC/OSC are increased by the addition of outside filtered air.

Fans Off _____ Time: _____

☐ 3.12 **IF** The Emergency Response Organization was activated,
THEN Provide turnover to the Technical Support Center using Enclosure 4.5 of this procedure.

Technical Support Center Activated _____ Time: _____

A. Turn over all emergency response procedures in use to the TSC.

☐ 3.13 **IF** An Unusual Event classification is being terminated,
THEN **REFER TO** Enclosure 4.6, (Emergency Classification Termination Criteria), of this procedure for termination guidance.

☐ 3.13.1 Verify that the Offsite Communicator has provided termination message to the offsite agencies.

NOTE:	The EP Section shall develop a written report, for signature by the Site Vice President, to the State Emergency Preparedness Agency, Oconee County EPD, and Pickens County EPD within 24 working hours of the event termination.

☐ 3.13.2 Notify Emergency Planning Section (Emergency Planning Duty person after hours) that the Unusual Event has been terminated.

☐ 3.13.3 Emergency Planning shall hold a critique following termination of any actual Unusual Event.

4. Enclosures

4.1 General Emergency

4.2 Site Area Emergency

4.3 Alert

4.4 Unusual Event

4.5 Operations Shift Manager to TSC Emergency Coordinator Turnover Sheet

4.6 Emergency Classification Termination Criteria

4.7 Condition A/Condition B Response Actions

4.8 ERO Pager Activation By Security

1. Immediate Actions

NOTE:	
	• State and County Agencies must be notified of event classification within **15 minutes** of Emergency Declaration.
	• Provide Offsite Communicator with declaration time.

☐ 1.1 **IF** It has been determined that an Emergency Action Level for an Initiating Condition has been met,

THEN Declare a **General Emergency**.

Time of Declaration: ______

☐ 1.2 Appoint a person to maintain the Emergency Coordinator Log **OR** maintain the log yourself.

NOTE:	
	• Remind the Control Room Offsite Communicator that Follow Up notifications (updates) are required at least every **60 Minutes** for this classification.
	• Condition A, Dam Failure (Keowee or Jocassee), **OR** Condition B also requires notification of the Georgia Emergency Management Agency and National Weather Service. Remind the Control Room Offsite Communicator to notify these agencies in addition to and after SC State, Oconee County, and Pickens County.

☐ 1.3 Appoint Control Room Offsite Communicator(s).

☐ 1.4 Provide the following Protective Action Recommendations for use by the Offsite Communicator to complete the Emergency Notification Form.

PROTECTIVE ACTION RECOMMENDATION	PICKENS COUNTY SECTORS							OCONEE COUNTY SECTORS						
	A0	A1	B1	C1	A2	B2	C2	A0	D1	E1	F1	D2	E2	F2
EVACUATE	X	X	X	X				X	X	X	X			
SHELTER					X	X	X					X	X	X

1.4.1 **IF** Condition A, Imminent or Actual Dam Failure (Keowee or Jocassee) exists,

THEN **REFER TO** Enclosure 4.7, (Condition A/Condition B Response Actions), Step 1.0, for additional Protective Action Recommendations.

> **NOTE:** Steps 1.6 - 1.13 may be started/completed while the Emergency Notification Form is being prepared by the Offsite Communicator.

☐ 1.5 Review and approve completed Emergency Notification Form.

1.5.1 Sign Emergency Notification Form.

> **NOTE:** Activate the Alternate TSC and OSC in the Oconee Office Building, Rooms 316 and 316A, if a fire in the Turbine Building, flooding conditions, security events, or onsite/offsite hazardous materials spill have occurred or area occurring.

☐ 1.6 Activate the Emergency Response Organization (ERO) by completing the following actions.

1.6.1 Activate ERO Pagers as follows:

> **NOTE:** Flooding/dam failure/earthquake conditions assume bridges may be impassable to reach emergency facilities. Provide the code below for these conditions.

☐ A. **IF** ERO activation for an Emergency (Blue Echo)is required,
THEN Press ERO Pager Activation Panel Button **1**.

☐ B. **IF** ERO activation for an Emergency affecting bridges (Blue Echo Bridges) is required,
THEN Press ERO Pager Activation Panel Button **2**.

☐ C. **IF** ERO activation for a Drill (Blue Delta) is required,
THEN Press ERO Pager Activation Panel Button **3**.

☐ D. **IF** ERO activation for a Drill affecting bridges (Blue Delta Bridges) is required,
THEN Press ERO Pager Activation Panel Button **4**.

☐ E. **IF** Alternate TSC/OSC will be used,
THEN Press ERO Pager Activation Panel Button **5**.

☐ F. **IF** A Security Event is in progress,
THEN Press ERO Pager Activation Panel Button **6**.

☐ 1.6.2 Wait one minute and repeat step 1.6.1.

☐ 1.6.3 Monitor ERO Pager and verify that message has been provided to the ERO.

☐ 1.6.4 Repeat steps 1.6.1 - 1.6.3 if message is not displayed on ERO Pager.

A. **REFER TO** Enclosure 4.8, (ERO Pager Activation By Security), if the ERO Pager is not activated by the completion of Steps 1.6.1 - 1.6.3.

☐ 1.6.5 **IF** ERO activation is after normal working hours,
THEN Contact Security at extension 3636 or 2309.

Security Officer Name ______________________

A. Request Security Officer to activate the CAN call list.

WARNING: Conducting Site Assembly during a Security Event may not be prudent.

☐ 1.7 Contact the Security Shift Supervisor.

1.7.1 Inform the Security Shift Supervisor that the ERO has been activated.

1.7.2 Discuss the need to conduct Site Assembly.

☐ 1.8 **IF** A Security Event does **NOT** exist,
OR A Security Event does exist and the Security Shift Supervisor agrees,
THEN Conduct Site Assembly per RP/0/B/1000/009, (Procedure For Site Assembly), Enclosure 4.1 and 4.3.

☐ 1.9 **IF** Area Radiation Monitors are in **ALARM**,
OR Steam Line Break has occurred,
THEN Contact shift RP and dispatch onsite monitoring teams.

NOTE:
- Remind the NRC Communicator to complete the NRC Briefing Form prior to contacting the NRC.
- An open line to the NRC may be required.

☐ 1.10 Appoint an SRO to notify the NRC immediately after notification of the Offsite Agencies but not later than **one (1) hour** after declaration of the emergency.

1.10.1 NRC Communicator (SRO) Name ______________________

NOTE: The NRC Communicator should be used to activate ERDS.

☐ 1.10.2 Start the Emergency Response Data System (ERDS) for unit(s) involved within **one (1) hour** of the emergency classification.

A. **REFER TO** RP/0/B/1000/003A, (ERDS Operation).

☐ 1.11 Evacuate all non-essential personnel from the site after personnel accountability has been reached.

1.11.1 **REFER TO** RP/0/B/1000/010, (Procedure For Emergency Evacuation/ Relocation Of Site Personnel).

☐ 1.12 **IF** Condition A, Imminent or Actual Dam Failure (Keowee or Jocassee),
OR Condition B (Keowee) exists,
THEN **REFER TO** Enclosure 4.7, (Condition A/Condition B Response Actions), Step 2.0 or 3.0, for additional response actions.

☐ 1.13 Notify the Unit Operations Coordinator/Duty person of emergency status.

☐ 1.14 Return to Step 3.0, (Subsequent Actions), of this procedure.

1. Immediate Actions

NOTE:	• State and County Agencies must be notified of event classification within **15 minutes** of Emergency Declaration. • Provide Offsite Communicator with declaration time.

☐ 1.1 **IF** It has been determined that an Emergency Action Level for an Initiating Condition has been met,
THEN Declare a **Site Area Emergency**.

Time of Declaration: ______

☐ 1.2 Appoint a person to maintain the Emergency Coordinator Log **OR** maintain the log yourself.

NOTE:	• Remind the Control Room Offsite Communicator that Follow Up notifications (updates) are required at least every **60 Minutes** for this classification. • Condition A, Dam Failure (Keowee or Jocassee), **OR** Condition B also requires notification of the Georgia Emergency Management Agency and National Weather Service. Remind the Control Room Offsite Communicator to notify these agencies in addition to and after SC State, Oconee County, and Pickens County.

☐ 1.3 Appoint Control Room Offsite Communicator(s).

☐ 1.4 Provide the Protective Action Recommendations from Enclosure 4.7, (Condition A/ Condition B Response Actions), Step 1.0, for use by the Offsite Communicator if a Condition A, Imminent or Actual Dam Failure, exists.

NOTE:	Steps 1.6 - 1.12 may be started/completed while the Emergency Notification Form is being prepared by the Offsite Communicator.

☐ 1.5 Review and approve completed Emergency Notification Form.

1.5.1 Sign Emergency Notification Form.

NOTE: Activate the Alternate TSC and OSC in the Oconee Office Building, Rooms 316 and 316A, if a fire in the Turbine Building, flooding conditions, security events, or onsite/offsite hazardous materials spill have occurred or area occurring.

☐ 1.6 Activate the Emergency Response Organization (ERO) by completing the following actions.

1.6.1 Activate ERO Pagers as follows:

NOTE: Flooding/dam failure/earthquake conditions assume bridges may be impassable to reach emergency facilities. Provide the code below for these conditions.

☐ A. **IF** ERO activation for an Emergency (Blue Echo)is required,
THEN Press ERO Pager Activation Panel Button **1**.

☐ B. **IF** ERO activation for an Emergency affecting bridges (Blue Echo Bridges) is required,
THEN Press ERO Pager Activation Panel Button **2**.

☐ C. **IF** ERO activation for a Drill (Blue Delta) is required,
THEN Press ERO Pager Activation Panel Button **3**.

☐ D. **IF** ERO activation for a Drill affecting bridges (Blue Delta Bridges) is required,
THEN Press ERO Pager Activation Panel Button **4**.

☐ E. **IF** Alternate TSC/OSC will be used,
THEN Press ERO Pager Activation Panel Button **5**.

☐ F. **IF** A Security Event is in progress,
THEN Press ERO Pager Activation Panel Button **6**.

☐ 1.6.2 Wait one minute and repeat step 1.6.1.

☐ 1.6.3 Monitor ERO Pager and verify that message has been provided to the ERO.

☐ 1.6.4 Repeat steps 1.6.1 - 1.6.3 if message is not displayed on ERO Pager.

A. **REFER TO** Enclosure 4.8, (ERO Pager Activation By Security), if the ERO Pager is not activated by the completion of Steps 1.6.1 - 1.6.3.

☐ 1.6.5 **IF** ERO activation is after normal working hours,
THEN Contact Security at extension 3636 or 2309.

Security Officer Name ______________________

A. Request Security Officer to activate the CAN call list.

WARNING: Conducting Site Assembly during a Security Event may not be prudent.

☐ 1.7 Contact the Security Shift Supervisor.

1.7.1 Inform the Security Shift Supervisor that the ERO has been activated.

1.7.2 Discuss the need to conduct Site Assembly.

☐ 1.8 **IF** A Security Event does **NOT** exist,
OR A Security Event does exist and the Security Shift Supervisor agrees,
THEN Conduct Site Assembly per RP/0/B/1000/009, (Procedure For Site Assembly), Enclosure 4.1 and 4.3.

☐ 1.9 **IF** Area Radiation Monitors are in **ALARM**,
OR Steam Line Break has occurred,
THEN Contact shift RP and dispatch onsite monitoring teams.

NOTE:
- Remind the NRC Communicator to complete the NRC Briefing Form prior to contacting the NRC.
- An open line to the NRC may be required.

☐ 1.10 Appoint an SRO to notify the NRC immediately after notification of the Offsite Agencies but not later than **one (1) hour** after declaration of the emergency.

1.10.1 NRC Communicator (SRO) Name ______________________

NOTE: The NRC Communicator should be used to activate ERDS.

☐ 1.10.2 Start the Emergency Response Data System (ERDS) for unit(s) involved within **one (1) hour** of the emergency classification.

A. **REFER TO** RP/0/B/1000/003A, (ERDS Operation).

☐ 1.11 **IF** Condition A, Imminent or Actual Dam Failure (Keowee or Jocassee),
OR Condition B (Keowee) exists,
THEN **REFER TO** Enclosure 4.7, (Condition A/Condition B Response Actions), Step 2.0 or 3.0, for additional response actions.

☐ 1.12 Notify the Unit Operations Coordinator/Duty person of emergency status.

☐ 1.13 Return to Step 3.0, (Subsequent Actions), of this procedure.

1. Immediate Actions

> **NOTE:**
> - State and County Agencies must be notified of event classification within **15 minutes** of Emergency Declaration.
> - Provide Offsite Communicator with declaration time.

☐ 1.1 **IF** It has been determined that an Emergency Action Level for an Initiating Condition has been met,
THEN Declare an **Alert**.

Time of Declaration: ______

☐ 1.2 Appoint a person to maintain the Emergency Coordinator Log **OR** maintain the log yourself.

> **NOTE:**
> - Remind the Control Room Offsite Communicator that Follow Up notifications (updates) are required at least every **60 minutes** for this classification.
> - Condition B for Keowee Hydro Project Dams/Dikes also requires notification of the Georgia Emergency Management Agency and National Weather Service. Remind the Control Room Offsite Communicator to notify these agencies in addition to and after SC State, Oconee County, and Pickens County.

☐ 1.3 Appoint Control Room Offsite Communicator(s).

> **NOTE:** Steps 1.5 - 1.11 may be started/completed while the Emergency Notification Form is being prepared by the Offsite Communicator.

☐ 1.4 Review and approve completed Emergency Notification Form.

1.4.1 Sign Emergency Notification Form.

NOTE: Activate the Alternate TSC and OSC in the Oconee Office Building, Rooms 316 and 316A, if a fire in the Turbine Building, flooding conditions, security events, or onsite/offsite hazardous materials spill have occurred or area occurring.

☐ 1.5 Activate the Emergency Response Organization (ERO) by completing the following actions.

1.5.1 Activate ERO Pagers as follows:

NOTE: Flooding/dam failure/earthquake conditions assume bridges may be impassable to reach emergency facilities. Provide the code below for these conditions.

☐ A. **IF** ERO activation for an Emergency (Blue Echo)is required,
THEN Press ERO Pager Activation Panel Button **1**.

☐ B. **IF** ERO activation for an Emergency affecting bridges (Blue Echo Bridges) is required,
THEN Press ERO Pager Activation Panel Button **2**.

☐ C. **IF** ERO activation for a Drill (Blue Delta) is required,
THEN Press ERO Pager Activation Panel Button **3**.

☐ D. **IF** ERO activation for a Drill affecting bridges (Blue Delta Bridges) is required,
THEN Press ERO Pager Activation Panel Button **4**.

☐ E. **IF** Alternate TSC/OSC will be used,
THEN Press ERO Pager Activation Panel Button **5**.

☐ F. **IF** A Security Event is in progress,
THEN Press ERO Pager Activation Panel Button **6**.

☐ 1.5.2 Wait one minute and repeat step 1.5.1.

☐ 1.5.3 Monitor ERO Pager and verify that message has been provided to the ERO.

☐ 1.5.4 Repeat steps 1.5.1 - 1.5.3 if message is not displayed on ERO Pager.

A. **REFER TO** Enclosure 4.8, (ERO Pager Activation By Security), if the ERO Pager is not activated by the completion of Steps 1.5.1 - 1.5.3.

☐ 1.5.5 **IF** ERO activation is after normal working hours,
THEN Contact Security at extension 3636 or 2309.

Security Officer Name ______________________

A. Request Security Officer to activate the CAN call list.

WARNING: Conducting Site Assembly during a Security Event may not be prudent.

☐ 1.6 Contact the Security Shift Supervisor.

1.6.1 Inform the Security Shift Supervisor that the ERO has been activated.

1.6.2 Discuss the need to conduct Site Assembly.

☐ 1.7 **IF** A Security Event does **NOT** exist,
OR A Security Event does exist and the Security Shift Supervisor agrees,
THEN Conduct Site Assembly per RP/0/B/1000/009, (Procedure For Site Assembly), Enclosure 4.1 and 4.3.

☐ 1.8 **IF** Area Radiation Monitors are in **ALARM**,
OR Steam Line Break has occurred,
THEN Contact shift RP and dispatch onsite monitoring teams

NOTE:
- Remind the NRC Communicator to complete the NRC Briefing Form prior to contacting the NRC.
- An open line to the NRC may be required.

☐ 1.9 Appoint an SRO to notify the NRC immediately after notification of the Offsite Agencies but not later than **one (1) hour** after declaration of the emergency.

1.9.1 NRC Communicator (SRO) Name ____________________

NOTE: The NRC Communicator should be used to activate ERDS.

☐ 1.9.2 Start the Emergency Response Data System (ERDS) for unit(s) involved within **one (1) hour** of the emergency classification.

A. **REFER TO** RP/0/B/1000/003A, (ERDS Operation).

☐ 1.10 **IF** Condition B at Keowee exists,
THEN **REFER TO** Enclosure 4.7, (Condition A/Condition B Response Actions), Step 3.0, for additional response actions.

☐ 1.11 Notify the Unit Operations Coordinator/Duty person of emergency status.

☐ 1.12 Return to Step 3.0, (Subsequent Actions), of this procedure.

1. Immediate Actions

NOTE:	• State and County Agencies must be notified of event classification within **15 minutes** of Emergency Declaration. • Provide Offsite Communicator with declaration time.

☐ 1.1 **IF** It has been determined that an Emergency Action Level for an Initiating Condition has been met,
THEN Declare an **Unusual Event**.

Time of Declaration: ______

☐ 1.2 Appoint a person to maintain the Emergency Coordinator Log **OR** maintain the log yourself.

NOTE:	• Remind the Control Room Offsite Communicator that an Initial Message and a Termination Message are required for this classification. No Follow Up Notifications (updates) are required unless requested by the Offsite Agencies. • Condition B for Keowee Hydro Project Dams/Dikes also requires notification of the Georgia Emergency Management Agency and National Weather Service. Remind the Control Room Offsite Communicator to notify these agencies in addition to and after SC State, Oconee County, and Pickens County.

☐ 1.3 Appoint Control Room Offsite Communicator(s).

NOTE:	Steps 1.5 - 1.11 may be started/completed while the Emergency Notification Form is being prepared by the Offsite Communicator.

☐ 1.4 Review and approve completed Emergency Notification Form.

1.4.1 Sign Emergency Notification Form.

☐ 1.5 **IF** Condition B at Keowee exists,
THEN **REFER TO** Enclosure 4.7, (Condition A/Condition B Response Actions), Step 3.0, for additional response actions.

NOTE:	• Activation of the ERO is **NOT** required for an Unusual Event Classification. • Activate the Alternate TSC and OSC in the Oconee Office Building, Rooms 316 and 316A, if a fire in the Turbine Building, flooding conditions, security events, or onsite/ offsite hazardous materials spills have occurred or are occurring.

☐ 1.6 **IF** Emergency Response Organization (ERO) activation is desired,
THEN Complete the following actions.

1.6.1 Activate ERO Pagers as follows:

NOTE:	Flooding/dam failure/earthquake conditions assume bridges may be impassable to reach emergency facilities. Provide the code below for these conditions.

☐ A. **IF** ERO activation for an Emergency (Blue Echo)is required,
THEN Press ERO Pager Activation Panel Button **1**.

☐ B. **IF** ERO activation for an Emergency affecting bridges (Blue Echo Bridges) is required,
THEN Press ERO Pager Activation Panel Button **2**.

☐ C. **IF** ERO activation for a Drill (Blue Delta) is required,
THEN Press ERO Pager Activation Panel Button **3**.

☐ D. **IF** ERO activation for a Drill affecting bridges (Blue Delta Bridges) is required,
THEN Press ERO Pager Activation Panel Button **4**.

☐ E. **IF** Alternate TSC/OSC will be used,
THEN Press ERO Pager Activation Panel Button **5**.

☐ F. **IF** A Security Event is in progress,
THEN Press ERO Pager Activation Panel Button **6**.

☐ 1.6.2 Wait one minute and repeat step 1.6.1.

☐ 1.6.3 Monitor ERO Pager and verify that message has been provided to the ERO.

☐ 1.6.4 Repeat steps 1.6.1 - 1.6.3 if message is not displayed on ERO Pager.

A. **REFER TO** Enclosure 4.8, (ERO Pager Activation By Security), if the ERO Pager is not activated by the completion of Steps 1.6.1 - 1.6.3.

☐ 1.6.5 **IF** ERO activation is after normal working hours,
THEN Contact Security at extension 3636 or 2309.

Security Officer Name ______________________

A. Request Security Officer to activate the CAN call list.

WARNING: Conducting Site Assembly during a Security Event may not be prudent.

☐ 1.7 Contact the Security Shift Supervisor.

1.7.1 Inform the Security Shift Supervisor that the ERO has been activated.

1.7.2 Discuss the need to conduct Site Assembly.

NOTE: Consider conducting a Site Assembly if a Hazardous Materials spill affecting personnel safety is involved; or, if personnel safety is a concern.

☐ 1.8 **IF** The Emergency Response Organization is needed to assist with the Unusual Event emergency activities,
AND A Security Event does **NOT** exist,
OR A Security Event does exist and the Security Shift Supervisor agrees,
THEN Conduct Site Assembly per RP/0/B/1000/009, (Procedure For Site Assembly), Enclosure 4.1 and 4.3.

☐ 1.8.1 Document the decision to conduct Site Assembly in the Control Room Emergency Coordinator Log.

☐ 1.9 **IF** Area Radiation Monitors are in **ALARM**,
OR Steam Line Break has occurred,
THEN Contact shift RP and dispatch onsite monitoring teams

NOTE:
- Remind the NRC Communicator to complete the NRC Briefing Form prior to contacting the NRC.
- An open line to the NRC may be required.

☐ 1.10 Appoint an SRO to notify the NRC immediately after notification of the Offsite Agencies but not later than **one (1) hour** after declaration of the emergency.

1.10.1 NRC Communicator (SRO) Name ______________________

☐ 1.11 Notify the Unit Operations Coordinator/Duty person of emergency status.

☐ 1.12 Return to Step 3.0, (Subsequent Actions), of this procedure.

Enclosure 4.5

Operations Shift Manager To TSC Emergency Coordinator Turnover Sheet

EMERGENCY CLASSIFICATION ____________________ TIME DECLARED ________
DESCRIPTION OF EVENT ____
____

Unit One Status:

Reactor Power __________ RCS Pressure __________ RCS Temperature ______________
Auxiliaries Being Supplied Power From __________ ES Channels Actuated ______________

MAJOR EQUIPMENT OUT OF SERVICE ___________________________________
____
____
____

JOBS IN PROGRESS ____
____
____

Unit Two Status:

Reactor Power __________ RCS Pressure __________ RCS Temperature ______________
Auxiliaries Being Supplied Power From __________ ES Channels Actuated ______________

MAJOR EQUIPMENT OUT OF SERVICE ___________________________________
____
____
____

JOBS IN PROGRESS ____
____
____

Unit Three Status:

Reactor Power __________ RCS Pressure __________ RCS Temperature ______________
Auxiliaries Being Supplied Power From __________ ES Channels Actuated ______________

MAJOR EQUIPMENT OUT OF SERVICE ___________________________________
____
____
____

JOBS IN PROGRESS ____
____
____

Operations Shift Manager To TSC Emergency Coordinator Turnover Sheet

Classification Procedure in Use:

RP/0/B/1000/002 (Control Room Emergency Coordinator Procedure)

Is RP/0/B/1000/03A, (ERDS Operation) in use? Yes ____ No ____ If Yes, Unit No. ____

Step No. ____

Is RP/0/B/1000/007, (Security) in use? Yes ____ No ____ If Yes, Step No. ____

Is RP/0/B/1000/016, (Medical) in use? Yes ____ No ____ If Yes, Step No. ____

Is RP/0/B/1000/017, (Spill Response) in use? Yes ____ No ____ If Yes, Step No. ____

Is RP/0/B/1000/022, (Fire/Flood) in use? Yes ____ No ____ If Yes, Step No. ____

IF Condition A, Dam Failure, has been declared for Keowee Hydro Project,

THEN Provide the following information to the TSC Emergency Coordinator:

- Status of Offsite Agency Notifications ________________________
- Recommendations made to offsite agencies ________________________
- Status of relocation of site personnel ________________________

What is the status of Site Assembly? (This question is only applicable for those times that the Emergency Response Organization is activated after hours, holidays, or weekends.)

____

____

____

Next message due to Offsite Agencies at Time: __________

Operations Shift Manager/CR ________________________ Time: __________

Emergency Coordinator/TSC ________________________ Time: __________

Enclosure 4.6

Emergency Classification Termination Criteria

IF The following guidelines **applicable to the present emergency condition** have been met or addressed,

THEN An emergency condition may be considered resolved when:

_____ 1. Existing conditions no longer meet the existing emergency classification criteria and it appears unlikely that conditions will deteriorate further.

_____ 2. Radiation levels in affected in-plant areas are stable or decreasing to below acceptable levels.

_____ 3. Releases of radioactive material to the environment greater than Technical Specifications are under control or have ceased.

_____ 4. The potential for an uncontrolled release of radioactive material is at an acceptably low level.

_____ 5. Containment pressure is within Technical Specification requirements.

_____ 6. Long-term core cooling is available.

_____ 7. The shutdown margin for the core has been verified.

_____ 8. A fire, flood, earthquake, or similar emergency condition is controlled or has ceased.

_____ 9. Offsite power is available per Technical Specification requirements.

_____ 10. All emergency action level notifications have been completed.

_____ 11. The Area Hydro Manager has been notified of termination of Condition B for Keowee Hydro Project.

- **REFER TO** Section 6 of the Emergency Telephone Directory, (Keowee Hydro Project Dam/Dike Notification).

_____ 12. The Regulatory Compliance Section has evaluated plant status with respect to Technical Specifications and recommends Emergency classification termination.

_____ _____ 13. Emergency terminated. Request the Control Room Offsite Communicator to complete an Emergency Notification Form for a Termination Message using guidance in RP/0/1000/015A, (Offsite Communications From The Control Room), and provide information to offsite agencies.

Date/Time Initial

- Return to Step 3.13.1.

1. Condition A Response - Immediate Actions

☐ 1.1 **IF** Condition A, Imminent or Actual Dam Failure (Keowee or Jocassee) exists,
THEN Perform the following actions:

☐ 1.1.1 Provide the following **protective action recommendations** to Oconee County and Pickens County for imminent/actual dam failure.

A. Provide the following recommendation for Emergency Notification Form Section 15 (B) Evacuate:

1. Move residents living downstream of the Keowee Hydro Project dams to higher ground.

B. Provide the following recommendation for Emergency Notification Form Section 15 (D) Other:

1. Prohibit traffic flow across bridges identified on your inundation maps until the danger has passed.

☐ 1.2 Return to applicable Enclosure (4.1 or 4.2).

☐ 1.2.1 **IF** A General Emergency has been declared,
THEN **GO TO** Step 1.5 of Enclosure 4.1, (General Emergency).

☐ 1.2.2 **IF** A Site Area Emergency has been declared,
THEN **GO TO** Step 1.5 of Enclosure 4.2, (Site Area Emergency).

2. Condition A Response - Subsequent Actions

☐ 2.1 Notify the Duke Power System Coordinator (Systems Operation Center) on the Control Room Dispatcher phone and provide information related to the event.

☐ 2.2 Relocate Keowee personnel to the Operational Support Center (OSC) if events occur where their safety could be affected.

☐ 2.2.1 **IF** Keowee personnel are relocated to the OSC,
THEN Notify the Duke Power System Coordinator (Systems Operation Center) on the Control Room Dispatcher phone.

NOTE:	A loss of offsite communications capabilities (Selective Signaling and the Wide Area Network - WAN) could occur within 1.5 hours after Keowee Hydro Dam failure. Rerouting of the Fiber Optic Network through Bad Creek should be started **as soon as possible**.

☐ 2.3 Notify Telecommunications Group in Charlotte to begin rerouting the Oconee Fiber Optic Network.

2.3.1 **REFER TO** Selective Signaling Section of the Emergency Telephone Directory (page 9).

☐ 2.4 Request Security to alert personnel at the Security Track/Firing Range and Building 8055 (Warehouse #5) to relocate to work areas inside the plant.

NOTE:	• Plant access road to the Oconee Complex could be impassable within **1.5 hours** if the Keowee Hydro Dam fails. A loss of the Little River Dam (Newry Dam) or Dikes A-D will take longer to affect this road. • PA Announcements can be made by the Control Room using the Office Page Override feature or Security.

☐ 2.5 Make a PA Announcement to relocate personnel at the following locations to the World Of Energy/Operations Training Center.

_________ Oconee Complex

_________ Oconee Garage

_________ Oconee Maintenance Training Facility

☐ 2.6 Dispatch operators to the SSF and establish communications.

☐ 2.7 Return to applicable Enclosure (4.1 or 4.2).

☐ 2.7.1 **IF** A General Emergency has been declared,
THEN **GO TO** Step 1.13 of Enclosure 4.1, (General Emergency).

☐ 2.7.2 **IF** A Site Area Emergency has been declared,
THEN **GO TO** Step 1.12 of Enclosure 4.2, (Site Area Emergency).

3. Condition B Response - Immediate Actions

☐ 3.1 **IF** Condition B at Keowee exists,
THEN Notify the Area Hydro Manager.

3.1.1 **REFER TO** Section 6 of the Emergency Telephone Directory, (Keowee Hydro Project Dam/Dike Notification).

☐ 3.2 Return to applicable Enclosure (4.1, or 4.2, or 4.3, or 4.4).

☐ 3.2.1 **IF** A General Emergency has been declared,
THEN **GO TO** Step 1.13 of Enclosure 4.1, (General Emergency).

☐ 3.2.2 **IF** A Site Area Emergency has been declared,
THEN **GO TO** Step 1.12 of Enclosure 4.2, (Site Area Emergency).

☐ 3.2.3 **IF** An Alert has been declared,
THEN **GO TO** Step 1.11 of Enclosure 4.3, (Alert).

☐ 3.2.4 **IF** An Unusual Event has been declared,
THEN **GO TO** Step 1.6 of Enclosure 4.4, (Unusual Event).

1. Symptoms

1.1 Activation of the ERO Pagers using the ERO Pager Activation Panel in the TSC was unsuccessful.

2. Immediate Actions

2.1 Activate the Emergency Response Organization (Technical Support Center, Operational Support Center, and Emergency Operations Facility) by completing the following actions.:

2.1.1 Contact Security.

A. Dial 3636 (Dial 2309 if no response is received).

Security Officer Name ____________________________

2.1.2 Read the following information to the Security:

A. The Emergency Response Organization (Technical Support Center, Operational Support Center, and Emergency Response Facility) is being activated for an emergency relating to Unit # _____.

NOTE:	Activate the Alternate TSC and OSC in the Oconee Office Building, Rooms 316 and 316A, if a fire in the Turbine Building, flooding conditions, security events, or onsite/ offsite hazardous materials spills have occurred or are occurring.

B. ______ Primary TSC/OSC will be used

<u>OR</u>

______ Alternate TSC/OSC will be used

C. This is a _______ Blue Delta (Drill) activation

<u>OR</u>

This is a _______ Blue Echo (Emergency) activation

NOTE:	Flooding/dam failure/earthquake conditions assume bridges may be impassable to reach emergency facilities. Provide the code below for these conditions.

D. This is a _______ Blue Delta Bridges (Drill) activation

OR

This is a _______ Blue Echo Bridges (Emergency) activation

(R06-97)

INFORMATION ONLY

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. RP/0/B/1000/015A

Revision No 3

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Offsite Communications From The Control Room

(4) Prepared By Donice Kelley Date 12/4/98

(5) Requires 10CFR50.59 evaluation?

- ☒ Yes (New procedure or revision with major changes)
- ☐ No (Revision with minor changes)
- ☐ No (To incorporate previously approved changes)

(6) Reviewed By W.P. Brandt (QR) Date 12/9/98

Cross-Disciplinary Review By ______ (QR)NA WB Date 12/9/98

Reactivity Mgmt. Review By ______ (QR)NA WB Date 12/9/98

(7) Additional Reviews

QA Review By ______ Date ______

Reviewed By ______ Date ______

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By M R Thorne Date 12-10-98

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification

- ☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
- ☐ Yes ☐ NA Listed enclosures attached?
- ☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
- ☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
- ☐ Yes ☐ NA Procedure requirements met?

OTC CONTROL COPY

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

SIMULATOR CONTROL COPY

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station	Procedure No. **RP/0/B/1000/015A**
Offsite Communications From The Control Room	Revision No. 002
Reference Use	Electronic Reference No. OX002WP7

Offsite Communications From The Control Room

NOTE:	This procedure is an implementing procedure to the Oconee Nuclear Site Emergency Plan and must be forwarded to Emergency Planning within three (3) working days of approval.

1. Symptoms

1.1 Events are in progress or have occurred which require activation of the Oconee Nuclear Site Emergency Plan and notification of offsite agencies.

2. Immediate Actions

NOTE:

- Personnel responding as Control Room Offsite Communicator shall have training prior to serving in this role to assure familiarity with location of equipment and procedure requirements.
- Enclosure 4.6, (Acronym Listing), provides a list of acronyms that are used throughout this procedure. Offsite Communicators should refer to this enclosure as needed.
- Lines left of procedure steps are used to indicate place in procedure. Check marks are acceptable in these blanks.
- Initial Notifications **MUST BE** communicated to Offsite Agencies within **15 minutes** of Emergency Classification.

2.1 Report to the affected unit Control Room at the direction of the Operations Shift Manager/Emergency Coordinator.

2.1.1 Obtain the portable phone located on the column in the Control Room. The antenna for this phone is required to be extended for it to operate.

2.2 Obtain a Yellow Folder from the Emergency Procedures Cart (located in Technical Support Center or Unit 3 Control Room Document Library).

_____ 2.2.1 Verify that the following items are included in the Yellow Folder:

_____ A. Emergency Telephone Directory

_____ B. Authentication Code List

_____ C. Emergency Notification Forms

_____ 2.2.2 Obtain the Emergency Action Level Guideline Manual

2.3 Complete the Emergency Notification Form by performing the following steps Enclosure 4.1, (Guidelines for Completion of Emergency Notification Form), is available for reference.

_____ 2.3.1 Review the OSM/Emergency Coordinator Log to determine plant conditions.

_____ 2.3.2 Complete information for Line 1.

A. Mark **A** for a Drill, **B** for an Actual Emergency.

B. Mark Initial.

NOTE:	Chronological numbering of messages is required until the event is terminated.

C. Identify this notification as Message Number 1.

_____ 2.3.3 Complete information for Line 2.

A. Identify Site as Oconee.

B. Identify the Unit affected by the event.

<u>**IF**</u> More than one unit is affected,
<u>**THEN**</u> Write **ALL** on this blank.

C. Write your Name on the Reported By blank.

_____ 2.3.4 Determine information needed to complete Lines 5, 6, 7, 8, 9, 10 from the OSM/Emergency Coordinator.

A. Mark the appropriate Emergency Classification for Line 5.

B. Complete information for Line 6.

1. Mark **A** (Emergency Declaration At:) and include the declaration time provided by the OSM/EC.

NOTE:	Do <u>**NOT**</u> use acronyms when completing Line 7, Emergency Description/Remarks.

C. Use the Emergency Action Level Guideline to write the description on Line 7..

D. <u>**IF**</u> This is a notification for an Unusual Event or Alert classification,
<u>**THEN**</u> Mark **B** (Stable) for Line 8.

E. **IF** This is a notification for a Site Area Emergency or General Emergency classification,
THEN Mark **C** (Degrading) for Line 8.

F. **IF** The reactor(s) is/are shutdown,
THEN Mark **A** (Shutdown) for Line 9 and include the Time/Date of shutdown.

NOTE:	Include the % Power for all units in this section (e.g.; U-1 0%, U-2 45%, U-3 100%) if more than one unit is affected by the event.

G. **IF** The reactor(s) is/are still at power,
THEN Mark **B** and include the % Power for Line 9.

H. **IF** No release is in progress,
THEN Mark **A** (None) for Line 10.

I. **IF** OSM determines that a potential release is in progress,
THEN Mark **B** (Potential) for Line 10.

J. **IF** A release is occurring,
THEN Mark **C** (Is Occurring) for Line 10.

K. **IF** OSM determines that a release has occurred,
THEN Mark **D** (Has Occurred) for Line 10.

_____ 2.3.5 Write Not Applicable across Lines 11-14.

_____ 2.3.6 Complete Line 15, Recommended Protective Actions.

A. **IF** This is a notification for an Unusual Event or Alert classification,
THEN Mark **A** (No Recommended Protective Actions).

B. **IF** This is a notification for a Site Area Emergency and a Condition A (Keowee or Jocassee) does **NOT** exist,
THEN Mark **A** (No Recommended Protective Actions).

C. **IF** This is a notification for a Site Area Emergency and Condition A (Keowee or Jocassee) does exist,
THEN Mark **B** (Evacuate), and include the following information:

- Move residents living downstream of the Keowee Hydro Project dams to higher ground.

AND

THEN Mark **D** (Other), and include the following information:

- Prohibit traffic flow across bridges identified on your inundation maps until the danger has passed.

IF This is a notification for a General Emergency Classification,
THEN Mark **B** (Evacuate) and include the following information:

- Oconee County: A0, D1, E1, F1;
  Pickens County: A0, A1, B1, C1

AND

THEN Mark **C** (Shelter), and include the following information:

- Oconee County: D2, E2, F2;
  Pickens County: A2, B2, C2

_____ 2.3.7 Provide completed Emergency Notification Form to the OSM/Emergency Coordinator for approval and completion of Line 16, Approved By.

NOTE:	Do **NOT** wait on the Fax to finish before proceeding to Step 2.5.

_____ 2.4 Copy the Emergency Notification Form, (Use Fax to copy) and fax the form to offsite agencies using Speed Dial 14 on the fax in the TSC, or Unit 1&2 Control Room, or the OSC, Enclosure 4.3, (Fax Operation) is available for reference.

2.5 Notify SC State/County agencies by using Selective Signaling.

_____ 2.5.1 **IF** Selective Signaling is unavailable,
THEN GOTO Step 2.6.

_____ 2.5.2 **IF** Selective Signaling is available,
THEN GOTO Step 2.5.3.

_____ 2.5.3 Dial ***4.**

_____ 2.5.4 Complete the information in Line 3 of the Emergency Notification Form.

A. Record Transmittal Time/Date whenever Selective Signaling Group Call number has been dialed and phone begins to ring.

> **NOTE:**
> - Oconee County EPD and Pickens County EPD are available during normal day work schedules (Monday-Friday). It is very important to know if all required agencies are on line.
> - Do <u>**NOT**</u> record agency names at this time.

_____ 2.5.5 Turn the Emergency Notification Form over and check off the State and County agencies as they answer. At a minimum, the message must be provided to the following agencies (24 hour warning points):

Oconee County LEC	(416)
Pickens County LEC	(410)
State Warning Point EPD	(518)

_____ 2.5.6 **<u>IF</u>** All required agencies did not respond to group call,
<u>THEN</u> Dial the Selective Signaling number for the applicable agency/agencies no more than two times.

Oconee County LEC	(416)
Pickens County LEC	(410)
State Warning Point EPD	(518)
Oconee County EPD*	(417)
Pickens County EPD*	(419)

* Not staffed after 1700 hours Monday - Friday

A. **<u>IF</u>** An agency does not respond to Selective Signaling call,
<u>THEN</u> GOTO Step 2.5.7 and complete the notification.

_____ 2.5.7 **<u>IF</u>** Offsite agency/agencies request authentication,
<u>THEN</u> Identify an agency to provide authentication code number.

A. Record Authentication number on Line 4.

B. Locate Authentication Code Number and corresponding Authentication Code Word.

C. Record applicable Authentication Code Word on Line 4 and provide to offsite agencies.

_____ 2.5.8 Read information from Emergency Notification Form to offsite agencies.

A. Read each line beginning with the number. For Example - "Line 1.... A.... This is a Drill.... This is an Initial Message.... Message Number 1."

B. Read each line distinctly and slowly.

_____ 2.5.9 Record Time/Name of agencies receiving notification (Lines 1-5 on back of Emergency Notification Form).

NOTE:	The duty officer for DHEC <u>will</u> call back after they are contacted by the State Warning Point.

A. Record Time/Name of DHEC contact on Line 6 on back of Emergency Notification Form.

_____ 2.5.10	**<u>IF</u>**	This is a Condition A or B notification for a Keowee Hydro Project dam/dike,
	<u>THEN</u>	Fax the Emergency Notification Form to GEMA and the NWS by using Speed Dial 27 on the fax in the TSC, or Unit 1&2 Control Room, or the OSC (Enclosure 4.3, Fax Operation, is available for reference).
_____ 2.5.11	**<u>IF</u>**	One or more agencies did not respond to Selective Signaling call,
	<u>THEN</u>	GOTO Step 2.6.
_____ 2.5.12	**<u>IF</u>**	All required State and County agencies have been notified,
	<u>THEN</u>	GOTO Step 2.9.

NOTE:	Only ROLM phones with outside dialing access <u>shall be</u> used to make offsite notifications if Selective Signaling is unavailable.

_____ 2.6	**<u>IF</u>**	Selective Signaling is unavailable,
	<u>THEN</u>	Notify SC State/County agencies using a ROLM phone.
_____ 2.6.1	**<u>IF</u>**	ROLM phone system is unavailable,
	<u>THEN</u>	GOTO Step 2.7.
_____ 2.6.2	**<u>IF</u>**	ROLM phone system is available,
	<u>THEN</u>	GOTO Step 2.6.3.

_____ 2.6.3 Use the Emergency Telephone Directory to determine phone number for offsite agency/agencies.

_____ 2.6.4 **IF** One or more offsite agencies have been notified using Selective Signaling,
THEN GOTO Step 2.6.6.

_____ 2.6.5 **IF** No offsite agencies have not been notified using Selective Signaling,
THEN GOTO Step 2.6.12.

_____ 2.6.6 Dial number as listed in Emergency Telephone Directory.

_____ 2.6.7 **IF** Offsite agency requests authentication,
THEN Perform the following steps:

A. Record Authentication Code Number on Line 4.

B. Locate Authentication Code Number and corresponding Code Word from the Authentication Code List.

C. Record applicable Authentication Code Word on Line 4 and provide to offsite agency.

_____ 2.6.8 Read information from Emergency Notification Form to offsite agency/agencies.

A. Read each line beginning with the number. For Example "Line 1....A....This is a Drill.... This is an Initial Message.... Message Number 1."

B. Read each line distinctly and slowly.

_____ 2.6.9 Record Time/Name of agency/agencies receiving notification (Lines 1-5 on back of the Emergency Notification Form).

NOTE: The duty officer for DHEC will call back after they are contacted by the State Warning Point.

_____ 2.6.10 **IF** The State Warning Point is notified using the ROLM phone system,
THEN Record Time/Name of DHEC contact on Line 6 on back of the Emergency Notification Form

_____ 2.6.11 **<u>IF</u>** All required State and County agencies have been notified,
<u>THEN</u> GOTO Step 2.9.

_____ 2.6.12 **<u>IF</u>** No offsite agencies have been notified using Selective Signaling,
<u>THEN</u> Perform the following steps:

_____ A. Use the Emergency Telephone Directory to determine phone number for offsite agency/agencies.

_____ B. Dial number as listed in Emergency Telephone Directory.

_____ C. Complete the information in Line 3 of the Emergency Notification Form.

1. Record Transmittal Time/Date whenever offsite agency number has been dialed and phone begins to ring.

NOTE:
- Oconee County EPD and Pickens County EPD are available during normal day work schedule (Monday-Friday).
- Do <u>**NOT**</u> record agency names at this time.

_____ D. Turn the Emergency Notification Form over and check off the State and County agencies as they answer. At a minimum, the message must be provided to the following agencies (24 hour warning points):

Oconee County LEC
Pickens County LEC
State Warning Point EPD

_____ E. **<u>IF</u>** Offsite agency requests authentication,
<u>THEN</u> Perform the following steps:

1. Record Authentication Code Number on Line 4.

2. Locate Authentication Code Number and corresponding Code Word from the Authentication Code List.

3. Record applicable Authentication Code Word on Line 4 and provide to offsite agency.

_____ F. Read information from Emergency Notification Form to offsite agency/agencies.

1. Read each line beginning with the number. For Example "Line 1.... A.... This is a Drill.... This is an Initial Message.... Message Number 1."

2. Read each line distinctly and slowly.

_____ G. Record Time/Name of agency/agencies receiving notification (Lines 1-5 on back of the Emergency Notification Form).

NOTE:	The duty officer for DHEC will call back after they are contacted by the State Warning Point.

1. Record Time/Name of DHEC contact on Line 6 on back of the Emergency Notification Form.

_____ H. Repeat Step 2.6.12., Step B and Steps 2.6.12, Step D - 2.6.12, Step G.1 until all required agencies have been notified.

_____ 2.6.13 **IF** This is a Condition A or B notification for a Keowee Hydro Project dam/dike,
THEN Fax the Emergency Notification Form to GEMA and the NWS by using Speed Dial 27 on the fax in the TSC, or Unit 1&2 Control Room, or the OSC (Enclosure 4.3, (Fax Operation), is available for reference).

_____ 2.6.14 **IF** All required State and County agencies have been notified,
THEN GOTO Step 2.9.

NOTE:	The portable phone located on the column in Unit 1/2 and Unit 3 Control Room is a dedicated Bell South line.

_____ 2.7 **IF** ROLM phone system is unavailable,
THEN Notify SC State/County agencies using the portable phone.

_____ 2.7.1 **IF** The portable phone is unavailable,
THEN GOTO Step 2.8.

_____ 2.7.2 **IF** The portable phone is available,
THEN GOTO Step 2.7.3.

_____ 2.7.3 Use the Emergency Telephone Directory to determine phone number for offsite agency/agencies.

_____ 2.7.4 **IF** One or more offsite agencies have been notified using the ROLM phone system or Selective Signaling,
THEN GOTO Step 2.7.6.

_____ 2.7.5 **IF** No offsite agencies have been notified using the ROLM phone system or Selective Signaling,
THEN GOTO Step 2.7.12.

_____ 2.7.6 Dial number as listed in Emergency Telephone Directory.

_____ 2.7.7 **IF** Offsite agency requests authentication,
THEN Perform the following steps:

A. Record Authentication Code Number on Line 4.

B. Locate Authentication Code Number and corresponding Code Word from the Authentication Code List.

C. Record applicable Authentication Code Word on Line 4 and provide to Offsite Agency.

_____ 2.7.8 Read information from Emergency Notification Form to offsite agency/agencies.

A. Read each line beginning with the number. For Example "Line 1....A....This is a Drill.... This is an Initial Message.... Message Number 1."

B. Read each line distinctly and slowly.

_____ 2.7.9 Record Time/Name of agency/agencies receiving notification (Lines 1-5 on back of the Emergency Notification Form).

NOTE:	The duty officer for DHEC will call back after they are contacted by the State Warning Point.

_____ 2.7.10 **IF** The State Warning Point is notified using the portable phone,
THEN Record Time/Name of DHEC contact on Line 6 on back of the Emergency Notification Form

2.7.11 **IF** All required State and County agencies have been notified,
THEN GOTO Step 2.9.

_____ 2.7.12 **IF** No offsite agencies have been notified using the ROLM phone system or Selective Signaling,

THEN Perform the following steps:

_____ A. Use the Emergency Telephone Directory to determine phone number for offsite agency/agencies.

_____ B. Dial number as listed in Emergency Telephone Directory.

_____ C. Complete the information in Line 3 of the Emergency Notification Form.

1. Record Transmittal Time/Date whenever offsite agency number has been dialed and phone begins to ring.

NOTE:
- Oconee County and Pickens County EPD are available during normal day work schedules (Monday-Friday).
- Do **NOT** record agency names at this time.

_____ D. Turn the Emergency Notification Form over and check off the State and County agencies as they answer. At a minimum, the message must be provided to the following agencies (24 hour warning points):

Oconee County LEC
Pickens County LEC
State Warning Point EPD

_____ E. **IF** Offsite agency requests authentication,
THEN Perform the following steps:

1. Record Authentication Code Number on Line 4.
2. Locate Authentication Code Number and corresponding Code Word from the Authentication Code List.
3. Record applicable Authentication Code Word on Line 4 and provide to offsite agency.

_____ F. Read information from Emergency Notification Form to offsite agency/agencies.

1. Read each line beginning with the number. For Example "Line 1.... A.... This is a Drill.... This is an Initial Message.... Message Number 1."

2. Read each line distinctly and slowly.

_____ G. Record Time/Name of agency/agencies receiving notification (Lines 1-5 on back of the Emergency Notification Form).

NOTE:	The duty officer for DHEC will call back after they are contacted by the State Warning Point.

1. Record Time/Name of DHEC contact on Line 6 on back of the Emergency Notification Form.

_____ H. Repeat Step 2.7.12.2 and Steps 2.7.12.4 - 2.7.12.7 until all required agencies have been notified.

_____ 2.7.13 **IF** This is a Condition A or B notification for a Keowee Hydro Project dam/dike,
THEN Fax the Emergency Notification Form to GEMA and the NWS by using Speed Dial 27 on the fax in the TSC, or Unit 1&2 Control Room, or the OSC (Enclosure 4.3, Fax Operation, is available for reference).

_____ 2.7.14 **IF** All required State and County agencies have been notified,
THEN GOTO Step 2.9.

NOTE:	The Offsite Radio is located in the radio console near the Fax in Unit 1&2 Control Room. The radio is designated as WQC699.

_____ 2.8 **IF** The portable phone is unavailable,
THEN Notify SC State/County agencies using the Offsite Radio.

_____ 2.8.1 **IF** One or more offsite agencies have been notified using the portable phone, ROLM phone system, or Selective Signaling,
THEN GOTO Step 2.8.3.

_____ 2.8.2 **IF** No offsite agencies have been notified using the portable phone, ROLM phone system, or Selective Signaling,
THEN GOTO Step 2.8.15.

_____ 2.8.3 Push SEL on WQC699 frequency panel.

_____ 2.8.4 Adjust volume control knob to a high setting.

_____ 2.8.5 Enter the applicable radio code for the offsite agency using the numeric key pad.

Oconee County LEC	32*
Pickens County LEC	35*
Pickens County EPD[1]	31*

* Not staffed after 1700 hours Monday - Friday or on weekends and holidays

A. Press MONITOR button to determine if the selected frequency is in use.

_____ 2.8.6 Depress FOOT PEDAL or XMIT button **AND** keep engaged while talking.

_____ 2.8.7 Call the offsite agency being contacted by using applicable Identifier. For Example - "Oconee Control Room to Oconee LEC".

Oconee County LEC	Oconee LEC
Pickens County LEC	Pickens LEC
Pickens County EPD	Pickens EOC
U 1&2 Control Room	Oconee Control Room

A. Release FOOT PEDAL or XMIT button to receive incoming response from offsite agency.

_____ 2.8.8 **IF** Offsite agency requests authentication,
THEN Perform the following steps:

A. Record Authentication Code Number on Line 4 of the Emergency Notification Form.

B. Locate Authentication Code Number and corresponding Code Word from the Authentication Code List.

C. Record applicable Authentication Code Word on Line 4 of the Emergency Notification Form and provide to offsite agency.

_____ 2.8.9 Read information from Emergency Notification Form to offsite agency/agencies.

A. Read each line beginning with the number. For Example "Line 1....A....This is a Drill.... This is an Initial Message.... Message Number 1."

B. Read each line distinctly and slowly.

_____ 2.8.10 Request Oconee County or Pickens County to Notify the SC State Warning Point.

A. Inform county that phone communications with SC State is not available.

_____ 2.8.11 Record Time/Call Letters of agency/agencies receiving notification (Lines 1,2, and 4 on back of the Emergency Notification Form).

Oconee County LEC	KNBE-488
Pickens County LEC	KNBZ-965
Pickens County EPD	KNBE-480

_____ 2.8.12 End radio transmission using Call Letters WQC699.

NOTE: The duty officer for DHEC will call back after they are contacted by the State WARNING Point.

2.8.13 **IF** The State Warning Point is notified by County agency/agencies,
THEN Record Time/Name of DHEC contact on Line 6 on back of the Emergency Notification Form

_____ 2.8.14 **IF** Oconee and Pickens County agencies have been notified,
THEN GOTO Step 2.9.

_____ 2.8.15 **IF** No offsite agencies have been notified using the portable phone, ROLM phone system, or Selective Signaling,
THEN Perform the following steps:

_____ A. Push SEL on WQC699 frequency panel.

_____ B. Adjust volume control knob to a high setting.

_____ C. Enter the group call radio code 30* using the numeric key pad.

1. Press MONITOR button to determine if the selected frequencies are in use.

_____ D. Depress FOOT PEDAL or XMIT button **AND** keep engaged while talking.

_____ E. Call the offsite agencies being contacted by using applicable Identifier. For Example "Oconee Control Room to Oconee LEC, Pickens LEC, and Pickens EOC".

Oconee County LEC	Oconee LEC
Pickens County LEC	Pickens LEC
Pickens County EPD	Pickens EOC
U 1&2 Control Room	Oconee Control Room

1. Release FOOT PEDAL or XMIT button to receive incoming response from offsite agency.

_____ F. Complete the information in Line 3 of the Emergency Notification Form.

1. Record Transmittal Time/Date whenever the group call number has been encoded and voice transmission begins.

NOTE:
- Pickens County EPD is available during normal day work schedules (Monday-Friday).
- Do **NOT** record Call Letters at this time.

_____ G. Turn the Emergency Notification Form over and check off the County agencies as they answer.

_____ H. **IF** Offsite agency requests authentication,
THEN Perform the following steps:

1. Record Authentication Code Number on Line 4 of the Emergency Notification Form.
2. Locate Authentication Code Number and corresponding Code Word from the Authentication Code List.
3. Record applicable Authentication Code Word on Line 4 of the Emergency Notification Form and provide to offsite agency.

_____ I. Read information from Emergency Notification Form to offsite agency/agencies.

1. Read each line beginning with the number. For Example "Line 1....A....This is a Drill.... This is an Initial Message.... Message Number 1."
2. Read each line distinctly and slowly.

_____ J. Request Oconee County or Pickens County to notify the SC State Warning Point.

1. Inform county that phone communications with SC State is not available.

_____ K. Record Time/Call Letters of agency/agencies receiving notification (Lines 1,2, and 4 on back of the Emergency Notification Form).

Oconee County LEC	KNBE-488
Pickens County LEC	KNBZ-965
Pickens County EPD	KNBE-480

_____ L. End radio transmission using Call Letters WQC699.

NOTE:	The duty officer for DHEC will call back after they are contacted by the State Warning Point.

M. <u>**IF**</u> The State Warning Point is notified by County agency/agencies.
<u>**THEN**</u> Record Time/Name of DHEC contact on Line 6 on the back of the Emergency Notification Form.

_____ N. <u>**IF**</u> Oconee and Pickens County agencies have been notified,
<u>**THEN**</u> GOTO Step 2.9.

_____ 2.9 Provide the OSM with a status of offsite notifications:

2.9.1 Provide a copy of the completed Emergency Notification Form to the OSM.

2.9.2 Identify the offsite agencies notified/not notified.

2.9.3 Identify any communications equipment problems.

2.9.4 Identify any offsite agency questions requiring information that was not included on the Emergency Notification Form.

A. Record OSM's response to question on a separate sheet of paper.

B. Have OSM approve response by signing and dating it.

C. Attach the question and answer sheet to the Emergency Notification Form used when the question was asked and provide to applicable agency/agencies.

D. Document the date and time answers were called back and the name of the agency contact receiving the information.

_____ 2.10 **IF** Communication equipment problems occur,
THEN Notify the Telecommunications Group and request immediate support.

_____ 2.11 **IF** Directed by the OSM,
THEN Assist the NRC ENS Communicator.

3. Subsequent Actions

_____ 3.1 **IF** A Termination notification is required for an emergency event,
THEN GOTO Step 3.5.

_____ 3.2 **IF** A Follow Up notification is required for an emergency event,
THEN GOTO Step 3.6.

_____ 3.3 **IF** An Emergency Event Classification is being up graded,
THEN GOTO Step 3.7.

_____ 3.4 **IF** The TSC Offsite Communicator is available, and additional notifications are not immediately required,
THEN GOTO Step 3.8.

_____ 3.5 Perform the following steps for a Termination notification.

3.5.1 Obtain a blank Emergency Notification Form.

NOTE: Do **NOT** mark Initial or Follow Up for a Termination notification.

_____ 3.5.2 Complete information for Line 1.

A. Mark **A** for a Drill, **B** for an Actual Emergency.

NOTE: Chronological numbering of messages is required until the event is terminated.

B. **IF** No follow up messages have been provided,
THEN Identify this notification as Message Number **2**.

C. **IF** Follow up messages have been provided,
THEN Identify this notification with the next number in sequence.

_____ 3.5.3 Complete information for Line 2.

A. Identify Site as Oconee.

B. Identify the Unit affected by the event.

IF	More than one unit is affected,
THEN	Write **ALL** on this blank.

C. Write your Name on the Reported By blank.

_____ 3.5.4 Complete information for Line 6.

A. Mark **B** (Termination At:) and include the Termination time provided by the OSM/EC.

_____ 3.5.5 Provide the completed Emergency Notification Form for the Termination notification message to the OSM/Emergency Coordinator for approval and completion of Line 16, Approved By.

_____ 3.5.6 Copy the Emergency Notification Form, (use Fax machine) and fax the form to offsite agencies using Speed Dial 14 on the fax in the TSC, or Unit 1&2 Control Room, or the OSC Enclosure 4.3, (Fax Operation), is available for reference.

_____ 3.5.7 Notify SC State/County agencies by repeating the applicable portions of Steps 2.5 - 2.9.

_____ 3.5.8 GOTO Step 3.8.

NOTE:

- Follow-Up notifications are required at least every **sixty (60) minutes** for an **Alert**, **Site Area Emergency**, or **General Emergency** Classification. Significant changes in plant conditions should be communicated as they occur. This frequency **may be** changed at the request of offsite agencies. A Follow-Up notification is not required for an Unusual Event unless requested.

- Do not delay sending a Follow-Up notification if all information is not available. Use the same information from the previous message sheet.

_____ 3.6 Perform the following steps for a Follow-Up notification.

3.6.1 Obtain a blank Emergency Notification Form.

NOTE:	**IF**	Information in Lines 8-14 has NOT changed,
	THEN	Complete information for Line 1-7 (Steps 3.6.2 - 3.6.6) and Lines 15-16 (Steps 3.6.14 - 3.6.15).

_____ 3.6.2 Complete information for Line 1.

A. Mark **A** for a Drill, **B** for an Actual Emergency.

B. Mark Follow-Up.

NOTE:	Chronological numbering of messages is required until the event is terminated.

C. Identify this notification's Message Number as the next number in sequence.

_____ 3.6.3 Complete information for Line 2.

A. Identify Site as Oconee.

B. Identify the Unit affected by the event.

IF More than one unit is affected,
THEN Write **ALL** on this blank.

C. Write your Name on the Reported By blank.

_____ 3.6.4 Complete information for Line 5.

A. Mark the same Emergency Classification that was included on the previous message sheet.

_____ 3.6.5 Complete information for Line 6.

A. Mark **A** (Emergency Declaration At:) and include the Time/Date from the previous message sheet.

NOTE:	**IF**	A **Condition B** exists for a Keowee Hydro Project Dam/Dike,
	THEN	Include this as additional information for Line 7.

_____ 3.6.6 Complete information for Line 7.

A. Repeat the information from the previous message sheet.

B. Include any additional information on events that may be of interest to offsite agencies that have occurred since the previous message sheet (For example: transportation of injured personnel offsite; non-essential personnel being evacuated; plant events that would be a lower emergency classification).

_____ 3.6.7 Complete information for Line 8.

A. Verify Plant Conditions with the OSM/Emergency Coordinator.

_____ B. **IF** Plant conditions have not changed since the previous message sheet,
THEN Repeat the information from the previous message sheet.

_____ C. **IF** Plant conditions have changed since the previous message sheet,
THEN Determine the plant conditions and Mark **A**, **B**, or **C** as appropriate.

_____ 3.6.8 Complete information for Line 9.

A. Verify Reactor Status with the OSM/Emergency Coordinator.

_____ B. **IF** Reactor status has not changed since the previous message sheet,
THEN Repeat the information from the previous message sheet.

_____ C. **IF** Reactor status has changed since the previous message sheet,
THEN Perform the following steps:

_____ 1. **IF** The reactor(s) is/are shutdown,
THEN Mark **A** (Shutdown) and include the Time/Date of shutdown.

_____ 2. **IF** The reactor(s) is/are still at power,
THEN Mark **B** and include the % Power for the unit(s).

_____ 3.6.9 Complete information for Line 10.

A. Verify the status of Emergency Releases with the OSM/Emergency Coordinator.

_____ B. **IF** No release is in progress,

THEN Mark **A** (None)

AND

THEN GOTO Step 3.6.10.

_____ C. **IF** A potential release is in progress,
THEN Mark **B** (Potential)

AND

THEN GOTO Step 3.6.10.

_____ D. **IF** A release is occurring,
THEN Mark **C** (Is Occurring)

AND

THEN GOTO Step 3.6.10.

_____ E. **IF** A release has occurred,
THEN Mark **D** (Has Occurred)

AND

THEN GOTO Step 3.6.10.

NOTE:	• RP will be requested to perform an Offsite Dose Calculation if an airborne release is in progress or has occurred. • The Offsite Dose Calculation print out resembles the Emergency Notification Form.

_____ 3.6.10 Complete information for Lines 11-14 as follows:

_____ A. **IF** An airborne release is not in progress,
THEN Write Not Applicable across Lines 11-14.

_____ B. **IF** An airborne release is in progress and RP has completed an Offsite Dose Calculation,
THEN Use the information from the Offsite Dose calculation print out to complete Lines 11-14.

_____ C. **IF** An airborne release is in progress and RP has not completed an Offsite Dose Calculation,
THEN Write Not Available across Lines 11-14.

_____ 3.6.11 Complete information for Line 15.

A. Repeat the Recommended Protective Actions from the previous message sheet.

_____ 3.6.12 Provide the completed Emergency Notification Form for the Follow-Up notification message to the OSM/Emergency Coordinator for approval and completion of Line 16, Approved By.

_____ 3.6.13 Complete information for Line 3.

A. Record the Transmittal Time/Date prior to placing the Emergency Notification Form in the fax.

_____ 3.6.14 Fax the form to offsite agencies using Speed Dial 14 on the fax in the TSC, or Unit 1 & 2 Control Room, or the OSC (Enclosure 4.3, (Fax Operation), is available for reference).

_____ 3.6.15 GOTO Step 3.8.

_____ 3.7 Perform the following steps for an emergency classification upgrade notification.

3.7.1 Obtain a blank Emergency Notification Form.

_____ 3.7.2 Complete information for Line 1.

A. Mark **A** for a Drill, **B** for an Actual Emergency.

B. Mark Initial.

NOTE:	Chronological numbering of messages is required until the event is terminated.

C. Identify this notification's Message Number as the next number in sequence.

_____ 3.7.3 Complete information for Line 2.

A. Identify Site as Oconee.

B. Identify the Unit affected by the event.

IF More than one unit is affected,
THEN Write **ALL** on this blank.

C. Write your Name on the Reported By blank.

_____ 3.7.4 Determine information needed to complete Lines 5, 6, 7, 8, 9, 10 from the OSM/Emergency Coordinator.

A. Mark the appropriate Emergency Classification for Line 5.

B. Complete information for Line 6.

1. Mark **A** (Emergency Declaration At:) and include the Declaration Time provided by the OSM/EC.

NOTE:
- Do **NOT** use acronyms when completing Line 7, Emergency Description/remarks.
- If a **Condition B** for a Hydro Project Dam/Dike exists, include this as additional information for Line 7.

C. Use the Emergency Action Level Guideline to write a description for Line 7.

D. **IF** This is a notification for an Alert classification,
THEN Mark **B** (Stable) for Line 8.

E. **IF** This is a notification for a Site Area Emergency or General Emergency classification,
THEN Mark **C** (Degrading) for Line 8.

F. **IF** The reactor(s) is/are shutdown,
THEN Mark **A** (Shutdown) for Line 9 and include the Time/Date of shutdown.

NOTE: Include the % Power for all units in this section (e.g.; U-1 0%, U-2 45%, U-3 100%) if more than one unit is affected by the event.

G. **IF** The reactor(s) is/are still at power,
THEN Mark **B** and include the % Power for Line 9.

H. **IF** No release is in progress,
THEN Mark **A** (None) for Line 10.

I. **IF** OSM determines that a potential release is in progress,
THEN Mark **B** (Potential) for Line 10.

J. **IF** A release is occurring,
THEN Mark **C** (Is Occurring) for Line 10.

K. **IF** OSM determines that a release has occurred,
THEN Mark **D** (Has Occurred) for Line 10.

_____ 3.7.5 Write Not Applicable across Lines 11-14.

_____ 3.7.6 Complete Line 15, Recommended Protective Actions.

A. **IF** This is a notification for an Alert classification,
THEN Mark **A** (No Recommended Protective Actions).

B. **IF** This is a notification for a Site Area Emergency and a Condition A (Keowee or Jocassee) does **NOT** exist,
THEN Mark **A** (No Recommended Protective Actions).

C. **IF** This is a notification for a Site Area Emergency and a Condition A (Keowee or Jocassee) **does** exist,
THEN Mark **B** (Evacuate), and include the following information:

- Move residents living downstream of the Keowee Hydro Project dams to higher ground.

AND

THEN Mark **D** (Other), and include the following information:

- Prohibit traffic flow across bridges identified on your inundation maps until the danger has passed.

D. **IF** This is a notification for a General Emergency Classification,
THEN Mark **B** (Evacuate) and include the following information:

- Oconee County: A0, D1, E1, F1;
  Pickens County: A0, A1, B1, C1

AND

THEN Mark **C** (Shelter), and include the following information:

- Oconee County: D2, E2, F2;
  Pickens County: A2, B2, C2

_____ 3.7.7 Provide completed Emergency Notification Form to the OSM/Emergency Coordinator for approval and completion of Line 16, Approved By.

_____ 3.7.8 Copy the Emergency Notification Form (use Fax to copy) and fax the form to offsite agencies using Speed Dial 14 on the fax in the TSC, or Unit 1&2 Control Room, or the OSC Enclosure 4.3, (Fax Operation), is available for reference.

_____ 3.7.9 Notify SC State/County agencies by repeating the applicable portions of Steps 2.5 - 2.9.

_____ 3.7.10 GOTO Step 3.8.

NOTE:	The TSC Offsite Communicator will be utilizing RP/0/B/1000/015B, (Offsite Communications from the Technical Support Center).

_____ 3.8 Conduct turnover with the TSC Offsite Communicator.

3.8.1 Prepare for turnover with TSC Offsite Communicator by completing Enclosure 4.2, (Control Room Offsite Communicator Turnover Sheet).

3.8.2 Provide completed Emergency Notification Forms to the TSC Offsite Communicator.

3.8.3 Review Enclosure 4.2, (Control Room Offsite Communicator Turnover Sheet), with the TSC Offsite Communicator.

A. Provide completed turnover sheet to TSC Offsite Communicator.

3.8.4 Provide the portable phone to the TSC Offsite Communicator.

3.8.5 Report to the OSM once turnover is completed.

A. Provide this completed procedure to the OSM.

4. Enclosures

4.1 Guidelines For Completion Of Emergency Notification Form

4.2 Control Room Offsite Communicator Turnover Sheet

4.3 Fax Operation

4.4 Condition A Notification Chart (Site Area Emergency)

4.5 Condition B Notification Chart (Unusual Event)

4.6 Acronym Listing

A. COMPLETING EMERGENCY NOTIFICATION FORMS:

Line 1 Mark **A** or **B.** Indicate whether the notification is the INITIAL (first notification of a designated emergency classification) or a FOLLOW-UP message. For notification of TERMINATION, leave INITIAL and FOLLOW-UP blank.

For MESSAGE NUMBER, chronological numbering is required.

Line 2 SITE:Oconee Nuclear Site.

UNIT:

IF	The event is applicable to one unit only,
THEN	Designate 1, 2, or 3 for the appropriate unit.
IF	More than one unit is involved in the event,
THEN	Enter **ALL.**

REPORTED BY: Write your name on this blank.

Line 3 TRANSMITTAL TIME/DATE: The time to be placed here is the time that the communicator begins to make the offsite call or fax (FAX) a Follow-Up notification. The time is given in military time (24 hour clock) using Eastern Time. Follow-Up notifications are based on the time entered here. Use the wall clock in the appropriate emergency response facility to determine the time to be used.

CONFIRMATION PHONE NUMBER: Circle the number preprinted on the form.

Line 4 AUTHENTICATION: Authentication is determined by providing a code word for a given number listed on the Authentication Code List. The Authentication Code List is located in a yellow folder along with the Emergency Telephone Directory. The yellow folder is located in the front of the emergency procedures cart. Approximately 100 code numbers and words are listed on the Authentication Code List. The offsite agencies may give a number and the communicator must provide the corresponding word that matches the number. Accept only one number to authenticate. The number and word are both written on the form. Counties and state may or may not require authentication on phone calls following the initial contact/notification.

Line 5 EMERGENCY CLASSIFICATION: Mark the correct classification. Do **NOT** mark any classification for a Termination notification.

Line 6A — Emergency Declaration At: The time of emergency declaration is the time that the OSM/Emergency Coordinator determines an Emergency Classification exists. This information is written on all Follow-Up notifications for the same emergency classification.

Line 6B — Termination At: The time the emergency was terminated is the time the OSM/Emergency Coordinator determines that the criteria for terminating an event has been met.

Line 7 — EMERGENCY DESCRIPTION/REMARKS: The wording written here describes the initiating event. Language should be easy to understand. Do Not use acronyms. Provide any additional information here that would be of interest to offsite agencies. For Example: 1) Transported injured person offsite; 2) Fire Brigade responded to a fire at the Unit 1 Transformer; 3) Non-essential personnel are being evacuated.

Line 8 — PLANT CONDITION: Obtain this information from the Operations Shift Manager. As a general rule the following guidelines apply:

IF An Unusual Event or Alert classification exists,
THEN Mark **B** (Stable).

IF A Site Area Emergency or General Emergency classification exists,
THEN Mark **C** (Degrading).

IF The OSM/Emergency Coordinator indicates that conditions are improving,
THEN Mark **A** (Improving).

Line 9 — REACTOR STATUS:

IF The reactor(s) is/are shutdown,
THEN Mark **A**, (Shutdown), and include the Time/Date of shutdown.

NOTE: Include the % Power for all units in this section (e.g.; U-1 0%, U-2 45%, U-3 100%) if more than one unit is affected by the event.

IF The reactor(s) is/are still at power,
THEN Mark **B**, and include the power level.

Line 10 EMERGENCY RELEASES: **IF** No releases are occurring, **THEN** Mark **A**, (None).

The OSM, with RP support, will provide information to complete B, C, and D.

IF The potential for a release of radioactive materials exists,
THEN Mark **B**, (Potential).

A release is any unplanned, quantifiable activity being released as a result of the event. An airborne release is considered to be in progress if the following occurs:

- Reactor building RIA monitors (47 or 48 or 49 reading greater than 1 cpm or1,3RIA57 or 1,2,3RIA58 are reading greater than 1 Rad/hr) **or** 2RIA57 reading greater than 1.6 Rad/hr) and greater than 1 pound pressure is in the containment building or actual containment breach is determined.
- Increase in activity monitored by vent RIA monitors 45 or 46.
- Steam generator tube leak monitored by RIA 40.

Review the SORENTO RIA MONITOR screen to display this information.

IF A release of radioactive materials is in progress,
THEN Mark **C**, (Is Occurring).

IF A release of radioactive materials has occurred
THEN Mark **D**, (Has Occurred).

NOTE: Information for Lines 11, 12, 13, and 14 **may be** left blank for the Initial notification messages (This includes change of classification and change in Protective Action Recommendations).

Line 11 TYPE OF RELEASE: All releases should be marked GROUND LEVEL since the vent stacks at ONS do not meet the criteria for elevated releases. Also, any unmonitored releases from the various structures at ONS (Reactor Building, Auxiliary Building, Hot Machine Shop, Interim Radwaste Facility, and Radwaste Facility) are considered to occur at ground level. Determine from OSM/RP/Radwaste Chemistry if the release is **A**, (Airborne) or **B**, (Liquid), and provide the time the release started and/or stopped.

Line 12 RELEASE MAGNITUDE: **IF** The release is airborne and radioactive,
THEN Mark **Curies per Sec.**

IF The release is liquid and radioactive,
THEN Mark **Curies.**

NORMAL OPERATING LIMITS: Mark BELOW or ABOVE based on the determination of RP shift personnel (airborne releases) and Radwaste Chemistry (liquid releases).

IF Offsite Dose Calculation has been performed by RP shift,,
THEN They will provide airborne release information for:
A, (Noble Gases); **B**, (Iodines); and, **C**, (Particulates).

IF The Offsite Dose Calculation is unavailable,
THEN Write NOT AVAILABLE across this portion of Line 12.

IF Radwaste Chemistry is available,
THEN They will provide liquid release information for:
A, (Noble Gases); **B**, (Iodines); and, **C**, (Particulates), and,
D, (Other).

NOTE: **IF** A hazardous spill is in progress,

THEN Information from an additional form should be communicated to identified offsite agencies as required by RP/0/B/1000/017 (Spill Response).

IF A hazardous material spill is in progress,
THEN Mark **D**, (Other), and write the name of the material being released.

Line 13 ESTIMATE OF PROJECTED OFFSITE DOSE:

IF A release that is being made is BELOW NORMAL OPERATING LIMITS,
THEN Write Not Required, Release Less Than Normal Operating Limits in large letters across this portion of Line 13.

IF Offsite Dose calculation has been performed,
THEN Provide information from the Offsite Dose Calculation printout in this portion of Line 13.

IF Offsite Dose Calculation has **NOT** been performed,
THEN Write Not Available in large letters across this portion of Line 13.

Line 14 METEOROLOGICAL DATA:

IF An Initial notification message is being prepared,
THEN Write Not Available in large letters across Line 14 items.

IF A Follow-Up notification message is being prepared,
THEN Determine the meteorological information from the control room instrumentation and include on the applicable blanks of this line.

Line 15 RECOMMENDED PROTECTIVE ACTIONS:

NOTE:	Once any protective action has been recommended, **all** subsequent notification forms will carry the same recommendation until a new recommendation is determined/required.

IF An Unusual Event or Alert classification exists,
THEN Mark **A**, (No Recommended Protective Actions).

IF A Site Area Emergency classification exists **AND** a Condition A for the Keowee Hydro Project dams/dikes does **NOT** exist,
THEN Mark **A**, (No Recommended Protective Actions).

IF A Site Area Emergency classification exists **AND** a Condition A for the Keowee Hydro Project dams/dikes **does** exist,
THEN Mark **B**, (Evacuate), and include the following information:.

- Move residents living downstream of the Keowee Hydro Project dams to higher ground.

AND

THEN Mark **D**, (Other), and include the following information:

- Prohibit traffic flow across bridges identified on your inundation maps until the danger has passed.

Line 15 RECOMMENDED PROTECTIVE ACTIONS (continued):

IF An Initial notification for a General Emergency classification is being prepared,
THEN Mark **B**, (Evacuate), and include the following information:

- Oconee County: A0, D1, E1, F1;
  Pickens County: A0, A1, B1, C1

AND

THEN Mark **C**, (Shelter), and include the following information:

- Oconee County: D2, E2, F2
  Pickens County: A2, B2, C2

IF The OSM determines that additional action is required,
THEN Mark **D**, (Other), and include any additional information provided by the OSM

Initial information being provided to offsite agencies for **B**, (Evacuate); **C**, (Shelter); and/or **D**, (Other), will be provided verbally over the phone (Selective Signaling, ROLM phone system, etc.). The OSM/Emergency Coordinator should verify that the information received by the State/County Emergency Preparedness Directors is consistent with the information on the Emergency Notification Form.

Line 16 APPROVED BY: OSM/Emergency Coordinator approval signature along with time and date of approval.

B. TERMINATION INSTRUCTIONS:

IF A Termination notification message is being prepared,
THEN Complete Emergency Notification Form Lines 1 - 6B and Line 16.

Enclosure 4.2

Control Room Offsite Communicator Turnover Sheet

Date ________________

Last Message Number ______

Next Message Due at ______________
Time

OFFSITE AGENCY NOTIFICATION STATUS

Agencies who have been notified:

System Used:

		Selective Signaling	ROLM Phone	Portable Phone	Offsite Radio
Pickens EPD	______	______	______	______	______
Pickens LEC	______	______	______	______	______
Oconee EPD	______	______	______	______	______
Oconee LEC	______	______	______	______	______
State Warning Point	______	______	______	______	______
State EOC	______	______	______	______	______
DHEC-NEP	______	______	______	______	______

Communication equipment problems experienced: ________________________________

____

____

____

____

Equipment Problems reported to Telecommunications: ________ Yes ________ No

NOTE:	This enclosure provides basic operating instructions for the primary faxs in the TSC, U-1/2 Control Room, and OSC. Refer to the Operator Manuals for detailed information.

1. TSC/Control Room/OSC

1.1 Make a copy of the approved Emergency Notification Form to be transmitted to offsite agencies/EOF. (Use Fax machine to copy)

1.2 Place the copy of the notification form face down in the Automatic Document Feeder.

1.3 Adjust the Document Guides to the width of the notification form.

1.4 Determine which Speed Dial Code will be used to transmit the notification form to applicable locations.

1.4.1 **IF** A notification for Condition A or B involving the Keowee Hydro Project is **NOT** required,
THEN Refer to Step 2.4, 3.5.6, 3.6.14, or 3.7.8 of this procedure.

1.4.2 **IF** A notification for Condition A or B involving the Keowee Hydro Project is required,
THEN Refer to Step2.5.10, 2.6.10, 2.6.13, 2.7.10, or 2.7.13 of this procedure.

1.5 Fax the notification form using the following method:

1.5.1 **Speed Dial Code**

A. Press the Speed Dial button located in the Telephone Keypad area of the Control Panel.

1.6 Transmission of the notification form will start automatically after the dialing operation is completed. Since this is a send operation to multiple faxs, the Fax scans the document(s) prior to automatic dialing..

1.7 At the completion of a send operation, the fax will print a Broadcast Report. This report will indicate the agencies that the notification form was telecopied to and whether the send operation was successful or unsuccessful.

1.8 The following Speed Dial Codes have been programmed into the fax in the TSC/Unit 1&2 Control Room/OSC:

Speed Dial Code	Agency/Location Sent To
01	NRC
02	Pickens County EPD
03	Oconee County EPD
04	SC State Warning Point
05	SEOC
06	DHEC-BSHWM
07	EOF
08	OSC
09	World Of Energy
10	Alternate TSC
11	Oconee Complex
12	C&F Materials
13	JIC
14	Dial Group
15	Dial Group
16	FEOC
17	Dial Group
18	Oconee County LEC
19	Safety Assurance
20	GO JIC
21	Security
25	National Weather Service
26	GEMA
27	Dial Group
29	Deal Group
30	ONS SRG/RC/EC
31	Dial Group

1.9 The following Dial Group Codes have been programmed into the fax in the TSC/Unit 1&2 Control Room/OSC:

Speed Dial Code	Agency/Location Sent To
14	Pickens County EPD Oconee County EPD SC State Warning Point Oconee County LEC EOF World Of Energy GO JIC
15	Pickens County EPD Oconee County EPD
17	Pickens County EPD Oconee County EPD SEOC EOF World Of Energy GO JIC
27	National Weather Service GEMA
29	EOF OSC
31	OSC Security

Condition a Notification Chart
(Site Area Emergency)

STATION PERSONNEL (1)
(864) 885-3327
3G (Radio)

OPS SHIFT MANAGER
885-3271

CIVIL AUTHORITIES (1)
*SC EMER PREP DIV
*OCONEE EPD
*PICKENS EPD
NAT WEATHER SERVICE
GEMA

(2)
SYSTEM COORDINATOR
(704) 382-4413
(704) 375-2308
(704) 375-2728

PUMP STORAGE HYDRO MGR (1)
D A JOHNSON
(864) 944-4001 (W)
(704) 884-4329 (H)
777-3215 (Pager)

(2)
MGR HYDRO GEN
B E TAYLOR
(704) 382-4466 (W)
(864) 654-3864 (H)
777-3243 (Pager)
(864) 420-5485 (Car)

E D BROCKMAN (Alt)
(704) 382-8588 (W)
(704) 739-2162 (H)
777-8772 (Pager)

(1)
HYDRO ELEC & WATER RES ENGNRNG
R R MILLER
(704) 382-7019 (W)
(704) 643-3995 (H)

J C EY (Alt)
(704) 382-6232 (W)
(704) 364-7591 (H)

(2)
FERC
J W GOTSMER
(770) 452-2361 (W)
(770) 682-0196 (H)

(3)
SENIOR VP POWER GEN
T C McMEEKIN
(704) 382-4600 (W)
(704) 483-3898 (H)

(3)
UPSTREAM PLANTS
JOCASSEE
(864) 944-1464

DOWNSTREAM PLANTS
N/A

(4)
DOWNSTREAM UTILITIES
HARTWELL DAM CONTROL ROOM
(706) 376-2110

ARMY CORPS OF ENGRS SAVANAH
(912) 652-5431

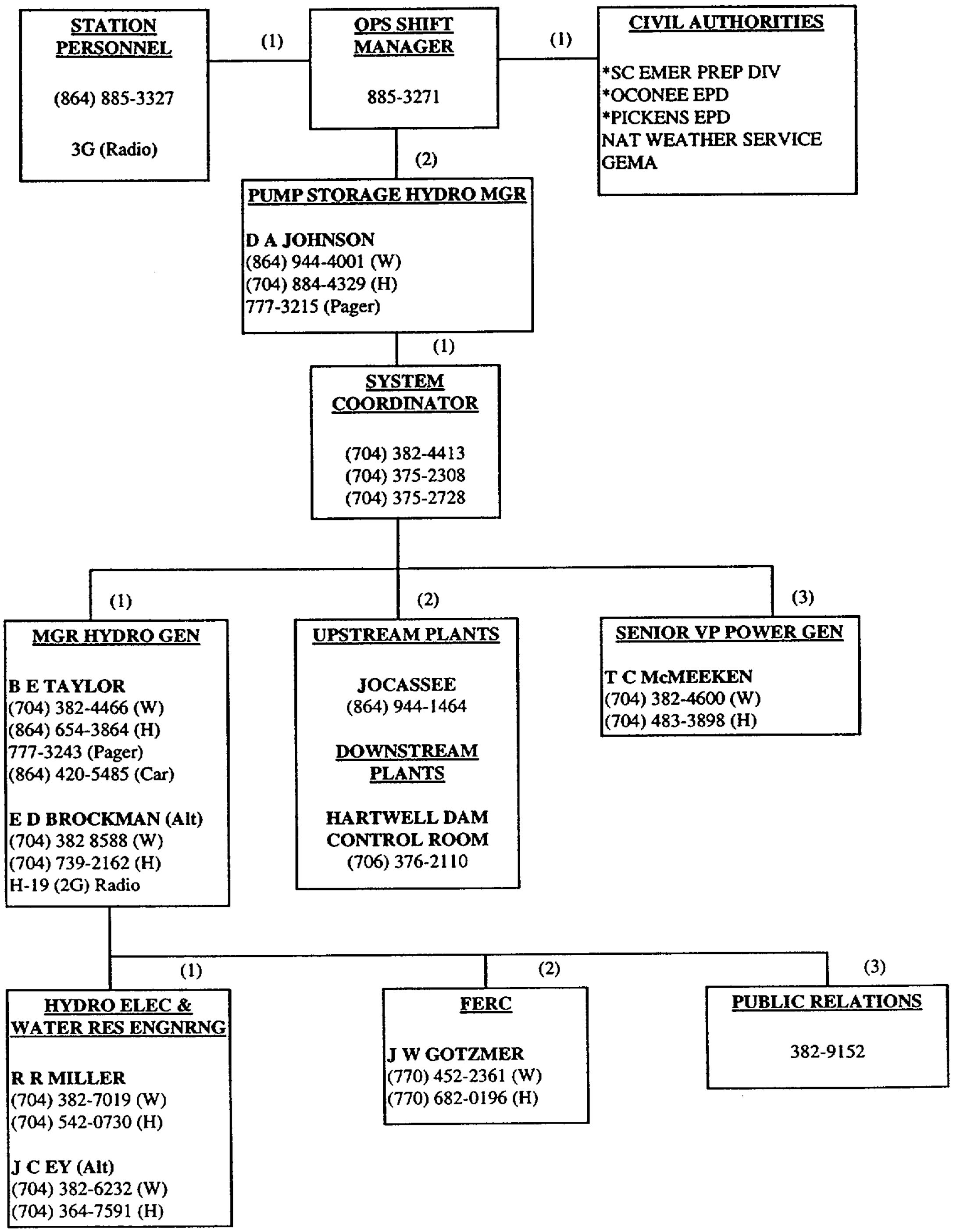
(5)
PUBLIC RELATIONS
382-9152

(1)
SC PSC
(803) 737-5100

() PRIORITY
* Notify by Selective Signaling

Enclosure 4.5

Condition B Notification Chart
(Unusual Event)



() PRIORITY
* Notify by Selective Signaling

The following listing of acronyms may be used within the body of this procedure or during conversations between the various Offsite Communicators. This list is provided as information to assist the Offsite Communicators.

CAN	Community Alert Network
CDEP	County Director of Emergency Preparedness
DHEC	Department of Health and Environmental Control
EC	Emergency Coordinator
ENS	Emergency Notification System
EOF	Emergency Operations Facility
EOFD	Emergency Operations Facility Director
EPD	Emergency Preparedness Division
ERO	Emergency Response Organization
FAX	Facsimile
FEOC	Forward Emergency Operations Center
GEMA	Georgia Emergency Management Agency
HPN	Health Physics Network
LEC	Law Enforcement Center
NEP	Nuclear Emergency Planning
NRC DSO	Nuclear Regulatory Commission, Director of Site Operations
NRC EOC	Nuclear Regulatory Commission, Emergency Operations Center
SDEP	State Director of Emergency Preparedness
SEOC	State Emergency Operations Center
SS	Selective Signaling
SWP	State Warning Point
TS	Technical Specifications
TSC	Technical Support Center

SRC-0

Facility: Oconee Date of Examination: 07-10/21-00

Examination Level: **SRO-U** Operating Test Number: 1

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Plant Parameter Verification	JPM CRO-040A (Bank-modified), Calculate Shutdown Margin with the Computer. (SRO ONLY) KA 2.1.7 [3.7/4.4] CFR 43.5/45.12/45.13 **Note:** This JPM will be conducted with B.1 Section JPM CRO-012, (Recovery of dropped rod)
	COLR/Tech. Spec Utilization	JPM NRC-005 (New) – Reactor Power Imbalance - Improved Tech Specs/COLR. KA 2.1.11 [3.0/3.8] CFR 43.2/45.13
A.2	Surveillance Testing	JPM NRC-004 (New) – Perform PT/1/A/0600/001, Enclosure 13.16, ICCM Subcooling Monitor Check. KA 2.2.12 [3.0/3.4] CFR 41.10/45.13, CFR 43.5/45.12/45.13
A.3	Radiation Control	SRO/RO – 2 Questions *Ability to perform procedures reduce excessive levels of radiation and guard against personnel exposure [CFR: 43.4/45.10] G2.3.10 (2.9/3.3)* Knowledge of 10 CFR 20 and related facility radiation control requirements. KA 2.3.1 [2.6/3.0] **Note:** These questions to be conducted with B.2 section JPM NLO-040 or 17.
A.4	Emergency Plan Implementation	SRO - JPM – Scenario event classification and protective action recommendations and/or classification upgrade. (SRO ONLY) KA 2.4.41 [2.3/4.1] CFR 43.5/45.11 **Note:** E-Plan classification to be conducted during C section simulator exams

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

CRO-40A/ADMIN A.1

CALCULATE SDM
WITH A DROPPED CONTROL ROD

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

CALCULATE SDM WITH A DROPPED CONTROL ROD

Alternate Path:

Yes, determine that 1% SDM does not exist and boration is required within 15 minutes.

Facility JPM #:

N/A

K/A Rating(s):

Gen 2.1.7 3.7/4.4

Task Standard:

PT/1/A/1103/15, Reactivity Balance Procedure is used to verify > 1% SDM with one inoperable (dropped) CR within 1 hour. Determine that 1% SDM does not exist and boration is required within 15 minutes.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/1103/15,Reactivity Balance Procedure
AP/1/A/1700/15, Dropped Control Rods
Improved Technical Specifications
- 3.1.4, Control Rod Group Alignment Limits
- 3.2.1, Regulating Rod Position Limits

Validation Time: 10 min. **Time Critical:** YES

===

Candidate: ____________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ____________________________ __________________________________/________
NAME SIGNATURE DATE

===

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC # SNAP _____
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

PT/1/A/1103/015, Reactivity Balance Procedure
OP/0/A/1105/009, Control Rod Drive System
Improved Technical Specifications
3.1.4, Control Rod Group Alignment Limits
3.2.1, Regulating Rod Position Limits

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.5.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

TIME CRITICAL

START TIME: _______

STEP 1: Within one hour verify > 1% SDM with allowance to the inoperable control rod. Perform PT/1/A/1103/15, Reactivity Balance Procedure. STANDARD: Obtain copy of PT/1/A/1103/15, Reactivity Balance Procedure. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 2 Determine proper enclosure to use. STANDARD: Enclosure 13.20, Shutdown Margin at Power, is chosen. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: Use Enclosure 13.21, Rod Position Limits at Power, 1 Inoperable Rod or 1 Dropped Rod – 4 Pump Flow. Verify available SDM is ≥ 1% ΔK/K by verifying that the control rod position and power level are within the acceptable region or the Restricted Region on the appropriate curve for the number of RCPs and Inoperable rods in Enclosure 13.21, Rod Position limits at Power. STANDARD: SDM is determined to be is ≤ 1% ΔK/K. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

<u>STEP 4</u>: Appropriate actions are taken per ITS 3.1.4, 3.1.5 and 3.2.1. <u>STANDARD</u>: Refer to ITS 3.1.4, 3.1.5 and 3.2.1 and determine that initiation of boration to restore SDM to within limits is required within 15 minutes. ***CUE: Inform student that an RO is commencing boration.*** <u>COMMENTS</u>: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	Step is necessary for the operator to select PT/1/A/1103/15, Reactivity Balance Calculations procedure to obtain correct enclosure to complete step three correctly.
2	Step is necessary for the operator to select to the Enclosure 13.20, Shutdown Margin at Power.
3	Step is necessary, the operator must interpret the 4 RCP curve to ensure adequate SDM.
4	Step is necessary, initiation of boration must be occur within 15 minutes to restore SDM.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.5.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

TIME CRITICAL

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.5.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

TIME CRITICAL

(R10-97) WHC/JMB/JPP

Duke Power Company Trng. (1) ID No AP/1/A/1700/015

PROCEDURE PROCESS RECORD

Revision No 4

LAN Location: SAROS

`EPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Dropped Control Rods

(4) Prepared By Dennis Gordon. Date 2/17/99

(5) Requires 10CFR50.59 evaluation?

- ☒ Yes (New procedure or revision with major changes)
- ☐ No (Revision with minor changes)
- ☐ No (To incorporate previously approved changes)

(6) Reviewed By Walter n. Barker (QR) Date 2/25/99

Cross-Disciplinary Review By ________ (QR)NA WB Date ________

Reactivity Mgmt. Review By Walter n. Barker (QR)NA ____ Date 2/25/99

(7) Additional Reviews

Reviewed By ________ Date ________

Reviewed By ________ Date ________

(8) Temporary Approval *(if necessary)*

By ________ (SRO/QR) Date ________

By ________ (QR) Date ________

(9) Approved By [signature] Date 3/8/99

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ________ Date ________

Compared with Control Copy ________ Date ________

Compared with Control Copy ________ Date ________

(11) Date(s) Performed ________

Work Order Number (WO#) ________

COMPLETION

(12) Procedure Completion Verification

- ☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
- ☐ Yes ☐ NA Listed enclosures attached?
- ☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
- ☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
- ☐ Yes ☐ NA Procedure requirements met?

Verified By ________ Date ________

(13) Procedure Completion Approved ________ Date ________

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Dropped Control Rods** **Continuous Use** **Reactivity Management Related**	Procedure No. AP/1/A/1700/015 Revision No. 004 Electronic Reference No. OX002RGS

DROPPED CONTROL RODS
Reactivity Management Related

TABLE OF CONTENTS

OCONEE NUCLEAR STATION

Dropped Control Rods

1. Purpose

This procedure provides the actions necessary to maintain the plant in a safe condition following a dropped control rod(s).

2. Symptoms

- "CRD POSITION ERROR" statalarm (1SA-2, B-10)
- "CRD SAFETY RODS NOT AT UPPER LIMIT" statalarm (1SA-2, C-10)
- Control Rod "IN-LIMIT" light on (Green light on Position Indication Panel)
- Control Rod "ASYMM. RODS" light on (Yellow light on Diamond).

3. Automatic Systems Actions

3.1 **IF** ICS is in Auto,

AND an "ASYMM. RODS" (Yellow Light on Diamond) occurs,

THEN an "OUT" inhibit at 60% power is established and the Reactor will runback to 55% power.

3.1.1 **IF** the "ASYMM. RODS" (Yellow Light on Diamond) clears,

THEN runback may stop before reaching 55% power.

4. Immediate Manual Actions

_____ 4.1 **IF** more than one Control Rod has dropped,

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 4.2 **IF** more than one Control Rod is misaligned > 9" (6%),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.3 **IF** due to a malfunction, a Control Rod Group is misaligned > 9" (6%) from the acceptable region of PT/1/A/0600/001, (Periodic Instrument Surveillance),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.4 **IF** a Control Rod has dropped on an approach to criticality,

OR a dropped Control Rod results in a return to subcriticality from a critical condition,

THEN manually insert all Control Rods to Group 1 at 50% WD.

5. Subsequent Actions

_____ 5.1 **IF** the Reactor has tripped,

THEN **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 5.2 Verify Reactor runback <60% Full Power is in progress.:

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

NOTE 5.2.1:	If a control rod drops while the ICS Diamond Panel is in MANUAL, an "Auto Inhibit" (safety rod) or "Sequence Enable" (regulating rod) alarm will prevent placing the ICS in automatic.

_____ 5.2.1 **IF** the Reactor has **NOT** runback,

THEN commence manual Reactor Power reduction to < 60% of the allowable thermal power for the RCP combination.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.3 **IF** operating with only three (3)RCPs,

THEN commence manual Reactor Power reduction to < 45% Full Power.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.4 Notify I&E to begin investigation for the cause of the Dropped Control Rod.

_____ 5.5 Initiate actions to meet the Required Action of ITS 3.1.4 by performing the following:

_____ 5.5.1 Within one hour verify > 1% SDM
with allowance for the inoperable control rod(s):

- Perform PT/1/A/1103/15, (Reactivity Balance Calculation).

_____ 5.5.2 Within two hours reduce Reactor Power < 60%
of the allowable thermal power for the RCP combination.

NOTE 5.5.3:	Notification to I&E should be made as soon as possible due to the complexity of resetting the RPS trip setpoints and the short ITS allowable time limits.

_____ 5.5.3 Notify I&E to reduce RPS Flux/Flow-Imbalance trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.5.4 Notify I&E to reduce RPS High Flux trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.6 **WHEN** Reactor Power is < 60%
of the allowable thermal power for the RCP combination,

THEN notify I&E to begin repair of the Dropped Control Rod.

_____ 5.7 **WHEN** I&E is ready to begin repairs on the Dropped Control Rod,

THEN Place the ICS Diamond control station in MANUAL,

AND permit I&E to repair Dropped Control Rod.

CAUTION:5.8	The Duty Reactor Engineer must evaluate the effects of local power distribution and the necessity for special maneuvering limits prior to the recovery of a dropped or an asymmetric Control Rod.

_____ 5.8 **WHEN** I&E has repaired the Dropped Control Rod,

THEN recover the Dropped Control Rod per OP/0/A/1105/009, (Control Rod Drive System).

END

Dropped Control Rods

Appendix

1. PIP # 0-O98-2734

END

(R06-97)

SR
Slm
NRC
115
JPP
JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. PT/1/A/1103/15

Revision No 51

PARATION

Station OCONEE NUCLEAR STATION

(3) Procedure Title Reactivity Balance Procedure (Unit 1)

INFORMATION ONLY

(4) Prepared By Stto Ceumo Date 7/28/99

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)
☐ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By TE Sanders (QR) Date 8/2/99

Cross-Disciplinary Review By Joe Price (QR) NA____ Date 8-16-99

Reactivity Mgmt. Review By TE Sanders (QR) NA____ Date 8/2/99

(7) Additional Reviews

Reviewed By ______________________ Date ________

Reviewed By ______________________ Date ________

(8) Temporary Approval *(if necessary)*

By ______________________ (SRO/QR) Date ________

By ______________________ (QR) Date ________

(9) Approved By [signature] Date 2/24/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______________________ Date ________

Compared with Control Copy ______________________ Date ________

Compared with Control Copy ______________________ Date ________

(11) Date(s) Performed ______________________

Work Order Number (WO#) ______________________

COMPLETION

(12) Procedure Completion Verification

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ______________________ Date ________

Procedure Completion Approved ______________________ Date ________

Remarks *(Attach additional pages, if necessary)*

FEB 2000
Received
Operator
Training

<table>
<tr><td rowspan="2">Duke Power Company
Oconee Nuclear Station

Reactivity Balance Procedure (Unit 1)</td><td>Procedure No.
PT/1/A/1103/15</td></tr>
<tr><td>Revision No.
51</td></tr>
<tr><td>* This procedure has the potential to affect Reactivity Management *
Continuous Use</td><td>Electronic Reference No.</td></tr>
</table>

Reactivity Balance Procedure

1. Purpose

1.1 To calculate the Boron concentration necessary to provide greater than 1% ΔK/K shutdown margin.

1.2 To calculate the actual shutdown margin when the reactor is shutdown.

1.3 To evaluate the available shutdown margin during power operation (e.g., in the event of an inoperable rod.)

1.4 To provide the minimum RCS Boron concentration required to ensure greater than 1% ΔK/K shutdown margin to perform the Control Rod Drive (CRD) patch verification (for initial startup following refueling).

1.5 To estimate the critical rod configuration or the critical Boron concentration prior to startup.

1.6 To provide a method for preventing inadvertent criticality using subcritical multiplication measurement.

1.7 To provide nominal APSR position.

2. References

2.1 Improved Tech Specs:
- 1.1, Definitions - Shutdown Margin
- 3.1.1, Shutdown Margin
- 3.1.4, Control Rod Group Alignment Limits
- 3.1.5, Safety Rod Position Limits
- 3.2.1, Regulating Rod Position Limits
- 3.3.9, Source Range Neutron Flux
- 3.9.1, Boron Concentration

2.2 Selected Licensee Commitments: 16.13.4, Reactivity Anomalies

2.3 Unit 1 - Physics Test Manual (PTM)

2.4 Unit 1 - Core Operating Limits Report (COLR)

2.5 Nuclear Systems Directive 304, Reactivity Management

3. Time Required

Two people - 1 hour for most Enclosures

4. Prerequisite Tests

None

5. Test Equipment

Personal computer (for computerized calculations)

6. Limits and Precautions

6.1 Operations uses the results of this procedure to make important operational decisions, therefore this procedure affects core reactivity.

6.2 Appropriate corrections have been made per this procedure, or actual plant conditions must be the same as the reference conditions stated on the appropriate enclosure(s).

6.3 Independent verification is required for each calculation performed. For hand calculations, this requires that two people separately complete the appropriate enclosures for the desired calculation to verify the results are in agreement. For computerized calculations, this requires that two people separately run the computer code(s) or verify the input.

7. Required Unit Status

None

8. Prerequisite System Conditions

None

9. Test Method

9.1 Shutdown Boron Concentration:

Calculated in Enclosure 13.1 or 13.2.
The shutdown Boron concentration provides a greater than 1.0% $\Delta K/K$ shutdown margin with the worst case stuck rod assumed to be out and with conservatism applied per standard practice for Babcock & Wilcox 177 fuel assembly reactors.

A reference shutdown Boron concentration is obtained based on the cycle burnup, rod positions and RCS temperature. The reactivity worths of Xenon, Samarium, and the inoperable rod penalty (if applicable) are converted into their equivalent Boron concentrations. (Credit is taken only for the minimum Xenon worth occurring in a specified time interval, which should not exceed 12 hours. The Shutdown Boron concentration is valid ONLY during that time interval. Due to inaccuracies in the Xenon models, .8 times the Xenon and Samarium worth are used unless the RCS is below 450°F, in which case .5 times the Xenon and Samarium worths are used. Xenon and Samarium worths may be assumed to be zero for conservatism.) These Boron concentrations are then applied to the reference Boron concentration to provide the <u>required Boron concentration for a greater than 1.0% ΔK/K shutdown margin (i.e., the shutdown Boron concentration).</u>

9.2 Shutdown Margin Calculation while Shutdown:

Calculated in Enclosure 13.1 or 13.2.
The shutdown margin is the amount of reactivity by which the reactor is shutdown. The worst case stuck rod is assumed to be out, and additional conservatisms are applied per standard practice for Babcock and Wilcox 177 fuel assembly reactors. If operating with a known inoperable rod, an additional penalty is applied to account for that rod. This penalty need not be applied when the reactor is shutdown if that rod can be confirmed to be fully inserted by redundant indications. The shutdown Boron concentration must first be found per 9.1. The actual Boron concentration is then subtracted from this concentration and the result converted to % ΔK/K. 1.0% ΔK/K is then subtracted from this value to obtain the shutdown margin, expressed in - % ΔK/K. A separate check for SSF RC Makeup System operability is performed, which takes no credit for Xenon, but does not require the stuck rod penalty. This limit is shown in Enclosure 13.23.

Following a shutdown, Control Rod Position at the time of Shutdown may be used with the Rod Position Limit curves (in Enclosure 13.21) to verify at least 1% ΔK/K shutdown margin for the first 3 hours following shutdown (provided RCS Temperature stays >= 532°F and boron does not decrease). This may be necessary for shutdowns with an inoperable rod, since the more conservative calculation method (in Enclosure 13.1, 13.2) may not show 1% ΔK/K shutdown margin immediately after shutdown. Boration should begin immediately to be able to show 1% ΔK/K shutdown margin using the calculation method.

9.3 Shutdown Margin at Power:

Verified in Enclosure 13.20.
While at power, the available shutdown margin may be verified to be >= 1% ΔK/K by using the Rod Position Limits curves. Operation in the "Acceptable Region" of these

curves ensures that the shutdown margin following a reactor trip will be >= 1% ΔK/K with the worst stuck rod out. There are curves for 3 and 4 RCP operation, and curves for 0 and 1 inoperable rod. A dropped rod is considered inoperable for the purposes of providing shutdown margin while at power.

9.4 Estimated Critical Rod Position:

Calculated in Enclosure 13.3.
The core excess reactivity is obtained based on the cycle burnup. The reactivity worths associated with Boron, Xenon, temperature correction (if RCS temperature not at 532°F) and Samarium are then obtained and summed with the core excess reactivity. The groups 5-7 positions are then determined for which the inserted rod worth when summed with all the above, yields a total core reactivity of 0.0% ΔK/K. The upper and lower rod position limits are then determined and the actual critical rod positions are recorded.

9.5 Estimated Critical Boron Concentration:

Calculated in Enclosure 13.4.
The core excess reactivity is obtained based on the cycle burnup. The reactivity worth associated with Xenon, temperature correction (if RCS temperature not at 532°F), Samarium and the desired critical rod positions are summed with the core excess reactivity. The Boron concentration is then determined for which its reactivity worth, when summed with all the above, yields a total core reactivity of 0.0% ΔK/K.

9.6 Subcritical Multiplication Measurement:

Performed in Enclosure 13.6.
With Group 1 at 50% wd, an initial source range (SR) count rate (C_0) is recorded. During control rod withdrawals, new counts (C) are recorded and used to calculate 1/M, or C_0/C. As criticality is approached, C/C_0 will approach infinity, and 1/M will approach zero. Plotting 1/M versus rod worth provides a rough indication of what rod position will yield a critical condition, and acts as an indication of premature criticality, or criticality more than 0.75% ΔK/K below the Estimated Critical Position calculated in 9.4.

10. Data Required

10.1 For Xenon Worth: cycle burnup and power history to time of last equilibrium xenon.

10.2 For Shutdown Boron Concentration/Shutdown Margin Calculation: Power, cycle burnup, RCS temperature, Group 1 and 8 positions, Xenon worth and the actual boron concentration.

10.3 For Estimated Critical Rod Configuration: RCS temperature, cycle burnup, present boron concentration, Xenon worth, and Samarium worth.

10.4 For Estimated Critical Boron Configuration: RCS temperature, cycle burnup, desired critical rod configuration, Xenon worth, and Samarium worth.

10.5 For Subcritical Multiplication Measurement: Control Rod position and source range (SR) count rate.

11. Acceptance Criteria

Independent/Separate verifications should agree within 10 ppmB (for Shutdown Boron or Estimated Critical Boron) or 5%wd (for Estimated Critical Position). The more conservative Shutdown Boron Concentration calculation shall be used to ensure at least a 1.0% $\Delta K/K$ shutdown margin.

12. Procedure

Complete, or refer to, the appropriate enclosure(s):

Shutdown Margin Calculation at power:
Enclosure 13.20

Shutdown Margin Calculation while shutdown:
Enclosure 13.1, "Shutdown Boron Concentration/Shutdown Margin Calculation,"
OR-
Enclosure 13.2 "Computerized Shutdown Margin Calculation"

Refueling Outage Boron Concentrations:
Enclosure 13.14, "Refueling Outage Boron Concentrations"

Estimated Critical Rod Position:
Enclosure 13.3, "Computerized Estimated Critical Rod Position Calculation"

Estimated Critical Boron Concentration:
Enclosure 13.4, "Computerized Estimated Critical Boron Calculation"

Subcritical Multiplication (1/M) Measurement:
Enclosure 13.6, "Subcritical Multiplication (1/M) Measurement"

Instructions for obtaining Xenon Prediction:
Enclosure 13.16, "Instructions for Obtaining Xenon Prediction"

Required Control Rod Group 8 Position:
Enclosure 13.15, "Required Group 8 Position and Designed Cycle Length"

Designed Cycle Length Information:
Enclosure 13.15, "Required Group 8 Position and Designed Cycle Length"

RCS Boron Concentration for SSF Operability:
Enclosure 13.23, "Minimum RCS Boron Concentration to Maintain SSF Operability"

Required Shutdown Margin:
Enclosure 13.18, "Shutdown Margin Requirements"

NOTE:	Only the appropriate completed enclosures need be attached to the procedure cover sheet to be submitted for procedure completion.

13. Enclosures

13.1 Shutdown Boron Concentration/Shutdown Margin Calculation

13.2 Computerized Shutdown Margin Calculation

13.2.1 Computerized Shutdown Margin Calculation Documentation

13.3 Computerized Estimated Critical Rod Position Configuration

13.4 Computerized Estimated Critical Boron Concentration

13.5 Deleted

13.6 Subcritical Multiplication (1/M) Measurement

13.7 Core Excess Reactivity vs. Burnup

13.8 Differential Boron Worth vs. Burnup

13.9 Inserted Control Rod Worth (for 1/M measurement of Groups 1-7)

13.10 Temperature Coefficient vs. RCS Boron Concentration

13.11 Shutdown Boron Concentration vs. Burnup (Group 1 @ 0% wd)

13.12 Shutdown Boron Concentration vs. Burnup (Group 1 @ 50% wd)

13.13 Inoperable Rod Penalty for Individual Inoperable Rod

13.14 Refueling Outage Boron Concentrations

13.15 Required Group 8 Position and Designed Cycle Length

13.16 Instructions for Obtaining Xenon Prediction

13.17 Power Defect vs. Reactor Power

13.18 Shutdown Margin Requirements

13.19 Control Rod Group Worths for Control Rod Drop Time Testing

13.20 Shutdown Margin Calculation at Power

13.21 Rod Position Limits at Power

13.22 Group 7 Control Rod Worth

13.23 Minimum RCS Boron Concentration to Maintain SSF Operability

Enclosure 13.20
Shutdown Margin Calculation at Power

Performed By: ____________________

NOTE:	A dropped rod is considered inoperable for the purpose of providing shutdown margin while at power.

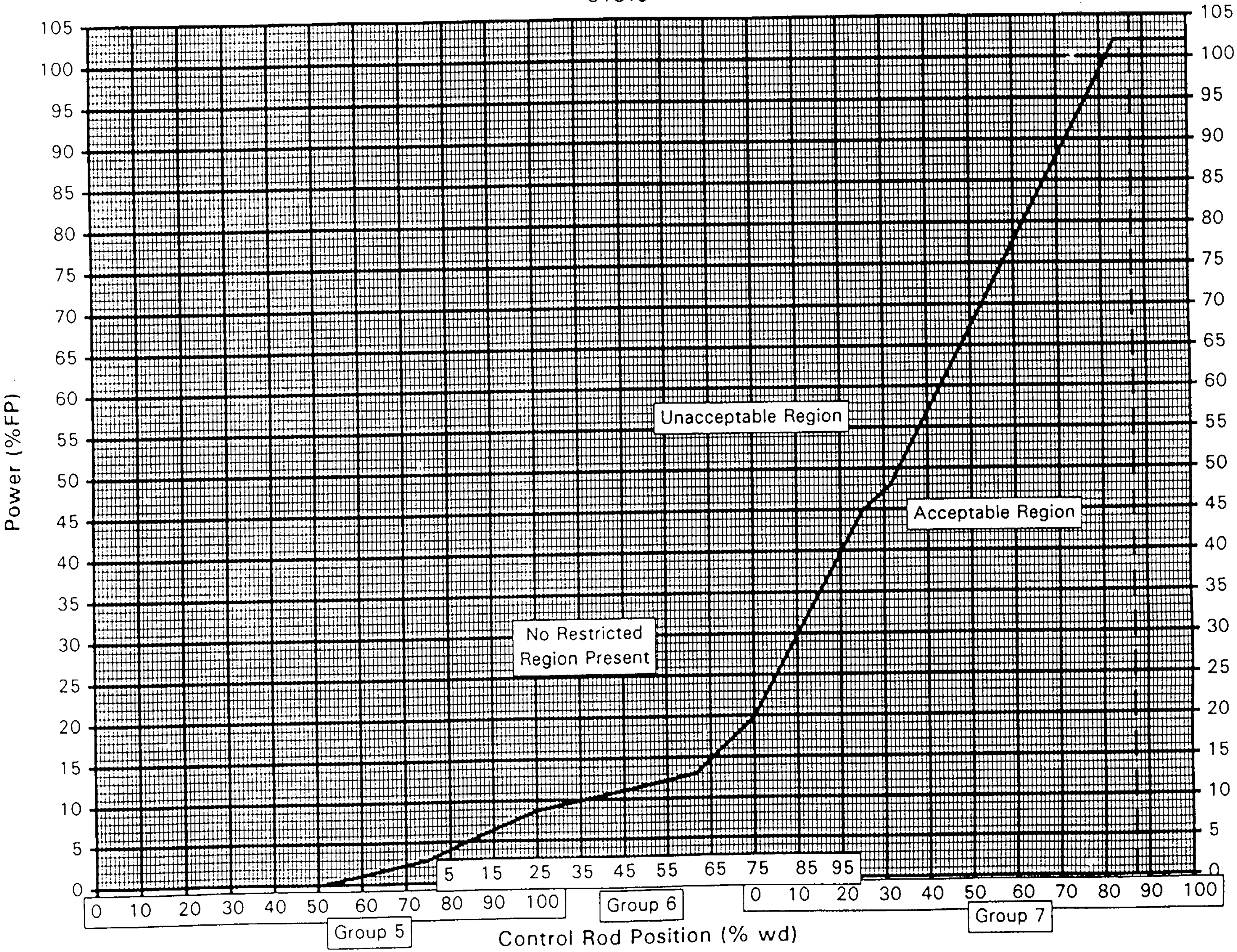
13.20.1 Verify one of the following:

____ ____ 13.20.1.a Available shutdown margin is ≥1% ΔK/K. This is shown by verifying that the control rod position and power level are within the Acceptable Region or the Restricted Region on the appropriate curve for the number of RC Pumps and Inoperable rods in Enclosure 13.21, Rod Position Limits at Power.
IV

- OR -

____ ____ 13.20.1.b Appropriate actions are taken per ITS 3.1.4, 3.1.5 and 3.2.1.
IV

Enclosure 13.21
Rod Position Limits at Power
1 Inoperable Rod or 1 Dropped Rod - 4 Pump Flow
01C19



3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 CONTROL ROD Group Alignment Limits

LCO 3.1.4 Each CONTROL ROD shall be OPERABLE and aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One trippable CONTROL ROD inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1 Restore CONTROL ROD alignment.	1 hour
	<u>OR</u>	
	A.2.1.1 Verify SDM is within the limit specified in the COLR.	1 hour <u>AND</u> Once per 12 hours thereafter
	<u>OR</u>	
	A.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER. <u>AND</u> A.2.3 Reduce the nuclear overpower trip setpoints, based on flux and flux/flow imbalance, to ≤ 65.5% of the ALLOWABLE THERMAL POWER. <u>AND</u> A.2.4 Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis.	2 hours 10 hours 72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	12 hours
C. More than one trippable CONTROL ROD inoperable, or not aligned within 6.5% of its group average height, or both.	C.1.1 Verify SDM is within the limit specified in the COLR. <u>OR</u>	1 hour (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	C.2	Be in MODE 3.	12 hours
D. One or more rods untrippable.	D.1.1	Verify SDM is within the limit specified in the COLR.	1 hour
	OR		
	D.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	D.2	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual CONTROL ROD positions are within 6.5% of their group average height.	12 hours
SR 3.1.4.2	Verify CONTROL ROD freedom of movement (trippability) by moving each individual CONTROL ROD that is not fully inserted by an amount in any direction sufficient to demonstrate the absence of thermal binding.	92 days
SR 3.1.4.3	Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is ≤ 1.66 seconds at reactor coolant full flow conditions or ≤ 1.40 seconds at no flow conditions from power interruption at the CONTROL ROD drive breakers to ¾ insertion (25% withdrawn position).	Prior to reactor criticality after each removal of the reactor vessel head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Safety Rod Position Limits

LCO 3.1.5 Each safety rod shall be fully withdrawn.

-------------------------------------NOTE-------------------------------------
Not required for any safety rod positioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One safety rod not fully withdrawn.	A.1 Withdraw the rod fully.	1 hour
	<u>OR</u>	
	A.2.1.1 Verify SDM is within the limit specified in the COLR.	1 hour
	<u>OR</u>	
	A.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2.2 Declare the rod inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. More than one safety rod not fully withdrawn.	B.1.1 Verify SDM is within the limit specified in the COLR.	1 hour
	<u>OR</u>	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	B.2 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each safety rod is fully withdrawn.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Regulating Rod Position Limits

LCO 3.2.1 Regulating rod groups shall be within the physical position, sequence, and overlap limits specified in the COLR.

-----------------------------------NOTE-----------------------------------
Not required for any regulating rod positioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Regulating rod groups sequence or overlap requirements not met.	A.1	Restore regulating rod groups to within limits.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Regulating rod groups positioned in restricted or unacceptable region.	B.1	-----------NOTE----------- Not applicable to regulating rod groups positioned in the restricted region. ------------------------------	
		Initiate boration to restore SDM to within the limits specified in the COLR.	15 minutes
	AND		
	B.2.1	Restore regulating rod groups to within acceptable region.	2 hours
	OR		
	B.2.2	Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group position limits.	2 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	12 hours
SR 3.2.1.2	Verify regulating rod groups meet the position limits as specified in the COLR.	12 hours
SR 3.2.1.3	Verify SDM to be within the limit as specified in the COLR.	Within 4 hours prior to achieving criticality

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NRC-005/ADMIN A.1

REACTOR POWER IMBALANCE
Improved Technical Specifications/COLR

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

Axial Power Imbalance

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

Gen 2.1.11 3.07/3.8

Task Standard:

Perform power imbalance within limits verification.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and Section 12.3
PT/0/A/1103/019, Backup Incore Detector System
Core Operating Limits Report

Validation Time: 25 min. **Time Critical:** NO

==

Candidate: ______________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________________ ___________________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and Section 12.3
PT/0/A/1103/019, Backup Incore Detector System
Core Operating Limits Report

READ TO OPERATOR

DIRECTIONS TO STUDENT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

The Unit has been operating at 100% power for 2 weeks.
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 has been completed up to page 6, Axial Power Imbalance Operating Limits.

INITIATING CUE:

The SRO directs you to perform the Axial Power Imbalance Operating Limits verification.
The Reactor calculation package is NOT running.

START TIME: _______

<table>
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STEP 1:
Verify Power imbalance within operational alarm limit in COLR when > 40% RTP.

IF Reactor calculation package is NOT running on computer, refer to Section 12.3.

STANDARD: When told Reactor Calculation package not running, refer to Section 12.3.

CUE: Tell candidate that the Reactor calculation package is not running.

COMMENTS:

</td>
<td>___ SAT

___ UNSAT</td>
</tr>
<tr>
<td>

STEP 2:

Axial Imbalance shall **NOT** exceed appropriate limit curve in COLR.
IF axial imbalance limit is exceeded, take immediate corrective action to
IF an acceptable imbalance is **NOT** achieved within 2 hours, reactor power shall be reduced until imbalance limits are met.
Refer to TS 3.2.2.

STANDARD: Candidate obtains the correct limit curve in COLR. This curve is located on page 12 of 31 (Oconee 1 Cycle 19) (Oconee 2 Cycle 18) (Oconee 3 Cycle 18)

NOTE: Later in JPM when imbalance calculation is made with the Incores a different enclosure from the COLR will be used.

CUE: Only Imbalance Surveillance is required for this JPM. Step 12.3.2 is not required.

COMMENTS:

</td>
<td>___ SAT

___ UNSAT</td>
</tr>
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STEP 3: Order of preference of measurement systems to determine axial imbalance and quadrant power tilt is as follows: A. Incore Detectors (Computer Reactor Calculation Package). B. Outcore Detectors (Power Range Outcore Detectors). C. Backup Incore Detectors. Refer to PT/10/A/1103/019 (Backup Incore Detector System). ***CUE: The Backup Incore detectors will be used for this determination.*** STANDARD: Candidate refers to PT/10/A/1103/019 (Backup Incore Detector System). COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 4: Verification of minimum Incore operability. **NOTE: Backup Incore Chart "A" points and information provided to the student.** STANDARD: **NOTE: Give student Backup Incore Chart "A" data sheet.** **CUE: Inform candidate that all points on Backup Incore Chart "A" are operable (no points are off scale or contain a note).** COMMENTS:	___ SAT ___ UNSAT
STEP 5: 12.2.1 Verify the reactor has been at steady state conditions (± 2% FP) for at least 30 minutes. STANDARD: The Candidate determines is reactor power is steady. ***CUE: Reactor power has been at 100% power for the past 2 weeks.*** COMMENTS:	___ SAT ___ UNSAT

<u>STEP 6</u>: Calculate axial imbalance per Enclosure 13.3 using operable recorder points identified on Enclosure 13.1. <u>STANDARD</u>: The candidate refers to and obtains a copy of Enclosures 13.1 and 13.3 The candidate performs calculation per Enclosure 13.3. **NOTE: Refer to completed enclosure 13.3.** <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT
<u>STEP 7</u>: Verify the calculated axial imbalance does not exceed the backup incore limits per 11.1. <u>STANDARD</u>: The candidate verifies the calculated axial imbalance does not exceed the backup incore limits per 11.1, (-18.7 / +18.7) the current Core Operating Limits Report (COLR) on the Backup Incore Setpoint Column of the (Error-Adjusted) "Operational Power Imbalance Setpoints" Table. <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: If either step 12.2.4 or 12.2.5 cannot be satisfied, notify the Unit Supervisor and take appropriate actions described in the applicable Technical Specifications as listed below Quadrant Power Tilt - ITS 3.2.3 Axial Power Imbalance - ITS 3.2.2 ***CUE: Inform candidate that for this JPM only imbalance will be checked.*** STANDARD: Candidate determines that step 12.2.4 is satisfied. COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
3	Step is necessary, because reference to the must use Backup Incore System procedure must be used to determine imbalance.
6	Step is necessary, because calculation is needed to determine imbalance.
7	Step is necessary, because imbalance must be compared to COLR to verify within limits.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

The Unit has been operating at 100% power for 2 weeks.
PT/1/A/0600/001, Periodic Instrument Surveillances, Enclosures 13.1 has been completed up to page 6, Axial Power Imbalance Operating Limits.

INITIATING CUE:

The SRO directs you to perform the Axial Power Imbalance Operating Limits verification.
The reactor calculation package is NOT running.

Enclosure 13.1
Required Backup Recorder Points For Calculating Axial Power Imbalance

* UNIT 1 ONLY *

From the following list of allowed point groupings for calculating axial power imbalance, three sets for which the points indicated are all operable and in calibration must be available.

NOTES: "A" recorder is on the Control Board; "B" is on the vertical board.
A set of recorder points for imbalance calculations consist of three operable points grouped together.

Any of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	3	G9	6		B	8	L6	6
A	2	G9	4		B	7	L6	4
A	1	G9	2		B	6	L6	2
A	22	D5	6		B	19	N4	6
A	24	D5	4		B	18	N4	4
A	14	D5	2		B	17	N4	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	6	E9	6		B	4	K5	6
A	5	E9	4		B	5	K5	4
A	4	E9	2		B	10	K5	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	18	F13	6		B	22	O6	6
A	15	F13	4		B	21	O6	4
A	13	F13	2		B	20	O6	2

Recorded By ______________________ Date ________________

BACKUP INCORE CHART “A”		
Point #	**%**	**Location**
1	195.1	G09-L2
2	203.0	G09-L4
3	196.1	G09-L6
4	214.1	E09-L2
5	226.6	E09-L4
6	209.6	E09-L6
7	189.4	L06-L4
8	196.8	L06-L6
9	185.7	M09-L2
10	180.7	K05-L2
11	187.5	G11-L2
12	179.7	E07-L2
13	212.3	F13-L2
14	213.2	D04-L2
15	210.4	F13-L4
16	209.5	F03-L6
17	212.5	N04-L2
18	197.6	F13-L6
19	196.7	N04-L6
20	199.3	O06-L2
21	200.2	O06-L4
22	199.4	O06-L6
23	196.8	D05-L6
24	197.7	D05-L4

Enclosure 13.3
Axial Power Imbalance Calculation Sheet

13.3.1 Using Enclosure 13.1 choose a set of backup recorder points for which all points are operable and record the point identifications (recorder, point number, detector location, and level) on the blanks provided below.

13.3.2 Record the backup recorder readings on the blank provided below.

13.3.3 Record current % FP on the blanks provided below using the priority of thermal power indications listed in PT/0/A/0600/001 (usually either power range NIs or from the results of PT/0/A/0205/05, Thermal Power Calculation).

13.3.4 Calculate imbalance for each of the three detector strings using the following formula:

$$\text{IMB} = \frac{\text{R (level 6 or 5)} - \text{R (level 2 or 3)}}{\text{R (level 6 or 5)} + \text{R (level 4)} + \text{R (level 2 or 3)}} \times \%\text{ FP}$$

where R = detector reading

13.3.5 The imbalance from the backup recorders is the average for the three detector strings, calculated as indicated below:

	RECORDER ID	POINT #	DETECTOR LOCATION	DETECTOR LEVEL	DETECTOR READING (R)	
I	A	3	609-L6	6 or 5	196.1	$\text{IMB}_{I} = \frac{(196.1 - 195.1)}{(196.1 + 203.0 + 195.1)} \times 100\ \%\text{FP} = .17\ \%\text{IMB}$
	A	2	609-L4	4	203.0	
	A	1	609-L2	2 or 3	195.1	
II	A	6	E9-L6	6 or 5	209.6	$\text{IMB}_{II} = \frac{(209.6 - 214.1)}{(209.6 + 226.6 + 214.1)} \times 100\ \%\text{FP} = -.69\ \%\text{IMB}$
	A	5	E9-L4	4	226.6	
	A	4	E9-L2	2 or 3	214.1	
III	A	18	F13-L6	6 or 5	197.6	$\text{IMB}_{III} = \frac{(197.6 - 212.3)}{(197.6 + 210.4 + 212.3)} \times 100\ \%\text{FP} = -2.37\ \%\text{IMB}$
	A	15	F13-~~L4~~	4	210.4	
	A	13	F13-L2	2 or 3	212.3	

TOTAL -2.89 %IMB

AVERAGE IMBALANCE = TOTAL/3 = -.96 %IMB

Calculated by ______ Date/Time ______ Verified by ______ Date/Time ______

BACKUP INCORE CHART "A"		
Point #	**%**	**Location**
1	195.1	G09-L2
2	203.0	G09-L4
3	196.1	G09-L6
4	214.1	E09-L2
5	226.6	E09-L4
6	209.6	E09-L6
7	189.4	L06-L4
8	196.8	L06-L6
9	185.7	M09-L2
10	180.7	K05-L2
11	187.5	G11-L2
12	179.7	E07-L2
13	212.3	F13-L2
14	213.2	D04-L2
15	210.4	F13-L4
16	209.5	F03-L6
17	212.5	N04-L2
18	197.6	F13-L6
19	196.7	N04-L6
20	199.3	O06-L2
21	200.2	O06-L4
22	199.4	O06-L6
23	196.8	D05-L6
24	197.7	D05-L4

Enclosure 13.1
Required Backup Recorder Points For Calculating Axial Power Imbalance

* UNIT 1 ONLY *

From the following list of allowed point groupings for calculating axial power imbalance, three sets for which the points indicated are all operable and in calibration must be available.

NOTES: "A" recorder is on the Control Board; "B" is on the vertical board.
A set of recorder points for imbalance calculations consist of three operable points grouped together.

Any of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	3	G9	6		B	8	L6	6
A	2	G9	4		B	7	L6	4
A	1	G9	2		B	6	L6	2
A	22	D5	6		B	19	N4	6
A	24	D5	4		B	18	N4	4
A	14	D5	2		B	17	N4	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	6	E9	6		B	4	K5	6
A	5	E9	4		B	5	K5	4
A	4	E9	2		B	10	K5	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	18	F13	6		B	22	O6	6
A	15	F13	4		B	21	O6	4
A	13	F13	2		B	20	O6	2

Recorded By ______________________ Date ________________

Enclosure 13.3
Axial Power Imbalance Calculation Sheet

13.3.1 Using Enclosure 13.1 choose a set of backup recorder points for which all points are operable and record the point identifications (recorder, point number, detector location, and level) on the blanks provided below.

13.3.2 Record the backup recorder readings on the blank provided below.

13.3.3 Record current % FP on the blanks provided below using the priority of thermal power indications listed in PT/0/A/0600/001 (usually either power range NIs or from the results of PT/0/A/0205/05, Thermal Power Calculation).

13.3.4 Calculate imbalance for each of the three detector strings using the following formula:

$$\text{IMB} = \frac{\text{R (level 6 or 5) - R (level 2 or 3)}}{\text{R (level 6 or 5) + R (level 4) + R (level 2 or 3)}} \times \text{\% FP}$$

where R = detector reading

13.3.5 The imbalance from the backup recorders is the average for the three detector strings, calculated as indicated below:

	RECORDER ID	POINT #	DETECTOR LOCATION	DETECTOR LEVEL	DETECTOR READING (R)		
I	______	______	______	6 or 5	______	IMB_{I}=	(_____ - _____) / (____ + ____ + ____) x ____%FP = ____%IMB
	______	______	______	4	______		
	______	______	______	2 or 3	______		
II	______	______	______	6 or 5	______	IMB_{II}=	(_____ - _____) / (____ + ____ + ____) x ____%FP = ____%IMB
	______	______	______	4	______		
	______	______	______	2 or 3	______		
III	______	______	______	6 or 5	______	IMB_{III}=	(_____ - _____) / (____ + ____ + ____) x ____%FP = ____%IMB
	______	______	______	4	______		
	______	______	______	2 or 3	______		

TOTAL _____%IMB

AVERAGE IMBALANCE = TOTAL/3 = ______%IMB

Calculated by ____________ Date/Time ________________ Verified by ____________ Date/Time ______________

(R06-97)

SR
Sum
NRC
115
JPP
JMB

Duke Power Company

PROCEDURE PROCESS RECORD

(1) ID No PT/1/A/0600/001

Revision No 217

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Periodic Instrument Surveillance

(4) Prepared By William M. Buchanan (Signature) William M. Buchanan Date 04/11/00

(5) Requires 10CFR50.59 evaluation?

☐ Yes (New procedure or revision with major changes)

☒ No (Revision with minor changes)

☐ No (To incorporate previously approved changes)

(6) Reviewed By [signature] (QR) Date 4/12/00

Cross-Disciplinary Review By ____ (QR)NA [initials] Date ____

Reactivity Mgmt. Review By ____ (QR)NA [initials] Date ____

(7) Additional Reviews

Reviewed By (IT) Alan Sweeney (Time Sync Only) Date 4/10/00

Reviewed By ____ Date ____

(8) Temporary Approval *(if necessary)*

By ____ (SRO/QR) Date ____

By ____ (QR) Date ____

(9) Approved By D B Coyle Date 4/12/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

(11) Date(s) Performed ____

Work Order Number (WO#) ____

COMPLETION

(12) Procedure Completion Verification:

☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?

☐ Yes ☐ NA Listed enclosures attached?

☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?

☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?

☐ Yes ☐ NA Procedure requirements met?

Verified By ____ Date ____

(13) Procedure Completion Approved ____ Date ____

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Periodic Instrument Surveillance** **Continuous Use**	Procedure No. PT/1/A/0600/001 Revision No. 217 Electronic Reference No. OX002WAS

Periodic Instrument Surveillance

1. Purpose

1.1 To periodically verify proper operation of various instruments and systems.

2. References

2.1 Technical Specifications (TS)

2.2 DPC/Oconee Nuclear Station Core Operating Limits Report (COLR)

2.3 UFSAR Chapter 16 Selected Licensee Commitments (SLC)

3. Time Required

3.1 90 minutes per shift

4. Prerequisite Tests

None

5. Test Equipment

None

6. Limits And Precautions

6.1 This procedure controls activities that have the potential to affect reactivity. Major changes to this procedure shall be reviewed by a Qualified Reviewer to determine if a cross-disciplinary review for Reactivity Management concerns is needed as required by NSD 304 (Reactivity Management).

6.2 Failure to meet the required conditions may be a violation of TS or SLC. If so, it must be reported immediately to OPS Duty Person, Superintendent of Operations, or Station Manager.

6.3 OP/0/A/1103/020 (Loss Of Computer) should be referred to upon a loss of computer or loss of a computer point or function needed for a surveillance.

6.4 When changing Modes, <u>ALL</u> TS, SLC, and Surveillance Requirements (SR) prior to changing Modes of operation shall be performed. <u>ALL</u> TS, SLC, and Surveillance Requirements (SR) (Semi-Daily, Daily, Weekly, and Monthly) must be initialed prior to changing modes. When the surveillance is completed, initial in the block for the shift you are working, regardless of the time of day, week or month, even if there is not a (N), (D), 1st day of month, etc., indicated in the block.

7. Required Unit Status

7.1 Surveillance of instrumentation per Enclosure "Mode 1 & 2" required:

Prior to entering Mode 2 from Mode 3

AND

During operation in either Mode 1 or Mode 2.

7.2 Surveillance of instrumentation per Enclosure "Mode 3" required:

Prior to entering Mode 3 from Mode 4

AND

During operation in Mode 3.

7.3 Surveillance of instrumentation per Enclosure "Mode 4" required:

Prior to entering Mode 4 from Mode 5

AND

During operation in Mode 4.

7.4 Surveillance of instrumentation per Enclosure "Mode 5" required:

Prior to entering Mode 5 from Mode 6

AND

During operation in Mode 5.

7.5 Surveillance of instrumentation per Enclosure "Mode 6" required:

Prior to entering Mode 6 from No Mode

AND

During operation in Mode 6.

7.6 Surveillance of instrumentation per Enclosure "No Mode" required:

During operation in No Mode.

7.7 Surveillance per Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)" is required when 1LT-5 < 50" and core contains <u>any</u> irradiated fuel.

7.7.1 Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)" is performed in parallel with Enclosure "Mode 5" or "Mode 6".

8. Prerequisite System Conditions

None

9. Test Method

9.1 Component checks will be made according to information given in the following enclosures:

Enclosure "Modes 1 & 2"

Enclosure "Mode 3"

Enclosure "Mode 4"

Enclosure "Mode 5"

Enclosure "Mode 6"

Enclosure "No Mode"

Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)".

10. Data Required

10.1 Data requirements specified in Enclosure "Modes 1 & 2", Enclosure "Mode 3", Enclosure "Mode 4", Enclosure "Mode 5", Enclosure "Mode 6", Enclosure "No Mode", or Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)".

11. Acceptance Criteria

11.1 Systems or components meet TS or SLC requirements applicable to surveillance step.

11.2 Any discrepancy noted during performance of this test shall show corrective action taken.

12. Procedure

12.1 As required, perform component checks according to schedule specified in the following enclosures:

- Enclosure "Modes 1 & 2"
- Enclosure "Mode 3"
- Enclosure "Mode 4"
- Enclosure "Mode 5"
- Enclosure "Mode 6"
- Enclosure "No Mode"
- Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)"

12.1.1 For surveillances required per TS and/or SLC: (D) indicates between 0730-1030 hours; (N) indicates between 1930-2230 hours.

- Example: "N/Tuesday" means signoff shall be completed sometime between Tuesday at 1930 hours and Tuesday at 2230 hours.

12.1.2 For surveillances <u>NOT</u> required per TS and/or SLC: (D) indicates between 0700-1900 hours; (N) indicates between 1900-0700 hours.

- Example: "N/Tuesday" means signoff shall be completed sometime between Tuesday at 1900 hours and Wednesday at 0700 hours.

12.1.3 Instrument operation may be checked either by reading appropriate recorder, gauge, etc., or by selecting computer point ID, where applicable.

12.2 <u>IF</u> any component supplying input to RPS or ES channels fails to meet its Required Condition (i.e., is "out of tolerance"), initiate the following action:

12.2.1 ES Instrument

A. Check other analog channels to see if any other channel is tripped.

NOTE:	For determining appropriate TS condition a tripped channel is considered inoperable.

B. <u>IF</u> no other analog channel is tripped, trip affected analog channel by placing instrument channel for affected parameter (RC pressure or RB pressure) in "TEST-OPERATE". Affected parameter(s) should be left in "TEST-OPERATE" until channel input(s) is repaired.

C. IF any other analog channel is tripped, do NOT trip affected channel. Initiate immediate action to have instrument repaired. Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits).

D. Immediate shutdown may be required.
- Refer to TS 3.3.5.
- Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits).

12.2.2 RPS Instrument

A. IF no other RPS channel is in MANUAL BYPASS or no other RPS channel contains a DUMMY BISTABLE, place affected RPS channel in MANUAL BYPASS. Initiate action to have instrument channel repaired.

NOTE:	For determining appropriate TS condition a tripped channel is considered inoperable.

B. IF another RPS channel is in MANUAL BYPASS or contains a DUMMY BISTABLE, trip affected RPS channel by placing any one of its instrument channels in "TEST-OPERATE" (for STAR Modules select "TEST"). Affected parameter(s) should be left in "TEST-OPERATE" (or "TEST") until channel input(s) is repaired. Initiate immediate action to have instrument channel repaired.

C. IF affected RPS channel is already in MANUAL BYPASS, do NOT trip affected RPS channel. Initiate action to have instrument channel repaired.

D. IF affected RPS channel contains a DUMMY BISTABLE and no other RPS channel is in MANUAL BYPASS, place affected RPS channel in MANUAL BYPASS. TS allows any one RPS channel to contain more than one DUMMY BISTABLE.

E. IF another RPS channel is tripped, do NOT trip affected RPS channel. Initiate immediate action to have instrument channel repaired. Tripping affected RPS channel will cause a reactor trip.

F. Immediate shutdown may be required.
- Refer to TS 3.3.1
- Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits)

12.2.3 Priority of Power Indications to Use for Surveillance.
(A = highest priority, G = lowest priority)

A. OAC Calculated Thermal Power Best.

- O1P2037 (Core Thermal Power Best (60 min avg)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0588 (Core Thermal Power Best (10 min. avg.)) - Transient in last 60 minutes or O1P2037 unavailable.
- O1P0889 (Core Thermal Power Best (snapshot)) - Transient in progress or O1P2037 and O1P0588 unavailable.

B. OAC Calculated Thermal Power Secondary (if above ≈ 25% power).

- O1P0587 (Core Thermal Power Secondary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0888 (Core Thermal Power Secondary (snapshot)) - Transient within the last 60 minutes or O1P0587 unavailable.

C. OAC Calculated Thermal Power Primary (if below ≈ 25% power).

- O1P0576 (Core Thermal Power Primary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0887 (Core Thermal Power Primary (snapshot)) - Transient within the last 60 minutes or O1P0576 unavailable.

D. OAC Calculated Thermal Power ΔT.

- O1P0575 (Core Ther Pwr From Delta Temp (10 min. avg.)) - Steady State Ops. (i.e., no transient in last 10 minutes)
- O1P0326 (Core Thermal Power From Delta T (snapshot)) - Transient within the last 10 minutes or O1P0575 unavailable.

E. Alternate method for determining (%) Reactor Power

$$\frac{\text{NI-5} + \text{NI-6} + \text{NI-7} + \text{NI-8}}{4} = \%\ \text{Rx Power (Avg)}$$

F. Hand-Calculated Thermal Power ΔT using Enclosure "Loop ΔT Vs Reactor Power".

G. Thermal Power from Nuclear Engineering using PT/0/A/0205/002 (Thermal Power Calculation).

12.3 Reactor Power Axial Imbalance and Quadrant Power Tilt

12.3.1 Axial Imbalance shall **NOT** exceed appropriate limit curve in COLR..

A. **IF** axial imbalance limit is exceeded, take immediate corrective action to achieve an acceptable imbalance.

B. **IF** an acceptable imbalance is **NOT** achieved within 2 hours, reactor power shall be reduced until imbalance limits are met. Refer to TS 3.2.2.

12.3.2 Quadrant Power Tilt (QPT) shall **NOT** exceed appropriate positive (+) limit in COLR.

A. **IF** QPT limit is exceeded, take immediate corrective action to achieve an acceptable QPT. Refer to TS 3.2.3. .

B. Alternate method for determining QPT:

$$QPT = 100\left[\frac{\text{power in any quadrant}}{\text{Avg. power of all quadrants}} - 1\right]$$

12.3.3 Order of preference of measurement systems to determine axial imbalance and quadrant power tilt is as follows:

A. Incore Detectors (Computer Reactor Calculation Package).

B. Outcore Detectors (Power Range Outcore Detectors).

C. Backup Incore Detectors. Refer to PT/0/A/1103/019 (Backup Incore Detector System).

12.3.4 **IF** at least one power range outcore detector is **NOT** operable in each quadrant, outcore detectors shall **NOT** be used to measure axial imbalance or quadrant power tilt.

12.3.5 **IF** Outcore Detectors (Power Range Outcore Detectors) are needed for tilt calculations, contact Rx Engineering group to perform PT/0/A/1103/018 (Excore Tilt Calculations).

12.3.6 IF Outcore Detectors (Power Range Outcore Detectors) are needed for imbalance calculations, refer to the following alternate method for determining (%) Reactor Power Axial Imbalance:

$$\frac{\text{NI-5*} + \text{NI-6*} + \text{NI-7*} + \text{NI-8*}}{4} = \text{\% Imbalance (Avg.)}$$

* Use Imbalance CR gauges reading for each NI.

12.3.7 IF Reactor Calculations package is **NOT** running, verify minimum incore detector operability requirements are met. Refer to PT/0/A/1103/019 (Backup Incore Detector System).

NOTE:	"Steady Conditions" defined as: Operating at a constant power level with no rod motion due to xenon and no plans to change power level in next 24 hours.

12.4 WHEN operating at a steady condition above 40% FP:

12.4.1 Control Rods should be positioned at or above dashed vertical lines designating Steady State Operating Bands in COLR.

A. Maneuvering restrictions on Control Rod and APSR movement in OP/1/A/1102/004 (Operation At Power) have priority over 24 hour time limit to resume operation in Steady State Operating Bands.

B. IF Control Rod position limits are exceeded, (i.e., operating in restricted region), corrective action shall be taken immediately to achieve an acceptable control rod position. TS 3.2.1 requires an acceptable control rod position be attained within 2 hours.

12.4.2 APSRs should be positioned as required per Enclosure "Required Group 8 Position" of PT/1/A/1103/015 (Reactivity Balance Procedure).

12.4.3 IF plant operating conditions or imbalance control requirements prevent steady operation within Control Rod Steady State Operating Bands, contact Systems Engineering/Reactor Group.

12.5 SASS (Smart Automatic Signal Selector) Auto Operation

12.5.1 SASS for Pzr level looks at Pzr level 1, 2, or 3. If level 1 or 2 fails, SASS will AUTO swap to Pzr level 3.

12.5.2 IF "AUTO" light is off, "MISMATCH" light is on, and "TRIP 'A'" or "TRIP 'B'" light is on, a SASS trip has occurred.

A. Controlling signal will be signal which does **NOT** have a "TRIP" light illuminated.

NOTE:	Failure to swap switch to valid signal could result in failed signal feeding through if SASS is reset before signal is repaired.

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbuttons for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.5.3 IF "AUTO" light is off and "MISMATCH" light is on, a mismatch has occurred.:

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.5.4 Initiate a Work Request to repair faulty signal.

12.5.5 Following repair of faulty signal, reset SASS by pushing "RESET" button. The following should occur:

A. SASS should swap to "AUTO". "AUTO" light should be illuminated, "TRIP 'A' or "TRIP 'B'" light and "MISMATCH" light should be off.

B. Controlling signal should remain unchanged.

12.6 SASS (Smart Automatic Signal Selector) Manual Operation

12.6.1 IF "MISMATCH" light is on and "TRIP 'A'" or "TRIP 'B'" light is on, a SASS trip has occurred.

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.6.2 IF "MISMATCH" light is on, a mismatch has occurred.:

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.6.3 Initiate a Work Request to repair faulty signal.

12.6.4 Following repair of faulty signal, reset SASS by pushing "RESET" button. The following should occur:

A. SASS should swap to "AUTO". "AUTO" light should be illuminated, "TRIP 'A'" or "TRIP 'B'" light and "MISMATCH" light should be off.

B. Controlling signal should remain unchanged.

12.7 AMSAC/DSS

12.7.1 Refer to the following indications to determine normal status of AMSAC/DSS:

- AMSAC CH 1 and CH 2 **NOT** actuated

  (O1D2928, O1D2929, 1SA-8 D-5/D-8)

- DSS CH 1 and CH 2 **NOT** actuated

  (O1D2930, O1D2931, 1SA-8 C-9/C-10)

- AMSAC/DSS CH 1 and CH 2 **NOT** bypassed

  (O1D2932, O1D2933, Indicating Lights on 1UB1)

- AMSAC/DSS Enabled

  (Indicating Light on 1UB1)

- AMSAC/DSS CH 1 **AND** CH 2 UPS Normal

  (O1D2934, O1D2935)

- "Sy Max" Programmable Controllers

CH 1 AMSAC/DSS	CH 2 AMSAC/DSS
RUN Light (ON)	RUN Light (ON)
HALT Light (OFF)	HALT Light (OFF)

12.7.2 AMSAC/DSS UPS (Uninterruptable Power Supply) has been upgraded with new firmware.

A. UPS will generate an alarm if noise is encountered on its power supply. However, it will automatically re-assess input power supply quality and, if transient has passed, it will reset and clear its alarm.

B. UPS will still generate UPS Trouble alarm. However, it will be more likely to clear automatically without operator intervention.

C. **IF** UPS Trouble alarm does **NOT** automatically reset, issue a Work Request.

12.7.3 IF all of the following conditions are met, AMSAC/DSS may be considered operable:

A. Surveillance requirements of SLC 16.7.2 (Anticipated Transients Without Scram) are satisfied.

B. AMSAC/DSS CH 1 and AMSAC/DSS CH 2 are enabled.

C. AMSAC/DSS CH 1 AND AMSAC/DSS CH 2 are capable of generating intended EFDW start signals, control rod drop signals, turbine trip signal, and TBV setpoint shift signal.

- To satisfy these criteria, all AMSAC/DSS circuitry (including input pressure switches/pressure transmitters, electrical isolation devices, logic circuits, programmable controllers, and uninterruptible power supplies) shall be functional and properly calibrated.

D. "Sy Max" Programmable Controllers "RUN" Lights (ON) and "HALT" Lights (OFF) for AMSAC/DSS CH 1 and AMSAC/DSS CH 2.

12.7.4 Inability of EFDW pumps, turbine trip circuit, or control rods to respond to an AMSAC/DSS signal does NOT constitute inoperability of AMSAC/DSS system. These malfunctions are governed by applicable TS.

- TS 3.7.5 and TS 3.3.14 for inoperable Emergency Feedwater Pumps or existing Initiation Circuitry.
- TS 3.3.15 for inoperable Turbine Stop Valve closure circuitry.
- TS 3.1.4 for inoperable control rod(s).

12.7.5 IF one or both channels of AMSAC/DSS are inoperable AND reactor is critical, refer to SLC 16.7.2. Notify Compliance of inoperabilities extending beyond seven days.

12.7.6 IF any AMSAC/DSS channel is inoperable or generates an invalid trip signal, bypass both AMSAC/DSS channels from control panel in AHU Room located on 6th floor above Units 1 & 2 CR.

A. IF reactor is critical, declare AMSAC/DSS system inoperable AND refer to SLC 16.7.2. Initiate a Work Request to repair affected channel.

12.7.7 WHEN AMSAC/DSS channel has been repaired, return AMSAC/DSS channels to service per Enclosure "Return To Service Of AMSAC/DSS".

12.8 MSLB

12.8.1 IF one or both trains of MSLB do NOT meet Surveillance Requirements:

A. Refer to TS 3.3.11, 3.3.12 and/or 3.3.13 for appropriate TS Condition for inoperability that is indicated.

B. IF entry into condition A of TS 3.3.11 indicated, Immediately Notify I&E to perform IP/0/A/0270/003 (Main Steam Line Break (MSLB) Loss Of An Analog Channel Trip/Restoration) to trip affected channel and prevent entry into condition B of TS 3.3.11.

C. Initiate a Priority Work Request.

D. Initiate a PIP and contact Accountable Systems Engineer.

12.9 Dixon Indicators

12.9.1 Dixons listed on Enclosure "Dixon Meter Information" are on an enhanced surveillance interval.

12.9.2 IF any Dixon listed on Enclosure "Dixon Meter Information" are found to be blinking with a reading of zero, no action is required unless a failure is suspected.

12.9.3 Dixons NOT listed on Enclosure "Dixon Meter Information" are on a surveillance interval. These Dixons have alternate methods of verifying input signal is valid.

12.10 WHEN a computer point needed for a surveillance is NOT available, refer to OP/0/A/1103/020 (Loss Of Computer).

13. Enclosures

13.1 Mode 1 & 2

13.2 Mode 3

13.3 Mode 4

13.4 Mode 5

13.5 Mode 6

13.6 No Mode

13.7 Minimum Temperature For Criticality Surveillance Sheet

13.8 RCS Pressure, Temperature, Heatup And Cooldown Rates Surveillance Sheet

13.9 Pzr Level For LTOP Surveillance Sheet

13.10 RCP Power Supply Verification

13.11 LPI Pump Power Supply Verification

13.12 Loop ΔT Vs Reactor Power

13.13 Gross Load Vs Reactor Power

13.14 Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)

13.15 Return To Service Of AMSAC/DSS

13.16 ICCM Subcooling Monitor Check

13.17 Surveillance Evaluation

13.18 Dixon Meter Information

13.19 Hot Lake Water Surveillance

NOTE:	If Reactor calculations package is NOT running on computer, section 12.3 contains guidance.
NOTE:	If Reactor calculations package is running properly on computer, NAS Loop Counter should differ by ≈ 24 every two hours.
NOTE:	Step 1.1 contains the priority of indications to use for (%) Reactor Power.

TIME	% Reactor Power	NAS Loop Counter O1P5504	INITIALS					
			Step 1	Step 2	Step 3	Step 4	Step 5	RCP Seal Leakoff Flow
2000						N/A		
2200						N/A		
0000								
0200						N/A		
0400						N/A		
0600						N/A		
0800						N/A		
1000						N/A		
1200						N/A		
1400						N/A		
1600						N/A		
1800						N/A		

1. <u>IF</u> Thermal Power Best indicates "Good", verify Core Thermal Power Indication (every 2 hours when Rx critical.)

1.1 Priority of Power Indications to Use for Surveillance (A = highest priority, G = lowest priority):

A. OAC Calculated Thermal Power Best.

- O1P2037 (Core Thermal Power Best (60 min avg)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0588 (Core Thermal Power Best (10 min. avg.)) - Transient in last 60 minutes or O1P2037 unavailable.
- O1P0889 (Core Thermal Power Best (snapshot)) - Transient in progress or O1P2037 and O1P0588 unavailable.

B. OAC Calculated Thermal Power Secondary if above ≈ 25% power.

- O1P0587 (Core Thermal Power Secondary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0888 (Core Thermal Power Secondary (snapshot)) - Transient within last 60 minutes or O1P0587 unavailable.

C. OAC Calculated Thermal Power Primary if below ≈ 25% power.

- O1P0576 (Core Thermal Power Primary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0887 (Core Thermal Power Primary (snapshot)) - Transient within last 60 minutes, or O1P0576 unavailable.

D. OAC Calculated Thermal Power Delta T.

- O1P0575 (Core Ther Pwr From Delta Temp (10 min. avg.)) - Steady State Ops. (i.e., no transient in last 10 minutes)
- O1P0326 (Core Thermal Power From Delta T (snapshot)) - Transient within last 10 minutes or O1P0575 unavailable.

E. Alternate method for determining (%) Reactor Power

$$\frac{NI\text{-}5 + NI\text{-}6 + NI\text{-}7 + NI\text{-}8}{4} = \%\ \text{Rx Power (Avg)}$$

F. Hand-Calculated Thermal Power Δ T using Enclosure "Loop ΔT Vs Reactor Power".

G. Thermal Power from Rx Engineering using PT/0/A/0205/002 (Thermal Power Calculation).

1.2 IF Thermal Power Best indicates "Bad", enter "NIS" (NOT In Service) and initials in appropriate block (s). Refer to OP/0/A/1103/020 (Loss Of Computer) for other actions.

NOTE:	If either step 1.3 or 1.4 is NOT satisfied, Duty Rx Engineer should perform verification of computer calculated TPB indication prior to calibrating NIs.

1.3 Verify Thermal Power Best (TPB) within ± 2.0% Rx Power of percent power from ΔT:

1.3.1 Refer to step 1.1 for priority of indications to determine percent power from ΔT,

OR

1.3.2 Average two RC Loop ΔTs from RC Loop ΔT gauge and using Enclosure "Loop ΔT Vs Reactor Power" determine percent power from ΔT.

1.4 Verify current Gross Load does NOT exceed value given by Enclosure "Gross Load Vs Reactor Power" for current Rx power level. Obtain Gross Load from O1P0963 OR Watt/Var meter if O1P0963 is NOT available.

2. Review Shift Turnover Sheet every 2 Hours.

2.1 Review Enclosure "Shift Turnover Sheet" to verify all turnover items updated and all required testing/surveillance items resulting from a degraded Mode per TS performed.

3. IF > 90% RTP and Steady State, AND fouling coefficient is less than 1.0, verify every 2 Hours O1P0576 (Core Thermal Power Primary (60 min avg)) does NOT exceed O1P0587 (Core Thermal Power Secondary (60 min avg)) by more than 0.2% RTP (i.e., O1P0576 < O1P0587 + 0.2).

   3.1 IF fouling coefficient is less than 1.0, AND IF O1P0576 (Core Thermal Power Primary (60 min avg)) exceeds O1P0587 (Core Thermal Power Secondary (60 min avg)) by more than 0.2% RTP, contact Duty Rx Engineer.

4. Obtain and call in Daily Dispatcher Readings.

CAUTION:	If LDST ≥ 130 °F, HPI System is inoperable.

5. Verify LDST temperature < 120 °F. (CP O1A1240)

6. Initial when verification of steps 1, 2, 3, 4, and 5 completed.

7. Procedure for Periodic Checks:

- Review all in-progress Surveillance Evaluation enclosures in Tech Spec R&R Book:
  - (N) Verify all corrective/compensatory actions still valid. (e.g., WRs, WOs, PIPs open; procedure change(s) NOT yet implemented)
- No surveillance or completion time exceeded.
- Update in-progress Surveillance Evaluations by one-lining, initialing, and dating as required (change WR numbers to WO numbers, update resolution times).
  - RO and SRO sign updated line per step 9.1 of Enclosure "Surveillance Evaluation".
- Close out Surveillance Evaluations no longer applicable (e.g., corrective actions completed, TS/SLC no longer applicable).
- Attach completed (closed out) Surveillance Evaluations to this procedure.

- Perform periodic checks as specified.
  - <u>IF</u> check can be performed as specified and is satisfied, initial appropriate block.
  - <u>IF</u> check <u>CANNOT</u> be performed as written or is <u>NOT</u> satisfied, perform the following:
    - <u>IF</u> Surveillance Evaluation is <u>NOT</u> outstanding for check, perform Enclosure "Surveillance Evaluation".
    - Record Surveillance Evaluation in effect in appropriate block for any periodic checks with Surveillance Evaluations issued.
    - Attach a <u>copy</u> of Surveillance Evaluation issued this shift to this procedure.
    - List Surveillance Evaluations in effect in Remarks section of Procedure Process Record.
  - Place Surveillance Evaluations initiated this shift in Tech Spec R&R Book.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.7.6.1 12 Hours SR 16.10.1.1 6 Hours	CST, UST, and HW	______ 1930-2100 ______ 0130-0300	______ 0730-0900 ______ 1330-1500	O1E2250 O1A0152	Verify Combined Inventory for EFW in Acceptable Operation Region of Enclosure "Combined Inventory for EFW" of OP/0/A/1108/001 (Curves And General Information).
SR 3.7.6.1 12 Hours	CST, UST, and HW	______ 1930-2100 ______ 0130-0300	______ 0730-0900 ______ 1330-1500	O1E2250 O1E2295	Verify UST level > 6 ft. (done on 6 hr frequency)
SR 3.4.1.3 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1970	Verify RCS total flow within limits in COLR. Mode 1 only, Steady State Operation
SR 3.2.2.1 12 Hours	Axial Power Imbalance Operating Limits	(N)	(D)	O1P0877	Verify Power imbalance within operational alarm limits in COLR when > 40% RTP. **IF** Reactor calculations package is **NOT** running on computer, refer to Section 12.3. **IF** % Rx Power Imbalance changes > 2% during Steady State Operations, contact Rx Engineering **IF** NI calibration is required under these conditions, contact Rx Engineering.
SR 3.2.3.1 7 Days	QPT	(N)	(D)	O1P0737 O1P0738 O1P0739 O1P0740	Verify QPT within limits in COLR when > 20% RTP. (done on 12 hr frequency) **IF** Reactor calculations package is **NOT** running on computer, refer to Section 12.3.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.3.1.1 12 Hours	RPS Instrumentation NI Power Range NI-5, 6, 7, 8, 9	(N)	(D)	O1A1544 O1A1545 O1A1546 O1A1547 O1A1548	Verify computer readouts agree within 2%. (4% in RPS Cab) **NOTE:** <u>IF</u> the channels are off scale, the channel check will only verify that they are off scale in the same direction. (TS Bases SR 3.3.1.1)
SR 3.3.1.2 24 Hours	RPS Instrumentation Heat Balance Check Power Range Amplifiers	(N)	(D)	O1P0889	Verify TPB does <u>NOT</u> exceed NI-5, 6, 7, 8 or 9 by more than 2% power. (done on 12 hr frequency) Calibrate NIs when TPB ≥ 2% above any two of the power range NIs. Do <u>NOT</u> exceed ≥ 4% in non-conservative direction. <u>IF</u> TPB indicates "Bad", contact Duty Rx Engineer to calculate core thermal power per PT/0/A/0205/002 (Thermal Power Calculation). Mode 1 only, <u>NOT</u> required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP.
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation RB Pressure Narrow Range	(N)	(D)	O1A1566 O1A1286 O1A1287	Verify computer readouts agree within 0.6 psi (2 psi in ES Cab). <u>IF</u> readouts differ by > 0.4 psi, issue a Priority "E" Work Request.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.6.4.1 12 Hours	Containment Pressure NR RB Pressure	(N)	(D)	O1A1566 O1A1286 O1A1287	Verify RB pressure ≥ -2.45 psig but ≤ 1.2 psig. **IF** > + 0.6 psig, depressurize RB prior to > + 0.8 psig per OP/1/A/1102/014 (RB Purge). **IF** ≤ -0.5 psig, notify MCE for operability evaluation. {PIP 98-3976 & OSC-4476}
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Pressure Narrow Range	(N)	(D)	O1A1688 O1A1689 O1A1690 O1A1691	Verify computer readouts agree within 26 psi (48 psi in RPS Cab).
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Temperature T_H	(N)	(D)	O1A1692 O1A1693 O1A1694 O1A1695	Verify computer readouts agree within 3°F (5°F in RPS Cab). **IF** any of CR RCS temperature selectors are changed, notify Rx Engineering to evaluate and update Enclosure "Loop ΔT Vs. Reactor Power" for new selected inputs.
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Flow	(N)	(D)	O1A1549 O1A0877 O1A1420 O1A1712	Verify total flow agrees within 4800 klbm/hr **AND** no computer alarms for high flow present.
SR 3.4.1.2 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1888 O1P1889	Verify RCS loop average temperature: < 580°F on OAC < 579.5°F Dixon indication (OAC unavailable) Mode 1 only, Steady State Operation When 3 RCPs operating, limits applied to loop with lowest loop average temperature for the condition where there is a 0°F ΔTc Setpoint.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.7.11.1 12 Hours	Pressurizer Temperature	(N)	(D)	O1E2298 O1E2299	Verify each temperature channel agrees within 12°F on computer or indicator.
SR 3.4.9.1 12 Hours	Pressurizer Level (Corrected)	(N)	(D)	O1E2275 O1E2276 O1E2277	Verify each level channel agrees within 9" between computer and recorder **OR** between indicator and recorder. Verify Pzr level ≤ 260".
SR 16.7.11.2 24 Hours TS 3.5.2	LDST Level	(N)	(D)	O1A1042 O1A1043	Verify redundant level channels on computer and gauge agrees within 2". (done on 12 hr frequency)
SR 3.3.11.1 12 Hours	MSLB Detection and MFW Isolation Instrumentation	(N)	(D)		Verify redundant outlet pressure channels for 1A and 1B SGs agree within 30 psig: "A" SG — "B" SG O1E2281 — O1E2282 O1E2283 — O1E2284 O1E2111 — O1E2112 **IF** required conditions **NOT** met, refer to step D (MSLB).
SR 3.7.8.3 24 Hours	ECCW		(D)	O1P0761	Verify average CCW inlet temperature ≤ 88°F. **IF** > 88°F, notify MSE for operability evaluation. {PIP 98-3976 & OSC-4476}
SR 3.4.1.1 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1609 O1P1620	Verify RCS loop pressure within limits in COLR. Mode 1 only, Steady State Operation. When 3 RCPs operating, limits applied to loop with highest pressure.
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation RC Pressure Wide Range	(N)	(D)	O1A1416 O1A1417 O1A1418	Verify computer readouts agree within 75 psi (100 psi in ES Cab).
SR 3.1.6.1 12 Hours	APSR Alignment Limits	(N)	(D)	GD60 REG	Verify position of each APSR within 6.5% of group average.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.2.1.1 12 Hours	Regulating Rod Position Limits	(N)	(D)	GD60 REG	Verify regulating rod groups within sequence and overlap limits in COLR.
SR 3.2.1.2 12 Hours	Regulating Rod Position Limits	(N)	(D)	GD60 REG	Verify regulating rod groups within position limits on curve in COLR.
SR 3.1.4.1 12 Hours	Control Rod Group Alignment Limits	(N)	(D)	GD60 REG GD60 SAFETY	Verify all Control Rods in each Group agree within ± 3.5% of group average. **IF** a Control Rod is > ± 3.5% of its Group average, refer to OP/0/A/1105/009 (Control Rod Drive System).
SR 3.1.5.1 12 Hours	Safety Rod Position Limits	(N)	(D)	GD60 SAFETY PI Panel	Verify each safety rod fully withdrawn.
SR 16.7.11.3 31 Days SLC 16.5.13	CBAST Temperature	(N)		O1A0784	Verify computer indication > 125°F. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)
SR 16.5.13.1 7 Days	CBAST	(N)		O1A0797	Verify equivalency of 1100 ft^3 of 11,000 ppm boron per OP/0/A/1108/001 (Curves And General Information). (done on 24 hr frequency)
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation ES Channels 7 & 8 RB 10 psig	(N)	(D)		Verify no trips present. Verify status annunciators operable (lamp test).
SR 3.3.1.1 12 Hours SLC 16.7.9	RPS Instrumentation RP RCP/Flux Trip	(N)	(D)		Verify no Dummy Bistable installed. Verify no trips present. Verify status annunciators operable (lamp test).
SR 3.3.1.1 12 Hours	RPS Instrumentation RB High Press Trip	(N)	(D)		Verify no Dummy Bistable installed. Verify no trips present. Verify status annunciators operable (lamp test).

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.1.7.1 12 Hours	Position Indicator Channels PI Panel	(N)	(D)		Verify all Relative Rod Position indications agree within 5% of Absolute Rod Position indications. **IF NOT**, notify Duty Rx Engineer for evaluation of core parameters and recommended actions.
SR 3.3.9.1 12 Hours	Source Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 SR agree within 1 decade. Mode 2 only
SR 3.3.10.1 12 Hours TS 3.3.8	Wide Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 agree within 3 LED Segments. Mode 2 only
SR 3.4.4.1 12 Hours	RCS Loops	(N)	(D)		Verify required RCPs (3 or 4) in operation with RCS flow indicated.
SR 3.5.4.2 7 Days TS 3.3.8 TS 3.5.4	BWST	(N)			Verify BWST level on ICCM Plasma Displays ≥ 47.0 ft. (done on 24 hr frequency) • A 1LT-BWST 1 • B 1LT-BWST 2 **AND** BWST level ≥ 46.0 ft. on both analog gauges on 1UB2 • BWST Level A • BWST Level B **IF** required conditions **NOT** met, BWST is inoperable.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.5.4.1 24 Hours TS 3.5.4	BWST Temperature	(N)			Verify BWST between ≥ 50°F and ≤ 92.5°F as read on Bailey Indicator. **IF** > 92.5°F, notify MSE for operability evaluation of RBS System. {PIP 98-3976 & OSC-4476} **IF** ≥ 102.5°F or < 50°F, BWST is inoperable.
SR 16.11.3.1 24 Hours	WG Decay Tk Disch Flow Recorder	(N)	(D)		Verify recorder (GWD CR033) indicates flow. Perform anytime during shift hours during GWD Tank releases.
SR 16.11.3.1 24 Hours SR 16.11.3.2 24 Hours	1RIA-35	(N)			Verify 1RIA-35 indicates > zero **AND** no low flow alarm present.
SR 16.11.3.12 24 Hours	1RIA-37		(D)		Perform source check on 1RIA-37.
SR 16.11.3.1 24 Hours	1RIA-38		(D)		Verify 1RIA-38 indicates > zero **AND** no fault alarm present.
SR 16.11.3.2 24 Hours	1RIA-40	(N)			Verify 1RIA-40 indicates > zero **AND** no low flow alarm present.
SR 16.11.3.2 24 Hours	1RIA-43, 44, 45	(N)			Verify the following: 1) 1RIA-43, 44, 45 indicate > zero. 2) Unit Vent Monitor has no low flow alarm. 3) Unit Vent Flow Recorder indicates on scale.
SR 3.4.15.1 12 Hours	RCS Leakage Detection Instrumentation	(N)	(D)		Verify 1RIA-47 indicates > zero **AND** no flow alarm present. **OR** Verify 1RIA-49 indicates > zero **AND** no flow alarm present.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.11.3.1 24 Hours SR 16.11.3.2 24 Hours	1&2RIA-54	(N)			Verify the following: 1) 1&2RIA-54 indicates > zero. 2) No low flow alarm present. 3) "NORMAL/BYPASS" switch in "Normal".
SR 3.5.1.1 12 Hours	CFTs	(N)	(D)		Verify 1CF-1 **AND** 1CF-2 fully open.
SR 16.7.10.1 12 Hours	CFT Instrumentation	(N)	(D)		Verify redundant level channels on each CFT agree within 0.3 ft.
SR 3.5.1.2 12 Hours	CFTs	(N)	(D)		Verify CFT levels between 12.56 ft and 13.44 ft.
SR 16.7.10.1 12 Hours	CFT Instrumentation	(N)	(D)		Verify redundant pressure channels on each CFT agree within 30 psi.
SR 3.5.1.3 12 Hours	CFTs	(N)	(D)		Verify CFT pressures between 575 psig and 625 psig.
SR 16.8.6.1 24 Hours	Lee/Central Alternate Power System		(D)		Verify status of LCTs by contacting Lee Steam Station CR. Operable (✓) ___ 4C ___ 5C ___ 6C **IF** two LCTs are **NOT** operable, refer to Maintenance Rule **AND** contact Switchyard Coordinator.
SR 3.7.8.1 12 Hours	ECCW System	(N)	(D)		Verify two Unit 1 ESV Pumps in operation. **IF** two Unit 1 ESV Pumps **NOT** in operation, Refer To TS 3.7.8 Bases for allowed Pump/Header combinations.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.7.8.2 24 Hours SR 16.9.7.1 12 Hours	ECCW	(N)	(D)		Verify Keowee lake level within limits per SLC 16.9.7. **NOTE:** Instrument error of 1.15 ft. must be added to the absolute lake levels found in SLC 16.9.7 if using a computer point to verify level. Absolute lake level can be determined at the Keowee Hydro Intake structure.
SR 3.4.13.1 72 Hours SLC 16.5.10	RCS Operational Leakage	(N)			<u>Evaluate</u> per PT/1/A/600/010 (Reactor Coolant Leakage) when at steady state for ≥ 12 hours. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)
SR 3.7.16.1 12 Hours SR 16.8.1.1 SR 16.8.1.2 SR 16.8.1.3 24 Hours	Room Temperatures Unit 1 Cable Rm. Unit 1 Equip. Rm. Unit 1&2 Control Rm.	(N) ________ ________ ________	(D) ________ ________ ________		Record and verify room temperatures within respective temperature limits: Unit 1 Cable Rm: ≤ 80°F Unit 1 Equip. Rm: ≤ 85°F Unit 1&2 Control Room: ≤ 80°F **<u>IF</u>** limit is exceeded, refer to OP/0/A/1104/019 (Control Room Ventilation System), notify Unit Coordinator **<u>AND</u>** refer to SLC 16.8.1 and TS 3.7.16.
SR 16.7.11.3 31 Days	BAMT Temperature	(N)			Have NLO verify normal readout at Chemical Addition Panel agrees with local readout within 5°F. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.5.4.3 7 Days	BWST	(N) Wednesday			Verify BWST concentration within limit in COLR. May be performed anytime during shift hours (1900-0700)
SR 16.7.11.3 31 Days	PORV and Safety Valve Flow Monitors	(N) Saturday			Verify power supply lights on **AND** verify flow monitor statalarm actuates from "TEST" switch. (done on 7 Day frequency) May be performed anytime during shift hours (1900-0700)
SR 3.5.1.5 31 Days	CFTs	(N) 1st Day of Month			Verify with NLO 1CF-1 **AND** 1CF-2 breakers open. May be performed anytime during shift hours (1900-0700)
SR 3.5.1.4 31 days	CFTs	(N) 1st Day of Month			Verify each CFT boron concentration within limit in COLR. May be performed anytime during shift hours (1900-0700)
SR 3.6.3.1 31 Days	1PR-1, 2, 3, 4, 5, 6	(N) 1st Day of Month			Verify with NLO 1PR-1 and 1PR-6 breakers open. Verify with I&E links open for 1PR-2, 1PR-3, 1PR-4, and 1PR-5. May be performed anytime during shift hours (1900-0700)

NOTE:	Remaining items may be performed anytime during shift hours.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	Pressurizer Level (Uncorrected)	(N)	(D)	O1E2301 O1E2303 O1E2305	Verify redundant level channels agree within 8" on computer.
TS 3.10.1	SFP Temperature	(N)	(D)	O1A0839	Verify SFP temperature ≤ 143°F. **<u>IF</u>** > 143°F, SSF RCMUP is inoperable. Contact Duty MSE Engineer.
TS 3.7.5	UST Temperature	(N)	(D)	O1A0122 O1A0123	Verify UST temperature ≤ 125°F. **<u>IF</u>** > 125°F, notify Unit Coordinator and refer to OP/1/A/1106/006 (Emergency FDW System) for EFDW operability.
SLC 16.7.2	AMSAC/DSS	(N)	(D)	O1D2928 THRU O1D2935	Verify no trips present **<u>AND</u>** status annunciators indicate operable channels. Verify 1SA-8 C-9/C-10/D-5/D-8 **<u>AND</u>** indicating lights on 1UB1. Refer to Section 12.7 for operability determinations.
SLC 16.7.3	SG "A" XSUR Level Redundant Level	(N)	(D)	O1A1213 O1E2052	Verify redundant levels agree within 3" when < 2% RTP. > 3" acceptable during momentary swings (< 30 sec).

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
SLC 16.7.3	SG "A" XSUR Level Minimum Level	(N)	(D)	O1A1213 O1E2052	Verify all XSUR OAC indications ≥ 23" when SG on level control. < 23" are acceptable during momentary swings (< 30 sec)
SLC 16.7.3	SG "B" XSUR Level Redundant Level	(N)	(D)	O1A1215 O1E2053	Verify redundant levels agree within 3" when Rx < 2% RTP. > 3" acceptable during momentary swings (< 30 sec).
SLC 16.7.3	SG "B" XSUR Level Minimum Level	(N)	(D)	O1A1215 O1E2053	Verify all XSUR OAC indications ≥ 23" when SG on level control. < 23" acceptable during momentary swings (< 30 sec).
	1A SG SU Levels	(N)	(D)	O1E2000 O1E2001	Verify redundant levels agree within 2" when Rx < 2% RTP. >2" acceptable during momentary swings (< 30 sec).
	1B SG SU Levels	(N)	(D)	O1E2005 O1E2006	Verify redundant levels agree within 2" when Rx < 2% RTP. >2" acceptable during momentary swings (< 30 sec).
SLC 16.7.5	1A SG OR Levels	(N)			Verify redundant operating range recorder level channels agree within 3%.
SLC 16.7.5	1B SG OR Levels	(N)			Verify redundant operating range recorder level channels agree within 3%.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	"A" SG Shell Temperatures	(N)		O1P1892 O1A0968 O1A0969 O1A0970 O1A0971 O1A0972	Verify O1P1892 agrees with manually calculated average of five "A" SG Shell Temperatures (O1A0968 – O1A0972) within $\pm$ 5°F. **IF** required conditions **NOT** met, issue a Work Request.
	"B" SG Shell Temperatures	(N)		O1P1893 O1A0973 O1A0974 O1A0975 O1A0976 O1A0977	Verify O1P1893 agrees with a manually calculated average of five "B" SG Shell Temperatures (O1A0973 – O1A0977) within $\pm$ 5°F. **IF** required conditions **NOT** met, issue a Work Request.
	Station Condenser ΔT		(D)	O1P1947 or O3P1947	**IF** CCW inlet temperature > 68°F, verify Station Condenser ΔT ≤ 22°F. **IF** Station Condenser ΔT is > 22°F, notify Unit Coordinator **OR** OPS Duty Person. **IF** O1P1947 and O3P1947 are OOS, perform the following: 1) Verify all Units with CCW flow have CP O*P1944 operable **OR** 2) Hourly Inlet/Outlet sheet performed per OP/0/A/1103/020 (Loss Of Computer).
	CCW	(N)	(D)	O1P0761	Verify average CCW inlet temperature ≤ 80°F. **IF** > 80°F, Perform Enclosure 13.19 "Hot Lake Water Surveillance". {OSC-2576}

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	Station CCW Discharge Temperature		(D)	O1P1945 or O3P1945	Verify Station 2 Hour Average CCW discharge temperature ≤ 100°F. **IF** Station 2 Hour Average CCW discharge temperature > 100°F, notify Unit Coordinator **OR** OPS Duty Person. **IF** O1P1945 and O3P1945 are OOS, perform the following: 1) Verify all Units with CCW flow have CP O*P1942 operable **OR** 2) Hourly Inlet/Outlet sheet performed per OP/0/A/1103/020 (Loss Of Computer).
TS 3.5.2	LDST Pressure	(N)	(D)	CR Gage O1A2191	Verify both LDST pressure/level relationships comply with OP/0/A/1108/001 (Curves And General Information).
	Room Temperatures Unit 1 Cable Rm. Unit 1 Equip. Rm. Unit 1&2 Control Rm.	(N) ________ ________ ________	(D) ________ ________ ________		Record and compare room temperatures to those taken on previous shift. A ≥ 3°F temperature increase observed from the previous shift may be an indication of a problem with the WC System. **IF** this is indicated **REFER TO** Enclosure "Control Room, Equipment Room, And Cable Room Temperature Troubleshooting Guide" of OP/0/A/1106/029 (Control Room, Equipment Room, And Cable Room Chillers).
	ES Channels 1 & 2 RC Press	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 3 & 4 RC Press	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 1 & 2 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	ES Channels 3 & 4 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 5 & 6 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP Low Press Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP RCP/Flux/Imb Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Temp Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP Press/Temp Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Press Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Flux Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
TS 3.10.1	RCS Boron Concentration	(N)	(D)		Verify RCS Boron Concentration greater than "Minimum RCS Boron Concentration to Maintain SSF Operability" curve of PT/1/A/1103/015 (Reactivity Balance). **IF** minimum concentration is **NOT** met, SSF RC MU Pump is inoperable. Contact Duty Rx Engineer.
	Control Rod Position	(N)	(D)		Verify limit lamps operable on Diamond and PI Panel.
TS 3.3.10 TS 3.3.8	Wide Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 agree within: 3 LED Segments when < 10% RTP **OR** 2 LED Segments when ≥ 10% RTP. Mode 1 only

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
TS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) __AND__ SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). __IF__ SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
TS 3.3.8	Subcooling Monitors Core	(N) Thursday			__IF__ > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F __AND__ (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. __IF__ (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	Subcooling Monitors Core	(N) Thursday			**IF** < 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' **AND** Train 'B') within -9 to +11°F. **IF** (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion
TS 3.3.8	ICC Level - Train 'A'		(D)		Verify "MALFUNCT 00" message on diagnostic page of ICCM Plasma Display. Verify "RVLIS/ICCM/RG 1.97 TRAIN A TROUBLE" annunciator (1SA-18/A-3) **NOT** in alarm. **IF** a "MALFUNCTION FF" message **OR** annunciator alarm is present, issue a Priority Work Request **AND** contact I&E to investigate problem to determine operability of ICC channel.
TS 3.3.8	ICC Core Cooling Train 'A'		(D)		Verify from Core Cooling core map ≥ 5 CETCs operable (do **NOT** indicate "FAIL").

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	ICC Level - Train 'B'		(D)		Verify "MALFUNCT 00" message on diagnostic page of ICCM Plasma Display. Verify "RVLIS/ICCM/RG 1.97 TRAIN B TROUBLE" annunciator (1SA-18/A-4) <u>NOT</u> in alarm. <u>**IF**</u> a "MALFUNCT FF" message <u>**OR**</u> annunciator alarm is present, issue a Priority Work Request <u>**AND**</u> contact I&E to investigate problem to determine operability of ICC channel.
TS 3.3.8	ICC Core Cooling Train 'B'		(D)		Verify from Core Cooling core map ≥ 5 CETCs operable (do <u>**NOT**</u> indicate "FAIL").
TS 3.3.5 TS 3.5.4	BWST Level Instrument ICCM Plasma Displays	(N)			Verify redundant indicators on ICCM Plasma Displays agree within 2 ft. <u>**IF**</u> required conditions <u>**NOT**</u> met, BWST is inoperable.
TS 3.5.4	BWST Level Instrument Analog Gauges on 1UB2	(N)			Verify redundant indicators on 1UB2 agree within 2 ft. <u>**IF**</u> required conditions <u>**NOT**</u> met, BWST is inoperable.
TS 3.3.14 TS 3.7.5	1A & 1B MD EFDW Pumps "OFF/AUTO/RUN" Lights	(N)	(D)		Verify lights energized. <u>**IF NOT**</u>, MD EFDW Pumps are inoperable and Auto Start capability is lost.
TS 3.3.8	EFDW Total Flow	(N)			Verify Train 'A' <u>**AND**</u> Train 'B' EFDW Hdr Flow to SG indicates < 20 gpm with no EFDWPs operating. (indicators fail high)

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8 TS 3.7.6	UST Level CR Gauges	(N)	(D)		Verify redundant indicators agree within 0.4 ft.
SLC 16.11.3	Sorrento Radiation Monitor Time Check	(N)	(D)		Verify current time on RIA CRT within ± 1 minute of current time on OAC CRT. **IF** > ± 1 minute, Contact IT to reset time.
TS 3.3.8	1RIA-57, 58	(N)			Verify 1RIA-57, 58 indicate between 5.0E-1 and 1.0E0 R/HR. Press "C/S" button. Verify no Area Monitor Fault alarm exists after check source is complete. Press "R/HR" button to return to normal.
TS 3.5.3 TS 3.7.7	LPI Cooler 'A' LPSW Flow Dixon Indicator	(N)	(D)	O1A2124	**IF** dixon indicator zero and blinking, verify computer point **NOT** reading a high negative value.
TS 3.5.3 TS 3.7.7	LPI Cooler 'B' LPSW Flow Dixon Indicator	(N)	(D)	O1A2125	**IF** dixon indicator zero and blinking, verify computer point **NOT** reading a high negative value.
TS 3.6.5	1A, 1B, 1C RBCU LPSW Flow (IN)	(N)	(D)		Verify each RBCU LPSW Flow (IN) ≥ 550 gpm. **IF** Auxiliary Cooling Coils **AND** 1B RBCU **BOTH** have flow established, verify ≥ 1100 gpm Inlet Flow to 1B RBCU. **IF** any RBCU LPSW Flow (IN) < required, enter LCO 3.0.3

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.5.3 TS 3.6.4 TS 3.6.5	RB Dome Temperature	(N)	(D)	O1A0043 Chart Recorder 1RBCCR0007	Verify highest RB Dome temperature ≤ 170°F. Verify lowest RB Dome temperature ≥ 90°F when 100 % RTP. **IF** either limit is exceeded, contact MSE for operability evaluation. (LPI and BS) **IF** > 175°F, contact CEN for operability evaluation. (Reactor Building)
TS 3.3.8	RB Post Accident Water Level Wide Range Indication	(N)		O1A1033 O1A1565	Verify Train 'A' **AND** Train 'B' Meters, Computer, and Recorder (1BS CR0085) agree within 0.5 ft.
TS 3.3.8	RB Post Accident Pressure Wide Range Indication	(N)		O1A1011 O1A1315	Verify Train 'A' **AND** Train 'B' Meters, Computer, and Recorder (1BS CR0085) agree within 6 psi.
TS 3.3.8 TS 3.4.15	RB Normal Sump	(N)	(D)		Verify Train 'A' **AND** Train 'B' Meters and Recorder (1LWDCR0095) agree within 1 ft.
TS 3.4.15	RB Normal Sump	(N)			Verify water level in RBNS on scale.
TS 3.3.8	RB Emerg Sump Narrow Range	(N)		O1A0050	Verify Train 'A' **AND** Train 'B' Meters, Computer and Recorder (1LWDCR0095) agree within 1 ft.
	RB Emergency Sump	(N)			Verify zero water level in RBES.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.10.1	RCP Seal Leakoff Flow	(N)	(D)		Verify RCP seal leakoff flow electronic display on **AND** display present. IF seal leakoff flow > 4 gpm on any RCP, increase seal leakoff flow surveillance to every 2 hours and initial on page 1. IF the SSF is **NOT** manned and seal leakoff flow > 4.7 gpm for 1A1, 1A2, 1B1 or 1B2 RCP, SSF RCMU Pump is inoperable. IF the SSF is manned and seal leakoff flow >6.0 gpm for 1A1 or 1B1 RCP, or > 4.7 gpm for 1A2 RCP, or > 5.5 gpm for 1B2 RCP, SSF RCMU Pump is inoperable.
	Loose Parts Monitor	(N)			Monitor all operable points on LPM. Test alarm circuitry per OP/1/A/1105/011 (Loose Parts Monitoring System).
	Event Recorders	(N)			Verify paper in all Events recorders.
	800 mHz Radio	Sunday (0100-0400) (N)			Test the backup radio communications with the System Operating Center (SOC) **AND** the Transmission Control Center (TCC). SOC code – 96 TCC code – 11 IF communications fail notify SPOC.
	Easterline Angus Charts	(N)			Stamp charts.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
SLC 16.8.5	125 VDC Ground Detection System Test	(N)	(D)		Perform Enclosure "125 VDC Ground Detection System Operation" of OP/1/A/1107/010 (Operation Of The Batteries And Battery Chargers). **IF** required conditions **CANNOT** be met, refer to SLC 16.8.5 for required actions
	SASS	(N)	(D)		Verify the following on SASS panels in ICS cabinet #8: 1) All "AUTO" lights on. 2) No "MISMATCH" lights on. 3) All "POWER" lights on.
SLC 16.9.6	Fire Alarm Cabinet	(N)			Verify "Power" **AND** "Run" LEDs are on **AND** no Trouble/Alarm lights present.
TS 3.10.1	SFP Level	(N)	(D)		**IF** all fuel in SFP subcritical ≥ 20 days, verify SFP level > - 2 ft. **IF** any fuel in SFP subcritical < 20 days, verify SFP level greater than Enclosure "Unit 1&2 Spent Fuel Pool Level Vs Temperature Curve (7-19 days))" of OP/0/A/1108/001 (Curves And General Information). **IF** limit exceeded, SSF RCMUP is inoperable.
	RCP Data Sheets	(N)			Complete RCP Data Sheets: • Sunday and Wednesday when at steady-state power. • Daily when changing Rx power **OR** RCS temperature.
SLC 16.11.3	RB Depressurization		(D)		**IF** RB depressurization is in progress, submit a RB Sample Request.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.11	MSLB Digital Channels	(N)	(D)		Verify with NLO all five power supply lamps lit. **IF** required conditions **NOT** met, refer to step D (MSLB).
ISFSI TS 4.1.1 ISFSI C of C 1.3.1 1.3.2	ISFSI Storage Facility		(D)		Verify notified by Security: 1) All Horizontal Storage Modules (HSM) ventilation screens (inlet and outlet) free of debris and no material accumulated between modules to block air flow. 2) All Roof Slab temperatures monitored which contain Spent Fuel < 260°F and < 80°F increase in 24 hrs. **IF** temperature limits are exceeded, issue a Work Request **AND** contact Rx Engineering.
	Unit 1 BWST	(N) Monday			Place in recirc per OP/1&2/A/1104/006 (SF Cooling System).
	LDST Level	(N) Saturday			Verify redundant level channels 1&2 CR Gage and Local Gage (1HPIPG0437) agree within 2".
	LDST Pressure	(N) Saturday			Verify redundant pressure channels 1&2 CR Gage and Local Gage (1HPIPG0438) agree within 1 psig.
	SSF Radio	(N)			Verify communications with CR via SSF radio. Use base station on Channel 2.

Duke Power Company Oconee Nuclear Station **Backup Incore Detector System** *** This procedure has the potential to affect Reactivity Management *** **Continuous Use**	Procedure No. PT/0/A/1103/019 Revision No. 4 Electronic Reference No.

Performed By ______________________________

Date ____________________

Backup Incore Detector System

1. Purpose

1.1 To verify the operable backup recorder points meet the minimum requirements for the incore instrumentation system upon loss of the incore system on the unit computer or loss of the unit computer.

1.2 To provide a method to calculate reactor power axial imbalance and quadrant power tilt using the backup incore detector system when the incore system is not available on the unit computer and one or more of the excore detectors are inoperable.

2. References

2.1 Improved Technical Specifications 3.2.2, Axial Power Imbalance
3.2.3, Quadrant Power Tilt

2.2 OP/0/A/1103/020, Loss of Computer

2.3 PT/1,2,3/A/0600/001, Periodic Instrument Surveillance

2.4 Unit Core Operating Limits Report (COLR)

2.5 NSD 304, Reactivity Management

2.6 Selected Licensee Commitment 16.7.8

3. Time Required

3.1 Verify Backup Incore Recorders operable - 10 minutes - 1 Operator or Reactor Engineer

3.2 Calculate Backup Tilt/Imbalance - 30 minutes - 2 Operators and/or Reactor Engineers

4. Prerequisite Tests

None

5. Test Equipment

Calculator

6. Limits and Precautions

6.1 This procedure has the potential to affect REACTIVITY MANAGEMENT, since the backup incore recorders are used to monitor reactivity.

6.2 If the incore system is not available on the unit computer and the backup recorder points are not operable per this procedure, then the reactor power shall be reduced below 80% of the power allowable for the existing reactor coolant pump combination within eight hours unless:

6.2.1 The incore system is restored on the unit computer.

or

6.2.2 The backup recorder points are restored to meet the minimum requirements for operability. (ref. SLC 16.7.8)

6.3 If the backup incore limits are exceeded then action must be taken per the applicable Technical Specifications as listed below:

6.3.1 Quadrant Power Tilt - ITS 3.2.3

6.3.2 Axial Power Imbalance - ITS 3.2.2

7. Required Plant Status

______7.1 Quadrant power tilt surveillances are required when the Unit is above 20% full power.

______7.2 Reactor power imbalance surveillances are required when the Unit is above 40% rated power.

8. Prerequisite System Conditions

Loss of incore system on the unit computer or loss of the unit computer

9. Test Method

9.1 The backup recorder points will be checked to identify which points are a) inoperable as indicated by off-scale readings or b) identified as inoperable or out of calibration during the last functional verification. The remaining operable points will be checked to verify the minimum number of detectors are operable to measure axial imbalance and quadrant power tilt as required.

9.2 Axial Imbalance and quadrant power tilt calculations may be performed using the operable backup recorder points.

10. Data Required

Incore Backup recorder point readings

11. Acceptance Criteria

11.1 The calculated axial imbalance is within the curve for the appropriate pump configuration shown in the current Core Operating Limits Report (COLR) on the Backup Incore Setpoint Column of the (Error-Adjusted) "Operational Power Imbalance Setpoints" Table.

11.2 The calculated quadrant power tilt is less than the value given in the current Core Operating Limits Report (COLR) on the Backup Incore row of the (Error-Adjusted) "Quadrant Power Tilt Setpoints" Table.

12. Procedure

NOTE:	If this procedure is being performed only to satisfy Section 1.1, then perform Section 12.1 and N/A Section 12.2.

_______12.1 Verification of Minimum Incore Detector Operability

______12.1.1 On Enclosure 13.1 and 13.2, place an "X" next to the backup recorder points which are inoperable as indicated by off-scale readings or notes attached to the recorders.

______12.1.2 Verify that all three required points on at least three detector strings are operable per instructions on Enclosure 13.1.

______12.1.3 Verify that all four required points on at least four sets (two sets in each axial core half) are operable per instructions on Enclosure 13.2.

______12.1.4 If either step 12.1.2 or 12.1.3 cannot be satisfied;

______12.1.4.1 Notify the Unit Supervisor.

______12.1.4.2 Take actions described in 6.2.

_______12.2 Calculation of Axial Imbalance and Quadrant Power Tilt

NOTE:	If this portion of the procedure is being performed to fulfill the requirements of PT/1,2,3/A/0600/01, Periodic Instrument Surveillance, then repeat the steps below every twelve hours as required and record the calculated values in that procedure.

______12.2.1 Verify the reactor has been at steady state conditions (±2% FP) for at least 30 minutes.

______12.2.2 Calculate axial imbalance per Enclosure 13.3 using operable recorder points identified on Enclosure 13.1.

______12.2.3 Calculate quadrant power tilt per Enclosure 13.4 using operable recorder points identified on Enclosure 13.2.

______12.2.4 Verify the calculated axial imbalance does not exceed the backup incore limits per 11.1.

______12.2.5 Verify the calculated quadrant power tilt does not exceed the backup incore limits per 11.2.

______12.2.6 If either step 12.2.4 or 12.2.5 cannot be satisfied, notify the Unit Supervisor and take appropriate actions described in the applicable Technical Specification as listed below:

Quadrant Power Tilt - ITS 3.2.3
Axial Power Imbalance - ITS 3.2.2.

13. Enclosures

13.1 Required Backup Recorder Points for Calculating Axial Power Imbalance

13.2 Required Backup Recorder Points for Calculating Quadrant Power Tilt

13.3 Axial Power Imbalance Calculation Sheet

13.4 Quadrant Power Tilt Calculation Sheet

Oconee 1 Cycle 19

Operational Power Imbalance Setpoints

	%FP	Full Incore	Backup Incore	Out of Core
4 Pumps	0	-31.5	-31.0	-31.5
	80	-31.5	-31.0	-31.5
	90	-29.7	-29.3	-29.7
	100	-19.1	-18.7	-19.1
	102	-17.0	-16.5	-17.0
	102	17.0	17.0	17.0
	100	19.1	18.7	19.1
	90	22.4	21.8	22.4
	80	23.1	22.3	23.1
	0	23.1	22.3	23.1
3 Pumps	0.0	-31.5	-31.0	-31.5
	63.30	-31.5	-	-31.5
	63.77	-	-31.0	-
	77.0	-17.0	-16.5	-17.0
	77.0	17.0	17.0	17.0
	71.99	-	22.3	-
	71.24	23.1	-	23.1
	0.0	23.1	22.3	23.1

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NRC-004/ADMIN A.2

ICCM Subcooling Margin Monitor Check

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

SUBCOOLING MONITOR CHECK

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

System: Conduct of Operations
K/A: G2.2.12
Rating: 3.0/3.8

Task Standard:

Perform Subcooling Monitor Check

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and 13.16

Validation Time: 10 min. **Time Critical:** NO

==

Candidate: ____________________________ Time Start : ________
NAME Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________ __________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC or SNAP # ________
2. Go to run, acknowledge alarms.
3. Verify accurate pressure/ temperature values
4. Freeze simulator.
5. Leave simulator in FREEZE to prevent values changing.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and 13.16

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit 1 is at 100% power
Today is Thursday
The time is 2000
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1 has been completed up page 21

INITIATING CUES:

The SRO directs you to complete PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1.

START TIME: ________

<table>
<tr><td>
<u>STEP 1</u>:

Verify Loop A and Loop B RCS WR Pressure and ICCM Plasma Display pressure agree with in 10 psig
"A" WR = 2114, ICCM = 2114
"B" WR = 2160, ICCM = 2150

<u>STANDARD</u>:
Locate and obtain Loop "A" RCS WR Pressure and ICCM Plasma Display pressure and ensure pressures agree within 10 psig

Locate and obtain Loop "B" RCS WR Pressure and ICCM Plasma Display pressure and ensure pressures agree within 10 psig

<u>COMMENTS</u>:
</td><td>
___ SAT

___ UNSAT
</td></tr>
<tr><td>
NOTE: Perform this step as the Initial Conditions indicate it is Thursday night shift.
<u>STEP 2</u>:

Obtain readings from:
<u>(25)</u> SCM Loop "A" (OAC)
<u>(24)</u> SCM Loop "A" (ICC)

<u>STANDARD</u>:
Locate and obtain Loop "A" SCM readings.

SCM Loop "A" (OAC) - SCM Loop "A" (ICC) = <u>(+1)</u>

Verify SCM Loops agree within -6 °F to +9 °F

Candidate determines that "A" is within the specified range

<u>COMMENTS</u>:
</td><td>
___ SAT

___ UNSAT
</td></tr>
</table>

STEP 2: Obtain readings from: **(18)** SCM Loop "B" (OAC) **(27)** SCM Loop "B" (ICC) STANDARD: Locate and obtain Loop "B" SCM readings. SCM Loop "B" (OAC) - SCM Loop "B" (ICC) = **(-9)** Verify SCM Loops agree within –6 °F to +9 °F Candidate determines that "B" is NOT within the specified range COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: Candidate determines that "B" is NOT within the specified range. STANDARD: Refers to Enclosure 13.16 (ICCM Subcooling Monitor Check) of PT/1/A/0600/001, Periodic Instrument Surveillance COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: Loop "B" Subcooling Monitor Obtain RCS Loop "A" pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below STANDARD: **(2026)** psig + 14.7 psi = **(2040.7)** psia COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: Using ASME Steam Tables **OR** OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below STANDARD: **(638)** °F COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6: Obtain RCS Loop "B" temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below. STANDARD: **(598)** °F ***CUE: Tell the operator to use 598°F instead of 600 as indicated on the simulator. This will allow the final calculation to within the proper range of +/- 5°F*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below. STANDARD: Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) = Correction. **(22)** °F = (638) °F – (598) °F - 18 °F COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: Verify ICC Train "B" SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4) STANDARD: Determine difference in ICCM and manually calculated SCM agrees within +/- 5°F ICC Train "B" SCM Loop **(27) – 22 = 5** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 9: Calculations performed in step 2.2 require independent verification. **CUE: another operator will perform verification calculations.** STANDARD: Sign the Performed By: ______________________ COMMENTS:	___ SAT ___ UNSAT

TIME END: ___________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
2	Step is necessary, to determine that the "B" SCM is not within the required range and is inoperable. Calculation is -9
3	Step is necessary, Refer to Enclosure 13.16 to perform Manual SCM calculation
4	Step is necessary, calculation of actual RCS pressure in psia to obtain correct saturation temperature
5	Step is necessary, obtain correct saturation temperature based on pressure (psia)
6	Step is necessary, obtain actual RCS Th temperature
7	Step is necessary, obtain actual Loop A SCM
8	Step is necessary, determine that ICCM Loop SCM agrees with the manual calculated SCM within +/- 5. Actual = 5

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit 1 is at 100% power
Today is Thursday
The time is 2000
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1 has been completed up page 21

INITIATING CUES:

The SRO directs you to complete PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) <u>AND</u> SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). <u>IF</u> SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			<u>IF</u> > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F <u>AND</u> (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. <u>IF</u> (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

_____ 2.2.2 Using ASME Steam Tables OR OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

_________ °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

_________ °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

_________°F = _________°F - _________°F - 18°F
(step 2.2.2) (step 2.2.3)

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop ________

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core AND ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core ________

ICC Train 'B' SCM Core ________

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

________________________ ________________________
Performed By IV By

1. Initial Conditions

_____ 1.1 Manual verification of ICCM Subcooling monitors required.

_____ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

_____ 2.1.2 Using ASME Steam Tables <u>OR</u> OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

__________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

__________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.1.2) (step 2.1.3)

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) **AND** SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). **IF** SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			**IF** > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F **AND** (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. **IF** (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

1. Initial Conditions

_____ 1.1 Manual verification of ICCM Subcooling monitors required.

_____ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

_____ 2.1.2 Using ASME Steam Tables <u>OR</u> OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

__________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

__________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.1.2) (step 2.1.3)

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

_____ 2.2.2 Using ASME Steam Tables OR OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

__________ °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

__________ °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.2.2) (step 2.2.3)

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop ________

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core AND ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core ________

ICC Train 'B' SCM Core ________

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

________________________ ________________________
Performed By IV By

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)	Loop A 2114 ICC 2114 Loop B 2160 ICC 2150	Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday		25-24 = 1 18-27 = -9	Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) AND SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). IF SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			IF > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F AND (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. IF (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

1. Initial Conditions

✓ 1.1 Manual verification of ICCM Subcooling monitors required.

✓ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

__________ psig + 14.7 psi =__________psia

_____ 2.1.2 Using ASME Steam Tables OR OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

__________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

__________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.1.2) (step 2.1.3)

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

2026 psig + 14.7 psi = 2040.7 psia

_____ 2.2.2 Using ASME Steam Tables OR OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

638 °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

CUE 598 °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

22 °F = 638 °F - 598 °F - 18°F
(step 2.2.2) (step 2.2.3)

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop 27

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core AND ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core _____

ICC Train 'B' SCM Core _____

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

Sign
Performed By

____
IV By

QUESTION NO. A.3 SRO/RO-Q1 G2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure [CFR: 43.4/45.10] (2.9/3.3) **REFERENCE ALLOWED**	

QUESTION:

You have been asked been asked to verify that 3RC-2 (PZR Spray Bypass) is properly back seated.

- Contamination levels on top of the PZR 500,000 dpm/cm^2
- Radiation levels are 30 mrem/hr β-γ general area

Q1: What RWP will you use to enter the RB for this job?

Q2: What dress requirements are required for this job?

Note: This information can be obtained from the RWPs in the U3 Change Room or from Shift/Unit RP crew.

ANSWER:

A1: RWP 3001, U3 RB Inspections and Valve Operations

A2: Per the RWP Dress Category and Task Description – Dress Category I, Cloth Hood, cloth coverall, cotton gloves, 2 pair of rubber gloves, booties, shoecovers, no personal outer clothing,. Secure gloves and booties (tape, Velcro, straps).

REFERENCE: Reference: Radiation Protection Policy Manual, NSD 507, RP-RPP

COMMENTS:

RADIATION WORK PERMIT # 3001 REV: 9 DATE/TIME: 03/30/00 13:36

OCONEE NUCLEAR STATION ACTIVATION DATE: 04/07/00 00:01

Job Title: U3 RX BLDG INSPECTIONS AND VALVE OPERATIONS

STANDING REQUIREMENTS FOR USE OF THIS RWP

EACH RADIATION WORKER IS RESPONSIBLE FOR:

- KNOWING THEIR WORK AREA DOSE RATES.
- FOLLOWING REQUIREMENTS OF THIS RWP.
- BEING ALARA.
- HOUSEKEEPING.
- WEARING A POCKET OR ELECTRONIC DOSIMETER AND A TLD.
- FOLLOWING POSTED REQUIREMENTS.
- REVIEWING AREA RADIOLOGICAL PLAN VIEW WHEN AVAILABLE PRIOR TO ENTRY.
- NOTIFYING RADIATION PROTECTION PRIOR TO SWEEPING, BRUSHING, GRINDING, WELDING, OR USE OF COMPRESSED AIR IN CONTAMINATED AREAS.
- FOLLOWING POSTED DRESS CATEGORY REQUIREMENTS.
- WEARING MODESTY GARMENTS WHEN NOT WEARING PERSONAL OUTER CLOTHING.
- MONITORING PERSONNEL/TOOL/EQUIPMENT REQUIRED WHEN LEAVING RCA OR CONTAMINATED RCZ.

DRESS CATEGORY AND TASK DESCRIPTION

D 1. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

H 2. WORK IN CONTAMINATED AREA.

I 3. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.

M 4. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

N 5. WORK IN CONTAMINATED AREA WHEN WET CONDITIONS OR POTENTIAL FOR WET CONDITIONS EXIST.

SPECIAL DOSIMETRY	RESPIRATORY

SPECIAL INSTRUCTIONS/PRECAUTIONS

* NOTIFY RP PRIOR TO START OF WORK
* USE HOSE CLAMPS TO SECURE HOSE/TUBING CONNECTIONS

COMMENTS

NOTIFY RP PRIOR TO ENTERING THE REACTOR BUILDING.
NOTIFY RP IF WORK AREA CONDITIONS OR JOB SCOPE CHANGES.
RESPIRATORY REQUIREMENTS WILL BE BASED ON TEDE / ALARA EVALUATIONS.
WORKERS TO INSTALL CATCH CONTAINMENTS / DRAIN RIGS TO PREVENT SPILLS WHILE DRAINING COMPONENTS.
DISPOSABLE (PLASTIC) BOOTIES SHALL BE WORN INSIDE NYLON (WASHABLE) BOOTIES FOR WORK IN WET CONDITIONS.

"EXTRA HIGH RADIATION AREA" DOSE RATES:
5000 MREM/HR HIGH CONTACT ON FLOOR OF DEEP END OF CANAL
UP TO 1000 MREM/HR GENERAL AREA IN DEEP END OF CANAL

ED (MG) SET POINTS
DOSE ALARM - 25 MREM
DOSE RATE ALARM - 100 MREM/HR

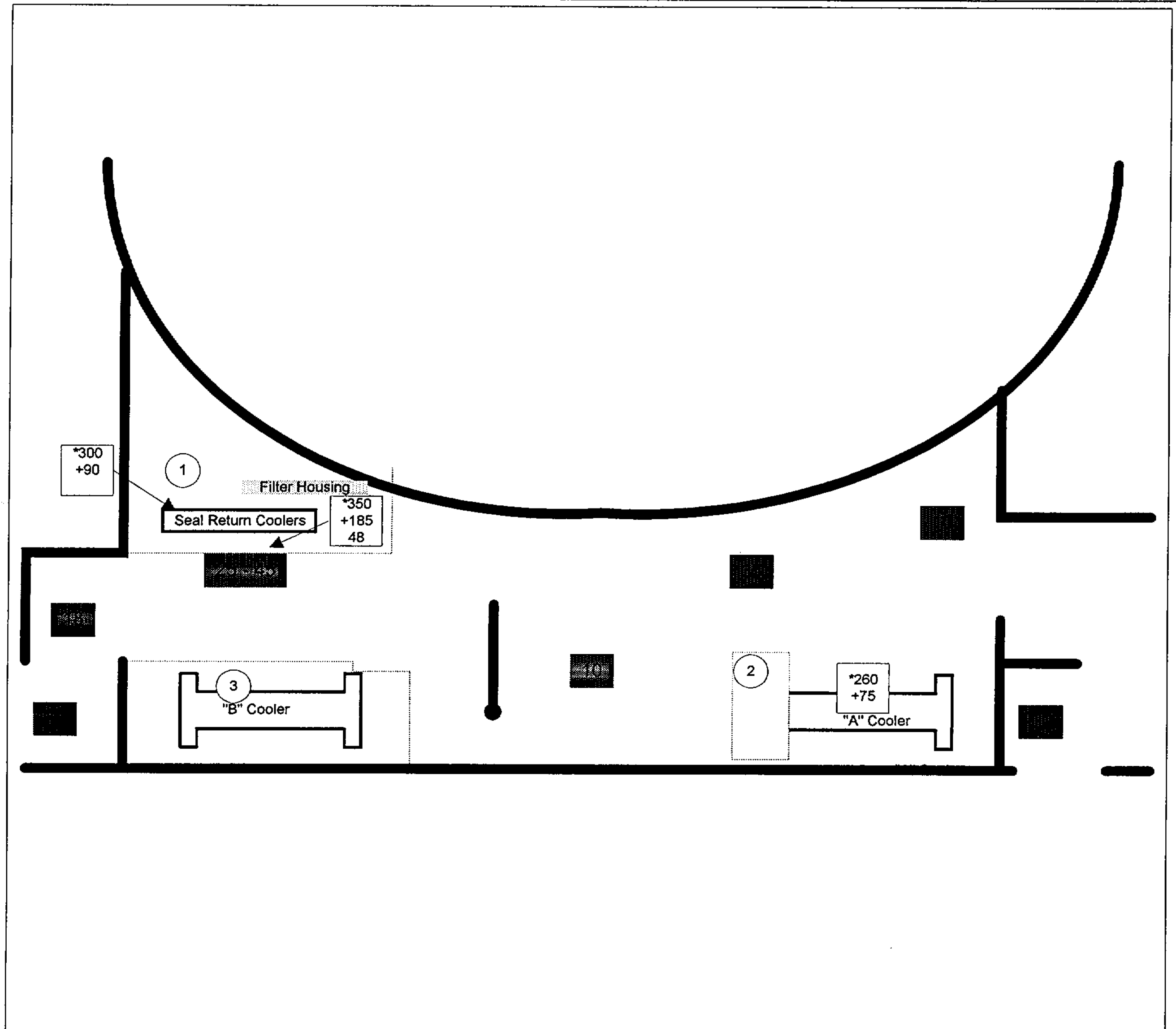
APPROVED BY: NRH1552	TERMINATED BY:
DATE/TIME: 03/30/00 13:35	DATE/TIME:

5.3.6 PROTECTIVE CLOTHING FOR EACH DRESS CATEGORY

DRESS CATEGORY	PROTECTIVE CLOTHING
A	None.
B	Surgical gloves.
C	Cotton and rubber gloves.
D	Cotton and rubber gloves, booties and shoecovers.
E	Labcoat, cotton and rubber or surgical gloves.
F	Labcoat, cotton and rubber gloves, booties and shoecovers.
G	Cloth hood, disposable coveralls, cotton and rubber gloves, booties and shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
H	Cloth hood, cloth coverall, cotton and rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
I	Cloth hood, cloth coverall, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
J	Cloth hood, cloth coverall, cotton gloves, 2 pair rubber gloves, booties, shoecovers, no personal outer clothing and additional outer booties or shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
K	Cloth hood, cloth coverall, disposable coveralls, cotton gloves, rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
L	Cloth hood, cloth coverall, disposable coveralls, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing and additional outer booties or shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
M	Cloth hood, 2 pair cloth coveralls, cotton gloves, 2 pair rubber gloves, 2 pair booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
N	Cloth hood, cloth coverall, wetsuit, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
O	Cloth hood, cloth coverall, bubble suit, cotton gloves, 2 pair rubber gloves, booties, shoecovers, no personal outer clothing and additional shoe covers or jump boots. Secure gloves and booties (tape, elastic, Velcro, straps).
Z	Special dress as required by Radiation Protection.

QUESTION NO. A.3 SRO/RO-Q2 G2.3.1 Radiation Exposure Limits [2.6/3.0] **REFERENCE ALLOWED**
QUESTION: Given the attached Oconee Nuclear Station VSDS Survey Report for Room 108: Concerning the area the Room 108 (U-1Decay Heat Removal / Seal Return) plan view 1. Describe the type of posting this area should have to warn radiation worker? Explain your answer. 2. Identify where you should stand while waiting for further direction from the Control Room during a work evaluation. Explain your answer.
ANSWER: 1. High Radiation Area because the dose rate (185 mrem/hour) at 30 centimeters is greater than 100 mrem/hour. This would be marked High radiation area signs. 2. Near the entrance because the general area radiation level is the lowest value at this location.
REFERENCE: NSD 507.8 Exposure and Contamination Control
COMMENTS:

Room 108 Decay Heat Removal / Seal Return	Survey # 050100-27	Date/Time: 07/01/2000 22:48



Comments: 1000/054 ROUTINE SURVEY, PALNVIEW UPDATED, ALL TAKEN IN CLEAN AREA WHERE < 100 CCPM.

Summary of Highest Readings

Smears	Air Samples & Wipes
2) 3621 DPM/100 CM2 B /γ 1) 1800 DPM/100 CM2 B /γ 3) 1465 DPM/100 CM2 B /γ	

Dose Rate

*150 - - Contact Reading
+75 - - 30 cm Reading
20 - - General Area

HS-50 | Hot Spot
Posting
Drip Bag

15 Smear | 15 Air Sample | RM | Wipe

Type = Monthly

RWP: 15

Reactor Power = 100%

Unless otherwise Noted, dose rate in mrem/hr.

Surveyor:

Reviewed by:

ENABLING OBJECTIVES (continued)

4. State the approval requirements for an individual at Duke Power Company to exceed the **basic** permissible exposure limit of 2.0 rem. (R4)

5. State the special dose limits established for the general public. (R5)

6. Describe the special dose control measures used to protect the fetus of a "declared" pregnant radiation worker. (R6)

7. Recognize that in "exceptional situations", it is possible to allow an adult radiation worker to receive additional exposure, apart from normal occupational exposure. (R7)

8. Define and describe the specific site area for each of the following terms relating to the control of station areas: (R8)

    8.1 Unrestricted Area

    8.2 Restricted Area

    8.3 Controlled Area

    8.4 Radiation Control Area (RCA)

    8.5 Radiation Control Zone (RCZ)

    8.6 Radiation Area (RA)

    8.7 High Radiation Area (HRA)

    8.8 Extra High Radiation Area (EHRA)

    8.9 Very High Radiation Area (VHRA)

    8.10 Airborne Radioactivity Area

    8.11 Hot Spot

    8.12 Significant Dose Contributor

    8.13 Low Exposure Waiting Area

    8.14 Contaminated Area

507.8 EXPOSURE AND CONTAMINATION CONTROL

DPC adheres to the conservative assumption that there is a risk associated with radiation exposure. Application of the As Low As Reasonably Achievable (ALARA) concept minimizes this risk. Nuclear facility management and all individuals who perform work at the facility share the goal of keeping dose ALARA. Individuals are expected to be knowledgeable in and practice exposure control techniques.

507.8.1 AS LOW AS REASONABLY ACHIEVABLE (ALARA) PROGRAM

The ALARA Program is designed to minimize dose. The ALARA Program is described in the System ALARA Manual, Section III. Some important ALARA Program components are:

- Holding pre-job and post-job briefs
- Pre-planning jobs
- Using training mock-ups
- Using engineering controls
- Removing sources of radiation exposure
- Applying lessons learned from industry events
- Providing job feedback
- Using ALARA Suggestion Forms

A. Planning for Tasks <500 mrem Total Exposure

- Use basic ALARA principles
- RP ALARA Group involvement is not required.

B. ALARA Planning for Tasks Greater Than or Equal to 500 mrem

All Work Order tasks greater than or equal to 500 mrem are planned and tracked using an ALARA package. The package consists of:

- ALARA Planning Worksheet (System ALARA Manual, Section IV)
- ALARA Briefing Checklist (System ALARA Manual, Section IV)
- Execution Team Post-Job ALARA Critique (System ALARA Manual, Section IV)
- RP ALARA Post-Job Critique (System ALARA Manual, Section IV)

C Dose Tracking

- Task supervisor and execution team has primary responsibility for tracking all exposures received.
- RP shall be contacted when received dose exceeds expected values.

D Post-Job Critique

- Provide problem and improvement ideas to RP on Execution Team Post-job Critique (when provided) or use ALARA Suggestion Form.

507.8.2 ACCESS CONTROL

Controls are in place to limit access to site and in-plant areas for security and radiological safety purposes. The majority of radiologically controlled areas are located within the RCA; however, Radiation Control Zones may also be established at locations outside of the RCA.

ENABLING OBJECTIVES (continued)

18. Describe the method used at Oconee to indicate whether an article is "clean" and can be unconditionally released from the RCA, or is above the contamination limit for unconditional release. (R18)
19. State the maximum contamination limit (in cpm) for personal clothing, body surfaces, and hand-held items for unconditional release from the RCA. (R19)
20. Describe how designated contaminated tools used inside the RCA are identified. (R20)
21. Concerning RWPs/SRWPs: (R21)
    - 21.1 Explain the purpose of RWPs and SRWPs.
    - 21.2 Explain the differences between RWPs and SRWPs.
    - 21.3 List the requirements for re-evaluating SRWPs.
    - 21.4 Understand that indivduals do not have the authority to deviate from RWP or SRWP requirements.
    - 21.5 Identify plant locations of SRWP information.
22. Define ALARA. (R22)
23. Identify the various methods available to aid in maintaining exposures ALARA. (R23)
24. Given a set of conditions, correctly apply the radiation protection practices addressed in this lesson plan. (R24) ✱

✱ P.1 CHE 11-18-97

RCA/RCZs are posted with warning signs that clearly identify the radiological hazard(s) and other pertinent access information. All individuals at a nuclear facility are expected to:

- Comply with RCA/RCZ entrance/exit requirements
- Read and comply with posted warning signs and/or barricades.
- Maintain the integrity of barricades after entering or exiting an RCA/RCZ or when working in close proximity to one.
- Regard RCZ ropes as if they are walls. Do not reach across or move ropes unless authorized by RP.
- Notify RP with questions or problems.

A. Definitions

Airborne Radioactivity Area - An area containing airborne radioactivity that is equal to or greater than 25% of 1 Weighted Derived Air Concentration (DAC).

Contaminated Area - An area where loose contamination equal to or greater than 1000 dpm/100cm^2 beta/gamma and/or 20 dpm/100cm^2 alpha exists.

Extra High Radiation Area - An area with a dose rate greater than 1000 mrem/hour at 30 centimeters. These areas are locked or guarded and require continuous RP coverage for entry. In areas that can not be reasonably locked, a flashing yellow light is used as a warning device.

High Radiation Area - An area with a dose rate greater than 100 mrem/hour at 30 centimeters.

Hot Spot - A localized source of radiation that is at least five times the general area dose rates, has a contact dose rate greater than 100 mrem/hour and/or is located where the potential for significant personnel exposure exists.

Low Exposure Waiting Area (LEWA) - An area where the dose rate is less than the general area. Usually, the lowest dose rate location in a room or area.

Protected Area - Area within the double fence around the plant. Access requires security identification.

Radiation Area - An area with a dose rate greater than 5 mrem/hour at 30 centimeters.

Radiation Control Area (RCA) - An area established within the Restricted Area to provide additional access control for radiological safety purposes. Requirements for entry are located in Section 507.5.

Radiation Control Zone (RCZ) - An area where specific radiological hazards exist and which are defined and controlled in accordance with 10CFR20 requirements. Requirements for entry are located in Section 507.5.

Radioactive Material - An area where radioactive materials are stored.

Restricted Area - Any area where access is controlled by the licensee for purposes of protecting individuals from exposure to radiation and radioactive materials. At DPC nuclear facilities, the Restricted Area includes the Reactor Building(s), Auxiliary Building(s), Turbine Building(s), Service Building and fenced area adjacent to the above buildings. At ONS, the Independent Spent Fuel Storage Installation, Radwaste Facility, and some warehouses are also in Restricted Areas.

Significant Dose Contributor - An area normally posted with a Significant Dose Contributor sign and florescent green ribbon to identify highest dose rate area(s) in a Radiation Area, High Radiation Area or Extra High Radiation Area.

Unrestricted Area - Any area where access is neither limited or controlled outside the site boundary fence.

Very High Radiation Area - An area with a dose rate greater than 500 rad/hour at 1 meter. These areas are locked at all times and requires continuous RP coverage for entry.

507.8.4 CONTAMINATION PREVENTION AND CONTROL METHODS

RP radioactive contamination controls are implemented to minimize contamination of personnel, areas and equipment. Surface contamination controls minimize possible inhalation or ingestion of radioactivity, skin dose from small particles of radioactivity and the spread of contamination to the environment. Some methods used to control contamination are:

A. Performing surveys, establishing RCZs and posting warning signs in areas where sources of contamination exist.

B. Limiting eating, drinking, storage of food, chewing and use of tobacco products to authorized areas.

- Persons with medical conditions, such as heart problems, are allowed to carry emergency medication inside RCZs, RP personnel should be contacted in advance for instructions.

C. Planning and performing work to minimize spread of contamination and reduce number of contaminated areas.

D. Decontaminating surfaces whenever practical. See Section 507.8.8 for policy.

E. Bagging and tagging contaminated material, equipment and tools. See Section 507.9.3 for policy.

F. Securing equipment/cords/hoses that cross contaminated RCZ boundaries.

G. Using protective clothing appropriately to prevent becoming contaminated:

- Adhere to protective clothing requirements specified on the SRWP or RWP unless authorized to deviate by RP.
- Consult RP if SRWP or RWP requirements should be changed.
- Use radiological protective clothing only in the RCA or RCZ unless otherwise specified by RP.
- Leave the work area if conditions change and cause specified protective clothing to be inappropriate or inadequate. Notify RP.
- Leave the work area if protective clothing becomes torn, soaked, untaped or unfastened. Notify RP.
- Control protective clothing during removal and place it in appropriate containers.
- Remove all protective clothing (except modesty garments) before exiting the RCZ boundary (unless directed otherwise by RP).

H. Catch Containment Program

Catch containments are used to prevent the spread of contamination that occurs as a result of uncontrolled plant system leaks. Routine inspections/audits of catch containments are performed to ensure containments are functioning properly and in good condition.

I. Radiological Respiratory Protection Program

The primary objective of the Respiratory Program is to minimize inhalation of airborne radioactive materials by individuals. The preferred method of achieving this objective is the use of engineering controls. Engineering controls are built into the nuclear facilities to remove airborne radioactive materials from the work environment. When additional engineering controls, such as local exhaust ventilation, containment or decontamination cannot be used or are not practical, the following methods are used to maintain dose ALARA.

- Increasing monitoring and access control
- Limiting exposure times
- Using respiratory protection equipment

F. Lab Coats

Lab coats are used for limited purposes such as performing analysis on radioactive samples or laundry operations. Lab coats provide only minimal protection from contamination and should not be used for most contamination control purposes.

G. Modesty Tops/Gym Shorts

These items are provided to be worn under protective clothing to protect personal underclothing and to provide body cover after protective clothing removal.

H. Facial Protection

- Goggles and face shields are provided to protect the eyes and face from contamination when working in close proximity to contaminated components and from the splashing of contaminated liquid. Goggles and face shields also provide protection from beta exposure.
- Facial protection (such as masks, face socks, disposable face shields) provides protection to the skin of the face from contamination.

I. Worker Suiting Up

- Obtain the required protective clothing.
- Remove all personal outer clothing, if required.

  Note: T-shirts are considered outer personal clothing unless worn as a undershirt.
- Put on gym shorts (and modesty top, if worn).
- Put on required coveralls.
- Place dosimetry on a neck strap under the coveralls or place dosimetry inside the coverall pocket with the TLD in front of the ED with the TLD beta window facing away from the body. Secure or tape the pocket opening.
- Put on booties over personal shoes (these may be worn inside or outside of the coveralls).
- Secure outer booties (tape, elastic, velcro, straps).
- Put on rubber shoe covers.
- Put on hood.
- Put on cotton gloves and then rubber gloves or surgical gloves.
- Secure gloves over sleeves of coveralls (tape, elastic, velcro, straps).
- Return unused protective clothing to storage location.

J. Removal of Protective Clothing

Do not throw protective clothing or equipment across the exit area; this can result in the spread of contamination.

- Remove all tape, elastic or velcro, if used (from wrists, ankles, etc.).
- Remove rubber shoe covers.
- Remove rubbers gloves by peeling them off inside out.
- Remove hood, taking care not to contaminate hair or face.
- Remove dosimetry.
- Remove the coveralls, peeling off inside out.
- Remove booties as you transfer to the step-off pad which is clean.
- Remove cotton gloves and proceed to nearest monitor wearing the gym shorts (and modesty top). Hand and foot monitoring is required at minimum before proceeding to change room.
- Monitor whole body to ensure you are not contaminated before dressing in personal clothes.
- In cases where more than one set of overalls are worn or multiple step off pads are used, follow the instructions of RP for dress out and removal.

507.8.6 PERSONNEL CONTAMINATION MONITORING

Admin Exam A.4 Emergency Plan (SRO)

Question #1:

Based on the event that just occurred on the simulator describe your actions as the Emergency Coordinator concerning classifying this event. Sequence all classification thresholds throughout the entire event.

ANSWER:
ALERT based on HPI Cooling established Fission Product Matrix total = 4

Question #2:
If condenser vacuum was lost with the Condenser rupture disc blown and the 1B SGTL increases to 70 gpm how does this affect the E-Plan classification and any PAGs that may apply?

ANSWER:
Upgrade to SAE based on Fission Product Matrix total = 7 (HPI Cooling (4)+ SGTL > 10 gpm with direct opening to the environment (3).
No PAGs are required.

(R06-97)

INFORMATION ONLY

(1) ID No. RP/0/B/1000/001

Duke Power Company
PROCEDURE PROCESS RECORD

Revision No 6

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Emergency Classification

(4) Prepared By Nora Kelley Date 2-16-99

(5) Requires 10CFR50.59 evaluation?
- ☒ Yes (New procedure or revision with major changes)
- ☐ No (Revision with minor changes)
- ☐ No (To incorporate previously approved changes)

(6) Reviewed By W. C. Brandt (QR) Date 2/25/99

Cross-Disciplinary Review By ______ (QR)NA WB Date 2/25/99

Reactivity Mgmt. Review By ______ (QR)NA WB Date 2/25/99

(7) Additional Reviews

QA Review By ______ Date ______

Reviewed By ______ Date ______

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By M R Thome Date 3-27-99

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification
- ☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
- ☐ Yes ☐ NA Listed enclosures attached?
- ☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
- ☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
- ☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Site **Emergency Classification** **Reference Use**	Procedure No. RP/0/B/1000/001 Revision No. 006 Electronic Reference No. OX002WOS

Emergency Classification

NOTE:	This procedure is an implementing procedure tot he Oconee Nuclear Site Emergency plan and must be forwarded to Emergency Planning within three (3) working days of approval.

1. Symptoms

1.1 This procedure describes the immediate actions to be taken to recognize and classify an emergency condition.

1.2 This procedure identifies the four emergency classifications and their corresponding Emergency Action Levels (EALs).

1.3 This procedure provides reporting requirements for non-emergency abnormal events.

1.4 The following guidance is to be used by the Emergency Coordinator/EOF Director in assessing emergency conditions:

1.4.1 The Emergency Coordinator/EOF Director shall review all applicable initiating events to ensure proper classification.

1.4.2 The BASIS Document (Volume A, Section D of the Emergency Plan) is available for review if any questions arise over proper classification.

1.4.3 **IF** An event occurs on more than one unit concurrently,
THEN The event with the higher classification will be classified on the Emergency Notification Form.

A. Information relating to the problem(s) on the other unit(s) will be captured on the emergency Notification Form as shown in RP/0/B/1000/015A, (Offsite Communications From The Control Room), RP/0/B/1000/015B, (Offsite Communications From The Technical Support Center) or RP/0/B/1000/015C, (Offsite Communications From The Emergency Operations Facility).

1.4.4 **IF** An event occurs,
AND A lower or higher plant operating mode is reached before the classification can be made,
THEN The classification shall be based on the mode that existed at the time the event occurred.

1.4.5 The Fission Product Barrier Matrix is applicable only to those events that occur at Hot Shutdown or higher.

A. An event that is recognized at Cold Shutdown or lower shall not be classified using the Fission Product Barrier Matrix.

1. Reference should be made to the additional enclosures that provide Emergency Action Levels for specific events (e.g., Severe Weather, Fire, Security).

1.5 IF A transient event should occur,

THEN Review the following guidance:

1.5.1 IF An Emergency Action Level (EAL) identifies a specific duration
AND The Emergency Coordinator/EOF Director assessment concludes that the specified duration is exceeded or will be exceeded, (i.e.; condition cannot be reasonably corrected before the duration elapses),
THEN Classify the event.

1.5.2 IF A plant condition exceeding EAL criteria is corrected before the specified duration time is exceeded,
THEN The event is **NOT** classified by that EAL.

A. Review lower severity EALs for possible applicability in these cases.

NOTE:	Reporting under 10CFR50.72 may be required for the following step. Such a condition could occur, for example, if a follow up evaluation of an abnormal condition uncovers evidence that the condition was more severe than earlier believed.

1.5.3 IF A plant condition exceeding EAL criteria is not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g.; as a result of routine log or record review)
AND The condition no longer exists,
THEN An emergency shall **NOT** be declared.

1.5.4 IF An emergency classification was warranted, but the plant condition has been corrected prior to declaration and notification,
THEN The Emergency Coordinator must consider the potential that the initiating condition (e.g.; Failure of Reactor Protection System) may have caused plant damage that warrants augmenting the on shift personnel through activation of the Emergency Response Organization.

A. <u>IF</u> An **Unusual Event** condition exists,
<u>THEN</u> Make the classification as required.

1. The event may be terminated in the same notification or as a separate termination notification.

B. <u>IF</u> **An Alert, Site Area Emergency, or General Emergency** condition exists,
<u>THEN</u> Make the classification as required,
<u>AND</u> Activate the Emergency Response Organization.

1.6 Emergency conditions shall be classified as soon as the Emergency Coordinator/EOF Director assessment determines that the Emergency Action Levels for the Initiating Condition have been exceeded.

2. Immediate Actions

2.1 Determine the operating mode that existed at the time the event occurred prior to any protection system or operator action initiated in response to the event.

2.2 <u>IF</u> The unit is at Hot Shutdown or higher
<u>AND</u> The condition/event affects fission product barriers,
<u>THEN</u> GOTO Enclosure 4.1, (Fission Product Barrier Matrix).

2.2.1 Review the criteria listed in Enclosure 4.1, (Fission Product Barrier Matrix) and make the determination if the event should be classified.

2.3 Review the listing of enclosures to determine if the event is applicable to one of the categories shown.

2.3.1 <u>IF</u> One or more categories are applicable to the event,
<u>THEN</u> Refer to the associated enclosures.

2.3.2 Review the EALs and determine if the event should be classified.

A. <u>IF</u> An EAL is applicable to the event,
<u>THEN</u> Classify the event as required.

2.4 <u>IF</u> The condition requires an emergency classification,
<u>THEN</u> GOTO RP/0/B/1000/002, (Control Room Emergency Coordinator Procedure).

3. Subsequent Actions

3.1 Continue to review the emergency conditions to assure the current classification continues to be applicable.

4. Enclosures

DETERMINE THE APPROPRIATE CLASSIFICATION USING THE TABLE BELOW: CIRCLE EALS CHOSEN. ADD POINTS TO CLASSIFY. (SEE NOTE BELOW)

RCS BARRIERS (BD 5-7)		FUEL CLAD BARRIERS (BD 8-9)		CONTAINMENT BARRIERS (BD 10-12)	
Potential Loss (4)	Loss (5)	Potential Loss (4)	Loss (5)	Potential Loss (1)	Loss (3)
RCS Leakrate > Makeup capacity of one HPI pump in normal makeup mode (approx. 160 gpm) with Letdown isolated.	RCS Leak rate > available makeup capacity as indicated by a loss of subcooling	Average of the 5 highest CETC ≥ 700° F	Average of the 5 highest CETC ≥ 1200° F	CETC ≥ 1200° F ≥ 15 minutes OR CETC ≥ 700° F ≥ 15 minutes with a valid RVLS reading 0"	Rapid unexplained containment pressure decrease after increase OR containment pressure or sump level not consistent with LOCA
SGTR > Makeup capacity of one HPI pump in normal makeup mode (approx. 160 gpm) with Letdown isolated.		Valid RVLS reading of 0"	Coolant activity ≥ 300 μCi/ml DEI	RB pressure ≥ 59 psig OR RB pressure ≥ 10 psig and no RBCU or RBS.	Failure of secondary side of SG results in a direct opening to the environment with P/S leakage ≥ 10 gpm in the same SG
Entry into the TSOR (Thermal Shock) operating range	1RIA 57/58 reading ≥ 1.0 R/hr 2 RIA 57 reading ≥ 1.6 R/hr 2 RIA 58 reading ≥ 1.0 R/hr 3RIA 57/58 reading ≥ 1.0 R/hr		Hours Since SD — RIA57/58 - R/hr 0 - < 0.5 — ≥ 300/150 0.5 - < 2.0 — ≥ 80/40 2.0 - 8.0 — ≥ 32/16	Hours Since SD — RIA57/58 - R/hr 0 - < 0.5 — ≥ 1800/860 0.5 - < 2.0 — ≥ 400/195 2.0 - 8.0 — ≥ 280/130	Failure of secondary side of SG results in a direct opening to the environment with P/S leakage ≥ 10 gpm in the other SG AND Feeding SG with secondary side failure from the affected unit
HPI Forced Cooling	RCS pressure spike ≥ 2750 psig			Hydrogen concentration ≥ 9%	Containment isolation is incomplete and a release path to the environment exists
Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment

UNUSUAL EVENT (1-3)	ALERT (4-6)	SITE AREA EMERGENCY (7-10)	GENERAL EMERGENCY (11-13)
OPERATING MODE: 1, 2, 3, 4 • Any potential loss of Containment • Any loss of containment	OPERATING MODE: 1, 2, 3, 4 • Any potential loss or loss of the Fuel Clad • Any potential loss or loss of the RCS	OPERATING MODE: 1, 2, 3, 4 • Loss of any two barriers • Loss of one barrier and potential loss of either RCS or Fuel Clad Barriers • Potential loss of both the RCS and Fuel Clad Barriers	OPERATING MODE: 1, 2, 3, 4 • Loss of any two barriers and potential loss of the third barrier • Loss of all three barriers
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4

NOTE: An event with multiple events could occur which would result in the conclusion that exceeding the loss or potential loss threshold is IMMINENT (i.e., within 1-3 hours). In this IMMINENT LOSS situation, use judgment and classify as if the thresholds are exceeded.

Enclosure 4.2
Systems Malfunctions

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. RCS LEAKAGE (BD 14) ========== OPERATING MODE: 1, 2, 3, 4 • Unidentified leakage ≥ 10 gpm • Pressure boundary leakage ≥ 10 gpm • Identified leakage ≥ 25 gpm 2. UNPLANNED LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/INDICATION IN CONTROL ROOM FOR > 15 MINUTES (BD 15) ========== OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND Loss of anunciators or indicators requires additional personnel (beyond normal shift complement) to safely operate the unit 3. INABILITY TO REACH REQUIRED SHUTDOWN WITHIN LIMITS (BD 16) ========== OPERATING MODE: 1, 2, 3, 4 • Required operating mode not reached within TS LCO action statement time (CONTINUED)	1. UNPLANNED LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/INDICATION IN CONTROL ROOM (BD 19) ========== OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND Loss of anunciators/indicators requires additional personnel (beyond normal shift complement) to safely operate the unit AND EITHER OF THE FOLLOWING: • Significant plant transient in progress OR • Loss of the OAC and ALL PAM indications (END)	1. INABILITY TO MONITOR A SIGNIFICANT TRANSIENT IN PROGRESS (BD 21) ========== OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND A significant transient is in progress AND Loss of the OAC and ALL PAM indications AND Inability to directly monitor any one of the following functions: 1. Subcriticality 2. Core Cooling 3. Heat Sink 4. RCS Integrity 5. Containment Integrity 6. RCS Inventory (END)	
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4. UNPLANNED LOSS OF ALL ONSITE OR OFFSITE COMMUNICATIONS (BD 17) == OPERATING MODE: All ♦ Loss of all onsite communications capability (ROLM system, PA system, Pager system, Onsite Radio system) affecting ability to perform routine operations ♦ Loss of all onsite communications capability (Selective signaling, NRC FTS lines, Offsite Radio System, AT&T line) affecting ability to communicate with offsite authorities. 5. FUEL CLAD DEGRADATION (BD 18) == OPERATING MODE: All: ♦ DEI - >5μCi/ml (END)			
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4			

Enclosure 4.3
Abnormal Rad Levels/Radiological Effluent

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. ANY UNPLANNED RELEASE OF GASEOUS OR LIQUID RADIOACTIVITY TO THE ENVIRONMENT THAT EXCEEDS TWO TIMES THE SLC LIMITS FOR 60 MINUTES OR LONGER (BD 23) ========== OPERATING MODE: All • Valid indication on radiation monitor RIA 33 of ≥ 4.06E+06 cpm for > 60 minutes (See Note 1) • Valid indication on radiation monitor RIA 45 of ≥ 1.33E+06 cpm for > 60 minutes (See Note 1) • Liquid effluent being released exceeds two times SLC 16.11.1 for > 60 minutes as determined by Chemistry Procedure • Gaseous effluent being released exceeds two times SLC 16.11.2 for > 60 minutes as determined by RP Procedure NOTE 1: If monitor reading is sustained for the time period indicated in the EAL AND the required assessments (procedure calculations) cannot be completed within this period, declaration must be made on the valid Radiation Monitor reading. (CONTINUED)	1. ANY UNPLANNED RELEASE OF GASEOUS OR LIQUID RADIOACTIVITY TO THE ENVIRONMENT THAT EXCEEDS 200 TIMES RADIOLOGICAL TECHNICAL SPECIFICATIONS FOR 15 MINUTES OR LONGER (BD 28) ========== OPERATING MODE: All • Valid indication on RIA 46 of ≥ 2.98E+04 cpm for >15 minutes (See Note 1) • RIA 33 HIGH Alarm AND Liquid effluent being released exceeds 200 times the level of SLC 16.11.1 for > 15 minutes as determined by Chemistry Procedure • Gaseous effluent being released exceeds 200 times the level of SLC 16.11.2 for >15 minutes as determined by RP Procedure 2. RELEASE OF RADIOACTIVE MATERIAL OR INCREASES IN RADIATION LEVELS THAT IMPEDES OPERATION OF SYSTEMS REQUIRED TO MAINTAIN SAFE OPERATION OR TO ESTABLISH OR MAINTAIN COLD SHUTDOWN (BD 30) ========== OPERATING MODE: All • Valid radiation reading ≥ 15 mRad/hr in CR, CAS OR Radwaste CR • Unplanned/unexpected valid area monitor readings exceed limits stated in Enclosure 4.9 (CONTINUED)	1. BOUNDARY DOSE RESULTING FROM ACTUAL/ IMMINENT RELEASE OF GASEOUS ACTIVITY (BD 32) ========== OPERATING MODE: All • Valid reading on RIA 46 of ≥ 2.98E+05 cpm for >15 minutes (See Note 2) • Valid reading on RIA 57 or 58 as shown on Enclosure 4.8 (See Note 2) • Dose calculations result in a dose projection at the site boundary of: ≥ 100 mRem TEDE or 500 mRem CDE adult thyroid • Field survey results indicate site boundary dose rates exceeding ≥100 mRad/hr expected to continue for more than one hour OR Analyses of field survey samples indicate adult thyroid dose commitment of ≥ 500 mRem CDE (3.84 E^{-7} µCi/ml) for one hour of inhalation NOTE 2: If actual Dose Assessment cannot be completed within 15 minutes, then the valid radiation monitor reading should be used for emergency classification. (CONTINUED)	1. BOUNDARY DOSE RESULTING FROM ACTUAL/ IMMINENT RELEASE OF GASEOUS ACTIVITY (BD 36) ========== OPERATING MODE: All • Valid reading on RIA 46 of ≥ 2.98E+06 cpm for ≥15 minutes (See Note 3) • Valid reading on RIA 57 or 58 as shown on Enclosure 4.8 (See Note 3) • Dose calculations result in a dose projection at the site boundary of: ≥ 1000 mRem TEDE OR ≥ 5000 mRem CDE adult thyroid • Field survey results indicate site boundary dose rates exceeding ≥1000 mRad/hr expected to continue for more than one hour OR Analyses of field survey samples indicate adult thyroid dose commitment of ≥ 5000 mRem CDE for one hour of inhalation NOTE 3: If actual Dose Assessment cannot be completed within 15 minutes, then the valid radiation monitor reading should be used for emergency classification. (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

Assumptions used for calculation of vent monitors RIA 45 & 46:

1. Average annual meteorology (1.672 E-6 sec/m^3), semi-elevated
2. Vent flow rate 65,000 cfm (average daily flow rate)
3. No credit is taken for vent filtration
4. One hour release duration for Unusual Event, 15 minute duration for Alert, Site Area Emergency, General Emergency
5. General Emergency PAGs are 1 rem TEDE and 5 rem CDE; Site Area Emergency determination is based on 10% of the General Emergency PAGs
6. Calculations for monitor readings are based on whole body dose
7. Standard ODCM guidance together with NUMARC guidance indicates that effluent releases are based on Technical Specification releases

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2. UNEXPECTED INCREASE IN PLANT RADIATION OR AIRBORNE CONCENTRATION (BD 25) ========== <u>OPERATING MODE:</u> All • LT 5 reading 14" and decreasing with makeup not keeping up with leakage <u>WITH</u> fuel in the core • Uncontrolled water level decrease in the SFP and fuel transfer canal with all irradiated fuel assemblies remaining covered by water • 1 R/hr radiation reading at one foot away from a damaged storage cask located at the ISFSI • Valid area monitor readings exceeds limits stated in Enclosure 4.9. (END)	3. MAJOR DAMAGE TO IRRADIATED FUEL OR LOSS OF WATER LEVEL THAT HAS OR WILL RESULT IN THE UNCOVERING OF IRRADIATED FUEL OUTSIDE THE REACTOR VESSEL (BD 31) ========== <u>OPERATING MODE:</u> All • Valid RIA 3, 6, 41, OR 49 HIGH Alarm • HIGH Alarm for portable area monitors on the main bridge or auxiliary bridge or SFP bridge • Report of visual observation of irradiated fuel uncovered • Operators determine water level drop in either the SFP or fuel transfer canal will exceed makeup capacity such that irradiated fuel will be uncovered (END)	2. LOSS OF WATER LEVEL IN THE REACTOR VESSEL THAT HAS OR WILL UNCOVER FUEL IN THE REACTOR VESSEL (BD 35) ========== <u>OPERATING MODE:</u> 5, 6 • Failure of heat sink causes loss of Cold Shutdown condition <u>AND</u> LT 5 indicates 0 inches after initiation of RCS makeup • Failure of heat sink causes loss of Cold Shutdown condition <u>AND</u> Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup NOTE: This Initiating Condition is also located in Enclosure 4.4, (Loss of Shutdown Functions). High radiation levels will also be seen with this condition. (END)	
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	

Enclosure 4.4
Loss of Shutdown Functions

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	1. FAILURE OF RPS TO COMPLETE OR INITIATE A Rx SCRAM (BD 39) == <u>OPERATING MODE</u> 1, 2, 3 ♦ Valid reactor trip signal received or required <u>WITHOUT</u> automatic scram <u>AND ONE OF THE FOLLOWING</u> DSS has inserted Control Rod Groups 5, 6, 7 <u>OR</u> Manual trip from the Control Room is successful and reactor power is less than 5% and decreasing 2. INABILITY TO MAINTAIN PLANT IN COLD SHUTDOWN (BD 41) == <u>OPERATING MODE:</u> 5, 6 ♦ Loss of LPI and/or LPSW <u>AND</u> Inability to maintain RCS temperature below 200° F as indicated by either of the following: RCS temperature at the LPI Pump Suction <u>OR</u> visual observation (END)	1. FAILURE OF RPS TO COMPLETE OR INITIATE A Rx SCRAM (BD 42) == <u>OPERATING MODE:</u> 1, 2 ♦ Valid reactor trip signal received or required <u>WITHOUT</u> automatic scram <u>AND</u> DSS has <u>NOT</u> inserted Control Rod Groups 5, 6, 7 <u>AND</u> Manual trip from the Control Room was <u>NOT</u> successful in reducing reactor power to less than 5% and decreasing 2. COMPLETE LOSS OF FUNCTION NEEDED TO ACHIEVE OR MAINTAIN HOT SHUTDOWN (BD 43) == <u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ Average of the 5 highest CETCs ≥1200° F shown on ICCM ♦ Unable to maintain reactor subcritical ♦ SSF feeding SG per EOP (CONTINUED)	1. FAILURE OF RPS TO COMPLETE AUTOMATIC SCRAM AND MANUAL SCRAM NOT SUCCESSFUL WITH INDICATION OF CORE DAMAGE (BD 45) == <u>OPERATING MODE:</u> 1, 2 ♦ Valid Rx trip signal received or required <u>WITHOUT</u> automatic scram <u>AND</u> Manual trip from the Control Room was <u>NOT</u> successful in reducing reactor power to < 5% and decreasing <u>AND</u> Average of the 5 highest CETCs ≥1200° F on ICCM (END)
	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
		3. LOSS OF WATER LEVEL IN THE REACTOR VESSEL THAT HAS OR WILL UNCOVER FUEL IN THE REACTOR VESSEL (BD 44) == OPERATING MODE: 5, 6 ♦ Failure of heat sink causes loss of Cold Shutdown conditions AND LT-5 indicates 0 inches after initiation of RCS makeup ♦ Failure of heat sink causes loss of Cold Shutdown conditions AND Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup (END)	
		INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. LOSS OF ALL OFFSITE POWER TO ESSENTIAL BUSSES FOR GREATER THAN 15 MINUTES (BD 47) == OPERATING MODE: All • Loss of all offsite AC power to both the Red and Yellow Busses for > 15 minutes AND Unit auxiliaries are being supplied from Keowee or CT5 2. UNPLANNED LOSS OF REQUIRED DC POWER FOR GREATER THAN 15 MINUTES (BD 48) == OPERATING MODE: 5, 6 • Unplanned loss of vital DC power to required DC busses as indicated by bus voltage less than 110 VDC AND Failure to restore power to at least one required DC bus within 15 minutes from the time of loss (END)	1. LOSS OF ALL OFFSITE AC POWER AND LOSS OF ALL ONSITE AC POWER TO ESSENTIAL BUSSES (BD 49) == OPERATING MODE: 5, 6 Defueled • MFB 1 and 2 de-energized AND Failure to restore power to at least one MFB within 15 minutes from the time of loss of both offsite and onsite AC power 2. AC POWER CAPABILITY TO ESSENTIAL BUSSES REDUCED TO A SINGLE SOURCE FOR GREATER THAN 15 MINUTES (BD 50) == OPERATING MODE: 1, 2, 3, 4 • AC power capability has been degraded to a single power source for > 15 minutes due to the loss of all but one of: Unit Normal Transformer Unit SU Transformer Another Unit SU Transformer CT4 CT5 (END)	1. LOSS OF ALL OFFSITE AC POWER AND LOSS OF ALL ONSITE AC POWER TO ESSENTIAL BUSSES (BD 51) == OPERATING MODE: 1, 2, 3, 4 • MFB 1 and 2 de-energized AND Failure to restore power to at least one MFB within 15 minutes from the time of loss of both offsite and onsite AC power 2. LOSS OF ALL VITAL DC POWER (BD 52) == OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of vital DC power to required DC busses as indicated by bus voltage less than 110 VDC AND Failure to restore power to at least one required DC bus within 15 minutes from the time of loss (END)	1. PROLONGED LOSS OF ALL OFFSITE POWER AND ONSITE AC POWER (BD 54) == OPERATING MODE: 1, 2, 3, 4 • MFB 1 and 2 de-energized AND SSF fails to maintain Hot Shutdown AND At least one of the following conditions exist: Restoration of power to at least one MFB within 4 hours is NOT likely OR Indications of continuing degradation of core cooling based on Fission Product Barrier monitoring (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. FIRES/EXPLOSIONS WITHIN THE PLANT (BD 57) ========== OPERATING MODE: All NOTE: Within the plant means Turbine Building, Auxiliary Building, Reactor Building, Keowee Hydro • Fire within the plant not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm • Unanticipated explosion within the plant resulting in visible damage to permanent structures/equipment 2. CONFIRMED SECURITY THREAT INDICATES POTENTIAL DEGRADATION IN THE LEVEL OF SAFETY OF PLANT (BD 58) ========== OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications • Discovery of bomb within plant protected area and outside security vital areas • Hostage/Extortion situation • Violent civil disturbance within the owner controlled area (END)	1. FIRE/EXPLOSION AFFECTING OPERABILITY OF PLANT SAFETY SYSTEMS REQUIRED TO ESTABLISH/MAINTAIN SAFE SHUTDOWN (BD 59) ========== OPERATING MODE: All NOTE: Only one train of a system needs to be affected or damaged in order to satisfy this condition. • Fire/explosions AND ONE OF THE FOLLOWING: Affected safety-related system parameter indications show degraded performance OR Plant personnel report visible damage to permanent structures or equipment required for safe shutdown 2. SECURITY EVENT IN A PLANT PROTECTED AREA (BD 60) ========== OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications • Intrusion into plant protected area by a hostile force • Bomb discovered in an area containing safety related equipment (END)	1. SECURITY EVENT IN A PLANT VITAL AREA (BD 61) ========== OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications • Intrusion into any of the following plant areas by a hostile force: Reactor Building Auxiliary Building Keowee Hydro • Bomb detonated in the following areas: • Keowee Hydro • Keowee Dam • ISFSI • Reactor Building • Auxiliary Building • SSF (END)	1. SECURITY EVENT RESULTING IN LOSS OF ABILITY TO REACH AND MAINTAIN COLD SHUTDOWN (BD 62) ========== OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications • Loss of physical control of the control room due to security event • Loss of physical control of the Aux Shutdown panel and the SSF due to a Security Event (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY

1. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING THE PROTECTED AREA (BD 64)
==========
OPERATING MODE: All
 	
• Tremor felt and valid alarm on the strong motion accelerograph
• Tornado striking within Protected Area Boundary
• Vehicle crash into plant structures/systems within the Protected Area Boundary
• Turbine failure resulting in casing penetration or damage to turbine or generator seals
 	
(CONTINUED)
	1. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING THE PLANT VITAL AREA (BD 69)
	==========
	OPERATING MODE: All
	
	• Tremor felt and seismic trigger actuates (0.05g)
	• Tornado, high winds, missiles resulting from turbine failure, vehicle crashes, or other catastrophic event
	
	AND ONE OF THE FOLLOWING:
	
	NOTE: Only one train of a safety-related system needs to be affected or damaged in order to satisfy these conditions.
	
	Visible damage to permanent structures or equipment required for safe shutdown of the unit
	OR
	Affected safety system parameter indications show degraded performance
	
	2. RELEASE OF TOXIC/FLAMMABLE GASES JEOPARDIZING SYSTEMS REQUIRED TO MAINTAIN SAFE OPERATION OR ESTABLISH/ MAINTAIN COLD SHUTDOWN (BD 71)
	==========
	OPERATING MODE: All
	
	• Report/detection of toxic gases in concentrations that will be life-threatening to plant personnel
	• Report/detection of flammable gases in concentrations that will affect the safe operation of the plant:
	  • Reactor Building
	  • Auxiliary Building
	  • Turbine Building
	  • Control Room
	
	(CONTINUED)
		1. CONTROL ROOM EVACUATION AND PLANT CONTROL CANNOT BE ESTABLISHED (BD 75)
		==========
		OPERATING MODE: All
		
		• Control Room evacuation has been initiated
		AND
		Control of the plant cannot be established from the Aux Shutdown Panel or the SSF within 15 minutes
		
		2. KEOWEE HYDRO DAM FAILURE (BD 76)
		==========
		OPERATING MODE: All
		
		• Imminent/actual dam failure (includes any of the following:
		  • Keowee Hydro Dam
		  • Little River Dam
		  • Dikes A, B, C, or D
		  • Intake Canal Dike
		
		3. OTHER CONDITIONS WARRANT DECLARATION OF SITE AREA EMERGENCY (BD 77)
		==========
		OPERATING MODE: All
		
		• Emergency Coordinator/EOF Director judgment
		
		(END)
			1. OTHER CONDITIONS WARRANT DECLARATION OF GENERAL EMERGENCY (BD 78)
			==========
			OPERATING MODE: All
			
			• Emergency Coordinator/EOF Director judgment indicates:
			
			Actual/imminent substantial core degradation with potential for loss of containment
			OR
			Potential for uncontrolled radionuclide releases that would result in a dose projection at the site boundary greater than 1000 mRem TEDE or 5000 mRem CDE Adult Thyroid
			
			(END)

INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING KEOWEE HYDRO (BD 66) OPERATING MODE: All • Reservoir elevation ≥ 807 feet with all spillway gates open and the lake elevation continues to rise • Seepage readings increase or decrease greatly or seepage water is carrying a significant amount of soil particles • New area of seepage or wetness, with large amounts of seepage water observed on dam, dam toe, or the abutments • Slide or other movement of the dam or abutments which could develop into a failure • Developing failure involving the powerhouse or appurtenant structures and the operator believes the safety of the structure is questionable 3. RELEASE OF TOXIC OR FLAMMABLE GASES DEEMED DETRIMENTAL TO SAFE OPERATION OF THE PLANT (BD 67) OPERATING MODE: All • Report/detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant • Report by local, county, state officials for potential evacuation of site personnel based on offsite event (CONTINUED)	3. TURBINE BUILDING FLOOD (BD 72) OPERATING MODE: All • Turbine Building flood requiring use of AP/1,2,3/A/1700/10, (Uncontrolled Flooding Of Turbine Building) 4. CONTROL ROOM EVACUATION HAS BEEN INITIATED (BD 73) OPERATING MODE: All • Evacuation of Control Room AND ONE OF THE FOLLOWING: Plant control IS established form the Aux Shutdown Panel or the SSF OR Plant control IS BEING established from the Aux Shutdown Panel or SSF 5. OTHER CONDITIONS WARRANT CLASSIFICATION OF AN ALERT (BD 74) OPERATING MODE: All • Emergency Coordinator judgment indicates that: Plant safety may be degraded AND Increased monitoring of plant functions is warranted (END)		
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4. OTHER CONDITIONS EXIST WHICH WARRANT DECLARATION OF AN UNUSUAL EVENT (BD 68) == OPERATING MODE: All • Emergency Coordinator determines potential degradation of level of safety has occurred (END)			
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4			

NOTE: IF Actual Dose Assessment **cannot** be completed within 15 minutes.
THEN The valid monitor reading should be used for Emergency Classification.

All RIA values are considered GREATER THAN or EQUAL TO

HOURS SINCE REACTOR TRIPPED	RIA 57 R/hr		RIA 58 R/hr*	
	Site Area Emergency	General Emergency	Site Area Emergency	General Emergency
0.0 - < 0.5	5.9E+003	5.9E+004	2.6E+003	2.6E+004
0.5 - < 1.0	2.6E+003	2.6E+004	1.1E+003	1.1E+004
1.0 - < 1.5	1.9E+003	1.9E+004	8.6E+002	8.6E+003
1.5 - < 2.0	1.9E+003	1.9E+004	8.5E+002	8.5E+003
2.0 - < 2.5	1.4E+003	1.4E+004	6.3E+002	6.3E+003
2.5 - < 3.0	1.2E+003	1.2E+004	5.7E+002	5.7E+003
3.0 - < 3.5	1.1E+003	1.1E+004	5.2E+002	5.2E+003
3.5 - < 4.0	1.0E+003	1.0E+004	4.8E+002	4.8E+003
4.0 - < 8.0	1.0E+003	1.0E+004	4.4E+002	4.4E+003

* RIA 58 is partially shielded

Assumptions used for calculation of high range in-containment monitors RIA 57 and 58:

1. Average annual meteorology (7.308 E^{-6} sec/m^3)
2. Design basis leakage (5.6 E^6 ml/hr)
3. One hour release duration
4. General Emergency PAGs are 1 rem TEDE and 5 rem CDE; Site Area Emergency determination is based on 10% of the General Emergency PAGs
5. Calculations for monitor readings are based on CDE because thyroid dose is limiting
6. No credit is taken for filtration
7. LOCA conditions are limiting and provide the more conservative reading

NOTE: This Initiating Condition is not intended to apply to anticipated temporary increases due to planned events (e.g.; incore detector movement, radwaste container movement, depleted resin transfers, etc.).

MONITOR NUMBER	UNITS 1, 2, 3	
	UNUSUAL EVENT 1000x NORMAL LEVELS mRAD/HR	ALERT mRAD/HR
RIA 7, Hot Machine Shop Elevation 796	150	≥ 5000
RIA 8, Hot Chemistry Lab Elevation 796	4200	≥ 5000
RIA 10, Primary Sample Hood Elevation 796	830	≥ 5000
RIA 11, Change Room Elevation 796	210	≥ 5000
RIA 12, Chem Mix Tank Elevation 783	800	≥ 5000
RIA 13, Waste Disposal Sink Elevation 771	650	≥ 5000
RIA 15, HPI Room Elevation 758	NOTE*	≥ 5000

NOTE: RIA 15 normal readings are approximately 9 mRad/hr on a daily basis. Applying 1000x normal readings would put this monitor greater than 5000 mRad/hr just for an Unusual Event. For this reason, an Unusual Event will <u>NOT</u> be declared for a reading less than 5000 mRad/hr.

1. List of Definitions and Acronymns

1.1 ALERT - Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

1.2 BOMB - A fused explosive device

1.3 CONDITION A - Failure is Imminent or Has Occurred - A failure at the dam has occurred or is about to occur and minutes to days may be allowed to respond dependent upon the proximity to the dam.

1.4 CONDITION B - Potentially Hazardous Situation is Developing - A situation where failure may develop, but preplanned actions taken during certain events (such as major floods, earthquakes, evidence of piping) may prevent or mitigate failure.

1.5 CIVIL DISTURBANCE - A group of ten (10) or more people violently protesting station operations or activities at the site.

1.6 EXPLOSION - A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. A sudden failure of a pressurized pipe/line could fit this definition.

1.7 EXTORTION - An attempt to cause an action at the station by threat of force.

1.8 FIRE - Combustion characterized by heat and light. Sources of smoke, such as slipping drive belts or overheated electrical equipment, do NOT constitute fires. Observation of flames is preferred but is NOT required if large quantities of smoke and heat are observed.

1.9 GENERAL EMERGENCY - Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guidelines exposure levels outside the Exclusion Area Boundary.

1.10 HOSTAGE - A person or object held as leverage against the station to ensure demands will be met by the station.

1.11 INTRUSION/INTRUDER - Suspected hostile individual present in a Protected Area without authorization.

1.12 INABILITY TO DIRECTLY MONITOR - Operational Aid Computer data points are unavailable or gauges/panel indications are NOT readily available to the operator.

1.13 PROTECTED AREA - Encompasses all Owner Controlled Areas within the security perimeter fence.

1.14 RUPTURED (As relates to Steam Generator) - Existence of Primary to Secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

1.15 SABOTAGE - Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment unavailable.

1.16 SAFETY-RELATED SYSTEMS AREA - Any area within the Protected Area which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

1.17 SIGNIFICANT TRANSIENT - An unplanned event involving one or more of the following:

(1) Automatic turbine runback > 25% thermal reactor power
(2) Electrical load rejection > 25% full electrical load
(3) Reactor Trip
(4) Safety Injection System Activation

1.18 SITE AREA EMERGENCY - Events are in process or have occurred which involve actual or likely major failures of plant functions needed for the protection of the public. Any releases are NOT expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels outside the Exclusion Area Boundary.

1.19 SELECTED LICENSEE COMMITMENT (SLC) - Chapter 16 of the FSAR

1.20 SITE BOUNDARY - That area, including the Protected Area, in which DPC has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius from the center of Unit 2).

1.21 TOXIC GAS - A gas that is dangerous to life or health by reason of inhalation or skin contact (e.g.; Chlorine)

1.22 UNCONTROLLED - Event is not the result of planned actions by the plant staff

1.23 UNPLANNED - An event or action is UNPLANNED if it is not the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

1.24 UNUSUAL EVENT - Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

1.25 **VALID** - An indication or report or condition is considered to be **VALID** when it is conclusively verified by: (1) an instrument channel check; or, (2) indications on related or redundant instrumentation; or, (3) by direct observation by plant personnel such that doubt related to the instrument's operability, the condition's existence, or the report's accuracy is removed. Implicit with this definition is the need for timely assessment.

1.26 **VIOLENT** - Force has been used in an attempt to injure site personnel or damage plant property.

1.27 **VISIBLE DAMAGE** - Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage: deformation due to heat or impact, denting, penetration, rupture,

Operating Modes Defined In Improved Technical Specifications

MODES

MODE	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER (a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	$\geq$0.99	> 5	NA
2	Startup	$\geq$0.99	$\leq$5	NA
3	Hot Standby	<0.99	NA	$\geq$250
4	Hot Shutdown (b)	< 0.99	NA	250 > T > 200
5	Cold Shutdown (b)	< 0.99	NA	$\leq$ 200
6	Refueling (c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

Facility: Oconee

Date of Examination: 07-10/21-00

Examination Level: **SRO-I**

Operating Test Number: 1

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Plant Parameter Verification	JPM CRO-040A (Bank-modified), Calculate Shutdown Margin with the Computer. (SRO ONLY) KA 2.1.7 [3.7/4.4] CFR 43.5/45.12/45.13 **Note:** This JPM will be conducted with B.1 Section JPM CRO-012, (Recovery of dropped rod)
	Plant Parameter Verification	JPM NRC-005 (New) – Reactor Power Imbalance - Improved Tech Specs/COLR. KA 2.1.11 [3.0/3.8] CFR 43.2/45.13
A.2	Surveillance Testing	JPM NRC-004 (New) – Perform PT/1/A/0600/001, Enclosure 13.16, ICCM Subcooling Monitor Check. KA 2.2.12 [3.0/3.4] CFR 41.10/45.13, CFR 43.5/45.12/45.13
A.3	Radiation Control	SRO/RO – 2 Questions *Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure [CFR: 43.4/45.10] G2.3.10 (2.9/3.3)* Knowledge of 10 CFR 20 and related facility radiation control requirements. KA 2.3.1 [2.6/3.0] **Note:** These questions to be conducted with B.2 section JPM NLO-040 or 17.
A.4	Emergency Plan Implementation	SRO - JPM – Scenario event classification and protective action recommendations and/or classification upgrade. (SRO ONLY) KA 2.4.41 [2.3/4.1] CFR 43.5/45.11 **Note:** E-Plan classification to be conducted during C section simulator exams

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

CRO-40A/ADMIN A.1

CALCULATE SDM
WITH A DROPPED CONTROL ROD

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

CALCULATE SDM WITH A DROPPED CONTROL ROD

Alternate Path:

Yes, determine that 1% SDM does not exist and boration is required within 15 minutes.

Facility JPM #:

N/A

K/A Rating(s):

Gen 2.1.7 3.7/4.4

Task Standard:

PT/1/A/1103/15, Reactivity Balance Procedure is used to verify > 1% SDM with one inoperable (dropped) CR within 1 hour. Determine that 1% SDM does not exist and boration is required within 15 minutes.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/1103/15,Reactivity Balance Procedure
AP/1/A/1700/15, Dropped Control Rods
Improved Technical Specifications
- 3.1.4, Control Rod Group Alignment Limits
- 3.2.1, Regulating Rod Position Limits

Validation Time: 10 min. **Time Critical:** YES

===

Candidate: ______________________________________ Time Start : ________
NAME Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________________ ___________________________________/________
NAME SIGNATURE DATE

===

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC # SNAP _____
2. Go to run, acknowledge alarms.
3. Freeze simulator.
4. Place simulator in run when directed by the examiner.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

PT/1/A/1103/015, Reactivity Balance Procedure
OP/0/A/1105/009, Control Rod Drive System
Improved Technical Specifications
3.1.4, Control Rod Group Alignment Limits
3.2.1, Regulating Rod Position Limits

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.5.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

TIME CRITICAL

START TIME: _______

STEP 1:	Within one hour verify > 1% SDM with allowance to the inoperable control rod. Perform PT/1/A/1103/15, Reactivity Balance Procedure.	**CRITICAL STEP** ___ SAT ___ UNSAT
STANDARD:	Obtain copy of PT/1/A/1103/15, Reactivity Balance Procedure.	
COMMENTS:		
STEP 2	Determine proper enclosure to use.	**CRITICAL STEP** ___ SAT ___ UNSAT
STANDARD:	Enclosure 13.20, Shutdown Margin at Power, is chosen.	
COMMENTS:		
STEP 3:	Use Enclosure 13.21, Rod Position Limits at Power, 1 Inoperable Rod or 1 Dropped Rod – 4 Pump Flow. Verify available SDM is ≥ 1% ΔK/K by verifying that the control rod position and power level are within the acceptable region or the Restricted Region on the appropriate curve for the number of RCPs and Inoperable rods in Enclosure 13.21, Rod Position limits at Power.	**CRITICAL STEP** ___ SAT ___ UNSAT
STANDARD:	SDM is determined to be is ≤ 1% ΔK/K.	
COMMENTS:		

STEP 4: Appropriate actions are taken per ITS 3.1.4, 3.1.5 and 3.2.1. STANDARD: Refer to ITS 3.1.4, 3.1.5 and 3.2.1 and determine that initiation of boration to restore SDM to within limits is required within 15 minutes. ***CUE: Inform student that an RO is commencing boration.*** COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

TIME STOP: _______

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	Step is necessary for the operator to select PT/1/A/1103/15, Reactivity Balance Calculations procedure to obtain correct enclosure to complete step three correctly.
2	Step is necessary for the operator to select to the Enclosure 13.20, Shutdown Margin at Power.
3	Step is necessary, the operator must interpret the 4 RCP curve to ensure adequate SDM.
4	Step is necessary, initiation of boration must be occur within 15 minutes to restore SDM.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.5.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

TIME CRITICAL

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

You are the Unit 1 OATC and Group 6 Rod 3 has dropped into the core. Reactor power is stable at ≈ 55%. AP/1/A/1700/15, Dropped Control Rods, is complete up to step 5.5.

INITIATING CUES:

The SRO in the Control Room directs you to continue with AP/1/A/1700/15, Dropped Control Rods.

TIME CRITICAL

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NRC-005/ADMIN A.1

REACTOR POWER IMBALANCE
Improved Technical Specifications/COLR

CANDIDATE ______________________________

EXAMINER ______________________________

**REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE**

Task:

Axial Power Imbalance

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

Gen 2.1.11 3.07/3.8

Task Standard:

Perform power imbalance within limits verification.

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and Section 12.3
PT/0/A/1103/019, Backup Incore Detector System
Core Operating Limits Report

Validation Time: 25 min. **Time Critical:** NO

==

Candidate: ______________________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ______________________________ __________________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

Tools/Equipment/Procedures Needed:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and Section 12.3
PT/0/A/1103/019, Backup Incore Detector System
Core Operating Limits Report

READ TO OPERATOR

DIRECTIONS TO STUDENT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

The Unit has been operating at 100% power for 2 weeks.
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 has been completed up to page 6, Axial Power Imbalance Operating Limits.

INITIATING CUE:

The SRO directs you to perform the Axial Power Imbalance Operating Limits verification.
The Reactor calculation package is NOT running.

START TIME: _______

STEP 1: Verify Power imbalance within operational alarm limit in COLR when > 40% RTP. IF Reactor calculation package is NOT running on computer, refer to Section 12.3. STANDARD: When told Reactor Calculation package not running, refer to Section 12.3. ***CUE: Tell candidate that the Reactor calculation package is not running.*** COMMENTS:	___ SAT ___ UNSAT
STEP 2: Axial Imbalance shall **NOT** exceed appropriate limit curve in COLR. IF axial imbalance limit is exceeded, take immediate corrective action to IF an acceptable imbalance is **NOT** achieved within 2 hours, reactor power shall be reduced until imbalance limits are met. Refer to TS 3.2.2. STANDARD: Candidate obtains the correct limit curve in COLR. This curve is located on page 12 of 31 (Oconee 1 Cycle 19) (Oconee 2 Cycle 18) (Oconee 3 Cycle 18) **NOTE: Later in JPM when imbalance calculation is made with the Incores a different enclosure from the COLR will be used.** ***CUE: Only Imbalance Surveillance is required for this JPM. Step 12.3.2 is not required.*** COMMENTS:	___ SAT ___ UNSAT

<table>
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<u>STEP 3</u>:

Order of preference of measurement systems to determine axial imbalance and quadrant power tilt is as follows:

A. Incore Detectors (Computer Reactor Calculation Package).

B. Outcore Detectors (Power Range Outcore Detectors).

C. Backup Incore Detectors. Refer to PT/10/A/1103/019 (Backup Incore Detector System).

CUE: The Backup Incore detectors will be used for this determination.

<u>STANDARD</u>:

Candidate refers to PT/10/A/1103/019 (Backup Incore Detector System).

<u>COMMENTS</u>:

</td>
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CRITICAL STEP

___ SAT

___ UNSAT

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STEP 4: Verification of minimum Incore operability. **NOTE: Backup Incore Chart "A" points and information provided to the student.** STANDARD: **NOTE: Give student Backup Incore Chart "A" data sheet.** **CUE: Inform candidate that all points on Backup Incore Chart "A" are operable (no points are off scale or contain a note).** COMMENTS:	___ SAT ___ UNSAT
STEP 5: 12.2.1 Verify the reactor has been at steady state conditions (± 2% FP) for at least 30 minutes. STANDARD: The Candidate determines is reactor power is steady. ***CUE: Reactor power has been at 100% power for the past 2 weeks.*** COMMENTS:	___ SAT ___ UNSAT

STEP 6: Calculate axial imbalance per Enclosure 13.3 using operable recorder points identified on Enclosure 13.1. STANDARD: The candidate refers to and obtains a copy of Enclosures 13.1 and 13.3 The candidate performs calculation per Enclosure 13.3. **NOTE: Refer to completed enclosure 13.3.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: Verify the calculated axial imbalance does not exceed the backup incore limits per 11.1. STANDARD: The candidate verifies the calculated axial imbalance does not exceed the backup incore limits per 11.1, (-18.7 / +18.7) the current Core Operating Limits Report (COLR) on the Backup Incore Setpoint Column of the (Error-Adjusted) "Operational Power Imbalance Setpoints" Table. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: If either step 12.2.4 or 12.2.5 cannot be satisfied, notify the Unit Supervisor and take appropriate actions described in the applicable Technical Specifications as listed below Quadrant Power Tilt - ITS 3.2.3 Axial Power Imbalance - ITS 3.2.2 ***CUE: Inform candidate that for this JPM only imbalance will be checked.*** STANDARD: Candidate determines that step 12.2.4 is satisfied. COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
3	Step is necessary, because reference to the must use Backup Incore System procedure must be used to determine imbalance.
6	Step is necessary, because calculation is needed to determine imbalance.
7	Step is necessary, because imbalance must be compared to COLR to verify within limits.

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

The Unit has been operating at 100% power for 2 weeks.
PT/1/A/0600/001, Periodic Instrument Surveillances, Enclosures 13.1 has been completed up to page 6, Axial Power Imbalance Operating Limits.

INITIATING CUE:

The SRO directs you to perform the Axial Power Imbalance Operating Limits verification.
The reactor calculation package is NOT running.

Enclosure 13.1
Required Backup Recorder Points For Calculating Axial Power Imbalance

* UNIT 1 ONLY *

From the following list of allowed point groupings for calculating axial power imbalance, three sets for which the points indicated are all operable and in calibration must be available.

NOTES: "A" recorder is on the Control Board; "B" is on the vertical board.
A set of recorder points for imbalance calculations consist of three operable points grouped together.

Any of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	3	G9	6		B	8	L6	6
A	2	G9	4		B	7	L6	4
A	1	G9	2		B	6	L6	2
A	22	D5	6		B	19	N4	6
A	24	D5	4		B	18	N4	4
A	14	D5	2		B	17	N4	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	6	E9	6		B	4	K5	6
A	5	E9	4		B	5	K5	4
A	4	E9	2		B	10	K5	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	18	F13	6		B	22	O6	6
A	15	F13	4		B	21	O6	4
A	13	F13	2		B	20	O6	2

Recorded By ______________________ Date ________________

BACKUP INCORE CHART "A"		
Point #	**%**	**Location**
1	195.1	G09-L2
2	203.0	G09-L4
3	196.1	G09-L6
4	214.1	E09-L2
5	226.6	E09-L4
6	209.6	E09-L6
7	189.4	L06-L4
8	196.8	L06-L6
9	185.7	M09-L2
10	180.7	K05-L2
11	187.5	G11-L2
12	179.7	E07-L2
13	212.3	F13-L2
14	213.2	D04-L2
15	210.4	F13-L4
16	209.5	F03-L6
17	212.5	N04-L2
18	197.6	F13-L6
19	196.7	N04-L6
20	199.3	O06-L2
21	200.2	O06-L4
22	199.4	O06-L6
23	196.8	D05-L6
24	197.7	D05-L4

Enclosure 13.3
Axial Power Imbalance Calculation Sheet

13.3.1 Using Enclosure 13.1 choose a set of backup recorder points for which all points are operable and record the point identifications (recorder, point number, detector location, and level) on the blanks provided below.

13.3.2 Record the backup recorder readings on the blank provided below.

13.3.3 Record current % FP on the blanks provided below using the priority of thermal power indications listed in PT/0/A/0600/001 (usually either power range NIs or from the results of PT/0/A/0205/05, Thermal Power Calculation).

13.3.4 Calculate imbalance for each of the three detector strings using the following formula:

$$\text{IMB} = \frac{\text{R (level 6 or 5)} - \text{R (level 2 or 3)}}{\text{R (level 6 or 5)} + \text{R (level 4)} + \text{R (level 2 or 3)}} \times \%\text{ FP}$$

where R = detector reading

13.3.5 The imbalance from the backup recorders is the average for the three detector strings, calculated as indicated below:

	RECORDER ID	POINT #	DETECTOR LOCATION	DETECTOR LEVEL	DETECTOR READING (R)
I	A	3	609-L6	6 or 5	196.1
	A	2	609-L4	4	203.0
	A	1	609-L2	2 or 3	195.1
II	A	6	E9-L6	6 or 5	209.6
	A	5	E9-L4	4	226.6
	A	4	E9-L2	2 or 3	214.1
III	A	18	F13-L6	6 or 5	197.6
	A	16	F13-~~L6~~L4	4	210.4
	A	13	F13-L2	2 or 3	212.3

$\text{IMB}_{\text{I}} = \frac{(196.1 - 195.1)}{(196.1 + 203.0 + 195.1)} \times 100 \%\text{FP} = .17 \%\text{IMB}$

$\text{IMB}_{\text{II}} = \frac{(209.6 - 214.1)}{(209.6 + 226.6 + 214.1)} \times 100 \%\text{FP} = -.69 \%\text{IMB}$

$\text{IMB}_{\text{III}} = \frac{(197.6 - 212.3)}{(197.6 + 210.4 + 212.3)} \times 100 \%\text{FP} = -2.37 \%\text{IMB}$

TOTAL -2.89 %IMB

AVERAGE IMBALANCE = TOTAL/3 = -.96 %IMB

Calculated by ______________ Date/Time ______________ Verified by ______________ Date/Time ______________

BACKUP INCORE CHART "A"		
Point #	%	Location
1	195.1	G09-L2
2	203.0	G09-L4
3	196.1	G09-L6
4	214.1	E09-L2
5	226.6	E09-L4
6	209.6	E09-L6
7	189.4	L06-L4
8	196.8	L06-L6
9	185.7	M09-L2
10	180.7	K05-L2
11	187.5	G11-L2
12	179.7	E07-L2
13	212.3	F13-L2
14	213.2	D04-L2
15	210.4	F13-L4
16	209.5	F03-L6
17	212.5	N04-L2
18	197.6	F13-L6
19	196.7	N04-L6
20	199.3	O06-L2
21	200.2	O06-L4
22	199.4	O06-L6
23	196.8	D05-L6
24	197.7	D05-L4

Enclosure 13.1
Required Backup Recorder Points For Calculating Axial Power Imbalance

* UNIT 1 ONLY *

From the following list of allowed point groupings for calculating axial power imbalance, three sets for which the points indicated are all operable and in calibration must be available.

NOTES: "A" recorder is on the Control Board; "B" is on the vertical board.
A set of recorder points for imbalance calculations consist of three operable points grouped together.

Any of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	3	G9	6		B	8	L6	6
A	2	G9	4		B	7	L6	4
A	1	G9	2		B	6	L6	2
A	22	D5	6		B	19	N4	6
A	24	D5	4		B	18	N4	4
A	14	D5	2		B	17	N4	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	6	E9	6		B	4	K5	6
A	5	E9	4		B	5	K5	4
A	4	E9	2		B	10	K5	2

Any one, but not more than one, of the following strings may be used:

Recorder Number	Point Number	Core Location	Level		Recorder Number	Point Number	Core Location	Level
A	18	F13	6		B	22	O6	6
A	15	F13	4		B	21	O6	4
A	13	F13	2		B	20	O6	2

Recorded By ______________________ Date ________________

Enclosure 13.3
Axial Power Imbalance Calculation Sheet

13.3.1 Using Enclosure 13.1 choose a set of backup recorder points for which all points are operable and record the point identifications (recorder, point number, detector location, and level) on the blanks provided below.

13.3.2 Record the backup recorder readings on the blank provided below.

13.3.3 Record current % FP on the blanks provided below using the priority of thermal power indications listed in PT/0/A/0600/001 (usually either power range NIs or from the results of PT/0/A/0205/05, Thermal Power Calculation).

13.3.4 Calculate imbalance for each of the three detector strings using the following formula:

$$IMB = \frac{R\text{ (level 6 or 5)} - R\text{ (level 2 or 3)}}{R\text{ (level 6 or 5)} + R\text{ (level 4)} + R\text{ (level 2 or 3)}} \times \%\,FP$$

where R = detector reading

13.3.5 The imbalance from the backup recorders is the average for the three detector strings, calculated as indicated below:

	RECORDER ID	POINT #	DETECTOR LOCATION	DETECTOR LEVEL	DETECTOR READING (R)	
I	______	______	______	6 or 5	______	$IMB_I = \frac{(\_\_\_\_ - \_\_\_\_)}{(\_\_\_\_ + \_\_\_\_ + \_\_\_\_)} \times$ ____ %FP = ____ %IMB
	______	______	______	4	______	
	______	______	______	2 or 3	______	
II	______	______	______	6 or 5	______	$IMB_{II} = \frac{(\_\_\_\_ - \_\_\_\_)}{(\_\_\_\_ + \_\_\_\_ + \_\_\_\_)} \times$ ____ %FP = ____ %IMB
	______	______	______	4	______	
	______	______	______	2 or 3	______	
III	______	______	______	6 or 5	______	$IMB_{III} = \frac{(\_\_\_\_ - \_\_\_\_)}{(\_\_\_\_ + \_\_\_\_ + \_\_\_\_)} \times$ ____ %FP = ____ %IMB
	______	______	______	4	______	
	______	______	______	2 or 3	______	

TOTAL ____ %IMB

AVERAGE IMBALANCE = TOTAL/3 = ______ %IMB

Calculated by ____________ Date/Time ________________ Verified by ____________ Date/Time ______________

(R06-97)

SR
Sum
NRC
115
JPP
JMB

Duke Power Company

PROCEDURE PROCESS RECORD

(1) ID No PT/1/A/0600/001

Revision No 217

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Periodic Instrument Surveillance

(4) Prepared By William M. Buchanan (Signature) [signature] Date 04/11/00

(5) Requires 10CFR50.59 evaluation?

☐ Yes (New procedure or revision with major changes)
☒ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By [signature] (QR) Date 4/12/00

Cross-Disciplinary Review By ____ (QR)NA [initials] Date ____

Reactivity Mgmt. Review By ____ (QR)NA [initials] Date ____

(7) Additional Reviews

Reviewed By (IT) Alan Sweeney (Time Sync Only) Date 4/10/00

Reviewed By ____ Date ____

(8) Temporary Approval *(if necessary)*

By ____ (SRO/QR) Date ____

By ____ (QR) Date ____

(9) Approved By [signature] Date 4/12/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

(11) Date(s) Performed ____

Work Order Number (WO#) ____

COMPLETION

(12) Procedure Completion Verification:

☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ____ Date ____

(13) Procedure Completion Approved ____ Date ____

(14) Remarks *(Attach additional pages, if necessary)*

APR 2000
Received
Operator
Training

Duke Power Company Oconee Nuclear Station **Periodic Instrument Surveillance** **Continuous Use**	Procedure No. PT/1/A/0600/001 Revision No. 217 Electronic Reference No. OX002WAS

Periodic Instrument Surveillance

1. Purpose

1.1 To periodically verify proper operation of various instruments and systems.

2. References

2.1 Technical Specifications (TS)

2.2 DPC/Oconee Nuclear Station Core Operating Limits Report (COLR)

2.3 UFSAR Chapter 16 Selected Licensee Commitments (SLC)

3. Time Required

3.1 90 minutes per shift

4. Prerequisite Tests

None

5. Test Equipment

None

6. Limits And Precautions

6.1 This procedure controls activities that have the potential to affect reactivity. Major changes to this procedure shall be reviewed by a Qualified Reviewer to determine if a cross-disciplinary review for Reactivity Management concerns is needed as required by NSD 304 (Reactivity Management).

6.2 Failure to meet the required conditions may be a violation of TS or SLC. If so, it must be reported immediately to OPS Duty Person, Superintendent of Operations, or Station Manager.

6.3 OP/0/A/1103/020 (Loss Of Computer) should be referred to upon a loss of computer or loss of a computer point or function needed for a surveillance.

6.4 When changing Modes, <u>ALL</u> TS, SLC, and Surveillance Requirements (SR) prior to changing Modes of operation shall be performed. <u>ALL</u> TS, SLC, and Surveillance Requirements (SR) (Semi-Daily, Daily, Weekly, and Monthly) must be initialed prior to changing modes. When the surveillance is completed, initial in the block for the shift you are working, regardless of the time of day, week or month, even if there is not a (N), (D), 1st day of month, etc., indicated in the block.

7. Required Unit Status

7.1 Surveillance of instrumentation per Enclosure "Mode 1 & 2" required:

Prior to entering Mode 2 from Mode 3

AND

During operation in either Mode 1 or Mode 2.

7.2 Surveillance of instrumentation per Enclosure "Mode 3" required:

Prior to entering Mode 3 from Mode 4

AND

During operation in Mode 3.

7.3 Surveillance of instrumentation per Enclosure "Mode 4" required:

Prior to entering Mode 4 from Mode 5

AND

During operation in Mode 4.

7.4 Surveillance of instrumentation per Enclosure "Mode 5" required:

Prior to entering Mode 5 from Mode 6

AND

During operation in Mode 5.

7.5 Surveillance of instrumentation per Enclosure "Mode 6" required:

Prior to entering Mode 6 from No Mode

AND

During operation in Mode 6.

7.6 Surveillance of instrumentation per Enclosure "No Mode" required:

During operation in No Mode.

7.7 Surveillance per Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)" is required when 1LT-5 < 50" and core contains <u>any</u> irradiated fuel.

7.7.1 Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)" is performed in parallel with Enclosure "Mode 5" or "Mode 6".

8. Prerequisite System Conditions

None

9. Test Method

9.1 Component checks will be made according to information given in the following enclosures:

Enclosure "Modes 1 & 2"

Enclosure "Mode 3"

Enclosure "Mode 4"

Enclosure "Mode 5"

Enclosure "Mode 6"

Enclosure "No Mode"

Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)".

10. Data Required

10.1 Data requirements specified in Enclosure "Modes 1 & 2", Enclosure "Mode 3", Enclosure "Mode 4", Enclosure "Mode 5", Enclosure "Mode 6", Enclosure "No Mode", or Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)".

11. Acceptance Criteria

11.1 Systems or components meet TS or SLC requirements applicable to surveillance step.

11.2 Any discrepancy noted during performance of this test shall show corrective action taken.

12. Procedure

12.1 As required, perform component checks according to schedule specified in the following enclosures:

- Enclosure "Modes 1 & 2"
- Enclosure "Mode 3"
- Enclosure "Mode 4"
- Enclosure "Mode 5"
- Enclosure "Mode 6"
- Enclosure "No Mode"
- Enclosure "Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)"

12.1.1 For surveillances required per TS and/or SLC: (D) indicates between 0730-1030 hours; (N) indicates between 1930-2230 hours.

- Example: "N/Tuesday" means signoff shall be completed sometime between Tuesday at 1930 hours and Tuesday at 2230 hours.

12.1.2 For surveillances NOT required per TS and/or SLC: (D) indicates between 0700-1900 hours; (N) indicates between 1900-0700 hours.

- Example: "N/Tuesday" means signoff shall be completed sometime between Tuesday at 1900 hours and Wednesday at 0700 hours.

12.1.3 Instrument operation may be checked either by reading appropriate recorder, gauge, etc., or by selecting computer point ID, where applicable.

12.2 IF any component supplying input to RPS or ES channels fails to meet its Required Condition (i.e., is "out of tolerance"), initiate the following action:

12.2.1 ES Instrument

A. Check other analog channels to see if any other channel is tripped.

NOTE:	For determining appropriate TS condition a tripped channel is considered inoperable.

B. IF no other analog channel is tripped, trip affected analog channel by placing instrument channel for affected parameter (RC pressure or RB pressure) in "TEST-OPERATE". Affected parameter(s) should be left in "TEST-OPERATE" until channel input(s) is repaired.

C. IF any other analog channel is tripped, do NOT trip affected channel. Initiate immediate action to have instrument repaired. Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits).

D. Immediate shutdown may be required.
- Refer to TS 3.3.5.
- Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits).

12.2.2 RPS Instrument

A. IF no other RPS channel is in MANUAL BYPASS or no other RPS channel contains a DUMMY BISTABLE, place affected RPS channel in MANUAL BYPASS. Initiate action to have instrument channel repaired.

NOTE:	For determining appropriate TS condition a tripped channel is considered inoperable.

B. IF another RPS channel is in MANUAL BYPASS or contains a DUMMY BISTABLE, trip affected RPS channel by placing any one of its instrument channels in "TEST-OPERATE" (for STAR Modules select "TEST"). Affected parameter(s) should be left in "TEST-OPERATE" (or "TEST") until channel input(s) is repaired. Initiate immediate action to have instrument channel repaired.

C. IF affected RPS channel is already in MANUAL BYPASS, do NOT trip affected RPS channel. Initiate action to have instrument channel repaired.

D. IF affected RPS channel contains a DUMMY BISTABLE and no other RPS channel is in MANUAL BYPASS, place affected RPS channel in MANUAL BYPASS. TS allows any one RPS channel to contain more than one DUMMY BISTABLE.

E. IF another RPS channel is tripped, do NOT trip affected RPS channel. Initiate immediate action to have instrument channel repaired. Tripping affected RPS channel will cause a reactor trip.

F. Immediate shutdown may be required.
- Refer to TS 3.3.1
- Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits)

12.2.3 Priority of Power Indications to Use for Surveillance.
(A = highest priority, G = lowest priority)

A. OAC Calculated Thermal Power Best.

- O1P2037 (Core Thermal Power Best (60 min avg)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0588 (Core Thermal Power Best (10 min. avg.)) - Transient in last 60 minutes or O1P2037 unavailable.
- O1P0889 (Core Thermal Power Best (snapshot)) - Transient in progress or O1P2037 and O1P0588 unavailable.

B. OAC Calculated Thermal Power Secondary (if above ≈ 25% power).

- O1P0587 (Core Thermal Power Secondary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0888 (Core Thermal Power Secondary (snapshot)) - Transient within the last 60 minutes or O1P0587 unavailable.

C. OAC Calculated Thermal Power Primary (if below ≈ 25% power).

- O1P0576 (Core Thermal Power Primary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0887 (Core Thermal Power Primary (snapshot)) - Transient within the last 60 minutes or O1P0576 unavailable.

D. OAC Calculated Thermal Power ΔT.

- O1P0575 (Core Ther Pwr From Delta Temp (10 min. avg.)) - Steady State Ops. (i.e., no transient in last 10 minutes)
- O1P0326 (Core Thermal Power From Delta T (snapshot)) - Transient within the last 10 minutes or O1P0575 unavailable.

E. Alternate method for determining (%) Reactor Power

$$\frac{NI\text{-}5 + NI\text{-}6 + NI\text{-}7 + NI\text{-}8}{4} = \%\ Rx\ Power\ (Avg)$$

F. Hand-Calculated Thermal Power ΔT using Enclosure "Loop ΔT Vs Reactor Power".

G. Thermal Power from Nuclear Engineering using PT/0/A/0205/002 (Thermal Power Calculation).

12.3 Reactor Power Axial Imbalance and Quadrant Power Tilt

12.3.1 Axial Imbalance shall **NOT** exceed appropriate limit curve in COLR..

A. **IF** axial imbalance limit is exceeded, take immediate corrective action to achieve an acceptable imbalance.

B. **IF** an acceptable imbalance is **NOT** achieved within 2 hours, reactor power shall be reduced until imbalance limits are met. Refer to TS 3.2.2.

12.3.2 Quadrant Power Tilt (QPT) shall **NOT** exceed appropriate positive (+) limit in COLR.

A. **IF** QPT limit is exceeded, take immediate corrective action to achieve an acceptable QPT. Refer to TS 3.2.3. .

B. Alternate method for determining QPT:

$$QPT = 100\left[\frac{\text{power in any quadrant}}{\text{Avg. power of all quadrants}} - 1\right]$$

12.3.3 Order of preference of measurement systems to determine axial imbalance and quadrant power tilt is as follows:

A. Incore Detectors (Computer Reactor Calculation Package).

B. Outcore Detectors (Power Range Outcore Detectors).

C. Backup Incore Detectors. Refer to PT/0/A/1103/019 (Backup Incore Detector System).

12.3.4 **IF** at least one power range outcore detector is **NOT** operable in each quadrant, outcore detectors shall **NOT** be used to measure axial imbalance or quadrant power tilt.

12.3.5 **IF** Outcore Detectors (Power Range Outcore Detectors) are needed for tilt calculations, contact Rx Engineering group to perform PT/0/A/1103/018 (Excore Tilt Calculations).

12.3.6 **IF** Outcore Detectors (Power Range Outcore Detectors) are needed for imbalance calculations, refer to the following alternate method for determining (%) Reactor Power Axial Imbalance:

$$\frac{\text{NI-5* + NI-6* + NI-7* + NI-8*}}{4} = \text{\% Imbalance (Avg.)}$$

* Use Imbalance CR gauges reading for each NI.

12.3.7 **IF** Reactor Calculations package is **NOT** running, verify minimum incore detector operability requirements are met. Refer to PT/0/A/1103/019 (Backup Incore Detector System).

NOTE: "Steady Conditions" defined as: Operating at a constant power level with no rod motion due to xenon and no plans to change power level in next 24 hours.

12.4 **WHEN** operating at a steady condition above 40% FP:

12.4.1 Control Rods should be positioned at or above dashed vertical lines designating Steady State Operating Bands in COLR.

A. Maneuvering restrictions on Control Rod and APSR movement in OP/1/A/1102/004 (Operation At Power) have priority over 24 hour time limit to resume operation in Steady State Operating Bands.

B. **IF** Control Rod position limits are exceeded, (i.e., operating in restricted region), corrective action shall be taken immediately to achieve an acceptable control rod position. TS 3.2.1 requires an acceptable control rod position be attained within 2 hours.

12.4.2 APSRs should be positioned as required per Enclosure "Required Group 8 Position" of PT/1/A/1103/015 (Reactivity Balance Procedure).

12.4.3 **IF** plant operating conditions or imbalance control requirements prevent steady operation within Control Rod Steady State Operating Bands, contact Systems Engineering/Reactor Group.

12.5 SASS (Smart Automatic Signal Selector) Auto Operation

12.5.1 SASS for Pzr level looks at Pzr level 1, 2, or 3. If level 1 or 2 fails, SASS will AUTO swap to Pzr level 3.

12.5.2 <u>IF</u> "AUTO" light is off, "MISMATCH" light is on, and "TRIP 'A'" or "TRIP 'B'" light is on, a SASS trip has occurred.

A. Controlling signal will be signal which does <u>**NOT**</u> have a "TRIP" light illuminated.

NOTE:	Failure to swap switch to valid signal could result in failed signal feeding through if SASS is reset before signal is repaired.

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbuttons for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.5.3 <u>IF</u> "AUTO" light is off and "MISMATCH" light is on, a mismatch has occurred.:

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.5.4 Initiate a Work Request to repair faulty signal.

12.5.5 Following repair of faulty signal, reset SASS by pushing "RESET" button. The following should occur:

A. SASS should swap to "AUTO". "AUTO" light should be illuminated, "TRIP 'A' or "TRIP 'B'" light and "MISMATCH" light should be off.

B. Controlling signal should remain unchanged.

12.6 SASS (Smart Automatic Signal Selector) Manual Operation

12.6.1 IF "MISMATCH" light is on and "TRIP 'A'" or "TRIP 'B'" light is on, a SASS trip has occurred.

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.6.2 IF "MISMATCH" light is on, a mismatch has occurred.:

A. Controlling signal will be signal selected from CR keyswitch (for parameters in ICS Cabinet #8).

B. Select valid signal as controlling signal by positioning CR keyswitch or pushbutton for Pzr level to valid signal (for parameters in ICS Cabinet #8).

12.6.3 Initiate a Work Request to repair faulty signal.

12.6.4 Following repair of faulty signal, reset SASS by pushing "RESET" button. The following should occur:

A. SASS should swap to "AUTO". "AUTO" light should be illuminated, "TRIP 'A'" or "TRIP 'B'" light and "MISMATCH" light should be off.

B. Controlling signal should remain unchanged.

12.7 AMSAC/DSS

12.7.1 Refer to the following indications to determine normal status of AMSAC/DSS:

- AMSAC CH 1 and CH 2 **NOT** actuated

  (O1D2928, O1D2929, 1SA-8 D-5/D-8)

- DSS CH 1 and CH 2 **NOT** actuated

  (O1D2930, O1D2931, 1SA-8 C-9/C-10)

- AMSAC/DSS CH 1 and CH 2 **NOT** bypassed

  (O1D2932, O1D2933, Indicating Lights on 1UB1)

- AMSAC/DSS Enabled

  (Indicating Light on 1UB1)

- AMSAC/DSS CH 1 **AND** CH 2 UPS Normal

  (O1D2934, O1D2935)

- "Sy Max" Programmable Controllers

CH 1 AMSAC/DSS	CH 2 AMSAC/DSS
RUN Light (ON)	RUN Light (ON)
HALT Light (OFF)	HALT Light (OFF)

12.7.2 AMSAC/DSS UPS (Uninterruptable Power Supply) has been upgraded with new firmware.

A. UPS will generate an alarm if noise is encountered on its power supply. However, it will automatically re-assess input power supply quality and, if transient has passed, it will reset and clear its alarm.

B. UPS will still generate UPS Trouble alarm. However, it will be more likely to clear automatically without operator intervention.

C. **IF** UPS Trouble alarm does **NOT** automatically reset, issue a Work Request.

12.7.3 IF all of the following conditions are met, AMSAC/DSS may be considered operable:

A. Surveillance requirements of SLC 16.7.2 (Anticipated Transients Without Scram) are satisfied.

B. AMSAC/DSS CH 1 and AMSAC/DSS CH 2 are enabled.

C. AMSAC/DSS CH 1 AND AMSAC/DSS CH 2 are capable of generating intended EFDW start signals, control rod drop signals, turbine trip signal, and TBV setpoint shift signal.

- To satisfy these criteria, all AMSAC/DSS circuitry (including input pressure switches/pressure transmitters, electrical isolation devices, logic circuits, programmable controllers, and uninterruptible power supplies) shall be functional and properly calibrated.

D. "Sy Max" Programmable Controllers "RUN" Lights (ON) and "HALT" Lights (OFF) for AMSAC/DSS CH 1 and AMSAC/DSS CH 2.

12.7.4 Inability of EFDW pumps, turbine trip circuit, or control rods to respond to an AMSAC/DSS signal does NOT constitute inoperability of AMSAC/DSS system. These malfunctions are governed by applicable TS.

- TS 3.7.5 and TS 3.3.14 for inoperable Emergency Feedwater Pumps or existing Initiation Circuitry.
- TS 3.3.15 for inoperable Turbine Stop Valve closure circuitry.
- TS 3.1.4 for inoperable control rod(s).

12.7.5 IF one or both channels of AMSAC/DSS are inoperable AND reactor is critical, refer to SLC 16.7.2. Notify Compliance of inoperabilities extending beyond seven days.

12.7.6 IF any AMSAC/DSS channel is inoperable or generates an invalid trip signal, bypass both AMSAC/DSS channels from control panel in AHU Room located on 6th floor above Units 1 & 2 CR.

A. IF reactor is critical, declare AMSAC/DSS system inoperable AND refer to SLC 16.7.2. Initiate a Work Request to repair affected channel.

12.7.7 WHEN AMSAC/DSS channel has been repaired, return AMSAC/DSS channels to service per Enclosure "Return To Service Of AMSAC/DSS".

12.8 MSLB

12.8.1 IF one or both trains of MSLB do NOT meet Surveillance Requirements:

A. Refer to TS 3.3.11, 3.3.12 and/or 3.3.13 for appropriate TS Condition for inoperability that is indicated.

B. IF entry into condition A of TS 3.3.11 indicated, Immediately Notify I&E to perform IP/0/A/0270/003 (Main Steam Line Break (MSLB) Loss Of An Analog Channel Trip/Restoration) to trip affected channel and prevent entry into condition B of TS 3.3.11.

C. Initiate a Priority Work Request.

D. Initiate a PIP and contact Accountable Systems Engineer.

12.9 Dixon Indicators

12.9.1 Dixons listed on Enclosure "Dixon Meter Information" are on an enhanced surveillance interval.

12.9.2 IF any Dixon listed on Enclosure "Dixon Meter Information" are found to be blinking with a reading of zero, no action is required unless a failure is suspected.

12.9.3 Dixons NOT listed on Enclosure "Dixon Meter Information" are on a surveillance interval. These Dixons have alternate methods of verifying input signal is valid.

12.10 WHEN a computer point needed for a surveillance is NOT available, refer to OP/0/A/1103/020 (Loss Of Computer).

13. Enclosures

13.1 Mode 1 & 2

13.2 Mode 3

13.3 Mode 4

13.4 Mode 5

13.5 Mode 6

13.6 No Mode

13.7 Minimum Temperature For Criticality Surveillance Sheet

13.8 RCS Pressure, Temperature, Heatup And Cooldown Rates Surveillance Sheet

13.9 Pzr Level For LTOP Surveillance Sheet

13.10 RCP Power Supply Verification

13.11 LPI Pump Power Supply Verification

13.12 Loop ΔT Vs Reactor Power

13.13 Gross Load Vs Reactor Power

13.14 Periodic Checks Schedule Sheet (RCS < 50" With Irradiated Fuel In Core)

13.15 Return To Service Of AMSAC/DSS

13.16 ICCM Subcooling Monitor Check

13.17 Surveillance Evaluation

13.18 Dixon Meter Information

13.19 Hot Lake Water Surveillance

NOTE: If Reactor calculations package is NOT running on computer, section 12.3 contains guidance.

NOTE: If Reactor calculations package is running properly on computer, NAS Loop Counter should differ by ≈ 24 every two hours.

NOTE: Step 1.1 contains the priority of indications to use for (%) Reactor Power.

TIME	% Reactor Power	NAS Loop Counter O1P5504	INITIALS					
			Step 1	Step 2	Step 3	Step 4	Step 5	RCP Seal Leakoff Flow
2000						N/A		
2200						N/A		
0000								
0200						N/A		
0400						N/A		
0600						N/A		
0800						N/A		
1000						N/A		
1200						N/A		
1400						N/A		
1600						N/A		
1800						N/A		

1. IF Thermal Power Best indicates "Good", verify Core Thermal Power Indication (every 2 hours when Rx critical.)

1.1 Priority of Power Indications to Use for Surveillance (A = highest priority, G = lowest priority):

A. OAC Calculated Thermal Power Best.

- O1P2037 (Core Thermal Power Best (60 min avg)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0588 (Core Thermal Power Best (10 min. avg.)) - Transient in last 60 minutes or O1P2037 unavailable.
- O1P0889 (Core Thermal Power Best (snapshot)) - Transient in progress or O1P2037 and O1P0588 unavailable.

B. OAC Calculated Thermal Power Secondary if above ≈ 25% power.

- O1P0587 (Core Thermal Power Secondary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0888 (Core Thermal Power Secondary (snapshot)) - Transient within last 60 minutes or O1P0587 unavailable.

C. OAC Calculated Thermal Power Primary if below ≈ 25% power.

- O1P0576 (Core Thermal Power Primary (60 min. avg.)) - Steady State Ops. (i.e., no transient in last 60 minutes)
- O1P0887 (Core Thermal Power Primary (snapshot)) - Transient within last 60 minutes, or O1P0576 unavailable.

D. OAC Calculated Thermal Power Delta T.

- O1P0575 (Core Ther Pwr From Delta Temp (10 min. avg.)) - Steady State Ops. (i.e., no transient in last 10 minutes)
- O1P0326 (Core Thermal Power From Delta T (snapshot)) - Transient within last 10 minutes or O1P0575 unavailable.

E. Alternate method for determining (%) Reactor Power

$$\frac{\text{NI-5} + \text{NI-6} + \text{NI-7} + \text{NI-8}}{4} = \text{\% Rx Power (Avg)}$$

F. Hand-Calculated Thermal Power Δ T using Enclosure "Loop ΔT Vs Reactor Power".

G. Thermal Power from Rx Engineering using PT/0/A/0205/002 (Thermal Power Calculation).

1.2 IF Thermal Power Best indicates "Bad", enter "NIS" (NOT In Service) and initials in appropriate block (s). Refer to OP/0/A/1103/020 (Loss Of Computer) for other actions.

NOTE:	If either step 1.3 or 1.4 is NOT satisfied, Duty Rx Engineer should perform verification of computer calculated TPB indication prior to calibrating NIs.

1.3 Verify Thermal Power Best (TPB) within ± 2.0% Rx Power of percent power from ΔT:

1.3.1 Refer to step 1.1 for priority of indications to determine percent power from ΔT,

OR

1.3.2 Average two RC Loop ΔTs from RC Loop ΔT gauge and using Enclosure "Loop ΔT Vs Reactor Power" determine percent power from ΔT.

1.4 Verify current Gross Load does NOT exceed value given by Enclosure "Gross Load Vs Reactor Power" for current Rx power level. Obtain Gross Load from O1P0963 OR Watt/Var meter if O1P0963 is NOT available.

2. Review Shift Turnover Sheet every 2 Hours.

2.1 Review Enclosure "Shift Turnover Sheet" to verify all turnover items updated and all required testing/surveillance items resulting from a degraded Mode per TS performed.

3. <u>IF</u> > 90% RTP and Steady State, <u>AND</u> fouling coefficient is less than 1.0, verify every 2 Hours O1P0576 (Core Thermal Power Primary (60 min avg)) does <u>NOT</u> exceed O1P0587 (Core Thermal Power Secondary (60 min avg)) by more than 0.2% RTP (i.e., O1P0576 < O1P0587 + 0.2).

   3.1 <u>IF</u> fouling coefficient is less than 1.0, <u>AND</u> <u>IF</u> O1P0576 (Core Thermal Power Primary (60 min avg)) exceeds O1P0587 (Core Thermal Power Secondary (60 min avg)) by more than 0.2% RTP, contact Duty Rx Engineer.

4. Obtain and call in Daily Dispatcher Readings.

CAUTION:	If LDST ≥ 130 °F, HPI System is inoperable.

5. Verify LDST temperature < 120 °F. (CP O1A1240)

6. Initial when verification of steps 1, 2, 3, 4, and 5 completed.

7. Procedure for Periodic Checks:

- Review all in-progress Surveillance Evaluation enclosures in Tech Spec R&R Book:
  - (N) Verify all corrective/compensatory actions still valid. (e.g., WRs, WOs, PIPs open; procedure change(s) <u>NOT</u> yet implemented)
- No surveillance or completion time exceeded.
- Update in-progress Surveillance Evaluations by one-lining, initialing, and dating as required (change WR numbers to WO numbers, update resolution times).
  - RO and SRO sign updated line per step 9.1 of Enclosure "Surveillance Evaluation".
- Close out Surveillance Evaluations no longer applicable (e.g., corrective actions completed, TS/SLC no longer applicable).
- Attach completed (closed out) Surveillance Evaluations to this procedure.

- Perform periodic checks as specified.
  - IF check can be performed as specified and is satisfied, initial appropriate block.
  - IF check CANNOT be performed as written or is NOT satisfied, perform the following:
    - IF Surveillance Evaluation is NOT outstanding for check, perform Enclosure "Surveillance Evaluation".
    - Record Surveillance Evaluation in effect in appropriate block for any periodic checks with Surveillance Evaluations issued.
    - Attach a copy of Surveillance Evaluation issued this shift to this procedure.
    - List Surveillance Evaluations in effect in Remarks section of Procedure Process Record.
  - Place Surveillance Evaluations initiated this shift in Tech Spec R&R Book.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.7.6.1 12 Hours SR 16.10.1.1 6 Hours	CST, UST, and HW	______ 1930-2100 ______ 0130-0300	______ 0730-0900 ______ 1330-1500	O1E2250 O1A0152	Verify Combined Inventory for EFW in Acceptable Operation Region of Enclosure "Combined Inventory for EFW" of OP/0/A/1108/001 (Curves And General Information).
SR 3.7.6.1 12 Hours	CST, UST, and HW	______ 1930-2100 ______ 0130-0300	______ 0730-0900 ______ 1330-1500	O1E2250 O1E2295	Verify UST level > 6 ft. (done on 6 hr frequency)
SR 3.4.1.3 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1970	Verify RCS total flow within limits in COLR. Mode 1 only, Steady State Operation
SR 3.2.2.1 12 Hours	Axial Power Imbalance Operating Limits	(N)	(D)	O1P0877	Verify Power imbalance within operational alarm limits in COLR when > 40% RTP. **IF** Reactor calculations package is **NOT** running on computer, refer to Section 12.3. **IF** % Rx Power Imbalance changes > 2% during Steady State Operations, contact Rx Engineering **IF** NI calibration is required under these conditions, contact Rx Engineering.
SR 3.2.3.1 7 Days	QPT	(N)	(D)	O1P0737 O1P0738 O1P0739 O1P0740	Verify QPT within limits in COLR when > 20% RTP. (done on 12 hr frequency) **IF** Reactor calculations package is **NOT** running on computer, refer to Section 12.3.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.3.1.1 12 Hours	RPS Instrumentation NI Power Range NI-5, 6, 7, 8, 9	(N)	(D)	O1A1544 O1A1545 O1A1546 O1A1547 O1A1548	Verify computer readouts agree within 2%. (4% in RPS Cab) **NOTE:** <u>IF</u> the channels are off scale, the channel check will only verify that they are off scale in the same direction. (TS Bases SR 3.3.1.1)
SR 3.3.1.2 24 Hours	RPS Instrumentation Heat Balance Check Power Range Amplifiers	(N)	(D)	O1P0889	Verify TPB does <u>NOT</u> exceed NI-5, 6, 7, 8 or 9 by more than 2% power. (done on 12 hr frequency) Calibrate NIs when TPB ≥ 2% above any two of the power range NIs. Do <u>NOT</u> exceed ≥ 4% in non-conservative direction. <u>IF</u> TPB indicates "Bad", contact Duty Rx Engineer to calculate core thermal power per PT/0/A/0205/002 (Thermal Power Calculation). Mode 1 only, <u>NOT</u> required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP.
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation RB Pressure Narrow Range	(N)	(D)	O1A1566 O1A1286 O1A1287	Verify computer readouts agree within 0.6 psi (2 psi in ES Cab). <u>IF</u> readouts differ by > 0.4 psi, issue a Priority "E" Work Request.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.6.4.1 12 Hours	Containment Pressure NR RB Pressure	(N)	(D)	O1A1566 O1A1286 O1A1287	Verify RB pressure ≥ -2.45 psig but ≤ 1.2 psig. **IF** > + 0.6 psig, depressurize RB prior to > + 0.8 psig per OP/1/A/1102/014 (RB Purge). **IF** ≤ -0.5 psig, notify MCE for operability evaluation. {PIP 98-3976 & OSC-4476}
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Pressure Narrow Range	(N)	(D)	O1A1688 O1A1689 O1A1690 O1A1691	Verify computer readouts agree within 26 psi (48 psi in RPS Cab).
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Temperature T_H	(N)	(D)	O1A1692 O1A1693 O1A1694 O1A1695	Verify computer readouts agree within 3°F (5°F in RPS Cab). **IF** any of CR RCS temperature selectors are changed, notify Rx Engineering to evaluate and update Enclosure "Loop ΔT Vs. Reactor Power" for new selected inputs.
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Flow	(N)	(D)	O1A1549 O1A0877 O1A1420 O1A1712	Verify total flow agrees within 4800 klbm/hr **AND** no computer alarms for high flow present.
SR 3.4.1.2 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1888 O1P1889	Verify RCS loop average temperature: < 580°F on OAC < 579.5°F Dixon indication (OAC unavailable) Mode 1 only, Steady State Operation When 3 RCPs operating, limits applied to loop with lowest loop average temperature for the condition where there is a 0°F ΔTc Setpoint.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.7.11.1 12 Hours	Pressurizer Temperature	(N)	(D)	O1E2298 O1E2299	Verify each temperature channel agrees within 12°F on computer or indicator.
SR 3.4.9.1 12 Hours	Pressurizer Level (Corrected)	(N)	(D)	O1E2275 O1E2276 O1E2277	Verify each level channel agrees within 9" between computer and recorder **OR** between indicator and recorder. Verify Pzr level ≤ 260".
SR 16.7.11.2 24 Hours TS 3.5.2	LDST Level	(N)	(D)	O1A1042 O1A1043	Verify redundant level channels on computer and gauge agrees within 2". (done on 12 hr frequency)
SR 3.3.11.1 12 Hours	MSLB Detection and MFW Isolation Instrumentation	(N)	(D)		Verify redundant outlet pressure channels for 1A and 1B SGs agree within 30 psig: "A" SG — "B" SG O1E2281 — O1E2282 O1E2283 — O1E2284 O1E2111 — O1E2112 **IF** required conditions **NOT** met, refer to step D (MSLB).
SR 3.7.8.3 24 Hours	ECCW		(D)	O1P0761	Verify average CCW inlet temperature ≤ 88°F. **IF** > 88°F, notify MSE for operability evaluation. {PIP 98-3976 & OSC-4476}
SR 3.4.1.1 12 Hours	RCS Pressure, Temperature, and Flow DNB Limits	(N)	(D)	O1P1609 O1P1620	Verify RCS loop pressure within limits in COLR. Mode 1 only, Steady State Operation. When 3 RCPs operating, limits applied to loop with highest pressure.
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation RC Pressure Wide Range	(N)	(D)	O1A1416 O1A1417 O1A1418	Verify computer readouts agree within 75 psi (100 psi in ES Cab).
SR 3.1.6.1 12 Hours	APSR Alignment Limits	(N)	(D)	GD60 REG	Verify position of each APSR within 6.5% of group average.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.2.1.1 12 Hours	Regulating Rod Position Limits	(N)	(D)	GD60 REG	Verify regulating rod groups within sequence and overlap limits in COLR.
SR 3.2.1.2 12 Hours	Regulating Rod Position Limits	(N)	(D)	GD60 REG	Verify regulating rod groups within position limits on curve in COLR.
SR 3.1.4.1 12 Hours	Control Rod Group Alignment Limits	(N)	(D)	GD60 REG GD60 SAFETY	Verify all Control Rods in each Group agree within ± 3.5% of group average. **IF** a Control Rod is > ± 3.5% of its Group average, refer to OP/0/A/1105/009 (Control Rod Drive System).
SR 3.1.5.1 12 Hours	Safety Rod Position Limits	(N)	(D)	GD60 SAFETY PI Panel	Verify each safety rod fully withdrawn.
SR 16.7.11.3 31 Days SLC 16.5.13	CBAST Temperature	(N)		O1A0784	Verify computer indication > 125°F. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)
SR 16.5.13.1 7 Days	CBAST	(N)		O1A0797	Verify equivalency of 1100 ft^3 of 11,000 ppm boron per OP/0/A/1108/001 (Curves And General Information). (done on 24 hr frequency)
SR 3.3.5.1 12 Hours	ESPS Analog Instrumentation ES Channels 7 & 8 RB 10 psig	(N)	(D)		Verify no trips present. Verify status annunciators operable (lamp test).
SR 3.3.1.1 12 Hours SLC 16.7.9	RPS Instrumentation RP RCP/Flux Trip	(N)	(D)		Verify no Dummy Bistable installed. Verify no trips present. Verify status annunciators operable (lamp test).
SR 3.3.1.1 12 Hours	RPS Instrumentation RB High Press Trip	(N)	(D)		Verify no Dummy Bistable installed. Verify no trips present. Verify status annunciators operable (lamp test).

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.1.7.1 12 Hours	Position Indicator Channels PI Panel	(N)	(D)		Verify all Relative Rod Position indications agree within 5% of Absolute Rod Position indications. **IF NOT**, notify Duty Rx Engineer for evaluation of core parameters and recommended actions.
SR 3.3.9.1 12 Hours	Source Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 SR agree within 1 decade. Mode 2 only
SR 3.3.10.1 12 Hours TS 3.3.8	Wide Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 agree within 3 LED Segments. Mode 2 only
SR 3.4.4.1 12 Hours	RCS Loops	(N)	(D)		Verify required RCPs (3 or 4) in operation with RCS flow indicated.
SR 3.5.4.2 7 Days TS 3.3.8 TS 3.5.4	BWST	(N)			Verify BWST level on ICCM Plasma Displays ≥ 47.0 ft. (done on 24 hr frequency) • A 1LT-BWST 1 • B 1LT-BWST 2 **AND** BWST level ≥ 46.0 ft. on both analog gauges on 1UB2 • BWST Level A • BWST Level B **IF** required conditions **NOT** met, BWST is inoperable.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.5.4.1 24 Hours TS 3.5.4	BWST Temperature	(N)			Verify BWST between ≥ 50°F and ≤ 92.5°F as read on Bailey Indicator. **IF** > 92.5°F, notify MSE for operability evaluation of RBS System. {PIP 98-3976 & OSC-4476} **IF** ≥ 102.5°F or < 50°F, BWST is inoperable.
SR 16.11.3.1 24 Hours	WG Decay Tk Disch Flow Recorder	(N)	(D)		Verify recorder (GWD CR033) indicates flow. Perform anytime during shift hours during GWD Tank releases.
SR 16.11.3.1 24 Hours SR 16.11.3.2 24 Hours	1RIA-35	(N)			Verify 1RIA-35 indicates > zero **AND** no low flow alarm present.
SR 16.11.3.12 24 Hours	1RIA-37		(D)		Perform source check on 1RIA-37.
SR 16.11.3.1 24 Hours	1RIA-38		(D)		Verify 1RIA-38 indicates > zero **AND** no fault alarm present.
SR 16.11.3.2 24 Hours	1RIA-40	(N)			Verify 1RIA-40 indicates > zero **AND** no low flow alarm present.
SR 16.11.3.2 24 Hours	1RIA-43, 44, 45	(N)			Verify the following: 1) 1RIA-43, 44, 45 indicate > zero. 2) Unit Vent Monitor has no low flow alarm. 3) Unit Vent Flow Recorder indicates on scale.
SR 3.4.15.1 12 Hours	RCS Leakage Detection Instrumentation	(N)	(D)		Verify 1RIA-47 indicates > zero **AND** no flow alarm present. **OR** Verify 1RIA-49 indicates > zero **AND** no flow alarm present.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.11.3.1 24 Hours SR 16.11.3.2 24 Hours	1&2RIA-54	(N)			Verify the following: 1) 1&2RIA-54 indicates > zero. 2) No low flow alarm present. 3) "NORMAL/BYPASS" switch in "Normal".
SR 3.5.1.1 12 Hours	CFTs	(N)	(D)		Verify 1CF-1 **AND** 1CF-2 fully open.
SR 16.7.10.1 12 Hours	CFT Instrumentation	(N)	(D)		Verify redundant level channels on each CFT agree within 0.3 ft.
SR 3.5.1.2 12 Hours	CFTs	(N)	(D)		Verify CFT levels between 12.56 ft and 13.44 ft.
SR 16.7.10.1 12 Hours	CFT Instrumentation	(N)	(D)		Verify redundant pressure channels on each CFT agree within 30 psi.
SR 3.5.1.3 12 Hours	CFTs	(N)	(D)		Verify CFT pressures between 575 psig and 625 psig.
SR 16.8.6.1 24 Hours	Lee/Central Alternate Power System		(D)		Verify status of LCTs by contacting Lee Steam Station CR. Operable (✓) ____ 4C ____ 5C ____ 6C **IF** two LCTs are **NOT** operable, refer to Maintenance Rule **AND** contact Switchyard Coordinator.
SR 3.7.8.1 12 Hours	ECCW System	(N)	(D)		Verify two Unit 1 ESV Pumps in operation. **IF** two Unit 1 ESV Pumps **NOT** in operation, Refer To TS 3.7.8 Bases for allowed Pump/Header combinations.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.7.8.2 24 Hours SR 16.9.7.1 12 Hours	ECCW	(N)	(D)		Verify Keowee lake level within limits per SLC 16.9.7. **NOTE:** Instrument error of 1.15 ft. must be added to the absolute lake levels found in SLC 16.9.7 if using a computer point to verify level. Absolute lake level can be determined at the Keowee Hydro Intake structure.
SR 3.4.13.1 72 Hours SLC 16.5.10	RCS Operational Leakage	(N)			<u>Evaluate</u> per PT/1/A/600/010 (Reactor Coolant Leakage) when at steady state for ≥ 12 hours. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)
SR 3.7.16.1 12 Hours SR 16.8.1.1 SR 16.8.1.2 SR 16.8.1.3 24 Hours	Room Temperatures Unit 1 Cable Rm. Unit 1 Equip. Rm. Unit 1&2 Control Rm.	(N) ______ ______ ______	(D) ______ ______ ______		Record and verify room temperatures within respective temperature limits: Unit 1 Cable Rm: ≤ 80°F Unit 1 Equip. Rm: ≤ 85°F Unit 1&2 Control Room: ≤ 80°F **<u>IF</u>** limit is exceeded, refer to OP/0/A/1104/019 (Control Room Ventilation System), notify Unit Coordinator **<u>AND</u>** refer to SLC 16.8.1 and TS 3.7.16.
SR 16.7.11.3 31 Days	BAMT Temperature	(N)			Have NLO verify normal readout at Chemical Addition Panel agrees with local readout within 5°F. (done on 24 hr frequency) May be performed anytime during shift hours (1900-0700)

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.5.4.3 7 Days	BWST	(N) Wednesday			Verify BWST concentration within limit in COLR. May be performed anytime during shift hours (1900-0700)
SR 16.7.11.3 31 Days	PORV and Safety Valve Flow Monitors	(N) Saturday			Verify power supply lights on **AND** verify flow monitor statalarm actuates from "TEST" switch. (done on 7 Day frequency) May be performed anytime during shift hours (1900-0700)
SR 3.5.1.5 31 Days	CFTs	(N) 1st Day of Month			Verify with NLO 1CF-1 **AND** 1CF-2 breakers open. May be performed anytime during shift hours (1900-0700)
SR 3.5.1.4 31 days	CFTs	(N) 1st Day of Month			Verify each CFT boron concentration within limit in COLR. May be performed anytime during shift hours (1900-0700)
SR 3.6.3.1 31 Days	1PR-1, 2, 3, 4, 5, 6	(N) 1st Day of Month			Verify with NLO 1PR-1 and 1PR-6 breakers open. Verify with I&E links open for 1PR-2, 1PR-3, 1PR-4, and 1PR-5. May be performed anytime during shift hours (1900-0700)

NOTE: Remaining items may be performed anytime during shift hours.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	Pressurizer Level (Uncorrected)	(N)	(D)	O1E2301 O1E2303 O1E2305	Verify redundant level channels agree within 8" on computer.
TS 3.10.1	SFP Temperature	(N)	(D)	O1A0839	Verify SFP temperature ≤ 143°F. **IF** > 143°F, SSF RCMUP is inoperable. Contact Duty MSE Engineer.
TS 3.7.5	UST Temperature	(N)	(D)	O1A0122 O1A0123	Verify UST temperature ≤ 125°F. **IF** > 125°F, notify Unit Coordinator and refer to OP/1/A/1106/006 (Emergency FDW System) for EFDW operability.
SLC 16.7.2	AMSAC/DSS	(N)	(D)	O1D2928 THRU O1D2935	Verify no trips present **AND** status annunciators indicate operable channels. Verify 1SA-8 C-9/C-10/D-5/D-8 **AND** indicating lights on 1UB1. Refer to Section 12.7 for operability determinations.
SLC 16.7.3	SG "A" XSUR Level Redundant Level	(N)	(D)	O1A1213 O1E2052	Verify redundant levels agree within 3" when < 2% RTP. > 3" acceptable during momentary swings (< 30 sec).

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
SLC 16.7.3	SG "A" XSUR Level Minimum Level	(N)	(D)	O1A1213 O1E2052	Verify all XSUR OAC indications ≥ 23" when SG on level control. < 23" are acceptable during momentary swings (< 30 sec)
SLC 16.7.3	SG "B" XSUR Level Redundant Level	(N)	(D)	O1A1215 O1E2053	Verify redundant levels agree within 3" when Rx < 2% RTP. > 3" acceptable during momentary swings (< 30 sec).
SLC 16.7.3	SG "B" XSUR Level Minimum Level	(N)	(D)	O1A1215 O1E2053	Verify all XSUR OAC indications ≥ 23" when SG on level control. < 23" acceptable during momentary swings (< 30 sec).
	1A SG SU Levels	(N)	(D)	O1E2000 O1E2001	Verify redundant levels agree within 2" when Rx < 2% RTP. >2" acceptable during momentary swings (< 30 sec).
	1B SG SU Levels	(N)	(D)	O1E2005 O1E2006	Verify redundant levels agree within 2" when Rx < 2% RTP. >2" acceptable during momentary swings (< 30 sec).
SLC 16.7.5	1A SG OR Levels	(N)			Verify redundant operating range recorder level channels agree within 3%.
SLC 16.7.5	1B SG OR Levels	(N)			Verify redundant operating range recorder level channels agree within 3%.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	"A" SG Shell Temperatures	(N)		O1P1892 O1A0968 O1A0969 O1A0970 O1A0971 O1A0972	Verify O1P1892 agrees with manually calculated average of five "A" SG Shell Temperatures (O1A0968 – O1A0972) within $\pm$ 5°F. **IF** required conditions **NOT** met, issue a Work Request.
	"B" SG Shell Temperatures	(N)		O1P1893 O1A0973 O1A0974 O1A0975 O1A0976 O1A0977	Verify O1P1893 agrees with a manually calculated average of five "B" SG Shell Temperatures (O1A0973 – O1A0977) within $\pm$ 5°F. **IF** required conditions **NOT** met, issue a Work Request.
	Station Condenser ΔT		(D)	O1P1947 or O3P1947	**IF** CCW inlet temperature > 68°F, verify Station Condenser ΔT ≤ 22°F. **IF** Station Condenser ΔT is > 22°F, notify Unit Coordinator **OR** OPS Duty Person. **IF** O1P1947 and O3P1947 are OOS, perform the following: 1) Verify all Units with CCW flow have CP O*P1944 operable **OR** 2) Hourly Inlet/Outlet sheet performed per OP/0/A/1103/020 (Loss Of Computer).
	CCW	(N)	(D)	O1P0761	Verify average CCW inlet temperature ≤ 80°F. **IF** > 80°F, Perform Enclosure 13.19 "Hot Lake Water Surveillance". {OSC-2576}

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	Station CCW Discharge Temperature		(D)	O1P1945 or O3P1945	Verify Station 2 Hour Average CCW discharge temperature ≤ 100°F. **IF** Station 2 Hour Average CCW discharge temperature > 100°F, notify Unit Coordinator **OR** OPS Duty Person. **IF** O1P1945 and O3P1945 are OOS, perform the following: 1) Verify all Units with CCW flow have CP O*P1942 operable **OR** 2) Hourly Inlet/Outlet sheet performed per OP/0/A/1103/020 (Loss Of Computer).
TS 3.5.2	LDST Pressure	(N)	(D)	CR Gage O1A2191	Verify both LDST pressure/level relationships comply with OP/0/A/1108/001 (Curves And General Information).
	Room Temperatures Unit 1 Cable Rm. Unit 1 Equip. Rm. Unit 1&2 Control Rm.	(N) ________ ________ ________	(D) ________ ________ ________		Record and compare room temperatures to those taken on previous shift. A ≥ 3°F temperature increase observed from the previous shift may be an indication of a problem with the WC System. **IF** this is indicated **REFER TO** Enclosure "Control Room, Equipment Room, And Cable Room Temperature Troubleshooting Guide" of OP/0/A/1106/029 (Control Room, Equipment Room, And Cable Room Chillers).
	ES Channels 1 & 2 RC Press	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 3 & 4 RC Press	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 1 & 2 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
	ES Channels 3 & 4 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	ES Channels 5 & 6 RB 4 psig	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP Low Press Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP RCP/Flux/Imb Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Temp Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP Press/Temp Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Press Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
	RP High Flux Trip RPS	(N)	(D)		Verify no trips present **AND** status annunciators indicate operable channel.
TS 3.10.1	RCS Boron Concentration	(N)	(D)		Verify RCS Boron Concentration greater than "Minimum RCS Boron Concentration to Maintain SSF Operability" curve of PT/1/A/1103/015 (Reactivity Balance). **IF** minimum concentration is **NOT** met, SSF RC MU Pump is inoperable. Contact Duty Rx Engineer.
	Control Rod Position	(N)	(D)		Verify limit lamps operable on Diamond and PI Panel.
TS 3.3.10 TS 3.3.8	Wide Range Neutron Flux	(N)	(D)		Verify NI-1, NI-2, NI-3 and NI-4 agree within: 3 LED Segments when < 10% RTP **OR** 2 LED Segments when ≥ 10% RTP. Mode 1 only

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
TS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) **AND** SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). **IF** SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
TS 3.3.8	Subcooling Monitors Core	(N) Thursday			**IF** > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F **AND** (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. **IF** (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	Subcooling Monitors Core	(N) Thursday			**IF** < 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' **AND** Train 'B') within -9 to +11°F. **IF** (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion
TS 3.3.8	ICC Level - Train 'A'		(D)		Verify "MALFUNCT 00" message on diagnostic page of ICCM Plasma Display. Verify "RVLIS/ICCM/RG 1.97 TRAIN A TROUBLE" annunciator (1SA-18/A-3) **NOT** in alarm. **IF** a "MALFUNCTION FF" message **OR** annunciator alarm is present, issue a Priority Work Request **AND** contact I&E to investigate problem to determine operability of ICC channel.
TS 3.3.8	ICC Core Cooling Train 'A'		(D)		Verify from Core Cooling core map ≥ 5 CETCs operable (do **NOT** indicate "FAIL").

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8	ICC Level - Train 'B'		(D)		Verify "MALFUNCT 00" message on diagnostic page of ICCM Plasma Display. Verify "RVLIS/ICCM/RG 1.97 TRAIN B TROUBLE" annunciator (1SA-18/A-4) **NOT** in alarm. **IF** a "MALFUNCT FF" message **OR** annunciator alarm is present, issue a Priority Work Request **AND** contact I&E to investigate problem to determine operability of ICC channel.
TS 3.3.8	ICC Core Cooling Train 'B'		(D)		Verify from Core Cooling core map ≥ 5 CETCs operable (do **NOT** indicate "FAIL").
TS 3.3.5 TS 3.5.4	BWST Level Instrument ICCM Plasma Displays	(N)			Verify redundant indicators on ICCM Plasma Displays agree within 2 ft. **IF** required conditions **NOT** met, BWST is inoperable.
TS 3.5.4	BWST Level Instrument Analog Gauges on 1UB2	(N)			Verify redundant indicators on 1UB2 agree within 2 ft. **IF** required conditions **NOT** met, BWST is inoperable.
TS 3.3.14 TS 3.7.5	1A & 1B MD EFDW Pumps "OFF/AUTO/RUN" Lights	(N)	(D)		Verify lights energized. **IF NOT**, MD EFDW Pumps are inoperable and Auto Start capability is lost.
TS 3.3.8	EFDW Total Flow	(N)			Verify Train 'A' **AND** Train 'B' EFDW Hdr Flow to SG indicates < 20 gpm with no EFDWPs operating. (indicators fail high)

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.8 TS 3.7.6	UST Level CR Gauges	(N)	(D)		Verify redundant indicators agree within 0.4 ft.
SLC 16.11.3	Sorrento Radiation Monitor Time Check	(N)	(D)		Verify current time on RIA CRT within ± 1 minute of current time on OAC CRT. **IF** > ± 1 minute, Contact IT to reset time.
TS 3.3.8	1RIA-57, 58	(N)			Verify 1RIA-57, 58 indicate between 5.0E-1 and 1.0E0 R/HR. Press "C/S" button. Verify no Area Monitor Fault alarm exists after check source is complete. Press "R/HR" button to return to normal.
TS 3.5.3 TS 3.7.7	LPI Cooler 'A' LPSW Flow Dixon Indicator	(N)	(D)	O1A2124	**IF** dixon indicator zero and blinking, verify computer point **NOT** reading a high negative value.
TS 3.5.3 TS 3.7.7	LPI.Cooler 'B' LPSW Flow Dixon Indicator	(N)	(D)	O1A2125	**IF** dixon indicator zero and blinking, verify computer point **NOT** reading a high negative value.
TS 3.6.5	1A, 1B, 1C RBCU LPSW Flow (IN)	(N)	(D)		Verify each RBCU LPSW Flow (IN) ≥ 550 gpm. **IF** Auxiliary Cooling Coils **AND** 1B RBCU **BOTH** have flow established, verify ≥ 1100 gpm Inlet Flow to 1B RBCU. **IF** any RBCU LPSW Flow (IN) < required, enter LCO 3.0.3

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.5.3 TS 3.6.4 TS 3.6.5	RB Dome Temperature	(N)	(D)	O1A0043 Chart Recorder 1RBCCR0007	Verify highest RB Dome temperature ≤ 170°F. Verify lowest RB Dome temperature ≥ 90°F when 100 % RTP. **IF** either limit is exceeded, contact MSE for operability evaluation. (LPI and BS) **IF** > 175°F, contact CEN for operability evaluation. (Reactor Building)
TS 3.3.8	RB Post Accident Water Level Wide Range Indication	(N)		O1A1033 O1A1565	Verify Train 'A' **AND** Train 'B' Meters, Computer, and Recorder (1BS CR0085) agree within 0.5 ft.
TS 3.3.8	RB Post Accident Pressure Wide Range Indication	(N)		O1A1011 O1A1315	Verify Train 'A' **AND** Train 'B' Meters, Computer, and Recorder (1BS CR0085) agree within 6 psi.
TS 3.3.8 TS 3.4.15	RB Normal Sump	(N)	(D)		Verify Train 'A' **AND** Train 'B' Meters and Recorder (1LWDCR0095) agree within 1 ft.
TS 3.4.15	RB Normal Sump	(N)			Verify water level in RBNS on scale.
TS 3.3.8	RB Emerg Sump Narrow Range	(N)		O1A0050	Verify Train 'A' **AND** Train 'B' Meters, Computer and Recorder (1LWDCR0095) agree within 1 ft.
	RB Emergency Sump	(N)			Verify zero water level in RBES.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.10.1	RCP Seal Leakoff Flow	(N)	(D)		Verify RCP seal leakoff flow electronic display on **AND** display present. **IF** seal leakoff flow > 4 gpm on any RCP, increase seal leakoff flow surveillance to every 2 hours and initial on page 1. **IF** the SSF is **NOT** manned and seal leakoff flow > 4.7 gpm for 1A1, 1A2, 1B1 or 1B2 RCP, SSF RCMU Pump is inoperable. **IF** the SSF is manned and seal leakoff flow >6.0 gpm for 1A1 or 1B1 RCP, or > 4.7 gpm for 1A2 RCP, or > 5.5 gpm for 1B2 RCP, SSF RCMU Pump is inoperable.
	Loose Parts Monitor	(N)			Monitor all operable points on LPM. Test alarm circuitry per OP/1/A/1105/011 (Loose Parts Monitoring System).
	Event Recorders	(N)			Verify paper in all Events recorders.
	800 mHz Radio	Sunday (0100-0400) (N)			Test the backup radio communications with the System Operating Center (SOC) **AND** the Transmission Control Center (TCC). SOC code – 96 TCC code – 11 **IF** communications fail notify SPOC.
	Easterline Angus Charts	(N)			Stamp charts.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
SLC 16.8.5	125 VDC Ground Detection System Test	(N)	(D)		Perform Enclosure "125 VDC Ground Detection System Operation" of OP/1/A/1107/010 (Operation Of The Batteries And Battery Chargers). **<u>IF</u>** required conditions **<u>CANNOT</u>** be met, refer to SLC 16.8.5 for required actions
	SASS	(N)	(D)		Verify the following on SASS panels in ICS cabinet #8: 1) All "AUTO" lights on. 2) No "MISMATCH" lights on. 3) All "POWER" lights on.
SLC 16.9.6	Fire Alarm Cabinet	(N)			Verify "Power" **<u>AND</u>** "Run" LEDs are on **<u>AND</u>** no Trouble/Alarm lights present.
TS 3.10.1	SFP Level	(N)	(D)		**<u>IF</u>** all fuel in SFP subcritical ≥ 20 days, verify SFP level > - 2 ft. **<u>IF</u>** any fuel in SFP subcritical < 20 days, verify SFP level greater than Enclosure "Unit 1&2 Spent Fuel Pool Level Vs Temperature Curve (7-19 days))" of OP/0/A/1108/001 (Curves And General Information). **<u>IF</u>** limit exceeded, SSF RCMUP is inoperable.
	RCP Data Sheets	(N)			Complete RCP Data Sheets: • Sunday and Wednesday when at steady-state power. • Daily when changing Rx power **<u>OR</u>** RCS temperature.
SLC 16.11.3	RB Depressurization		(D)		**<u>IF</u>** RB depressurization is in progress, submit a RB Sample Request.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
TS 3.3.11	MSLB Digital Channels	(N)	(D)		Verify with NLO all five power supply lamps lit. **IF** required conditions **NOT** met, refer to step D (MSLB).
ISFSI TS 4.1.1 ISFSI C of C 1.3.1 1.3.2	ISFSI Storage Facility		(D)		Verify notified by Security: 1) All Horizontal Storage Modules (HSM) ventilation screens (inlet and outlet) free of debris and no material accumulated between modules to block air flow. 2) All Roof Slab temperatures monitored which contain Spent Fuel < 260°F and < 80°F increase in 24 hrs. **IF** temperature limits are exceeded, issue a Work Request **AND** contact Rx Engineering.
	Unit 1 BWST	(N) Monday			Place in recirc per OP/1&2/A/1104/006 (SF Cooling System).
	LDST Level	(N) Saturday			Verify redundant level channels 1&2 CR Gage and Local Gage (1HPIPG0437) agree within 2".
	LDST Pressure	(N) Saturday			Verify redundant pressure channels 1&2 CR Gage and Local Gage (1HPIPG0438) agree within 1 psig.
	SSF Radio	(N)			Verify communications with CR via SSF radio. Use base station on Channel 2.

Duke Power Company Oconee Nuclear Station **Backup Incore Detector System** *** This procedure has the potential to affect Reactivity Management *** **Continuous Use**	Procedure No. PT/0/A/1103/019 Revision No. 4 Electronic Reference No.

Performed By ______________________________

Date ____________________

Backup Incore Detector System

1. Purpose

1.1 To verify the operable backup recorder points meet the minimum requirements for the incore instrumentation system upon loss of the incore system on the unit computer or loss of the unit computer.

1.2 To provide a method to calculate reactor power axial imbalance and quadrant power tilt using the backup incore detector system when the incore system is not available on the unit computer and one or more of the excore detectors are inoperable.

2. References

2.1 Improved Technical Specifications 3.2.2, Axial Power Imbalance
3.2.3, Quadrant Power Tilt

2.2 OP/0/A/1103/020, Loss of Computer

2.3 PT/1,2,3/A/0600/001, Periodic Instrument Surveillance

2.4 Unit Core Operating Limits Report (COLR)

2.5 NSD 304, Reactivity Management

2.6 Selected Licensee Commitment 16.7.8

3. Time Required

3.1 Verify Backup Incore Recorders operable - 10 minutes - 1 Operator or Reactor Engineer

3.2 Calculate Backup Tilt/Imbalance - 30 minutes - 2 Operators and/or Reactor Engineers

4. Prerequisite Tests

None

5. Test Equipment

Calculator

6. Limits and Precautions

6.1 This procedure has the potential to affect REACTIVITY MANAGEMENT, since the backup incore recorders are used to monitor reactivity.

6.2 If the incore system is not available on the unit computer and the backup recorder points are not operable per this procedure, then the reactor power shall be reduced below 80% of the power allowable for the existing reactor coolant pump combination within eight hours unless:

6.2.1 The incore system is restored on the unit computer.

or

6.2.2 The backup recorder points are restored to meet the minimum requirements for operability. (ref. SLC 16.7.8)

6.3 If the backup incore limits are exceeded then action must be taken per the applicable Technical Specifications as listed below:

6.3.1 Quadrant Power Tilt - ITS 3.2.3

6.3.2 Axial Power Imbalance - ITS 3.2.2

7. Required Plant Status

_____ 7.1 Quadrant power tilt surveillances are required when the Unit is above 20% full power.

_____ 7.2 Reactor power imbalance surveillances are required when the Unit is above 40% rated power.

8. Prerequisite System Conditions

Loss of incore system on the unit computer or loss of the unit computer

9. Test Method

9.1 The backup recorder points will be checked to identify which points are a) inoperable as indicated by off-scale readings or b) identified as inoperable or out of calibration during the last functional verification. The remaining operable points will be checked to verify the minimum number of detectors are operable to measure axial imbalance and quadrant power tilt as required.

9.2 Axial Imbalance and quadrant power tilt calculations may be performed using the operable backup recorder points.

10. Data Required

Incore Backup recorder point readings

11. Acceptance Criteria

11.1 The calculated axial imbalance is within the curve for the appropriate pump configuration shown in the current Core Operating Limits Report (COLR) on the Backup Incore Setpoint Column of the (Error-Adjusted) "Operational Power Imbalance Setpoints" Table.

11.2 The calculated quadrant power tilt is less than the value given in the current Core Operating Limits Report (COLR) on the Backup Incore row of the (Error-Adjusted) "Quadrant Power Tilt Setpoints" Table.

12. Procedure

NOTE:	If this procedure is being performed only to satisfy Section 1.1, then perform Section 12.1 and N/A Section 12.2.

_____12.1 Verification of Minimum Incore Detector Operability

_____12.1.1 On Enclosure 13.1 and 13.2, place an "X" next to the backup recorder points which are inoperable as indicated by off-scale readings or notes attached to the recorders.

_____12.1.2 Verify that all three required points on at least three detector strings are operable per instructions on Enclosure 13.1.

_____12.1.3 Verify that all four required points on at least four sets (two sets in each axial core half) are operable per instructions on Enclosure 13.2.

_____12.1.4 If either step 12.1.2 or 12.1.3 cannot be satisfied;

_____12.1.4.1 Notify the Unit Supervisor.

_____12.1.4.2 Take actions described in 6.2.

_____12.2 Calculation of Axial Imbalance and Quadrant Power Tilt

NOTE:	If this portion of the procedure is being performed to fulfill the requirements of PT/1,2,3/A/0600/01, Periodic Instrument Surveillance, then repeat the steps below every twelve hours as required and record the calculated values in that procedure.

_____12.2.1 Verify the reactor has been at steady state conditions (±2% FP) for at least 30 minutes.

_____12.2.2 Calculate axial imbalance per Enclosure 13.3 using operable recorder points identified on Enclosure 13.1.

_____12.2.3 Calculate quadrant power tilt per Enclosure 13.4 using operable recorder points identified on Enclosure 13.2.

_____12.2.4 Verify the calculated axial imbalance does not exceed the backup incore limits per 11.1.

_____12.2.5 Verify the calculated quadrant power tilt does not exceed the backup incore limits per 11.2.

_____12.2.6 If either step 12.2.4 or 12.2.5 cannot be satisfied, notify the Unit Supervisor and take appropriate actions described in the applicable Technical Specification as listed below:

Quadrant Power Tilt - ITS 3.2.3
Axial Power Imbalance - ITS 3.2.2.

13. Enclosures

13.1 Required Backup Recorder Points for Calculating Axial Power Imbalance

13.2 Required Backup Recorder Points for Calculating Quadrant Power Tilt

13.3 Axial Power Imbalance Calculation Sheet

13.4 Quadrant Power Tilt Calculation Sheet

Oconee 1 Cycle 19

Operational Power Imbalance Setpoints

	%FP	Full Incore	Backup Incore	Out of Core
4 Pumps	0	-31.5	-31.0	-31.5
	80	-31.5	-31.0	-31.5
	90	-29.7	-29.3	-29.7
	100	-19.1	-18.7	-19.1
	102	-17.0	-16.5	-17.0
	102	17.0	17.0	17.0
	100	19.1	18.7	19.1
	90	22.4	21.8	22.4
	80	23.1	22.3	23.1
	0	23.1	22.3	23.1
3 Pumps	0.0	-31.5	-31.0	-31.5
	63.30	-31.5	-	-31.5
	63.77	-	-31.0	-
	77.0	-17.0	-16.5	-17.0
	77.0	17.0	17.0	17.0
	71.99	-	22.3	-
	71.24	23.1	-	23.1
	0.0	23.1	22.3	23.1

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

NRC-004/ADMIN A.2

ICCM Subcooling Margin Monitor Check

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

SUBCOOLING MONITOR CHECK

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

System: Conduct of Operations
K/A: G2.2.12
Rating: 3.0/3.8

Task Standard:

Perform Subcooling Monitor Check

Preferred Evaluation Location:

Simulator __X__ In-Plant ______

Preferred Evaluation Method:

Perform __X__ Simulate ______

References:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and 13.16

Validation Time: 10 min. **Time Critical:** NO

==

Candidate: ______________________________
NAME

Time Start : ________
Time Finish: ________

Performance Rating: SAT ______ UNSAT ______ Question Grade ______ Performance Time ______

Examiner: ________________________ ____________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

SIMULATOR OPERATOR INSTRUCTIONS:

SIMULATOR OPERATOR INSTRUCTIONS:

1. Recall IC or SNAP # ________
2. Go to run, acknowledge alarms.
3. Verify accurate pressure/ temperature values
4. Freeze simulator.
5. Leave simulator in FREEZE to prevent values changing.

SIMULATOR OPERATOR INSTRUCTIONS:

NONE

ANY NOTES/INSTRUCTIONS TO THE BOOTH OPERATOR SHOULD BE LISTED HERE, AND AT THE BOTTOM OF THE STEP BLOCK FOR WHICH IT APPLIES.

Tools/Equipment/Procedures Needed:

PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosures 13.1 and 13.16

READ TO OPERATOR

DIRECTION TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit 1 is at 100% power
Today is Thursday
The time is 2000
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1 has been completed up page 21

INITIATING CUES:

The SRO directs you to complete PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1.

START TIME: ________

STEP 1: Verify Loop A and Loop B RCS WR Pressure and ICCM Plasma Display pressure agree with in 10 psig **"A" WR = 2114, ICCM = 2114** **"B" WR = 2160, ICCM = 2150** STANDARD: Locate and obtain Loop "A" RCS WR Pressure and ICCM Plasma Display pressure and ensure pressures agree within 10 psig Locate and obtain Loop "B" RCS WR Pressure and ICCM Plasma Display pressure and ensure pressures agree within 10 psig COMMENTS:	___ SAT ___ UNSAT
NOTE: Perform this step as the Initial Conditions indicate it is Thursday night shift. STEP 2: Obtain readings from: __(25)__ SCM Loop "A" (OAC) __(24)__ SCM Loop "A" (ICC) STANDARD: Locate and obtain Loop "A" SCM readings. SCM Loop "A" (OAC) - SCM Loop "A" (ICC) = __(+1)__ Verify SCM Loops agree within -6 °F to +9 °F Candidate determines that "A" is within the specified range COMMENTS:	___ SAT ___ UNSAT

STEP 2: Obtain readings from: __(18)__ SCM Loop "B" (OAC) __(27)__ SCM Loop "B" (ICC) STANDARD: Locate and obtain Loop "B" SCM readings. SCM Loop "B" (OAC) - SCM Loop "B" (ICC) = __(-9)__ Verify SCM Loops agree within –6 °F to +9 °F Candidate determines that "B" is NOT within the specified range COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 3: Candidate determines that "B" is NOT within the specified range. STANDARD: Refers to Enclosure 13.16 (ICCM Subcooling Monitor Check) of PT/1/A/0600/001, Periodic Instrument Surveillance COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: Loop "B" Subcooling Monitor Obtain RCS Loop "A" pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below STANDARD: **(2026)** psig + 14.7 psi = **(2040.7)** psia COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: Using ASME Steam Tables **OR** OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below STANDARD: **(638)** °F COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6: Obtain RCS Loop "B" temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below. STANDARD: **(598)** °F ***CUE: Tell the operator to use 598°F instead of 600 as indicated on the simulator. This will allow the final calculation to within the proper range of +/- 5°F*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below. STANDARD: Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) = Correction. **(22)** °F = (638) °F – (598) °F - 18 °F COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 8: Verify ICC Train "B" SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4) STANDARD: Determine difference in ICCM and manually calculated SCM agrees within +/- 5°F ICC Train "B" SCM Loop **(27) – 22 = 5** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 9: Calculations performed in step 2.2 require independent verification. **CUE: another operator will perform verification calculations.** STANDARD: Sign the Performed By: ____________________ COMMENTS:	___ SAT ___ UNSAT

TIME END: __________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
2	Step is necessary, to determine that the "B" SCM is not within the required range and is inoperable. Calculation is -9
3	Step is necessary, Refer to Enclosure 13.16 to perform Manual SCM calculation
4	Step is necessary, calculation of actual RCS pressure in psia to obtain correct saturation temperature
5	Step is necessary, obtain correct saturation temperature based on pressure (psia)
6	Step is necessary, obtain actual RCS Th temperature
7	Step is necessary, obtain actual Loop A SCM
8	Step is necessary, determine that ICCM Loop SCM agrees with the manual calculated SCM within +/- 5. Actual = 5

CANDIDATE CUE SHEET
(TO BE RETURNED TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

Unit 1 is at 100% power
Today is Thursday
The time is 2000
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1 has been completed up page 21

INITIATING CUES:

The SRO directs you to complete PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) AND SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). IF SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			IF > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F AND (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. IF (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

_____ 2.2.2 Using ASME Steam Tables <u>OR</u> OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

__________ °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

__________ °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.2.2) (step 2.2.3)

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop ________

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core <u>AND</u> ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core ________

ICC Train 'B' SCM Core ________

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

________________________ ________________________
Performed By IV By

1. Initial Conditions

_____ 1.1 Manual verification of ICCM Subcooling monitors required.

_____ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

_____ 2.1.2 Using ASME Steam Tables <u>OR</u> OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

__________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

__________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.1.2) (step 2.1.3)

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

__________ psig + 14.7 psi =_________psia

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)		Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday			Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) **AND** SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). **IF** SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			**IF** > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F **AND** (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. **IF** (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

1. Initial Conditions

_____ 1.1 Manual verification of ICCM Subcooling monitors required.

_____ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

__________ psig + 14.7 psi =__________psia

_____ 2.1.2 Using ASME Steam Tables **OR** OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

__________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

__________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

__________°F = __________°F - __________°F - 18°F
(step 2.1.2) (step 2.1.3)

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

__________ psig + 14.7 psi =__________psia

_____ 2.2.2 Using ASME Steam Tables **OR** OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

_________ °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

_________ °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

_________°F = _________°F - _________°F - 18°F
(step 2.2.2) (step 2.2.3)

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop _________

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core **AND** ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core _________

ICC Train 'B' SCM Core _________

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

_______________________ Performed By

_______________________ IV By

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ITS 3.3.8	Digital RCS WR Press and ICCM Plasma Display RCS Pressure	(N)	(D)	Loop A 2114 ICC 2114 Loop B 2160 ICC 2150	Verify both loop "A" RCS pressures agree within 10 psig. Verify both loop "B" RCS pressures agree within 10 psig.
ITS 3.3.8	Subcooling Monitors Loop 'A' and Loop 'B'	(N) Thursday		25-24 = 1 18-27 = -9	Verify SCM Loops agree within –6 to +9°F: SCM Loop 'A' (OAC) minus SCM Loop 'A' (ICC) AND SCM Loop 'B' (OAC) minus SCM Loop 'B' (ICC). IF SCM Loop A (OAC) and/or SCM Loop B (OAC) is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check" and initial after satisfactory completion.
ITS 3.3.8	Subcooling Monitors Core	(N) Thursday			IF > 50% RTP, verify (OAC) Subcool Margin Core minus (ICC) Subcool Margin Core (Train 'A' and Train 'B') within +1 to +21°F AND (ICC) Subcool Margin Core (Train 'A' and Train 'B') read positive. IF (OAC) Subcool Margin Core is OOS, perform Enclosure 13.16 "ICCM Subcooling Monitor Check". Initial after satisfactory completion.

1. Initial Conditions

✓ 1.1 Manual verification of ICCM Subcooling monitors required.

✓ 1.2 Review Limits and Precautions.

2. Procedure

2.1 Loop 'A' Subcooling Monitor

_____ 2.1.1 Obtain RCS Loop 'A' pressure reading from computer point O1A1416 (RCS Loop A WR Press 1) and document below.

_________ psig + 14.7 psi = _________ psia

_____ 2.1.2 Using ASME Steam Tables OR OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.1.1 and document below.

_________ °F

_____ 2.1.3 Obtain RCS Loop 'A' temperature reading from computer point O1E2010 (RC Outlet Temp A) and document below.

_________ °F

_____ 2.1.4 Calculate subcooling margin using RCS temperature in step 2.1.3 and saturation temperature in step 2.1.2 and formula below.

Calculated SCM = Saturation Temp (step 2.1.2) - RCS Temperature (step 2.1.3) - Correction

_________ °F = _________ °F - _________ °F - 18°F
(step 2.1.2) (step 2.1.3)

_____ 2.1.5 Verify ICC Train 'A' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.1.4).

ICC Train 'A' SCM Loop ________

2.2 Loop 'B' Subcooling Monitor

_____ 2.2.1 Obtain RCS Loop 'B' pressure reading from computer point O1A1417 (RCS Loop B WR Press 1) and document below.

2026 psig + 14.7 psi = 2040.7 psia

_____ 2.2.2 Using ASME Steam Tables OR OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below.

638 °F

_____ 2.2.3 Obtain RCS Loop 'B' temperature reading from computer point O1E2011 (RC Outlet Temp B) and document below.

CUE 598 °F

_____ 2.2.4 Calculate subcooling margin using RCS temperature in step 2.2.3 and saturation temperature in step 2.2.2 and formula below.

Calculated SCM = Saturation Temp (step 2.2.2) - RCS Temperature (step 2.2.3) - Correction

22 °F = 638 °F - 598 °F - 18°F
(step 2.2.2) (step 2.2.3)

_____ 2.2.5 Verify ICC Train 'B' SCM Loop agrees within ± 5 °F of calculated subcooling margin (step 2.2.4)

ICC Train 'B' SCM Loop 27

2.3 Core 'A' and 'B' Subcooling Monitors

_____ 2.3.1 Verify ICC Train 'A' SCM Core AND ICC Train 'B' SCM Core within required conditions below.

ICC Train 'A' SCM Core within ± 5 °F of ICC Train 'B' SCM Core.

ICC Train 'A' SCM Core ________

ICC Train 'B' SCM Core ________

2.4 Calculations performed in Steps 2.1 and 2.2 require independent verification. Document individuals performing this enclosure.

Sign
Performed By

____
IV By

QUESTION NO. A.3 SRO/RO-Q1 G2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure [CFR: 43.4/45.10] (2.9/3.3) **REFERENCE ALLOWED**
QUESTION: You have been asked been asked to verify that 3RC-2 (PZR Spray Bypass) is properly back seated. • Contamination levels on top of the PZR 500,000 dpm/cm^2 • Radiation levels are 30 mrem/hr β-γ general area Q1: What RWP will you use to enter the RB for this job? Q2: What dress requirements are required for this job? Note: This information can be obtained from the RWPs in the U3 Change Room or from Shift/Unit RP crew.
ANSWER: A1: RWP 3001, U3 RB Inspections and Valve Operations A2: Per the RWP Dress Category and Task Description – Dress Category I, Cloth Hood, cloth coverall, cotton gloves, 2 pair of rubber gloves, booties, shoecovers, no personal outer clothing,. Secure gloves and booties (tape, Velcro, straps).
REFERENCE: Reference: Radiation Protection Policy Manual, NSD 507, RP-RPP
COMMENTS:

RADIATION WORK PERMIT # 3001 REV: 9 DATE/TIME: 03/30/00 13:36
OCONEE NUCLEAR STATION ACTIVATION DATE: 04/07/00 00:01

Job Title: U3 RX BLDG INSPECTIONS AND VALVE OPERATIONS

STANDING REQUIREMENTS FOR USE OF THIS RWP
EACH RADIATION WORKER IS RESPONSIBLE FOR:

- KNOWING THEIR WORK AREA DOSE RATES.
- FOLLOWING REQUIREMENTS OF THIS RWP.
- BEING ALARA.
- HOUSEKEEPING.
- WEARING A POCKET OR ELECTRONIC DOSIMETER AND A TLD.
- FOLLOWING POSTED REQUIREMENTS.
- REVIEWING AREA RADIOLOGICAL PLAN VIEW WHEN AVAILABLE PRIOR TO ENTRY.
- NOTIFYING RADIATION PROTECTION PRIOR TO SWEEPING, BRUSHING, GRINDING, WELDING, OR USE OF COMPRESSED AIR IN CONTAMINATED AREAS.
- FOLLOWING POSTED DRESS CATEGORY REQUIREMENTS.
- WEARING MODESTY GARMENTS WHEN NOT WEARING PERSONAL OUTER CLOTHING.
- MONITORING PERSONNEL/TOOL/EQUIPMENT REQUIRED WHEN LEAVING RCA OR CONTAMINATED RCZ.

DRESS CATEGORY AND TASK DESCRIPTION

D 1. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.
H 2. WORK IN CONTAMINATED AREA.
I 3. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.
M 4. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.
N 5. WORK IN CONTAMINATED AREA WHEN WET CONDITIONS OR POTENTIAL FOR WET CONDITIONS EXIST.

SPECIAL DOSIMETRY	RESPIRATORY

SPECIAL INSTRUCTIONS/PRECAUTIONS

* NOTIFY RP PRIOR TO START OF WORK
* USE HOSE CLAMPS TO SECURE HOSE/TUBING CONNECTIONS

COMMENTS

NOTIFY RP PRIOR TO ENTERING THE REACTOR BUILDING.
NOTIFY RP IF WORK AREA CONDITIONS OR JOB SCOPE CHANGES.
RESPIRATORY REQUIREMENTS WILL BE BASED ON TEDE / ALARA EVALUATIONS.
WORKERS TO INSTALL CATCH CONTAINMENTS / DRAIN RIGS TO PREVENT SPILLS WHILE DRAINING COMPONENTS.
DISPOSABLE (PLASTIC) BOOTIES SHALL BE WORN INSIDE NYLON (WASHABLE) BOOTIES FOR WORK IN WET CONDITIONS.

"EXTRA HIGH RADIATION AREA" DOSE RATES:
5000 MREM/HR HIGH CONTACT ON FLOOR OF DEEP END OF CANAL
UP TO 1000 MREM/HR GENERAL AREA IN DEEP END OF CANAL

ED (MG) SET POINTS
DOSE ALARM - 25 MREM
DOSE RATE ALARM - 100 MREM/HR

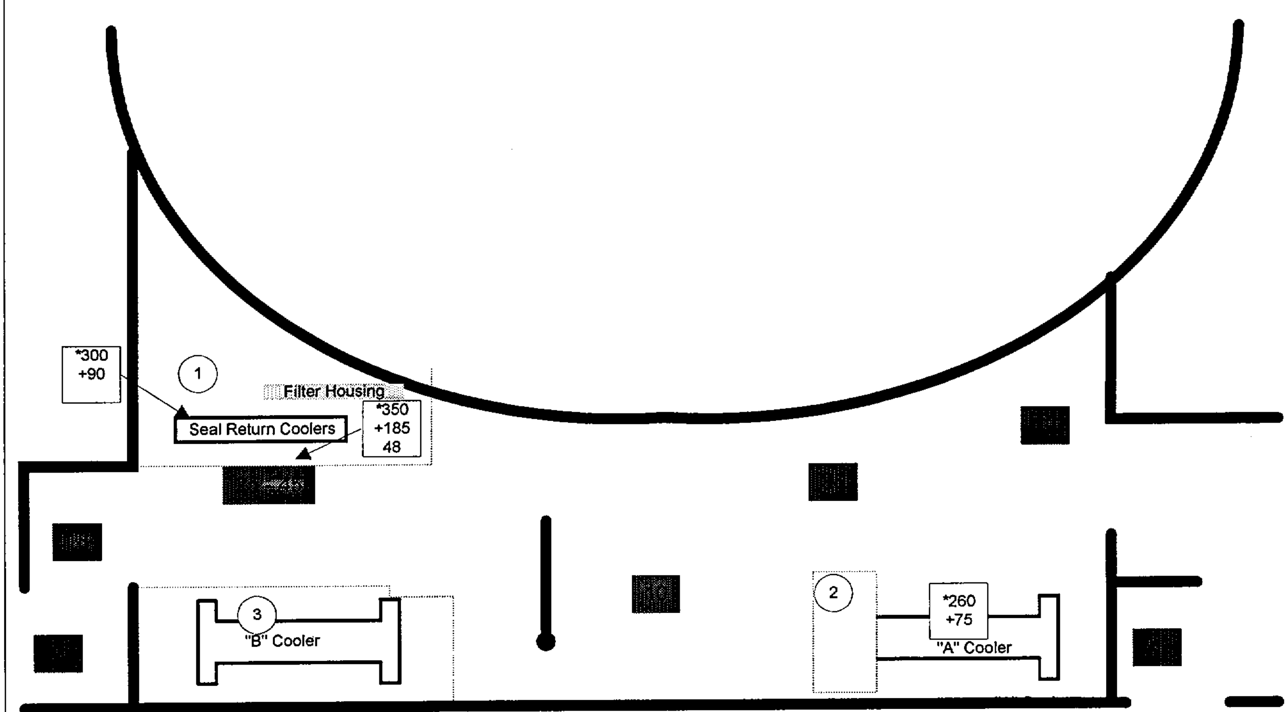
APPROVED BY: NRW1552 DATE/TIME: 03/30/00 13:35	TERMINATED BY: DATE/TIME:

5.3.6 PROTECTIVE CLOTHING FOR EACH DRESS CATEGORY

DRESS CATEGORY	PROTECTIVE CLOTHING
A	None.
B	Surgical gloves.
C	Cotton and rubber gloves.
D	Cotton and rubber gloves, booties and shoecovers.
E	Labcoat, cotton and rubber or surgical gloves.
F	Labcoat, cotton and rubber gloves, booties and shoecovers.
G	Cloth hood, disposable coveralls, cotton and rubber gloves, booties and shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
H	Cloth hood, cloth coverall, cotton and rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
I	Cloth hood, cloth coverall, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
J	Cloth hood, cloth coverall, cotton gloves, 2 pair rubber gloves, booties, shoecovers, no personal outer clothing and additional outer booties or shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
K	Cloth hood, cloth coverall, disposable coveralls, cotton gloves, rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
L	Cloth hood, cloth coverall, disposable coveralls, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing and additional outer booties or shoecovers. Secure gloves and booties (tape, elastic, Velcro, straps).
M	Cloth hood, 2 pair cloth coveralls, cotton gloves, 2 pair rubber gloves, 2 pair booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
N	Cloth hood, cloth coverall, wetsuit, cotton gloves, 2 pair rubber gloves, booties and shoecovers, no personal outer clothing. Secure gloves and booties (tape, elastic, Velcro, straps).
O	Cloth hood, cloth coverall, bubble suit, cotton gloves, 2 pair rubber gloves, booties, shoecovers, no personal outer clothing and additional shoe covers or jump boots. Secure gloves and booties (tape, elastic, Velcro, straps).
Z	Special dress as required by Radiation Protection.

QUESTION NO. A.3 SRO/RO-Q2 G2.3.1 Radiation Exposure Limits [2.6/3.0] **REFERENCE ALLOWED**
QUESTION: Given the attached Oconee Nuclear Station VSDS Survey Report for Room 108: Concerning the area the Room 108 (U-1Decay Heat Removal / Seal Return) plan view 1. Describe the type of posting this area should have to warn radiation worker? Explain your answer. 2. Identify where you should stand while waiting for further direction from the Control Room during a work evaluation. Explain your answer.
ANSWER: 1. High Radiation Area because the dose rate (185 mrem/hour) at 30 centimeters is greater than 100 mrem/hour. This would be marked High radiation area signs. 2. Near the entrance because the general area radiation level is the lowest value at this location.
REFERENCE: NSD 507.8 Exposure and Contamination Control
COMMENTS:

Room 108 Decay Heat Removal / Seal Return	Survey # 050100-27	Date/Time: 07/01/2000 22:48



Comments: 1000/054 ROUTINE SURVEY, PALNVIEW UPDATED, ALL TAKEN IN CLEAN AREA WHERE < 100 CCPM.

Summary of Highest Readings

Smears	Air Samples & Wipes
2) 3621 DPM/100 CM2 B /γ 1) 1800 DPM/100 CM2 B /γ 3) 1465 DPM/100 CM2 B /γ	

Dose Rate

*150 - - Contact Reading
+75 - - 30 cm Reading
20 - - General Area

HS-50 Hot Spot
Posting
Drip Bag

15 Smear
15 Air Sample
RM
Wipe

Type = Monthly

RWP: 15

Reactor Power = 100%

*Unless otherwise Noted, dose rate in mrem/hr.

Surveyor:

Reviewed by:

ENABLING OBJECTIVES (continued)

4. State the approval requirements for an individual at Duke Power Company to exceed the **basic** permissible exposure limit of 2.0 rem. (R4)

5. State the special dose limits established for the general public. (R5)

6. Describe the special dose control measures used to protect the fetus of a "declared" pregnant radiation worker. (R6)

7. Recognize that in "exceptional situations", it is possible to allow an adult radiation worker to receive additional exposure, apart from normal occupational exposure. (R7)

8. Define and describe the specific site area for each of the following terms relating to the control of station areas: (R8)

   8.1 Unrestricted Area

   8.2 Restricted Area

   8.3 Controlled Area

   8.4 Radiation Control Area (RCA)

   8.5 Radiation Control Zone (RCZ)

   8.6 Radiation Area (RA)

   8.7 High Radiation Area (HRA)

   8.8 Extra High Radiation Area (EHRA)

   8.9 Very High Radiation Area (VHRA)

   8.10 Airborne Radioactivity Area

   8.11 Hot Spot

   8.12 Significant Dose Contributor

   8.13 Low Exposure Waiting Area

   8.14 Contaminated Area

507.8 EXPOSURE AND CONTAMINATION CONTROL

DPC adheres to the conservative assumption that there is a risk associated with radiation exposure. Application of the As Low As Reasonably Achievable (ALARA) concept minimizes this risk. Nuclear facility management and all individuals who perform work at the facility share the goal of keeping dose ALARA. Individuals are expected to be knowledgeable in and practice exposure control techniques.

507.8.1 AS LOW AS REASONABLY ACHIEVABLE (ALARA) PROGRAM

The ALARA Program is designed to minimize dose. The ALARA Program is described in the System ALARA Manual, Section III. Some important ALARA Program components are:

- Holding pre-job and post-job briefs
- Pre-planning jobs
- Using training mock-ups
- Using engineering controls
- Removing sources of radiation exposure
- Applying lessons learned from industry events
- Providing job feedback
- Using ALARA Suggestion Forms

A. Planning for Tasks <500 mrem Total Exposure

- Use basic ALARA principles
- RP ALARA Group involvement is not required.

B. ALARA Planning for Tasks Greater Than or Equal to 500 mrem

All Work Order tasks greater than or equal to 500 mrem are planned and tracked using an ALARA package. The package consists of:

- ALARA Planning Worksheet (System ALARA Manual, Section IV)
- ALARA Briefing Checklist (System ALARA Manual, Section IV)
- Execution Team Post-Job ALARA Critique (System ALARA Manual, Section IV)
- RP ALARA Post-Job Critique (System ALARA Manual, Section IV)

C Dose Tracking

- Task supervisor and execution team has primary responsibility for tracking all exposures received.
- RP shall be contacted when received dose exceeds expected values.

D Post-Job Critique

- Provide problem and improvement ideas to RP on Execution Team Post-job Critique (when provided) or use ALARA Suggestion Form.

507.8.2 ACCESS CONTROL

Controls are in place to limit access to site and in-plant areas for security and radiological safety purposes. The majority of radiologically controlled areas are located within the RCA; however, Radiation Control Zones may also be established at locations outside of the RCA.

ENABLING OBJECTIVES (continued)

18. Describe the method used at Oconee to indicate whether an article is "clean" and can be unconditionally released from the RCA, or is above the contamination limit for unconditional release. (R18)

19. State the maximum contamination limit (in cpm) for personal clothing, body surfaces, and hand-held items for unconditional release from the RCA. (R19)

20. Describe how designated contaminated tools used inside the RCA are identified. (R20)

21. Concerning RWPs/SRWPs: (R21)

    21.1 Explain the purpose of RWPs and SRWPs.

    21.2 Explain the differences between RWPs and SRWPs.

    21.3 List the requirements for re-evaluating SRWPs.

    21.4 Understand that indivduals do not have the authority to deviate from RWP or SRWP requirements.

    21.5 Identify plant locations of SRWP information.

22. Define ALARA. (R22)

23. Identify the various methods available to aid in maintaining exposures ALARA. (R23)

24. Given a set of conditions, correctly apply the radiation protection practices addressed in this lesson plan. (R24) *

* P.1 CHE 11-18-97

RCA/RCZs are posted with warning signs that clearly identify the radiological hazard(s) and other pertinent access information. All individuals at a nuclear facility are expected to:

- Comply with RCA/RCZ entrance/exit requirements
- Read and comply with posted warning signs and/or barricades.
- Maintain the integrity of barricades after entering or exiting an RCA/RCZ or when working in close proximity to one.
- Regard RCZ ropes as if they are walls. Do not reach across or move ropes unless authorized by RP.
- Notify RP with questions or problems.

A. **Definitions**

Airborne Radioactivity Area - An area containing airborne radioactivity that is equal to or greater than 25% of 1 Weighted Derived Air Concentration (DAC).

Contaminated Area - An area where loose contamination equal to or greater than 1000 dpm/100cm^2 beta/gamma and/or 20 dpm/100cm^2 alpha exists.

Extra High Radiation Area - An area with a dose rate greater than 1000 mrem/hour at 30 centimeters. These areas are locked or guarded and require continuous RP coverage for entry. In areas that can not be reasonably locked, a flashing yellow light is used as a warning device.

High Radiation Area - An area with a dose rate greater than 100 mrem/hour at 30 centimeters.

Hot Spot - A localized source of radiation that is at least five times the general area dose rates, has a contact dose rate greater than 100 mrem/hour and/or is located where the potential for significant personnel exposure exists.

Low Exposure Waiting Area (LEWA) - An area where the dose rate is less than the general area. Usually, the lowest dose rate location in a room or area.

Protected Area - Area within the double fence around the plant. Access requires security identification.

Radiation Area - An area with a dose rate greater than 5 mrem/hour at 30 centimeters.

Radiation Control Area (RCA) - An area established within the Restricted Area to provide additional access control for radiological safety purposes. Requirements for entry are located in Section 507.5.

Radiation Control Zone (RCZ) - An area where specific radiological hazards exist and which are defined and controlled in accordance with 10CFR20 requirements. Requirements for entry are located in Section 507.5.

Radioactive Material - An area where radioactive materials are stored.

Restricted Area - Any area where access is controlled by the licensee for purposes of protecting individuals from exposure to radiation and radioactive materials. At DPC nuclear facilities, the Restricted Area includes the Reactor Building(s), Auxiliary Building(s), Turbine Building(s), Service Building and fenced area adjacent to the above buildings. At ONS, the Independent Spent Fuel Storage Installation, Radwaste Facility, and some warehouses are also in Restricted Areas.

Significant Dose Contributor - An area normally posted with a Significant Dose Contributor sign and florescent green ribbon to identify highest dose rate area(s) in a Radiation Area, High Radiation Area or Extra High Radiation Area.

Unrestricted Area - Any area where access is neither limited or controlled outside the site boundary fence.

Very High Radiation Area - An area with a dose rate greater than 500 rad/hour at 1 meter. These areas are locked at all times and requires continuous RP coverage for entry.

507.8.4 CONTAMINATION PREVENTION AND CONTROL METHODS

RP radioactive contamination controls are implemented to minimize contamination of personnel, areas and equipment. Surface contamination controls minimize possible inhalation or ingestion of radioactivity, skin dose from small particles of radioactivity and the spread of contamination to the environment. Some methods used to control contamination are:

A. Performing surveys, establishing RCZs and posting warning signs in areas where sources of contamination exist.

B. Limiting eating, drinking, storage of food, chewing and use of tobacco products to authorized areas.

- Persons with medical conditions, such as heart problems, are allowed to carry emergency medication inside RCZs, RP personnel should be contacted in advance for instructions.

C. Planning and performing work to minimize spread of contamination and reduce number of contaminated areas.

D. Decontaminating surfaces whenever practical. See Section 507.8.8 for policy.

E. Bagging and tagging contaminated material, equipment and tools. See Section 507.9.3 for policy.

F. Securing equipment/cords/hoses that cross contaminated RCZ boundaries.

G. Using protective clothing appropriately to prevent becoming contaminated:

- Adhere to protective clothing requirements specified on the SRWP or RWP unless authorized to deviate by RP.
- Consult RP if SRWP or RWP requirements should be changed.
- Use radiological protective clothing only in the RCA or RCZ unless otherwise specified by RP.
- Leave the work area if conditions change and cause specified protective clothing to be inappropriate or inadequate. Notify RP.
- Leave the work area if protective clothing becomes torn, soaked, untaped or unfastened. Notify RP.
- Control protective clothing during removal and place it in appropriate containers.
- Remove all protective clothing (except modesty garments) before exiting the RCZ boundary (unless directed otherwise by RP).

H. Catch Containment Program

Catch containments are used to prevent the spread of contamination that occurs as a result of uncontrolled plant system leaks. Routine inspections/audits of catch containments are performed to ensure containments are functioning properly and in good condition.

I. Radiological Respiratory Protection Program

The primary objective of the Respiratory Program is to minimize inhalation of airborne radioactive materials by individuals. The preferred method of achieving this objective is the use of engineering controls. Engineering controls are built into the nuclear facilities to remove airborne radioactive materials from the work environment. When additional engineering controls, such as local exhaust ventilation, containment or decontamination cannot be used or are not practical, the following methods are used to maintain dose ALARA.

- Increasing monitoring and access control
- Limiting exposure times
- Using respiratory protection equipment

F. Lab Coats

Lab coats are used for limited purposes such as performing analysis on radioactive samples or laundry operations. Lab coats provide only minimal protection from contamination and should not be used for most contamination control purposes.

G. Modesty Tops/Gym Shorts

These items are provided to be worn under protective clothing to protect personal underclothing and to provide body cover after protective clothing removal.

H. Facial Protection

- Goggles and face shields are provided to protect the eyes and face from contamination when working in close proximity to contaminated components and from the splashing of contaminated liquid. Goggles and face shields also provide protection from beta exposure.
- Facial protection (such as masks, face socks, disposable face shields) provides protection to the skin of the face from contamination.

I. Worker Suiting Up

- Obtain the required protective clothing.
- Remove all personal outer clothing, if required.

  Note: T-shirts are considered outer personal clothing unless worn as a undershirt.
- Put on gym shorts (and modesty top, if worn).
- Put on required coveralls.
- Place dosimetry on a neck strap under the coveralls or place dosimetry inside the coverall pocket with the TLD in front of the ED with the TLD beta window facing away from the body. Secure or tape the pocket opening.
- Put on booties over personal shoes (these may be worn inside or outside of the coveralls).
- Secure outer booties (tape, elastic, velcro, straps).
- Put on rubber shoe covers.
- Put on hood.
- Put on cotton gloves and then rubber gloves or surgical gloves.
- Secure gloves over sleeves of coveralls (tape, elastic, velcro, straps).
- Return unused protective clothing to storage location.

J. Removal of Protective Clothing

Do not throw protective clothing or equipment across the exit area; this can result in the spread of contamination.

- Remove all tape, elastic or velcro, if used (from wrists, ankles, etc.).
- Remove rubber shoe covers.
- Remove rubbers gloves by peeling them off inside out.
- Remove hood, taking care not to contaminate hair or face.
- Remove dosimetry.
- Remove the coveralls, peeling off inside out.
- Remove booties as you transfer to the step-off pad which is clean.
- Remove cotton gloves and proceed to nearest monitor wearing the gym shorts (and modesty top). Hand and foot monitoring is required at minimum before proceeding to change room.
- Monitor whole body to ensure you are not contaminated before dressing in personal clothes.
- In cases where more than one set of overalls are worn or multiple step off pads are used, follow the instructions of RP for dress out and removal.

507.8.6 PERSONNEL CONTAMINATION MONITORING

Admin Exam A.4 Emergency Plan (SRO)

Question #1:

Based on the event that just occurred on the simulator describe your actions as the Emergency Coordinator concerning classifying this event. Sequence all classification thresholds throughout the entire event.

ANSWER:
ALERT based on HPI Cooling established Fission Product Matrix total = 4

Question #2:
If condenser vacuum was lost with the Condenser rupture disc blown and the 1B SGTL increases to 70 gpm how does this affect the E-Plan classification and any PAGs that may apply?

ANSWER:
Upgrade to SAE based on Fission Product Matrix total = 7 (HPI Cooling (4)+ SGTL > 10 gpm with direct opening to the environment (3).
No PAGs are required.

(R06-97)

INFORMATION ONLY

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. RP/0/B/1000/001

Revision No 6

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Emergency Classification

(4) Prepared By Donnie Kelley Date 2-16-99

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)

☐ No (Revision with minor changes)

☐ No (To incorporate previously approved changes)

(6) Reviewed By W. C. Brandt (QR) Date 2/25/99

Cross-Disciplinary Review By ______ (QR)NA WCB Date 2/25/99

Reactivity Mgmt. Review By ______ (QR)NA WCB Date 2/25/99

(7) Additional Reviews

QA Review By ______ Date ______

Reviewed By ______ Date ______

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By M R Thorne Date 3-27-99

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?

☐ Yes ☐ NA Listed enclosures attached?

☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?

☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?

☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Site **Emergency Classification** **Reference Use**	Procedure No. RP/0/B/1000/001 Revision No. 006 Electronic Reference No. OX002WOS

Emergency Classification

NOTE:	This procedure is an implementing procedure tot he Oconee Nuclear Site Emergency plan and must be forwarded to Emergency Planning within three (3) working days of approval.

1. Symptoms

1.1 This procedure describes the immediate actions to be taken to recognize and classify an emergency condition.

1.2 This procedure identifies the four emergency classifications and their corresponding Emergency Action Levels (EALs).

1.3 This procedure provides reporting requirements for non-emergency abnormal events.

1.4 The following guidance is to be used by the Emergency Coordinator/EOF Director in assessing emergency conditions:

1.4.1 The Emergency Coordinator/EOF Director shall review all applicable initiating events to ensure proper classification.

1.4.2 The BASIS Document (Volume A, Section D of the Emergency Plan) is available for review if any questions arise over proper classification.

1.4.3 **IF** An event occurs on more than one unit concurrently,
THEN The event with the higher classification will be classified on the Emergency Notification Form.

A. Information relating to the problem(s) on the other unit(s) will be captured on the emergency Notification Form as shown in RP/0/B/1000/015A, (Offsite Communications From The Control Room), RP/0/B/1000/015B, (Offsite Communications From The Technical Support Center) or RP/0/B/1000/015C, (Offsite Communications From The Emergency Operations Facility).

1.4.4 **IF** An event occurs,
AND A lower or higher plant operating mode is reached before the classification can be made,
THEN The classification shall be based on the mode that existed at the time the event occurred.

1.4.5 The Fission Product Barrier Matrix is applicable only to those events that occur at Hot Shutdown or higher.

A. An event that is recognized at Cold Shutdown or lower shall not be classified using the Fission Product Barrier Matrix.

1. Reference should be made to the additional enclosures that provide Emergency Action Levels for specific events (e.g., Severe Weather, Fire, Security).

1.5 **IF** A transient event should occur,

THEN Review the following guidance:

1.5.1 **IF** An Emergency Action Level (EAL) identifies a specific duration
AND The Emergency Coordinator/EOF Director assessment concludes that the specified duration is exceeded or will be exceeded, (i.e.; condition cannot be reasonably corrected before the duration elapses),
THEN Classify the event.

1.5.2 **IF** A plant condition exceeding EAL criteria is corrected before the specified duration time is exceeded,
THEN The event is **NOT** classified by that EAL.

A. Review lower severity EALs for possible applicability in these cases.

NOTE:	Reporting under 10CFR50.72 may be required for the following step. Such a condition could occur, for example, if a follow up evaluation of an abnormal condition uncovers evidence that the condition was more severe than earlier believed.

1.5.3 **IF** A plant condition exceeding EAL criteria is not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g.; as a result of routine log or record review)
AND The condition no longer exists,
THEN An emergency shall **NOT** be declared.

1.5.4 **IF** An emergency classification was warranted, but the plant condition has been corrected prior to declaration and notification,
THEN The Emergency Coordinator must consider the potential that the initiating condition (e.g.; Failure of Reactor Protection System) may have caused plant damage that warrants augmenting the on shift personnel through activation of the Emergency Response Organization.

A. **IF** An **Unusual Event** condition exists,
THEN Make the classification as required.

1. The event may be terminated in the same notification or as a separate termination notification.

B. **IF** **An Alert, Site Area Emergency**, or **General Emergency** condition exists,
THEN Make the classification as required,
AND Activate the Emergency Response Organization.

1.6 Emergency conditions shall be classified as soon as the Emergency Coordinator/EOF Director assessment determines that the Emergency Action Levels for the Initiating Condition have been exceeded.

2. Immediate Actions

2.1 Determine the operating mode that existed at the time the event occurred prior to any protection system or operator action initiated in response to the event.

2.2 **IF** The unit is at Hot Shutdown or higher
AND The condition/event affects fission product barriers,
THEN GOTO Enclosure 4.1, (Fission Product Barrier Matrix).

2.2.1 Review the criteria listed in Enclosure 4.1, (Fission Product Barrier Matrix) and make the determination if the event should be classified.

2.3 Review the listing of enclosures to determine if the event is applicable to one of the categories shown.

2.3.1 **IF** One or more categories are applicable to the event,
THEN Refer to the associated enclosures.

2.3.2 Review the EALs and determine if the event should be classified.

A. **IF** An EAL is applicable to the event,
THEN Classify the event as required.

2.4 **IF** The condition requires an emergency classification,
THEN GOTO RP/0/B/1000/002, (Control Room Emergency Coordinator Procedure).

3. Subsequent Actions

3.1 Continue to review the emergency conditions to assure the current classification continues to be applicable.

4. Enclosures

DETERMINE THE APPROPRIATE CLASSIFICATION USING THE TABLE BELOW: CIRCLE EALS CHOSEN. ADD POINTS TO CLASSIFY. (SEE NOTE BELOW)

RCS BARRIERS (BD 5-7)		FUEL CLAD BARRIERS (BD 8-9)		CONTAINMENT BARRIERS (BD 10-12)	
Potential Loss (4)	Loss (5)	Potential Loss (4)	Loss (5)	Potential Loss (1)	Loss (3)
RCS Leakrate > Makeup capacity of one HPI pump in normal makeup mode (approx. 160 gpm) with Letdown isolated.	RCS Leak rate > available makeup capacity as indicated by a loss of subcooling	Average of the 5 highest CETC ≥ 700° F	Average of the 5 highest CETC ≥ 1200° F	CETC ≥ 1200° F ≥ 15 minutes OR CETC ≥ 700° F ≥ 15 minutes with a valid RVLS reading 0"	Rapid unexplained containment pressure decrease after increase OR containment pressure or sump level not consistent with LOCA
SGTR > Makeup capacity of one HPI pump in normal makeup mode (approx. 160 gpm) with Letdown isolated.		Valid RVLS reading of 0"	Coolant activity ≥ 300 µCi/ml DEI	RB pressure ≥ 59 psig OR RB pressure ≥ 10 psig and no RBCU or RBS.	Failure of secondary side of SG results in a direct opening to the environment with P/S leakage ≥ 10 gpm in the same SG
Entry into the TSOR (Thermal Shock) operating range	1RIA 57/58 reading ≥ 1.0 R/hr 2 RIA 57 reading ≥ 1.6 R/hr 2 RIA 58 reading ≥ 1.0 R/hr 3RIA 57/58 reading ≥ 1.0 R/hr		Hours Since SD — RIA57/58 - R/hr 0 - < 0.5 — ≥ 300/150 0.5 - < 2.0 — ≥ 80/40 2.0 - 8.0 — ≥ 32/16	Hours Since SD — RIA57/58 - R/hr 0 - < 0.5 — ≥ 1800/860 0.5 - < 2.0 — ≥ 400/195 2.0 - 8.0 — ≥ 280/130	Failure of secondary side of SG results in a direct opening to the environment with P/S leakage ≥ 10 gpm in the other SG AND Feeding SG with secondary side failure from the affected unit
HPI Forced Cooling	RCS pressure spike ≥ 2750 psig			Hydrogen concentration ≥ 9%	Containment isolation is incomplete and a release path to the environment exists
Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment

UNUSUAL EVENT (1-3)	ALERT (4-6)	SITE AREA EMERGENCY (7-10)	GENERAL EMERGENCY (11-13)
OPERATING MODE: 1, 2, 3, 4 • Any potential loss of Containment • Any loss of containment	OPERATING MODE: 1, 2, 3, 4 • Any potential loss or loss of the Fuel Clad • Any potential loss or loss of the RCS	OPERATING MODE: 1, 2, 3, 4 • Loss of any two barriers • Loss of one barrier and potential loss of either RCS or Fuel Clad Barriers • Potential loss of both the RCS and Fuel Clad Barriers	OPERATING MODE: 1, 2, 3, 4 • Loss of any two barriers and potential loss of the third barrier • Loss of all three barriers
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4

NOTE: An event with multiple events could occur which would result in the conclusion that exceeding the loss or potential loss threshold is <u>IMMINENT</u> (i.e., within 1-3 hours). In this IMMINENT LOSS situation, use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. RCS LEAKAGE (BD 14) == OPERATING MODE: 1, 2, 3, 4 • Unidentified leakage ≥ 10 gpm • Pressure boundary leakage ≥ 10 gpm • Identified leakage ≥ 25 gpm 2. UNPLANNED LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/INDICATION IN CONTROL ROOM FOR > 15 MINUTES (BD 15) == OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND Loss of anunciators or indicators requires additional personnel (beyond normal shift complement) to safely operate the unit 3. INABILITY TO REACH REQUIRED SHUTDOWN WITHIN LIMITS (BD 16) == OPERATING MODE: 1, 2, 3, 4 • Required operating mode not reached within TS LCO action statement time (CONTINUED)	1. UNPLANNED LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/INDICATION IN CONTROL ROOM (BD 19) == OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND Loss of anunciators/indicators requires additional personnel (beyond normal shift complement) to safely operate the unit AND EITHER OF THE FOLLOWING: • Significant plant transient in progress OR • Loss of the OAC and ALL PAM indications (END)	1. INABILITY TO MONITOR A SIGNIFICANT TRANSIENT IN PROGRESS (BD 21) == OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND A significant transient is in progress AND Loss of the OAC and ALL PAM indications AND Inability to directly monitor any one of the following functions: 1. Subcriticality 2. Core Cooling 3. Heat Sink 4. RCS Integrity 5. Containment Integrity 6. RCS Inventory (END)	
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4. UNPLANNED LOSS OF ALL ONSITE OR OFFSITE COMMUNICATIONS (BD 17) == OPERATING MODE: All ♦ Loss of all onsite communications capability (ROLM system, PA system, Pager system, Onsite Radio system) affecting ability to perform routine operations ♦ Loss of all onsite communications capability (Selective signaling, NRC FTS lines, Offsite Radio System, AT&T line) affecting ability to communicate with offsite authorities. 5. FUEL CLAD DEGRADATION (BD 18) == OPERATING MODE: All: ♦ DEI - >5µCi/ml (END)			
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4			

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. ANY UNPLANNED RELEASE OF GASEOUS OR LIQUID RADIOACTIVITY TO THE ENVIRONMENT THAT EXCEEDS TWO TIMES THE SLC LIMITS FOR 60 MINUTES OR LONGER (BD 23) === OPERATING MODE: All ♦ Valid indication on radiation monitor RIA 33 of ≥ 4.06E+06 cpm for > 60 minutes (See Note 1) ♦ Valid indication on radiation monitor RIA 45 of ≥ 1.33E+06 cpm for > 60 minutes (See Note 1) ♦ Liquid effluent being released exceeds two times SLC 16.11.1 for > 60 minutes as determined by Chemistry Procedure ♦ Gaseous effluent being released exceeds two times SLC 16.11.2 for > 60 minutes as determined by RP Procedure NOTE 1: If monitor reading is sustained for the time period indicated in the EAL AND the required assessments (procedure calculations) cannot be completed within this period, declaration must be made on the valid Radiation Monitor reading. (CONTINUED)	1. ANY UNPLANNED RELEASE OF GASEOUS OR LIQUID RADIOACTIVITY TO THE ENVIRONMENT THAT EXCEEDS 200 TIMES RADIOLOGICAL TECHNICAL SPECIFICATIONS FOR 15 MINUTES OR LONGER (BD 28) === OPERATING MODE: All ♦ Valid indication on RIA 46 of ≥ 2.98E+04 cpm for >15 minutes (See Note 1) ♦ RIA 33 HIGH Alarm AND Liquid effluent being released exceeds 200 times the level of SLC 16.11.1 for > 15 minutes as determined by Chemistry Procedure ♦ Gaseous effluent being released exceeds 200 times the level of SLC 16.11.2 for >15 minutes as determined by RP Procedure 2. RELEASE OF RADIOACTIVE MATERIAL OR INCREASES IN RADIATION LEVELS THAT IMPEDES OPERATION OF SYSTEMS REQUIRED TO MAINTAIN SAFE OPERATION OR TO ESTABLISH OR MAINTAIN COLD SHUTDOWN (BD 30) === OPERATING MODE: All ♦ Valid radiation reading ≥ 15 mRad/hr in CR, CAS OR Radwaste CR ♦ Unplanned/unexpected valid area monitor readings exceed limits stated in Enclosure 4.9 (CONTINUED)	1. BOUNDARY DOSE RESULTING FROM ACTUAL/ IMMINENT RELEASE OF GASEOUS ACTIVITY (BD 32) === OPERATING MODE: All ♦ Valid reading on RIA 46 of ≥ 2.98E+05 cpm for >15 minutes (See Note 2) ♦ Valid reading on RIA 57 or 58 as shown on Enclosure 4.8 (See Note 2) ♦ Dose calculations result in a dose projection at the site boundary of: ≥ 100 mRem TEDE or 500 mRem CDE adult thyroid ♦ Field survey results indicate site boundary dose rates exceeding ≥100 mRad/hr expected to continue for more than one hour OR Analyses of field survey samples indicate adult thyroid dose commitment of ≥ 500 mRem CDE (3.84 E^{-7} µCi/ml) for one hour of inhalation NOTE 2: If actual Dose Assessment cannot be completed within 15 minutes, then the valid radiation monitor reading should be used for emergency classification. (CONTINUED)	1. BOUNDARY DOSE RESULTING FROM ACTUAL/ IMMINENT RELEASE OF GASEOUS ACTIVITY (BD 36) === OPERATING MODE: All ♦ Valid reading on RIA 46 of ≥ 2.98E+06 cpm for ≥15 minutes (See Note 3) ♦ Valid reading on RIA 57 or 58 as shown on Enclosure 4.8 (See Note 3) ♦ Dose calculations result in a dose projection at the site boundary of: ≥ 1000 mRem TEDE OR ≥ 5000 mRem CDE adult thyroid ♦ Field survey results indicate site boundary dose rates exceeding ≥1000 mRad/hr expected to continue for more than one hour OR Analyses of field survey samples indicate adult thyroid dose commitment of ≥ 5000 mRem CDE for one hour of inhalation NOTE 3: If actual Dose Assessment cannot be completed within 15 minutes, then the valid radiation monitor reading should be used for emergency classification. (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

Assumptions used for calculation of vent monitors RIA 45 & 46:

1. Average annual meteorology (1.672 E-6 sec/m^3), semi-elevated
2. Vent flow rate 65,000 cfm (average daily flow rate)
3. No credit is taken for vent filtration
4. One hour release duration for Unusual Event, 15 minute duration for Alert, Site Area Emergency, General Emergency
5. General Emergency PAGs are 1 rem TEDE and 5 rem CDE; Site Area Emergency determination is based on 10% of the General Emergency PAGs
6. Calculations for monitor readings are based on whole body dose
7. Standard ODCM guidance together with NUMARC guidance indicates that effluent releases are based on Technical Specification releases

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2. UNEXPECTED INCREASE IN PLANT RADIATION OR AIRBORNE CONCENTRATION (BD 25) == OPERATING MODE: All ♦ LT 5 reading 14" and decreasing with makeup not keeping up with leakage WITH fuel in the core ♦ Uncontrolled water level decrease in the SFP and fuel transfer canal with all irradiated fuel assemblies remaining covered by water ♦ 1 R/hr radiation reading at one foot away from a damaged storage cask located at the ISFSI ♦ Valid area monitor readings exceeds limits stated in Enclosure 4.9. (END)	3. MAJOR DAMAGE TO IRRADIATED FUEL OR LOSS OF WATER LEVEL THAT HAS OR WILL RESULT IN THE UNCOVERING OF IRRADIATED FUEL OUTSIDE THE REACTOR VESSEL (BD 31) == OPERATING MODE: All ♦ Valid RIA 3, 6, 41, OR 49 HIGH Alarm ♦ HIGH Alarm for portable area monitors on the main bridge or auxiliary bridge or SFP bridge ♦ Report of visual observation of irradiated fuel uncovered ♦ Operators determine water level drop in either the SFP or fuel transfer canal will exceed makeup capacity such that irradiated fuel will be uncovered (END)	2. LOSS OF WATER LEVEL IN THE REACTOR VESSEL THAT HAS OR WILL UNCOVER FUEL IN THE REACTOR VESSEL (BD 35) == OPERATING MODE: 5, 6 ♦ Failure of heat sink causes loss of Cold Shutdown condition AND LT 5 indicates 0 inches after initiation of RCS makeup ♦ Failure of heat sink causes loss of Cold Shutdown condition AND Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup NOTE: This Initiating Condition is also located in Enclosure 4.4, (Loss of Shutdown Functions). High radiation levels will also be seen with this condition. (END)	
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	

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<tr><th>UNUSUAL EVENT</th><th>ALERT</th><th>SITE AREA EMERGENCY</th><th>GENERAL EMERGENCY</th></tr>
<tr><td></td>
<td>1. FAILURE OF RPS TO COMPLETE OR INITIATE A Rx SCRAM (BD 39)
==========
OPERATING MODE 1, 2, 3

• Valid reactor trip signal received or required WITHOUT automatic scram

AND ONE OF THE FOLLOWING

DSS has inserted Control Rod Groups 5, 6, 7

OR

Manual trip from the Control Room is successful and reactor power is less than 5% and decreasing

2. INABILITY TO MAINTAIN PLANT IN COLD SHUTDOWN (BD 41)
==========
OPERATING MODE: 5, 6

• Loss of LPI and/or LPSW

AND

Inability to maintain RCS temperature below 200° F as indicated by either of the following:

RCS temperature at the LPI Pump Suction OR visual observation

(END)</td>
<td>1. FAILURE OF RPS TO COMPLETE OR INITIATE A Rx SCRAM (BD 42)
==========
OPERATING MODE: 1, 2

• Valid reactor trip signal received or required WITHOUT automatic scram

AND

DSS has NOT inserted Control Rod Groups 5, 6, 7

AND

Manual trip from the Control Room was NOT successful in reducing reactor power to less than 5% and decreasing

2. COMPLETE LOSS OF FUNCTION NEEDED TO ACHIEVE OR MAINTAIN HOT SHUTDOWN (BD 43)
==========
OPERATING MODE: 1, 2, 3, 4

• Average of the 5 highest CETCs ≥1200° F shown on ICCM

• Unable to maintain reactor subcritical

• SSF feeding SG per EOP

(CONTINUED)</td>
<td>1. FAILURE OF RPS TO COMPLETE AUTOMATIC SCRAM AND MANUAL SCRAM NOT SUCCESSFUL WITH INDICATION OF CORE DAMAGE (BD 45)
==========
OPERATING MODE: 1, 2

• Valid Rx trip signal received or required WITHOUT automatic scram

AND

Manual trip from the Control Room was NOT successful in reducing reactor power to < 5% and decreasing

AND

Average of the 5 highest CETCs ≥1200° F on ICCM

(END)</td></tr>
<tr><td></td>
<td>INITIAL NOTIFICATION REQUIREMENTS:
SEE EMERGENCY TELEPHONE DIRECTORY

NOTIFY 1, 2, 3, 4</td>
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NOTIFY 1, 2, 3, 4</td></tr>
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UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
		3. LOSS OF WATER LEVEL IN THE REACTOR VESSEL THAT HAS OR WILL UNCOVER FUEL IN THE REACTOR VESSEL (BD 44) == OPERATING MODE: 5, 6 ♦ Failure of heat sink causes loss of Cold Shutdown conditions AND LT-5 indicates 0 inches after initiation of RCS makeup ♦ Failure of heat sink causes loss of Cold Shutdown conditions AND Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup (END)	
		INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. LOSS OF ALL OFFSITE POWER TO ESSENTIAL BUSSES FOR GREATER THAN 15 MINUTES (BD 47) == <u>OPERATING MODE:</u> All ♦ Loss of all offsite AC power to both the Red and Yellow Busses for > 15 minutes <u>AND</u> Unit auxiliaries are being supplied from Keowee or CT5 2. UNPLANNED LOSS OF REQUIRED DC POWER FOR GREATER THAN 15 MINUTES (BD 48) == <u>OPERATING MODE:</u> 5, 6 ♦ Unplanned loss of vital DC power to required DC busses as indicated by bus voltage less than 110 VDC <u>AND</u> Failure to restore power to at least one required DC bus within 15 minutes from the time of loss (END)	1. LOSS OF ALL OFFSITE AC POWER AND LOSS OF ALL ONSITE AC POWER TO ESSENTIAL BUSSES (BD 49) == <u>OPERATING MODE:</u> 5, 6 Defueled ♦ MFB 1 and 2 de-energized <u>AND</u> Failure to restore power to at least one MFB within 15 minutes from the time of loss of both offsite and onsite AC power 2. AC POWER CAPABILITY TO ESSENTIAL BUSSES REDUCED TO A SINGLE SOURCE FOR GREATER THAN 15 MINUTES (BD 50) == <u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ AC power capability has been degraded to a single power source for > 15 minutes due to the loss of all but one of: Unit Normal Transformer Unit SU Transformer Another Unit SU Transformer CT4 CT5 (END)	1. LOSS OF ALL OFFSITE AC POWER AND LOSS OF ALL ONSITE AC POWER TO ESSENTIAL BUSSES (BD 51) == <u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ MFB 1 and 2 de-energized <u>AND</u> Failure to restore power to at least one MFB within 15 minutes from the time of loss of both offsite and onsite AC power 2. LOSS OF ALL VITAL DC POWER (BD 52) == <u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ Unplanned loss of vital DC power to required DC busses as indicated by bus voltage less than 110 VDC <u>AND</u> Failure to restore power to at least one required DC bus within 15 minutes from the time of loss (END)	1. PROLONGED LOSS OF ALL OFFSITE POWER AND ONSITE AC POWER (BD 54) == <u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ MFB 1 and 2 de-energized <u>AND</u> SSF fails to maintain Hot Shutdown <u>AND</u> At least one of the following conditions exist: Restoration of power to at least one MFB within 4 hours is <u>NOT</u> likely <u>OR</u> Indications of continuing degradation of core cooling based on Fission Product Barrier monitoring (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. FIRES/EXPLOSIONS WITHIN THE PLANT (BD 57) == OPERATING MODE: All NOTE: Within the plant means Turbine Building, Auxiliary Building, Reactor Building, Keowee Hydro ♦ Fire within the plant not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm ♦ Unanticipated explosion within the plant resulting in visible damage to permanent structures/equipment 2. CONFIRMED SECURITY THREAT INDICATES POTENTIAL DEGRADATION IN THE LEVEL OF SAFETY OF PLANT (BD 58) == OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications ♦ Discovery of bomb within plant protected area and outside security vital areas ♦ Hostage/Extortion situation ♦ Violent civil disturbance within the owner controlled area (END)	1. FIRE/EXPLOSION AFFECTING OPERABILITY OF PLANT SAFETY SYSTEMS REQUIRED TO ESTABLISH/MAINTAIN SAFE SHUTDOWN (BD 59) == OPERATING MODE: All NOTE: Only one train of a system needs to be affected or damaged in order to satisfy this condition. ♦ Fire/explosions AND ONE OF THE FOLLOWING: Affected safety-related system parameter indications show degraded performance OR Plant personnel report visible damage to permanent structures or equipment required for safe shutdown 2. SECURITY EVENT IN A PLANT PROTECTED AREA (BD 60) == OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications ♦ Intrusion into plant protected area by a hostile force ♦ Bomb discovered in an area containing safety related equipment (END)	1. SECURITY EVENT IN A PLANT VITAL AREA (BD 61) == OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications ♦ Intrusion into any of the following plant areas by a hostile force: Reactor Building Auxiliary Building Keowee Hydro ♦ Bomb detonated in the following areas: • Keowee Hydro • Keowee Dam • ISFSI • Reactor Building • Auxiliary Building • SSF (END)	1. SECURITY EVENT RESULTING IN LOSS OF ABILITY TO REACH AND MAINTAIN COLD SHUTDOWN (BD 62) == OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications ♦ Loss of physical control of the control room due to security event ♦ Loss of physical control of the Aux Shutdown panel and the SSF due to a Security Event (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

Enclosure 4.7
Natural Disasters, Hazards and Other Conditions Affecting Plant Safety

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING THE PROTECTED AREA (BD 64) ========== OPERATING MODE: All ♦ Tremor felt and valid alarm on the strong motion accelerograph ♦ Tornado striking within Protected Area Boundary ♦ Vehicle crash into plant structures/systems within the Protected Area Boundary ♦ Turbine failure resulting in casing penetration or damage to turbine or generator seals (CONTINUED)	1. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING THE PLANT VITAL AREA (BD 69) ========== OPERATING MODE: All ♦ Tremor felt and seismic trigger actuates (0.05g) ♦ Tornado, high winds, missiles resulting from turbine failure, vehicle crashes, or other catastrophic event AND ONE OF THE FOLLOWING: NOTE: Only one train of a safety-related system needs to be affected or damaged in order to satisfy these conditions. Visible damage to permanent structures or equipment required for safe shutdown of the unit OR Affected safety system parameter indications show degraded performance 2. RELEASE OF TOXIC/FLAMMABLE GASES JEOPARDIZING SYSTEMS REQUIRED TO MAINTAIN SAFE OPERATION OR ESTABLISH/ MAINTAIN COLD SHUTDOWN (BD 71) ========== OPERATING MODE: All ♦ Report/detection of toxic gases in concentrations that will be life-threatening to plant personnel ♦ Report/detection of flammable gases in concentrations that will affect the safe operation of the plant: • Reactor Building • Auxiliary Building • Turbine Building • Control Room (CONTINUED)	1. CONTROL ROOM EVACUATION AND PLANT CONTROL CANNOT BE ESTABLISHED (BD 75) ========== OPERATING MODE: All ♦ Control Room evacuation has been initiated AND Control of the plant cannot be established from the Aux Shutdown Panel or the SSF within 15 minutes 2. KEOWEE HYDRO DAM FAILURE (BD 76) ========== OPERATING MODE: All ♦ Imminent/actual dam failure (includes any of the following: • Keowee Hydro Dam • Little River Dam • Dikes A, B, C, or D • Intake Canal Dike 3. OTHER CONDITIONS WARRANT DECLARATION OF SITE AREA EMERGENCY (BD 77) ========== OPERATING MODE: All ♦ Emergency Coordinator/EOF Director judgment (END)	1. OTHER CONDITIONS WARRANT DECLARATION OF GENERAL EMERGENCY (BD 78) ========== OPERATING MODE: All ♦ Emergency Coordinator/EOF Director judgment indicates: Actual/imminent substantial core degradation with potential for loss of containment OR Potential for uncontrolled radionuclide releases that would result in a dose projection at the site boundary greater than 1000 mRem TEDE or 5000 mRem CDE Adult Thyroid (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING KEOWEE HYDRO (BD 66) == OPERATING MODE: All ♦ Reservoir elevation ≥ 807 feet with all spillway gates open and the lake elevation continues to rise ♦ Seepage readings increase or decrease greatly or seepage water is carrying a significant amount of soil particles ♦ New area of seepage or wetness, with large amounts of seepage water observed on dam, dam toe, or the abutments ♦ Slide or other movement of the dam or abutments which could develop into a failure ♦ Developing failure involving the powerhouse or appurtenant structures and the operator believes the safety of the structure is questionable 3. RELEASE OF TOXIC OR FLAMMABLE GASES DEEMED DETRIMENTAL TO SAFE OPERATION OF THE PLANT (BD 67) == OPERATING MODE: All ♦ Report/detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant ♦ Report by local, county, state officials for potential evacuation of site personnel based on offsite event (CONTINUED)	3. TURBINE BUILDING FLOOD (BD 72) == OPERATING MODE: All ♦ Turbine Building flood requiring use of AP/1,2,3/A/1700/10, (Uncontrolled Flooding Of Turbine Building) 4. CONTROL ROOM EVACUATION HAS BEEN INITIATED (BD 73) == OPERATING MODE: All ♦ Evacuation of Control Room AND ONE OF THE FOLLOWING: Plant control IS established form the Aux Shutdown Panel or the SSF OR Plant control IS BEING established from the Aux Shutdown Panel or SSF 5. OTHER CONDITIONS WARRANT CLASSIFICATION OF AN ALERT (BD 74) == OPERATING MODE: All ♦ Emergency Coordinator judgment indicates that: Plant safety may be degraded AND Increased monitoring of plant functions is warranted (END)		
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4. OTHER CONDITIONS EXIST WHICH WARRANT DECLARATION OF AN UNUSUAL EVENT (BD 68) == OPERATING MODE: All • Emergency Coordinator determines potential degradation of level of safety has occurred (END)			
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4			

NOTE: IF Actual Dose Assessment **cannot be** completed within 15 minutes.
THEN The valid monitor reading should be used for Emergency Classification.

All RIA values are considered GREATER THAN or EQUAL TO

HOURS SINCE REACTOR TRIPPED	RIA 57 R/hr		RIA 58 R/hr*	
	Site Area Emergency	General Emergency	Site Area Emergency	General Emergency
0.0 - < 0.5	5.9E+003	5.9E+004	2.6E+003	2.6E+004
0.5 - < 1.0	2.6E+003	2.6E+004	1.1E+003	1.1E+004
1.0 - < 1.5	1.9E+003	1.9E+004	8.6E+002	8.6E+003
1.5 - < 2.0	1.9E+003	1.9E+004	8.5E+002	8.5E+003
2.0 - < 2.5	1.4E+003	1.4E+004	6.3E+002	6.3E+003
2.5 - < 3.0	1.2E+003	1.2E+004	5.7E+002	5.7E+003
3.0 - < 3.5	1.1E+003	1.1E+004	5.2E+002	5.2E+003
3.5 - < 4.0	1.0E+003	1.0E+004	4.8E+002	4.8E+003
4.0 - < 8.0	1.0E+003	1.0E+004	4.4E+002	4.4E+003

* RIA 58 is partially shielded

Assumptions used for calculation of high range in-containment monitors RIA 57 and 58:

1. Average annual meteorology (7.308 E^{-6} sec/m^3)
2. Design basis leakage (5.6 E^{6} ml/hr)
3. One hour release duration
4. General Emergency PAGs are 1 rem TEDE and 5 rem CDE; Site Area Emergency determination is based on 10% of the General Emergency PAGs
5. Calculations for monitor readings are based on CDE because thyroid dose is limiting
6. No credit is taken for filtration
7. LOCA conditions are limiting and provide the more conservative reading

NOTE: This Initiating Condition is not intended to apply to anticipated temporary increases due to planned events (e.g.; incore detector movement, radwaste container movement, depleted resin transfers, etc.).

MONITOR NUMBER	UNITS 1, 2, 3	
	UNUSUAL EVENT 1000x NORMAL LEVELS mRAD/HR	ALERT mRAD/HR
RIA 7, Hot Machine Shop Elevation 796	150	≥ 5000
RIA 8, Hot Chemistry Lab Elevation 796	4200	≥ 5000
RIA 10, Primary Sample Hood Elevation 796	830	≥ 5000
RIA 11, Change Room Elevation 796	210	≥ 5000
RIA 12, Chem Mix Tank Elevation 783	800	≥ 5000
RIA 13, Waste Disposal Sink Elevation 771	650	≥ 5000
RIA 15, HPI Room Elevation 758	NOTE*	≥ 5000

NOTE: RIA 15 normal readings are approximately 9 mRad/hr on a daily basis. Applying 1000x normal readings would put this monitor greater than 5000 mRad/hr just for an Unusual Event. For this reason, an Unusual Event will <u>NOT</u> be declared for a reading less than 5000 mRad/hr.

1. List of Definitions and Acronymns

1.1 ALERT - Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

1.2 BOMB - A fused explosive device

1.3 CONDITION A - Failure is Imminent or Has Occurred - A failure at the dam has occurred or is about to occur and minutes to days may be allowed to respond dependent upon the proximity to the dam.

1.4 CONDITION B - Potentially Hazardous Situation is Developing - A situation where failure may develop, but preplanned actions taken during certain events (such as major floods, earthquakes, evidence of piping) may prevent or mitigate failure.

1.5 CIVIL DISTURBANCE - A group of ten (10) or more people violently protesting station operations or activities at the site.

1.6 EXPLOSION - A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. A sudden failure of a pressurized pipe/line could fit this definition.

1.7 EXTORTION - An attempt to cause an action at the station by threat of force.

1.8 FIRE - Combustion characterized by heat and light. Sources of smoke, such as slipping drive belts or overheated electrical equipment, do NOT constitute fires. Observation of flames is preferred but is NOT required if large quantities of smoke and heat are observed.

1.9 GENERAL EMERGENCY - Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guidelines exposure levels outside the Exclusion Area Boundary.

1.10 HOSTAGE - A person or object held as leverage against the station to ensure demands will be met by the station.

1.11 INTRUSION/INTRUDER - Suspected hostile individual present in a Protected Area without authorization.

1.12 INABILITY TO DIRECTLY MONITOR - Operational Aid Computer data points are unavailable or gauges/panel indications are NOT readily available to the operator.

1.13 PROTECTED AREA - Encompasses all Owner Controlled Areas within the security perimeter fence.

1.14 RUPTURED (As relates to Steam Generator) - Existence of Primary to Secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

1.15 SABOTAGE - Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment unavailable.

1.16 SAFETY-RELATED SYSTEMS AREA - Any area within the Protected Area which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

1.17 SIGNIFICANT TRANSIENT - An unplanned event involving one or more of the following:

(1) Automatic turbine runback > 25% thermal reactor power
(2) Electrical load rejection > 25% full electrical load
(3) Reactor Trip
(4) Safety Injection System Activation

1.18 SITE AREA EMERGENCY - Events are in process or have occurred which involve actual or likely major failures of plant functions needed for the protection of the public. Any releases are NOT expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels outside the Exclusion Area Boundary.

1.19 SELECTED LICENSEE COMMITMENT (SLC) - Chapter 16 of the FSAR

1.20 SITE BOUNDARY - That area, including the Protected Area, in which DPC has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius from the center of Unit 2).

1.21 TOXIC GAS - A gas that is dangerous to life or health by reason of inhalation or skin contact (e.g.; Chlorine)

1.22 UNCONTROLLED - Event is not the result of planned actions by the plant staff

1.23 UNPLANNED - An event or action is UNPLANNED if it is not the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

1.24 UNUSUAL EVENT - Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

1.25 **VALID** - An indication or report or condition is considered to be **VALID** when it is conclusively verified by: (1) an instrument channel check; or, (2) indications on related or redundant instrumentation; or, (3) by direct observation by plant personnel such that doubt related to the instrument's operability, the condition's existence, or the report's accuracy is removed. Implicit with this definition is the need for timely assessment.

1.26 **VIOLENT** - Force has been used in an attempt to injure site personnel or damage plant property.

1.27 **VISIBLE DAMAGE** - Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage: deformation due to heat or impact, denting, penetration, rupture,

MODES

MODE	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER (a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 250
4	Hot Shutdown (b)	< 0.99	NA	$250 > T > 200$
5	Cold Shutdown (b)	< 0.99	NA	≤ 200
6	Refueling (c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

INITIAL SUBMITTAL

OCONEE EXAM 2000-301
50-269, 270, AND 287/2000-301

JULY 10 - 14, 18, 19, AND 20, 2000

INITIAL SUBMITTAL

SIMULATOR EXAM 1

SIMULATOR
Exam 1

Oconee
2000

NRC Copy

Facility: Oconee Scenario No.: 1 Op-Test No.: 1

Examiners: Operators:

Objectives:
The candidates will operate the simulator during all events described in the scenario as if it is actually Oconee Unit 1. During the exam the candidates will demonstrate appropriate licensed operator knowledge and abilities that will ensure safe operation of the facility during all aspects of operation. During the exam the candidates will use the following operating techniques to ensure safe plant operations and ensure health and safety of the general public is maintained at all times: proper procedure usage, communications, conservative decision making, reactivity management, equipment control and manipulation, and team skills.

Initial Conditions: Unit 1 75% power - 400 EFPD, Unit 2 100%, Unit 3 100%

Turnover:
- Operation at 75% per SOC for system load demand
- "1B" OTSG SGTL = 20 gpd (OP/1106/31 conditions have been evaluated)
- PCB-21 Gen Output Breaker open (low gas alarm occurred last shift – Transmission should add gas this shift).
- 1B CFT pressure low statalarm received at turnover - N2 makeup to 1B CFT required OP/1104/01, CF System, in progress
- GWD Vent header cross-connected – Unit 1 has the GWD header

Event No.	Malf. No.	Event Type*	Event Description
1. Pre-Insert	Override		Block MSLB Circuitry
2. Pre-Insert	Override	C, BOP/All	Under-voltage “27” relay failure for HWP
3. Pre-Insert	Override		1FDW-42 and 44 (FDW Startup Control and Block) (Failed open)
4. Pre-Insert	Override		PCB-21 Open
5. Pre-Insert	Override	N, ALL	1 B CFT pressure low - Initiate N2 makeup to 1B CFT (Pressure increase)

1	MPI091	I, BOP	Failure of RPS Channel "A" pressure transmitter (Failed Low) (SRO: Tech Spec)
2	Override	N, ALL	1B CFT pressure low – N2 makeup
3	MPI350	C, BOP SRO, TS	1B CFT water leak > TS (SRO: Tech Spec LCO-shutdown requirement)
4	MPI320 MCR040	C,OATC	Inability for CRD insertion in automatic during shutdown.
4A		R,OATC	Manual CRD power decrease
5	MSI051	I, OATC	Turbine Header Pressure transmitter fails high (Manual or automatic reactor trip)
6	MCS051	C,OATC	"1B" TBV fail 90% open (During Event #5)
7	Override	C,BOP/ OATC	1MS-26 ("B" TBV Block) failed open (Breaker failure) (During Event #5)
8	MEL080 MSS330	M, ALL	Load Rejection, loss of power (<3 sec. RCP remain in operation. Secondary lost except Pre-Insert #2)(Pre-Insert #4) (CT B.2.2 & B.2.3) TD EFDW Pump fails to start
9	MSS260	C,OATC	"1A" MDEFDWP trip
10		M, ALL	Establish HPI Cooling (CT B.1.6)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Op-Test No.: 01 Scenario No.: 01 Event No.: 1

Event Description:
FAILURE of RPS Channel "A" pressure transmitter.

Time	Position	Applicant's Actions or Behavior
	OATC	Respond to statalarms 1SA1 / A1, (RP Channel A Trip) A2, (RP Channel A Low Pressure Trip) A5, (RP Channel A Pressure/Temperature Trip). 1. 1SA1 / A1 • Check instrument channel to determine cause of trip • Check instrument parameters to verify validity of trip signal. • If faulty trip initiate action to repair • Refer to PT/1/A/0600/001 2. 1SA1 / A2 • Check instrumentation to verify low pressure. • If faulty trip initiate action to repair • Refer to ITS 3.3 • Refer to PT/1/A/0600/001 3. 1SA1 / A5 • Check instrumentation to verify low pressure and/or high temperature. • If faulty trip initiate action to repair • Refer to ITS 3.3 • Refer to PT/1/A/0600/001: ➢ Obtain RPS cabinet key, obtain Manual Bypass key, and place Channel "A" in manual bypass ➢ SRO refers to TS 3.3.1
	SRO	Refer to ITS 3.3.1, determines that the Condition statement is met if 3 RPS channels are operable.
	SRO / BOP	Refer to PT/1/A/0600/001
	SRO / BOP	Diagnoses that RPS Channel A fails to meet its required condition. Obtain RPS cabinet and Manual Bypass key

Op-Test No.:	01	Scenario No.:	01	Event No.:	1

Event Description:
FAILURE of RPS Channel "A" pressure transmitter.

Time	Position	Applicant's Actions or Behavior
	SRO / BOP	Determines that no other RPS channel is in MANUAL BYPASS or contains a DUMMY BISTABLE.
	BOP	Place RPS channel A in Manual Bypass (key switch)
	SRO / BOP	Initiate immediate action to have instrument channel repaired.

Op-Test No.: 01	Scenario No.: 01	Event No.: 2

Event Description:
Restore N_2 pressure CFT B to normal
~~Ramp 1B CFT level to the LOW-LEVEL ALARM.~~ Initiate makeup to 1B CFT (Level Increase).

Time	Position	Applicant's Actions or Behavior
	BOP	Respond to statalarm 1SA8 / A12, (Core Flood Tank "B" Pressure High/Low). Turnover item.
	SRO / BOP	Refer to OP/1/A/1104/01. (Enclosure 4.7)
	SRO / BOP	Determine cause of alarm and correct. Initiate N2 makeup to “B” CFT
	BOP	Dispatch NLO to Open 1N-137 (N2 to CFT Block) **TIME COMPRESS**
	BOP	Open 1N-299 (N2 Fill to 1B CFT).
	BOP	Monitor Core Flood Tank pressure increase and increase to ≈ 600 psig.
	BOP	Close 1N-299 (N2 Fill to 1B CFT).

Op-Test No.:	01	Scenario No.:	01	Event No.:	3

Event Description:
Increase CFT LEAKAGE to > TS (SRO: Tech Spec LCO – shutdown requirement).

Time	Position	Applicant's Actions or Behavior
	BOP	Respond to statalarm 1SA9 / A6, (Reactor Building Normal Sump Level High/Low). Respond to statalarm 1SA8 / A12, (Core Flood Tank "B" Pressure High/Low). Respond to statalarm 1SA8 / B12, (Core Flood Tank "B" Level High/Low).
	BOP / SRO	Statalarm 1SA8 / B12: • Determine cause of alarm.
	BOP / SRO	Diagnose that "1B" CFT water level is low and decreasing at a rate > makeup capability. The probable source of leakage into the RB is the CFT. (No RB RIA in alarm).
	SRO	Refer to TS 3.5.1: • (<575 psig or <1010 ft³ / 12.56 ft) 1 HOUR. restore operable CFT => MODE 3 in 12 hours and ≤800 psig RCS pressure 18 hours. • Initiate shutdown • (<575 psig or <1010 ft³ / 12.56 ft) 1 HOUR. restore operable CFT => MODE 3 in 12 hours and ≤800 psig RCS pressure 18 hours. • Initiate shutdown (By direction of Ops Duty person shutdown at 3%/minute) • Refer to OP/1102/04, Operations at Power

Op-Test No.: 01	Scenario No.: 01	Event No.: 4/4A

Event Description:

INABILITY for CRD insertion in automatic during shutdown.
During required shutdown CRDs fail to insert

- Tave increase (High statalarm received), RCS pressure increase, PZR spray actuates
- Diamond is placed into manual and power is decreased in manual.

Time	Position	Applicant's Actions or Behavior
	OATC	Acknowledges statalarm 1SA2 / B4, RC Average Temp High/Low • Compare Loop A Th (Loop A Tave) indication with Loop B Th (Loop B Tave) indication for instrumentation failure. Determine that no failure has occurred.
	OATC / BOP	Observes and reports that: • Tave increase (1SA2 / B4, RC Average Temp High/Low statalarm received) • RCS pressure increase, PZR level increase, LDST level increase.
	OATC	Diagnoses that CRDs fail to insert. • Places Diamond in Manual and decreases reactor power/Tave to match FDW and Unit load. • Continues the shutdown with the Diamond in Manual. NOTE: When Diamond is placed to HAND the FDW will take Tave control at setpoint

Op-Test No.:	01	Scenario No.:	01	Event No.:	5, 6, 7

Event Description:
Turbine Header Pressure transmitter FAILS HIGH.
"1B" TBV FAIL 90% OPEN.
1MS-26 ("B" TBV Block) FAILED OPEN (Breaker failure).
(Feed path: 1FDW-42 and 44, FDW Startup Control and Block, Failed OPEN.)

NOTE: A reactor trip could occur if steam pressure is not returned to setpoint due to Main Turbine trip If Main FDWP's discharge pressure decreases below 800 psig.

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	Statalarm 1SA2 / A9, MS Pressure High/Low • Check Main Steam pressure indication for high or low pressure. Statalarm 1SA2 / A12, ICS Tracking • Monitor plant parameters and stabilize plant with ICS in manual (Decrease Turbine Master, decrease FDW Masters • Determine cause of unit being in track. Statalarm 1SA2 / C12, H/A Station on Manual • Determine which station has transferred to manual and assume manual control. • Determine cause of H/A station in manual.
	OATC / BOP	Observe and report: • ICS in Track • TBV's open
	OATC	Diagnose reactor power increase (due to failure) • Insert CRD's to control reactor power. • Increase and maintain Tave at setpoint.
	OATC	Diagnose that MSCV's open and Mwe increases and actual THP decreases. • Decrease turbine demand to close MSCV's and control THP at 885 psig. • TBV"s will open and reclose as valves start to control on SG outlet pressure.

Op-Test No.: 01 Scenario No.: 01 Event No.: 5, 6, 7

Event Description:
Turbine Header Pressure transmitter FAILS HIGH.
"1B" TBV FAIL 90% OPEN.
1MS-26 ("B" TBV Block) FAILED OPEN (Breaker failure).
(Feed path: 1FDW-42 and 44, FDW Startup Control and Block, Failed OPEN.)

NOTE: A reactor trip could occur if steam pressure is not returned to setpoint due to Main Turbine trip If Main FDWP's discharge pressure decreases below 800 psig.

Time	Position	Applicant's Actions or Behavior
	OATC	Diagnose that "B" TBV has failed open (90%) creating a steam path to the condenser. • Attempt to isolate the failed open TBV by closing 1MS-26 (B TBV Block).
	OATC	Diagnose that 1MS-26 is failed open. • Dispatch the NLO to manually close 1MS-26.

Op-Test No.: 01	Scenario No.: 01	Event No.: 8

Event Description:

Load rejection, reactor trip, and LOSS of power (<3 sec. RCP's remain in operation. Secondary lost except Under voltage 27-relay FAILURE for HWP.)

Time	Position	Applicant's Actions or Behavior
	BOP / SRO	Diagnose loss of power. • Refer to AP1/A/1700/011, Loss of Power
	BOP	IF CC and HPI Seal Injection are lost to the RCP's THEN: • Establish RCP seal flow with the SSF makeup pump within 10 minutes.
	BOP	IF IA Header pressure < 90 psig • Direct Unit 3 to start the Diesel Air Compressor
	BOP	Transfer to Section 503, Unit Assessment of AP/1700/11, Loss of Power • Perform Section 503 to restore equipment in parallel with performing the EOP
	OATC	Refer to AP/1700/19, Loss of Main FDW NOTE: Crew will only refer to AP/1700/19 if EFDW is not operating properly.

Op-Test No.:	01	Scenario No.:	01	Event No.:	8

Event Description:

Load rejection, reactor trip, and LOSS of power (<3 sec. RCP's remain in operation. Secondary lost except Under voltage 27-relay FAILURE for HWP.)

Time	Position	Applicant's Actions or Behavior
	SRO / OATC / BOP	Refer to EP/1/A/1800/001 • Perform IMA and Symptom check • Verify reactor tripped • All Power Range NI's < 5% and decreasing • Turbine Tripped • All Turbine Stop Valves closed • Both Generator Output Breakers open • TBV's controlling as expected • Verify RCP seal injection available. IF CC and HPI Seal Injection are lost to the RCP's THEN: Establish RCP seal flow with the SSF makeup pump within 10 minutes (AP/1/1700/025, Standby Shutdown Facility Emergency Operating Procedure).
	SRO / OATC	Diagnose that RCS heat transfer is or has been excessive. • Transfer to Section 503, Excessive Heat Transfer
	SRO / OATC / BOP	Check or Verify: • PZR level < 80 inches • HPI pumps operating • SG levels NOT > 96% • SG's NOT isolated
	SRO / OATC / BOP	Diagnose that overcooling has NOT been stopped • Refer to Rule #6, Main Steam Line Break Actions:
	SRO / OATC / BOP	Perform Rule # 6, Main Steam Line Break Actions: • Secure MD EFDWP's • Initiate both trains of MSLB Isolation circuit

Op-Test No.: 01	Scenario No.: 01	Event No.: 8

Event Description:

Load rejection, reactor trip, and LOSS of power (<3 sec. RCP's remain in operation. Secondary lost except Under voltage 27-relay FAILURE for HWP.)

Time	Position	Applicant's Actions or Behavior
		• Ensure both FDWPT's tripped • Close EFDW control valve on the affected SG • Close Main and SU FDW block valves • If subcooling margin > 5° F throttle HPI header flow to maintain RCS P/T • If "B" SG isolated place Air Ejectors on Aux Steam Header • Stabilize RCS temperature and adjust TBV's to maintain CETC's constant.
	SRO / OATC / BOP	Diagnose the overcooling is caused by: • Failed open SU and Main FDW Block valves and that the HWP's and CBP's are running. • Stop the HWP's and CBP's.

Op-Test No.:	01	Scenario No.:	01	Event No.:	9/10

Event Description:

"1A" MDEFDWP trips.

- The only feedwater source to the operable "A" OTSG is "A" MD EFDWP.
- When this A MD EFDWP trips the unit will experience a total loss of all FDW and Loss of heat sink
- Cross-connection of another units EFDW system will not be allowed.

Time	Position	Applicant's Actions or Behavior
	SRO / OATC	Diagnose that the only feedwater source to the operable "A" OTSG is "A" MD EFDWP has been lost. • Parallel Actions - Transfer to EOP Section 502, Loss of Heat Transfer. • AP/1700/19 Section 501, Establishing EFDW
	SRO / OATC / BOP	Transfer to EOP Section 502, Loss of Heat Transfer: • Determine that HPI cooling is available • Reduce operating RCP's to one
	SRO / OATC / BOP	Initiate HPI cooling: • Open HP-24 and 25 aligning the HPI pump suction from BWST and open HP-26, "A" HP Injection and verify HP-27, 1B HP Injection is open. • Verify either A or B HPI pump operating and start C HPI pump • Ensure adequate HPI flow • Open 1RC-4, PZR Relief Block • Open 1RC-66, PORV • Manually deenergize all PZR heaters • Throttle HPI flow to maintain subcooling ≈20°F

Examination #1 Overview:

Initial Conditions: Unit 1 75% power - 400 EFPD, Unit 2 100%, Unit 3 100%

Turnover:

- "1B" OTSG SGTL = 20 gpd (OP/1106/31 conditions has been evaluated)
- PCB-21 Gen Output Breaker open (low gas alarm occurred last shift – Transmission should add gas this shift).
- 1B CFT pressure low statalarm actuated just prior to turnover. N2 makeup is required. OP/1104/01 in progress.
- GWD Vent header cross-connected – Unit 1 has the GWD header

Events

1. **RPS Channel "1A" Pressure transmitter fails low: (BOP/I)**

Statalarms received 1SA-1/A1,2,5. ARG information:
SRO - Refer to TS 3.3. TS LCO remains met due to 3 operable channels.
BOP - Refer to PT/600/01, Instrument Surveillance, Place "A" RPS Channel in Manual Bypass due inoperable instrument.

When "1A" Channel RPS is placed into Manual Bypass this event is completed
TIME = 5 minutes

2. **"1B" CFT N2 low pressure: (ALL/N)**

Turnover item - Statalarm 1SA-8/B12 (CFT B Pressure Low) has actuated prior to turnover.

BOP Refer to OP/1104/01, CFT and add N2 to regain normal operating pressure.

When Statalarm 1SA-8/B12 (CFT B Pressure Low) is cleared this event is complete

3. **Water leak in 1B CFT - rate to exceed TS level limit. (BOP/C) (SRO/TS)**

"1B" CFT level and pressure deceases.
BOP - Secure N2 addition.
SRO/BOP - Determine that leak rate is > capacity of makeup.

Statalarm 1SA-9/A6 (RBNS level high)
Statalarm 1SA-8/A12 (CFT B Pressure low)
Statalarm 1SA-8/B12 (CFT B level low)
Crew will determine leak source (No RB RIA in alarm)
T=5 min.
Crew determines the "1B" CFT is the probable source of leakage into the RB.
SRO - Refer to TS 3.5.1:(<575 psig or < 1010 ft^3 / 12.56 ft) 1 HOUR restore operable CFT=> MODE 3 in 12 hours and ≤800 psig RCS pressure 18 hours.
When SRO has made determination to shutdown per ITS this event is completed
TIME = 10 minutes TOTAL 25 min.

4. During required shutdown CRDs fail to insert (OATC/R)

Neutron error = 0%
Tave increase (High statalarm received), RCS pressure increase, PZR spray may actuate
OATC - Diamond is placed into manual and power is decreased in manual

When power is decreased~5% in manual this event is completed
TIME = 10 minutes TOTAL 35 min.

5. THP fails high (OATC/I)

1SA-2/A9, MS pressure high/low
1SA-2/A12, ICS Tracking
1SA-2/C12, ICS H/A Station on Manual (Turbine master reverts to manual in 5 seconds)

Reactor power increase-Operator will insert CRDs to control power increase and maintain Tave at setpoint.

Before transferring to manual, the turbine master will open MSCVs-MWe increase, Actual THP decreases. Operator will decrease turbine demand to close MSCVs and control actual THP~885 psig.

Note: A reactor trip could occur if steam pressure is not returned to setpoint as the Main Turbine will trip if MFDWPs discharge pressure decreases below 800 psig. If the T/G trips then a loss of power will occur as one Generator Output Breaker is open causing a 1 sec delay on the rapid bus transfer.

Turbine Bypass Valves (TBV) will open and then reclose as valves start to control on SG Outlet pressure.

When the operator stabilizes the unit with ICS in manual or reactor trip/loss of power occurs this event is complete
TIME = 5 minutes TOTAL 40 min.

6. "B" TBV failed 90% open (OATC/C)

This will create a steam path to the condenser.
OATC - will attempt to isolate the failed open TBV by closing 1MS-26 (B TBV Block). Steam path see Pre-Insert, FDW-42 and 44 failed open.

When the operator diagnoses B TBV failed open this event is completed
TIME = 5 minutes TOTAL 40 min.

7. **1MS-26 failed open (BOP-OATC/C)**

1MS-26 will be failed open and cannot be electrically operated from the control room. The crew should dispatch an NLO to manually close 1MS-26.

When the operator attempts to close MS-26 this event is completed
TIME=5 minutes TOTAL 40 min.

8. **Loss of Power (ALL/M)**

After THP failure is mitigated (reactor trip does not occur) the unit will experience a load rejection with ICS in manual. This will cause a short loss of power (<3 seconds) when the Aux. Loads transfer to the startup transformer via a break before make connection.

RCPs will remain in operation.
TD EFDWP fails to start
BOP - AP/1700/11, Loss of Power is entered
EOP is entered-IMA, Subsequent Actions, Parallel Actions, Section 503, Excessive Heat Transfer.

When the crew determines that the OTSG cannot be isolated the crew should perform RULE #6, Main Steam Line Break Actions.

The secondary side pumps should trip following the loss of power due to the undervoltage 27-relay. Failure of the 1C HWP relay will allow the HWP breaker to remain closed and when power is restored the HWP will remain operating.

The 1B CBP will also remain in operation as it receives a time delayed automatic start signal on low FDWP Suction pressure.

This establishes a driving head for condensate to be delivered to the "1B" OTSG and promotes a RCS overcooling event that can only be isolated when the operating secondary HWP and CBP is secured.

When the HWP and CBP is isolated this event is completed
TIME=10-20 minutes TOTAL 50-60 min.

9. **Loss of A MDEFDW (OATC/C)**

The only feedwater source to the operable "A" OTSG is "A" MD EFDWP. When this EFDWP trips the unit will experience a total loss of all FDW. Cross-connection of another units EFDW system will not be allowed.
SRO - EOP transfer to Section 502, Loss of Heat Transfer
OATC - RCS temperature and pressure will increase and HPI Cooling will be established when RCS pressure increases to 2300 psig

When the "A" MD EFDWP trip is diagnosed and the decision to establish HPI Cooling is made this event is completed
TIME=10-15 minutes TOTAL 60-65 min.

10. Establish HPI Cooling (ALL/M)

RCS temperature and pressure will increase and HPI Cooling will be established before RCS pressure increases to 2300 psig

When HPI Cooling is established or at the Examiners request the event and the exam is completed. Exam complete

<u>TIME=1-15 minutes TOTAL 70-75 min.</u>

E-Plan Classification:

<u>SRO Admin A.4</u>

ALERT based on HPI Cooling established Fission Product Matrix total = 4

Follow-up Question:
If condenser vacuum was lost with the Condenser rupture disc blown and the 1B SGTL increases to 70 gpm how does this affect the E-Plan classification and any PAGs that may apply?
ANASWER – upgrade to SAE based on Fission Product Matrix total = 7 (HPI Cooling (4)+ SGTL > 10 gpm 'with direct opening to the environment (3).
Jo PAGs are required

Simulator Scenario Report

Scenario : EXAM 1 SIM FILES
Report Date : 05-07-2000

AOR
Upd - 22,ZWISEVNT(13) = False,"WIS EVENT LOGIC BUFFERS"
Upd - 27,HPM804G = 1490,"MASS OF GAS IN CFT-B "
Switch - P1420TC,SET,BREAK,"START UP FDW BLOCK 1FDW-42 B : CLOSE"
Switch - P1420TN,SET,MAKE,"START UP FDW BLOCK 1FDW-42 B : OPEN"
Switch - P1B92SE,SET,BREAK,"MSLB ISOLATION TRAIN 1A ENABLE"
Switch - P1B94SE,SET,BREAK,"MSLB ISOLATION TRAIN 1B ENABLE"
Light - P1711LR,SET,ON,"LOOP B START UP FW VALVE MASTER AUTO : LEFT RED"
Switch - P1711TW,SET,BREAK,"LOOP B START UP FW VALVE MASTER LOWER"
Switch - P1711TR,SET,MAKE,"LOOP B START UP FW VALVE MASTER RAISE"
Switch - P1711SA,SET,BREAK,"LOOP B START UP FW VALVE MASTER AUTO"
Light - P3223RR,SET,ON,"HOTWELL PUMP 1C RED"
Light - P3223MX,SET,OFF,"HOTWELL PUMP 1C BRIGHT WHITE"
Light - P3223LW,SET,OFF,"HOTWELL PUMP 1C DIM WHITE"
Switch - P2104TT,SET,MAKE,"GENERATOR # 1 PCB21 TRIP"
T01 : 00:00:00
Malf - MPI091,Set,0,"RP-A RC PRESSURE TRANSMITTER Y1PT17P (1700-2500)"
T02 : 00:00:00
Upd - 19,HPM804 = 75000,"MASS OF LIQUID IN CFT-B "
Malf - MPI350,Set,10,"WTR LEAK FR CORE FL TK "B" TO ATMOSPHERE"
T03 : 00:00:00
Malf - MSI051,Set,100,"TRB HDR PRESSURE TRANSMITTER (A) Y1PT30P (600-1200)"
Malf - MSI061,Set,100,"TRB HDR PRESSURE TRANSMITTER (B) Y1PT31P (600-1200)"
Switch - P2302TC,SET,BREAK,"TURB BY PASS BLOCK B MS-26 : CLOSE"
Switch - P2302TN,SET,MAKE,"TURB BY PASS BLOCK B MS-26 : OPEN"
Switch - P1714SA,SET,BREAK,"LOOP B TURBINE BY-PASS VALVE MASTER AUTO"
Malf - MCS015,Set,95,"MS TURBINE BYPASS VALVE DEMAND B (INPUT 0. - 100%)"
Switch - P1711SM,SET,MAKE,"LOOP B START UP FW VALVE MASTER MANUAL"
Malf - MPS190,Set,100,"RC-1 (SPRAY VALVE) FAILS AS IS"
T04 : 00:00:00
Malf - MEL080,Set,100,"LOAD REJECTION (PCB 20 AND 21 TRIP)"
Malf - MSS330,Set,100,"EFWPT FAILS TO START"
T05 : 00:00:00
Malf - MSS260,Set,100,"MDEFWP A FAILS TO START"

E09 : "P1641PW [Reset MCR040, Inablity to insert CRDs]"
Event Enabled : Yes
Malf - MCR040,Reset,0,"INABILITY TO INSERT CONTROL RODS"

E10 : "NIPLP5A < 73.5 [Activate MCR040 & Block N error]"
Event Enabled : Yes
Malf - MPI320,Set,100,"BLOCK NEUTRON ERROR TO CRD"
Malf - MCR040,Set,100,"INABILITY TO INSERT CONTROL RODS"

E11 : "P3422RR [C-10 CLOSED]"
Event Enabled : Yes
Pot - P7508A1,SET,0,"C-10 VALVE DEMAND FROM CONTROLLER OUTPUT BLOCK FB03"

Switch - P7508TM,SET,MAKE,"C-10 CONTROLLER IN MANUAL MODE"

E12 : "VRCD010V = 0 [Start C HWP]"
Event Enabled : Yes Tie Timer : Yes
Switch - P3223WB,SET,MAKE,"HOTWELL PUMP 1C START"
Switch - P3223WU,SET,BREAK,"HOTWELL PUMP 1C RUN"
Switch - P3223WF,SET,BREAK,"HOTWELL PUMP 1C OFF"
Switch - P3223WA,SET,BREAK,"HOTWELL PUMP 1C AUTO"
T12 : 00:00:03
Pot - P7508A1,SET,100,"C-10 VALVE DEMAND FROM CONTROLLER OUTPUT BLOCK FB03"

E13 : "VRCD010V = 0 [Reopen C-10]"
Event Enabled : Yes Tie Timer : Yes
Upd - 21,ZWISEVNT(13) = True,"WIS EVENT LOGIC BUFFERS"
T13 : 00:00:20
Light - P3223RR,RESET,OFF,"HOTWELL PUMP 1C RED"
Light - P3223MX,RESET,OFF,"HOTWELL PUMP 1C BRIGHT WHITE"
Light - P3223LW,RESET,OFF,"HOTWELL PUMP 1C DIM WHITE"
Switch - P3223WB,RESET,BREAK,"HOTWELL PUMP 1C START"
Switch - P3223WU,RESET,BREAK,"HOTWELL PUMP 1C RUN"
Switch - P3223WF,RESET,BREAK,"HOTWELL PUMP 1C OFF"
Switch - P3223WA,RESET,BREAK,"HOTWELL PUMP 1C AUTO"
Pot - P7508A1,RESET,0,"C-10 VALVE DEMAND FROM CONTROLLER OUTPUT BLOCK FB03"
Switch - P7508TM,RESET,BREAK,"C-10 CONTROLLER IN MANUAL MODE"

Items not tied to any Event or Timer
Upd - 1,VRMS028D,"MS-28 INSTR. CONTROL "
Upd - 2,VRMS028,"MS-28 POSITION "
Upd - 3,XVRMS028,"SET TRUE TO OVERRIDE MS "
Upd - 4,VLCR026D,"A MAIN VACUUM PUMP SUCT"
Upd - 5,VLCR028D,"B MAIN VACUUM PUMP SUCT"
Upd - 6,VLCR030D,"C MAIN VACUUM PUMP SUCT"
Upd - 8,ZED01ML(1),"1X1 FDR BKR "
Upd - 9,ZED01ML(2),"1X2 FDR BKR "
Upd - 10,ZED01ML(3),"1X3 FDR BKR "
Upd - 11,ZED01ML(4),"1X4 FDR BKR "
Upd - 13,ZEDLCM1,"600V LOAD CENTER 1X1 ST "
Upd - 14,ZEDLCM2,"600V LOAD CENTER 1X2 ST "
Upd - 15,ZEDLCM3,"600V LOAD CENTER 1X3 ST "
Upd - 16,ZEDLCM4,"600V LOAD CENTER 1X4 ST "
Upd - 24,VRCD010V,"VALVE C-10 GREATER THAN 10% OPEN"
Upd - 25,VRCD010,"C- 10 INLET TO POWDEX DEMINERALIZER VALVE"

C. IF any other analog channel is tripped, do **NOT** trip affected channel. Initiate immediate action to have instrument repaired. Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits).

D. Immediate shutdown may be required.
- Refer to TS 3.3.5.
- Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits).

12.2.2 RPS Instrument

A. IF no other RPS channel is in MANUAL BYPASS or no other RPS channel contains a DUMMY BISTABLE, place affected RPS channel in MANUAL BYPASS. Initiate action to have instrument channel repaired.

NOTE:	For determining appropriate TS condition a tripped channel is considered inoperable.

B. IF another RPS channel is in MANUAL BYPASS or contains a DUMMY BISTABLE, trip affected RPS channel by placing any one of its instrument channels in "TEST-OPERATE" (for STAR Modules select "TEST"). Affected parameter(s) should be left in "TEST-OPERATE" (or "TEST") until channel input(s) is repaired. Initiate immediate action to have instrument channel repaired.

C. IF affected RPS channel is already in MANUAL BYPASS, do **NOT** trip affected RPS channel. Initiate action to have instrument channel repaired.

D. IF affected RPS channel contains a DUMMY BISTABLE and no other RPS channel is in MANUAL BYPASS, place affected RPS channel in MANUAL BYPASS. TS allows any one RPS channel to contain more than one DUMMY BISTABLE.

E. IF another RPS channel is tripped, do **NOT** trip affected RPS channel. Initiate immediate action to have instrument channel repaired. Tripping affected RPS channel will cause a reactor trip.

F. Immediate shutdown may be required.
- Refer to TS 3.3.1
- Refer to OMP 1-4 (Actions To Be Taken In Case Of Exceeding Limits)

3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation

LCO 3.3.1 Three channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required channel inoperable.	A.1 Place channel in trip.	1 hour
B. Two or more required channels inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1 Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately
C. As required by Required Action B.1 and referenced in Table 3.3.1-1.	C.1 Be in MODE 3. AND C.2 Open all control rod drive (CRD) trip breakers.	12 hours 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action B.1 and referenced in Table 3.3.1-1.	D.1 Open all CRD trip breakers.	6 hours
E. As required by Required Action B.1 and referenced in Table 3.3.1-1.	E.1 Reduce THERMAL POWER < 30% RTP.	6 hours
F. As required by Required Action B.1 and referenced in Table 3.3.1-1.	F.1 Reduce THERMAL POWER < 2% RTP.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.

SURVEILLANCE	FREQUENCY
SR 3.3.1.1 Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	------------------------NOTE------------------------ Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Compare results of calorimetric heat balance calculation to the power range channel output and adjust power range channel output if calorimetric exceeds power range channel output by ≥ 2% RTP.	24 hours
SR 3.3.1.3	------------------------NOTE------------------------ Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Compare out of core measured AXIAL POWER IMBALANCE (API_0) to incore measured AXIAL POWER IMBALANCE (API_1) as follows: $(RTP/TP)(API_0 - API_1)$ = imbalance error Adjust power range channel output if the absolute difference between the power range and incore measurements is ≥ 2% RTP.	31 days
SR 3.3.1.4	Perform CHANNEL FUNCTIONAL TEST.	45 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	-------------------------NOTE------------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months

Table 3.3.1-1 (page 1 of 1)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower				
a. High Setpoint	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5	≤ 105.5% RTP
b. Low Setpoint	2[(b)],3[(b)] 4[(b)],5[(b)]	D	SR 3.3.1.1 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 618°F
3. RCS High Pressure	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 2355 psig
4. RCS Low Pressure	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 1800 psig
5. RCS Variable Low Pressure	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
6. Reactor Building High Pressure	1,2,3[(c)]	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 4 psig
7. Reactor Coolant Pump to Power	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	>2% RTP with ≤ 2 pumps operating
8. Nuclear Overpower Flux/Flow Imbalance	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
9. Main Turbine Trip (Hydraulic Fluid Pressure)	≥ 30% RTP	E	SR 3.3.1.4 SR 3.3.1.5	≥ 800 psig
10. Loss of Main Feedwater Pumps (Hydraulic Oil Pressure)	≥ 2% RTP	F	SR 3.3.1.4 SR 3.3.1.5	≥ 75 psig
11. Shutdown Bypass RCS High Pressure	2[(b)],3[(b)] 4[(b)],5[(b)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

A-12

CORE FLOOD TANK "B" PRESSURE HIGH/LOW

1. Alarm Setpoint

1.1 High Alarm - 615 psig increasing pressure.

1.2 Low Alarm - 585 psig decreasing pressure.

2. Automatic Action

None

3. Manual Action

3.1 Refer to OP/1/A/1104/01 (Core Flooding System) to adjust pressure as necessary.

3.2 Determine cause of alarm and correct.

4. Alarm Sources and References

4.1 OEE-118-17 & 18.

4.2 IP/0/A/201/1B (Core Flood Tank Pressure and Temperature Instruments Calibration).

4.3 1PT-39 (CF1-PT3) and/or 1PT-40P (CF1-PT4).

4.4 Technical Specification 3.5.1.

(R06-97)

SR
Sim
NRC
106
115
JPP
JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No OP/1/A/1104/001

Revision No 44

REPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Core Flooding System

(4) Prepared By Dennis L. Masteller (Signature) Dennis L. Masteller Date 2/24/00

(5) Requires 10CFR50.59 evaluation?

X Yes (New procedure or revision with major changes)
☐ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By ____ (QR) Date 03/13/00

Cross-Disciplinary Review By ____ (QR)NA ____ Date ____

Reactivity Mgmt. Review By ____ (QR)NA ____ Date 03/13/00

(7) Additional Reviews

Reviewed By ____ Date ____

Reviewed By ____ Date ____

(8) Temporary Approval *(if necessary)*

By ____ (SRO/QR) Date ____

By ____ (QR) Date ____

(9) Approved By ____ Date 3/14/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

Compared with Control Copy ____ Date ____

(11) Date(s) Performed ____

Work Order Number (WO#) ____

MAR 2000
Received
Operator
Training

COMPLETION

(12) Procedure Completion Verification:

☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ____ Date ____

(13) Procedure Completion Approved ____ Date ____

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Core Flooding System** **Multiple Use**	Procedure No. OP/1/A/1104/001 Revision No. 044 Electronic Reference No. OX002VLR

Core Flooding System

1. Purpose

To describe the procedures for operating the CF System:

- Filling CFTs.
- Establishing CFT ES conditions.
- Makeup to CFTs with water or nitrogen.
- Initial setup of CFT pressurization rates.
- Lowering CFT pressure when ES is required.
- Depressurization of CFTs during shutdown conditions.
- Bleeding and draining CFTs.

2. Limits And Precautions

2.1 This procedure affects core reactivity. {6}

2.2 LTOP must be established on CFTs prior to RCS < 325°F.

2.3 Following a change in CFT inventory, CFT boron should be verified between 2500 - 3750 ppm.

2.4 If CFT inleakage exists, a sampling program should be established to ensure ≥ 2500 ppm is maintained. In this case makeup to CFT should be ≥ 2700 ppm.

2.5 Pressure increase in CFTs should <u>NOT</u> exceed 100 psig in 15 minutes (≈ 6.6 psig/min) unless Nitrogen Heater in operation.

2.6 Do <u>NOT</u> increase nitrogen header pressure while adding nitrogen to CFTs.

3. Procedure

3.1 Refer to one of the following enclosures to initially fill CFTs:

- Enclosure "Filling CFTs Using HPI Pump".
- Enclosure "Filling CFTs Using 1A Bleed Transfer Pump".

3.2 Refer to Enclosure "Establishing ES Conditions In Core Flooding System" to establish ES conditions in CFTs during startup.

3.3 Refer to Enclosure "Pressure Makeup To CFTs Using Nitrogen" to add nitrogen to CFTs.

3.4 Refer to Enclosure "CFT Pressurization Rate Check" to set CFT nitrogen pressurization rate at ≤ 100 psig per 15 minutes.

3.5 Refer to Enclosure "CFT Nitrogen Heater Operation" to set CFT nitrogen pressurization rate at > 100 psig per 15 minutes.

3.6 Refer to one of the following enclosures to adjust CFT levels:

- Enclosure "Makeup To CFTs Using HP Boric Acid Pump And DW".
- Enclosure "Non-Routine Adjustment Of CFT Levels".

3.7 Refer to Enclosure "CFT Sampling" to sample CFTs.

3.8 Refer to one of the following enclosures to adjust CFT pressures:

- Enclosure "Lowering CFT Pressure".
- Enclosure "Depressurization Of CFTs".

3.9 Refer to one of the following enclosures to lower CFT levels:

- Enclosure "Discharge Of CFTs Into RCS".
- Enclosure "Draining Of CFTs To MWHUT".
- Enclosure "Draining Of CFTs To 1A BHUT".

4. Enclosures

4.1 Filling CFTs Using HPI Pump

4.2 Filling CFTs Using 1A Bleed Transfer Pump

4.3 Boron Concentration Calculation

4.4 Valve Checklist

4.5 Verification Valve Checklist

4.6 Establishing ES Conditions In Core Flooding System

Continuous Use

1. Initial Conditions

_____ 1.1 High pressure nitrogen header in service.

_____ 1.2 Review Limits and Precautions.

2. Procedure

_____ 2.1 Open 1N-137 (CFTs Supply). (A-2-Hallway)

2.2 **IF** required to increase pressure in 1A CFT:

_____ 2.2.1 Open 1N-298 (N_2 FILL CORE FLOOD TANK 1A).

_____ 2.2.2 Monitor 1A CFT pressure.

_____ 2.2.3 **WHEN** pressurization of 1A CFT is complete, Close 1N-298 (N_2 FILL CORE FLOOD TANK 1A).

2.3 **IF** required to increase pressure in 1B CFT:

_____ 2.3.1 Open 1N-299 (N_2 FILL CORE FLOOD TANK 1B).

_____ 2.3.2 Monitor 1B CFT pressure.

_____ 2.3.3 **WHEN** pressurization of 1B CFT is complete, Close 1N-299 (N_2 FILL CORE FLOOD TANK 1B).

_____ 2.4 Close 1N-137 (CFTs Supply). (A-2-Hallway)

2.5 Verify CFT pressures stable:

_____ • 1A CFT.

_____ • 1B CFT.

B-12

CORE FLOOD TANK "B" LEVEL HIGH/LOW

1. Alarm Setpoint

1.1 High Alarm - 13.3 feet increasing level.

1.2 Low Alarm - 12.7 feet decreasing level.

2. Automatic Action

None

3. Manual Action

3.1 Refer to OP/1/A/1104/01 (Core Flooding System) to adjust level as necessary.

3.2 Determine cause of alarm and correct.

4. Alarm Sources and References

4.1 OEE-118-17 & 18.

4.2 IP/0/A/201/1A (Core Flood Tank Level Instruments Calibration).

4.3 1LT-13P (CF2-LT3) and/or 1LT-14 (CF2-LT4).

4.4 Technical Specification 3.5.1.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Core Flood Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CFT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One CFT inoperable for reasons other than Condition A.	B.1 Restore CFT to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours
	AND	
	C.2 Reduce RCS pressure to $\leq$ 800 psig.	18 hours
D. Two CFTs inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each CFT isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each CFT is $\geq$ 1010 ft^3 and $\leq$ 1070 ft^3.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is $\geq$ 575 psig and $\leq$ 625 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each CFT is within the limit specified in the COLR.	31 days <u>AND</u> ----------NOTE------------ Only required to be performed for affected CFT ----------------------------- Once within 12 hours after each solution volume increase of $\geq$ 80 gallons that is not the result of addition from a borated water source that meets CFT boron concentration requirements.
SR 3.5.1.5	Verify power is removed from each CFT isolation valve operator.	31 days

(R06-97)

OK
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106
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JMB

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PROCEDURE PROCESS RECORD

(1) ID No OP/1/A/1102/004

Revision No 085

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title OPERATION AT POWER

(4) Prepared By Ridings, R. (Signature) [signature] Date 2/17/00

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)
☐ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By [signature] (QR) Date 2/17/00

Cross-Disciplinary Review By (I&E) Doug Pepper (QR)NA Date 2-21-00

Reactivity Mgmt. Review By [signature] (QR)NA Date 2/17/00

(7) Additional Reviews

Reviewed By (Rx Eng) JE Sanders Date 2/21/00

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By [signature] Date 2/21/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification:

☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*
~~~

Operation At Power

1. Purpose

This procedure provides the following:

- Procedural guidance for operation of Unit 1 to escalate and reduce Rx power with the Turbine Generator on line.
- Maneuvering restrictions for Rx power ramp rates and CRD/APSR movement rates.
- Special instructions for power operation with < four RCPs.

2. Limits and Precautions

2.1 Use of this procedure will affect core reactivity.

2.2 Unit 1 should be operated within the limits of Technical Specifications (TS) at all times.

- When operating under LCO 3.0.3 of TS, shutdown rate should be determined by Operations such that the required actions can be achieved in a controlled manner.
- IF conditions require a faster shutdown rate up to and including a Rx Trip, refer to OMP 1-4 (Actions To Be Taken In The Case Of Exceeding Limits).

2.3 Reactor Power maneuvering restrictions are found in PT/0/A/1103/020 (Power Maneuvering Guidelines).

2.4 Power Imbalance should be maintained per PT/1/A/600/001 (Periodic Instrument Surveillance).

2.5 IF reactor power will be increased ≥ 10%,

OR

IF reactor power will be reduced ≥ 40% with reactor remaining critical, Reactor Engineering should be notified to develop Maneuvering Plan.

- When possible, notification should occur at least 24 hours prior to planned changes in power. {6}

2.6 IF an unscheduled power reduction occurs, an investigation should be conducted and corrective action taken prior to power level increase.

2.7 CRD group positions should be maintained at same level to minimize power tilts.

2.8 Condenser ΔT shall be ≤ 22°F with CCW temperature > 68°F.

2.9 CCW discharge effluent temperature shall **NOT** exceed 100°F for more than two hours.

NOTE:	Non-conservative NIs is Core Thermal Power (CTP) > NIs.

2.10 **IF** any two of four power range NIs > 2% non-conservative, calibration is required to prevent exceeding safety limits.

- Prior to initiating power changes > 5% CTP or CRD changes > 15% Rod Index, NI calibration should be checked.
- 15 minutes after reaching steady conditions, NI calibration should be checked.
- In no case, should ≥ 4% in the non-conservative direction be exceeded.
- **IF** any two of the four NIs are **NOT** within ±2% of CTP during power level increases, CTP increase should be stopped and NIs calibrated. {10}

2.11 Pressurizer Heaters should be in "AUTO" during system transients.

2.12 Pressurizer spray should be in "AUTO" during system transients.

- 1RC-1 should be in AUTO and 1RC-3 should be open.
- Anytime 1RC-1 is **NOT** in AUTO and 1RC-3 is throttled, an R&R should be completed per OMP 2-18 (Tagout Removal and Restoration Procedure).

2.13 **IF** Pressurizer level decreases to 200" with Rx > 15% CTP immediate action should be taken to return Pressurizer level to normal. Pressurizer level should be limited to ≤ 260".

2.14 Primary and Secondary Chemistry limits should be maintained as established by the Chemistry Manual.

2.15 **IF** Purification or Deborating IX is exhausted the applicable inlet valves should be closed and appropriate out of service stickers and R&R sheets issued.

2.16 Powdex should be 100% in even if a load reduction or three HWPs operation is required. Operation < 100% Powdex flow is acceptable for short time intervals.

2.17 Both MSRH Drain Tanks should be routed to the Hotwell for power changes > 6% FP to minimize feeding increased Secondary contaminants to Steam Generators.

2.18 Reactor power should be reduced ≈ 4% below allowable power when performing any operation that could cause a power swing.

2.19 The following UST temperature limits apply when EFDW is required: {1}

2.19.1 UST ≤ 125°F with Rx > 30% power

2.19.2 UST ≤ 125°F two hours after a Rx trip

2.19.3 UST ≤ 125°F two hours after reducing Rx < 30% power.

2.19.4 UST ≤ 145°F during Rx startup below 30% power.

2.20 **IF** load is reduced > 100 MW when ≥ 600 MW, VARs should be maintained within envelope specified in OP/1/A/1106/001 (Turbine Generator) until cold gas temperature has stabilized.

2.21 Operate individual coolers of the Second Cooler Group on the Main Transformer as needed to keep oil temperature from exceeding 75°C.

- When Main Transformer is energized and its oil temperature is < 50°C, only one cooler group should be operated (9 pumps/fans).

2.22 RCS boron may affect SSF RC M/U system operability per TS 3.10.1.

- RCS boron should be maintained > minimum required for SSF operability per PT/1/A/1103/015 (Reactivity Balance).

2.23 RATE SET (%/MIN / %HR) should be 0.0 when **NOT** actually changing power. This could prevent unanticipated Rx power changes if ICS goes into TRACK.

2.24 FDW flow to either SG should **NOT** exceed 5.7 x 10^6 lbm/hr.

- This limit applies to both 3 and 4 RCP operation.
- Computer points O1P0840 and O1P0841 most accurately indicate feedwater flow.
- Other computer points and flow gauges are **NOT** compensated and will indicate higher than actual FDW flow if FDW Fouling Coefficient is < one. {4}

2.25 At Rx power levels in the range of 35-85%, small (< 4.5%) power swings may occur.

- These swings are caused by adjustment of SG orifice plates causing steam flow oscillations in SG downcomer region.
- These swings have been evaluated and are considered normal.
- ICS is tuned to dampen the effects on CRD motion; however, if cyclic CRD motion is observed appropriate steps should be taken to correct problems as necessary. {8}

3. Enclosures

3.1 Power Escalation

3.2 Power Reduction

3.3 Special Instructions For < 4 RCP Operation

3.4 End Of Cycle APSR Withdrawal

3.5 Special Instructions For Operation With Reduced T_{AVE}

3.6 Operation Of CRD Group 8 To Control Rx Power Imbalance

Appendix

1. Initial Conditions

_____ 1.1 **IF** changing reactor power ≥ 10%, verify Maneuvering Plan is available for use. {6}

_____ 1.2 Review Limits and Precautions.

2. Procedure

2.1 **IF** > 6% CTP change is planned, refer to OP/1/A/1106/014 (Moisture Separator Reheater) to ensure Moisture Separator drains are routed to Hotwell.

CAUTION:	Special attention should be given to the selection of rate of power escalation. Before changing RATE SET the rate %/MIN or %/HR should be selected.

2.2 Limit rate of Rx power increase per PT/0/A/1103/020 (Power Maneuvering Guidelines).

_____ 2.2.1 Refer to Maneuvering Plan to view Reactor Engineering guidelines for power escalation. {6}

CAUTION:	If any CRD Groups are in the restricted region, boration of the RCS must begin within 15 minutes per TS 3.2.1. Operation in the restricted region is limited to 2 Hours.

2.3 Maintain CRD Groups 5-8 within required position limits per PT/1/A/0600/001 (Periodic Instrument Surveillance).

2.4 Maintain Core Power Imbalance **AND** Quadrant Power Tilt per PT/1/A/0600/001 (Periodic Instrument Surveillance).

_____ 2.5 At ≈ 225 MWe, transfer auxiliaries from CT-1 to 1T per OP/1/A/1107/002 (Normal Power).

_____ 2.6 Prior to exceeding 30% Reactor power, verify **OR** reduce UST temperature to ≤ 125°F. {1}

_____ 2.7 At ≈ 30% CTP, maintain steady state conditions **AND** perform the following:

_____ _____ • **IF** any two of the four NIs are **NOT** within ±2% of CTP, have I&E calibrate all NIs to ±2% CTP

AND resume power increase after calibration. {10}

• Reset "OUTPUT MEMORY" on "MAIN TURBINE TRIP BYPASS BISTABLE" for each of the four RPS Channels:

_____ _____ 'A' RPS Channel

_____ _____ 'B' RPS Channel

_____ _____ 'C' RPS Channel

_____ _____ 'D' RPS Channel.

_____ 2.8 Start second HWP.

_____ 2.8.1 Verify **OR** place standby HWP switch in "AUTO".

_____ 2.9 At ≈ 300 MWe, start 'D' Heater Drain Pumps per OP/1/A/1106/002 (Condensate and Feedwater System).

_____ 2.10 Begin second FDWPT startup per OP/1/A/1106/002 (Condensate and Feedwater System).

NOTE:	Rx power increase may continue while performing this step, sequence is NOT required.

_____ 2.11 Monitor computer point O1P3317 (CBP DELTA PRESSURE). {9}

_____ 2.11.1 At ≤ 400 psig Δp start second CBP.

_____ 2.11.2 Ensure standby CBP switch in "AUTO".

_____ 2.12 Prior to exceeding 60% CTP place second FDWPT in service.

NOTE:	Rx power increase may continue while performing this step, sequence is <u>NOT</u> required.

2.13 Verify the following valves indicate "CLOSED" at Heater Panel: (T-5-J18)

_____ 1HD-298 (HTR 1F1 DRAIN LVL CONTROL BYP)

_____ 1HD-303 (HTR 1F2 DRAIN LVL CONTROL BYP)

_____ 1HD-308 (HTR 1F3 DRAIN LVL CONTROL BYP).

1. **<u>IF</u>** any valve **<u>DOES NOT</u>** indicate "CLOSED", perform the following:

   a. Verify Emergency High Level Statalarm does **<u>NOT</u>** exist for associated 'F' Heater with Open valve(s).

   b. Close valve(s) as necessary to accomplish Step 2.13 above.

CAUTION:
- <u>Never</u> depress the upper toggle switch (Contact Buffer will Trip).
- The lower toggle switch is always used to reset Contact Buffers.

2.14 Reset **<u>OR</u>** verify reset, the following:

- 'A' RPS Channel

_____ _____ "1A FDWPT" Contact Buffer

_____ _____ "1B FDWPT" Contact Buffer

_____ _____ "MAIN FEED PUMP TRIP BYPASS BISTABLE" output memory.

- 'B' RPS Channel

_____ _____ "1A FDWPT" Contact Buffer

_____ _____ "1B FDWPT" Contact Buffer

_____ _____ "MAIN FEED PUMP TRIP BYPASS BISTABLE" output memory.

- 'C' RPS Channel

_____ _____ "1A FDWPT" Contact Buffer

_____ _____ "1B FDWPT" Contact Buffer

_____ _____ "(MAIN FEED PUMP TRIP BYPASS BISTABLE" output memory.

- 'D' RPS Channel

_____ _____ "1A FDWPT" Contact Buffer

_____ _____ "1B FDWPT" Contact Buffer

_____ _____ "MAIN FEED PUMP TRIP BYPASS BISTABLE" output memory.

2.15 After both FDWPT are operating with suction flow > 2300 gpm, place the following in "AUTO":

_____ • 1FDW-53 (1A FDWP RECIRC CONTROL)

_____ • 1FDW-65 (1B FDWP RECIRC CONTROL).

_____ 2.16 <u>IF</u> required, start additional CCW pumps per OP/1/A/1104/012 (CCW System).

NOTE:	Rx power increase may continue to meet conditions of this PT; step sequence is <u>NOT</u> required.

_____ 2.17 <u>WHEN</u> conditions are met, perform "Control Rod Movement (Startup)" enclosure of PT/1/A/0600/015 (Control Rod Movement).

CAUTION:	• Rx power increase may continue while machine gas pressure is being increased. • Refer to Capability Curve of OP/1/A/1106/001 (Turbine Generator) to ensure Generator operating limits are <u>NOT</u> exceeded while increasing pressure.

_____ 2.18 <u>WHEN</u> the load reaches 500 MWe, <u>begin</u> raising Machine Gas pressure to 60 psig per OP/1/A/1106/017 (Hydrogen System).

NOTE: 1. Power escalation may continue while Generator Cold Gas temperature is being adjusted.

2. Adjustments to 1C-58 (H_2 COOLER CONTROLLER) setpoint should be made in small increments to avoid Condensate System swings.

_____ 2.19 Adjust "AUTO" setpoint for 1C-58 as needed to maintain Generator Cold Gas temperature at ≈ 40°C as indicated by computer points O1A1148-O1A1151.

_____ 2.20 **IF** required, at ≈ 500 MWe notify Reactor Engineering to complete PT/0/A/0302/006 (Review and Control of Incore Instrumentation Signals). {2}

_____ _____ 2.21 At 65% CTP, maintain steady state conditions for ≈ 15 minutes for NI calibration check.

_____ 2.22 At ≈ 70% CTP, start third Hotwell Pump.

_____ 2.23 At ≈ 70% power, perform "Steam Extraction Check Valve Test" enclosure of PT/1/A/0290/005 (Secondary System Protection Test).

2.24 **IF** operating with < 4 RCPs, perform the following prior to exceeding 70% FP:

_____ • Have I&E reset high flux limiter for applicable overpower trip setpoints per Enclosure "Special Instructions for < 4 RCP Operation".

_____ • Adjust the high flux alarm setpoints as required. (The alarm setpoints are adjusted on the NI Recorder).

_____ • Perform pre-job brief stressing parameters to be monitored during power increase **AND** other factors to consider per Enclosure "Special Instructions for < 4 RC Pump Operation".

NOTE: The following step does **NOT** have to be done in sequence.

_____ 2.25 **WHEN** both FDWPTs hydraulic oil pressure is > 190 psig, verify FDWPTs Aux Oil Pumps off **AND** switches are in "AUTO". (Continue)

NOTE:	
	• The following valves will be signed off in OP/1/A/1106/014 (Moisture Separator Reheater).
	• The following step does **NOT** have to be done in sequence.

_____ 2.26 **WHEN** High Pressure Turbine Exhaust pressure is ≈ 118 psig, complete OP/1/A/1106/014 (Moisture Separator Reheater) startup of Moisture Separator Reheater by verifying:

- 1MS-77 (MS TO 1A1 SSRH) Open
- 1MS-78 (MS TO 1A2 SSRH) Open
- 1MS-80 (MS TO 1B1 SSRH) Open
- 1MS-81 (MS TO 1B2 SSRH) Open

_____ 2.27 **WHEN** ≥ 85% CTP, begin startup of 'E' Heater Drain Pumps per OP/1/A/1106/002 (Condensate and Feedwater System). {7}

2.28 **IF** required, stop power increase at ≈ 90% CTP (indicated by Thermal Power Best) **AND** calibrate NIs to Thermal Power Best.

NOTE:	Remaining steps in this enclosure may be performed in any sequence.

_____ _____ 2.29 Increase CORE THERMAL POWER DEMAND (CTPD) SET to final core power desired.

_____ 2.30 **Prior** to calibrating NIs **OR** exceeding 95% CTP, notify Reactor Engineer to evaluate Feedwater Fouling Coefficient of Thermal Power Secondary calculation. {3}

_____ 2.31 Place the "1A FDWP SEAL INJ PUMP" in "NORM".

_____ 2.32 Place the "1B FDWP SEAL INJECTION PUMP" in "NORM".

2.33 **WHEN** Rx power is stable:

_____ 2.33.1 Notify Secondary Chemistry of intent to feed Moisture Separator Drains forward.

_____ 2.33.2 Align Moisture Separator Drain Tanks to feed forward per OP/1/A/1106/014 (Moisture Separator Reheater).

_____ 2.34 At ≈ 100% CTP, maintain steady state conditions for ≈ 15 minutes for NI calibration check.

1. Initial Conditions

_____ 1.1 **IF** reactor power will be reduced ≥ 40% with reactor remaining on line, notify Reactor Engineering to develop Maneuvering Plan.

_____ 1.2 Auxiliary Steam Header pressurized **OR** contingencies in place.

_____ 1.3 **IF** required, NRC notified per the requirements of OMP 1-10 (Usage and Testing the Emergency Notification System (Red Phone)).

_____ 1.4 Review Limits and Precautions.

2. Procedure

_____ 2.1 Refer to OP/1/A/1106/001 (Turbine Generator) to ensure operating limits are **NOT** exceeded during shutdown.

_____ 2.2 **IF** reducing reactor power > 6%, then refer to OP/1/A/1106/014 (Moisture Separator Reheater) to ensure Moisture Separator drains are routed to hotwell.

_____ 2.3 **IF** available, refer to Maneuvering Plan to view Reactor Engineering guidelines for power decrease. {6}.

_____ 2.4 Notify System Operations Center (SOC) of load reduction.

_____ 2.5 **IF** required, advise plant personnel of load reduction.

2.6 Start the following:

- _____ Place "1A FDWP SEAL INJ PUMP" switch to "START".
- _____ Place "1B FDWP SEAL INJECTION PUMP" switch to "START".

NOTE:	If Unit 1 will be taken off-line, target CTP should be ≈ 25% CTP

2.7 Begin CTP reduction to desired power level per the following:

_____ 2.7.1 Select "HOLD".

2.7.2 Select desired shutdown rate (one of the following):

- _____ RATE %/MIN
- _____ RATE %/HR.

_____ 2.7.3 Select desired rate of power reduction on RATE SET.

_____ 2.7.4 Select CTPD SET power level.

_____ 2.7.5 Release "HOLD".

2.8 **WHEN** ≤ 80% CTP: {7}

_____ • Stop 1E1 Heater Drain Pump.

_____ A. Verify **OR** place in AUTO 1HD-254 (1E1 HD PUMP RECIRC).

_____ • Stop 1E2 Heater Drain Pump.

_____ B. Verify **OR** place in AUTO 1HD-276 (1E2 HD PUMP RECIRC).

NOTE:	1B FDWPT should be taken out of service first. This is due to differences in High Pressure Discharge Trip Setpoints.

_____ 2.9 Start Aux Oil Pump on both FDWPTs.

_____ 2.10 At ≈ 550 MWe, shut down one FDWPT per OP/1/A/1106/002 (Condensate and Feedwater System).

NOTE:	1C CBP should remain running, if possible. {5}

_____ 2.11 At ≈ 450 Mwe, stop all but one CBP. {9}

_____ • Ensure control switch for one of the secured CBPs is in "AUTO".

_____ 2.12 At ≈ 400 MWe, stop 'D' HD Pumps per OP/1/A/1106/002 (Condensate and Feedwater System).

_____ 2.13 At ≈ 325 MWe, stop all but two HWPs.

_____ • Place control switch for the secured HWP in "AUTO".

_____ 2.14 At ≈ 225 MWe, transfer Auxiliaries from 1T to CT-1 per OP/1/A/1107/002 (Normal Power).

_____ 2.15 Stop all but one HWP.

_____ • Verify **OR** place control switch for one of the secured HWPs in "AUTO".

_____ 2.16 Verify SSF Operability per Enclosure "Minimum RCS Boron Concentration to Maintain " of PT/1/A/1103/015 (Reactivity Balance).

_____ 2.17 **<u>IF</u>** Turbine **<u>OR</u>** Reactor shutdown is required, **<u>GO TO</u>** OP/1/A/1102/010 (Controlling Procedure For Unit Shutdown).

1. Procedure

1.1 If conditions permit, log the current quadrant power tilt and the position of the ΔT_C controller prior to securing a RC Pump during power operations.

> **NOTE:** Instructions for performing OAC trends are located in "Working With Trends" enclosure of OP/0/A/1103/020A (Operator Aid Computer Use).

1.2 Digitally trend the following data at one minute intervals:

Point ID		Description
1.0	O1P0889	CORE THERMAL POWER BEST
2.0	O1P0877	INCORE IMBALANCE
3.0	O1A0059	CONTROL ROD GROUP 7 POSN
4.0	O1A0060	CONTROL ROD GROUP 8 POSN
5.0	O1P0737	INCORE TILT QUADRANT W-X
6.0	O1P0738	INCORE TILT QUADRANT X-Y
7.0	O1P0739	INCORE TILT QUADRANT Y-Z
8.0	O1P0740	INCORE TILT QUADRANT Z-W
9.0	O1P0828	RC COLD LEG A1 TEMP (1 MIN AVG)
10.0	O1P0829	RC COLD LEG A2 TEMP (1 MIN AVG)
11.0	O1P0830	RC COLD LEG B1 TEMP (1 MIN AVG)
12.0	O1P0831	RC COLD LEG B2 TEMP (1 MIN AVG)

_____ 1.3 Notify the Reactor Engineer to reset the Feedwater Venturi Fouling Coefficient of the Thermal Power Secondary Calculation to 1.0 prior to calibrating the NIs.

_____ 1.3.1 Calibrate NIs to Thermal Power Best.

1.4 Follow PT/1/A/0600/001 (Periodic Instrument Surveillance) limits on control rod position and Power Imbalance. The 100% Power Imbalance curves also apply for runs at reduced power.

NOTE:	If Quadrant Power Tilt problems do **NOT** exist, it is **NOT** necessary to reset the high flux RPS trip setpoint for three RC Pump operation.

_____ 1.5 Notify I&E of the potential for severe tilt problems **AND** the need to adjust flux/flow imbalance trip setpoints per requirements of TS 3.2.3.

1.6 Perform the following:

_____ 1.6.1 Adjust the ICS high flux limiter to 72%. This provides control protection to minimize a trip on flux/Flow/Imb **OR** high flux in the event of an operating transient.

_____ 1.6.2 **WHEN** the ICS high flux limiter is reduced, adjust the associated alarm setpoint to ≈ 2% less than the high flux limiter. (The alarm setpoint is adjusted on the NI Recorder).

_____ 1.6.3 Anytime the ICS high flux limiter is reduced, note on Turnover Sheet.

1.7 Keep Auxiliary Steam available to the FDWP turbines. 'D' bleed pressure may **NOT** be high enough to run the FDWP turbines.

1.8 Operating with three RC Pumps may cause a quadrant power tilt. The Steam Generator Load Ratio (ΔT_C) Controller can be used to minimize the magnitude of the tilt. If the tilt is on the 'A' side, quadrant W-X **OR** X-Y, adjust the ΔT_C Controller clockwise to make 'A' hot. If the tilt is on the 'B' side, quadrants Y-Z **OR** Z-W, adjust the ΔT_C controller counter clockwise to make 'B' hot.

The ΔT_C controller should be adjusted in small increments such as 1/2 degree F changes. All adjustments to the controller should be logged so that the recorded data can be accurately analyzed by the Reactor Group to determine the causes and results of unusual core power distributions.

1.9 If 1SSH-9 (SSH DISCH. CTRL BYPASS) is being used to maintain Steam Seal Header pressure, throttle the valve during the load reduction to secure an RC Pump.

NOTE: RCS pressure decrease in the loop with two RC Pumps running is expected.

1.10 RCS pressure decrease in the loop with two RC Pumps running may cause acceptance criteria of PT/1/A/0600/001 (Periodic Instrument Surveillance) **NOT** to be met. Note this on PT/1/A/0600/001 (Periodic Instrument Surveillance). Be aware of the effect of the indicated pressure on the margin to trip setpoint for the Reactor Protective System on the following:

1. Pressure/Temperature Trip
2. Low Pressure Trip
3. High Pressure Trip

(R06-97)

INFORMATION ONLY

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. RP/0/B/1000/001

Revision No 6

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Emergency Classification

(4) Prepared By Donica Kelley Date 2-16-99

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)
☐ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By W.C. Brandt (QR) Date 2/25/99

Cross-Disciplinary Review By ______ (QR)NA WB Date 2/25/99

Reactivity Mgmt. Review By ______ (QR)NA WB Date 2/25/99

(7) Additional Reviews

QA Review By ______ Date ______

Reviewed By ______ Date ______

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By M R Thorne Date 3-27-99

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______

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(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
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☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Site **Emergency Classification** **Reference Use**	Procedure No. RP/0/B/1000/001 Revision No. 006 Electronic Reference No. OX002WOS

Emergency Classification

NOTE:	This procedure is an implementing procedure tot he Oconee Nuclear Site Emergency plan and must be forwarded to Emergency Planning within three (3) working days of approval.

1. Symptoms

1.1 This procedure describes the immediate actions to be taken to recognize and classify an emergency condition.

1.2 This procedure identifies the four emergency classifications and their corresponding Emergency Action Levels (EALs).

1.3 This procedure provides reporting requirements for non-emergency abnormal events.

1.4 The following guidance is to be used by the Emergency Coordinator/EOF Director in assessing emergency conditions:

1.4.1 The Emergency Coordinator/EOF Director shall review all applicable initiating events to ensure proper classification.

1.4.2 The BASIS Document (Volume A, Section D of the Emergency Plan) is available for review if any questions arise over proper classification.

1.4.3 **IF** An event occurs on more than one unit concurrently,
THEN The event with the higher classification will be classified on the Emergency Notification Form.

A. Information relating to the problem(s) on the other unit(s) will be captured on the emergency Notification Form as shown in RP/0/B/1000/015A, (Offsite Communications From The Control Room), RP/0/B/1000/015B, (Offsite Communications From The Technical Support Center) or RP/0/B/1000/015C, (Offsite Communications From The Emergency Operations Facility).

1.4.4 **IF** An event occurs,
AND A lower or higher plant operating mode is reached before the classification can be made,
THEN The classification shall be based on the mode that existed at the time the event occurred.

1.4.5 The Fission Product Barrier Matrix is applicable only to those events that occur at Hot Shutdown or higher.

A. An event that is recognized at Cold Shutdown or lower shall not be classified using the Fission Product Barrier Matrix.

1. Reference should be made to the additional enclosures that provide Emergency Action Levels for specific events (e.g., Severe Weather, Fire, Security).

1.5 **IF** A transient event should occur,

THEN Review the following guidance:

1.5.1 **IF** An Emergency Action Level (EAL) identifies a specific duration
AND The Emergency Coordinator/EOF Director assessment concludes that the specified duration is exceeded or will be exceeded, (i.e.; condition cannot be reasonably corrected before the duration elapses),
THEN Classify the event.

1.5.2 **IF** A plant condition exceeding EAL criteria is corrected before the specified duration time is exceeded,
THEN The event is **NOT** classified by that EAL.

A. Review lower severity EALs for possible applicability in these cases.

NOTE:	Reporting under 10CFR50.72 may be required for the following step. Such a condition could occur, for example, if a follow up evaluation of an abnormal condition uncovers evidence that the condition was more severe than earlier believed.

1.5.3 **IF** A plant condition exceeding EAL criteria is not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g.; as a result of routine log or record review)
AND The condition no longer exists,
THEN An emergency shall **NOT** be declared.

1.5.4 **IF** An emergency classification was warranted, but the plant condition has been corrected prior to declaration and notification,
THEN The Emergency Coordinator must consider the potential that the initiating condition (e.g.; Failure of Reactor Protection System) may have caused plant damage that warrants augmenting the on shift personnel through activation of the Emergency Response Organization.

A. **IF** An **Unusual Event** condition exists,
THEN Make the classification as required.

1. The event may be terminated in the same notification or as a separate termination notification.

B. **IF** **An Alert, Site Area Emergency**, or **General Emergency** condition exists,
THEN Make the classification as required,
AND Activate the Emergency Response Organization.

1.6 Emergency conditions shall be classified as soon as the Emergency Coordinator/EOF Director assessment determines that the Emergency Action Levels for the Initiating Condition have been exceeded.

2. Immediate Actions

2.1 Determine the operating mode that existed at the time the event occurred prior to any protection system or operator action initiated in response to the event.

2.2 **IF** The unit is at Hot Shutdown or higher
AND The condition/event affects fission product barriers,
THEN GOTO Enclosure 4.1, (Fission Product Barrier Matrix).

2.2.1 Review the criteria listed in Enclosure 4.1, (Fission Product Barrier Matrix) and make the determination if the event should be classified.

2.3 Review the listing of enclosures to determine if the event is applicable to one of the categories shown.

2.3.1 **IF** One or more categories are applicable to the event,
THEN Refer to the associated enclosures.

2.3.2 Review the EALs and determine if the event should be classified.

A. **IF** An EAL is applicable to the event,
THEN Classify the event as required.

2.4 **IF** The condition requires an emergency classification,
THEN GOTO RP/0/B/1000/002, (Control Room Emergency Coordinator Procedure).

3. Subsequent Actions

3.1 Continue to review the emergency conditions to assure the current classification continues to be applicable.

4. Enclosures

DETERMINE THE APPROPRIATE CLASSIFICATION USING THE TABLE BELOW: CIRCLE EALS CHOSEN. ADD POINTS TO CLASSIFY. (SEE NOTE BELOW)

RCS BARRIERS (BD 5-7)		FUEL CLAD BARRIERS (BD 8-9)		CONTAINMENT BARRIERS (BD 10-12)	
Potential Loss (4)	**Loss (5)**	**Potential Loss (4)**	**Loss (5)**	**Potential Loss (1)**	**Loss (3)**
RCS Leakrate > Makeup capacity of one HPI pump in normal makeup mode (approx. 160 gpm) with Letdown isolated.	RCS Leak rate > available makeup capacity as indicated by a loss of subcooling	Average of the 5 highest CETC ≥ 700° F	Average of the 5 highest CETC ≥ 1200° F	CETC ≥ 1200° F ≥ 15 minutes **OR** CETC ≥ 700° F ≥ 15 minutes with a valid RVLS reading 0"	Rapid unexplained containment pressure decrease after increase **OR** containment pressure or sump level not consistent with LOCA
SGTR > Makeup capacity of one HPI pump in normal makeup mode (approx. 160 gpm) with Letdown isolated.		Valid RVLS reading of 0"	Coolant activity ≥ 300 μCi/ml DEI	RB pressure ≥ 59 psig **OR** RB pressure ≥ 10 psig and no RBCU or RBS	Failure of secondary side of SG results in a direct opening to the environment with P/S leakage ≥ 10 gpm in the same SG
Entry into the TSOR (Thermal Shock) operating range	1RIA 57/58 reading ≥ 1.0 R/hr 2 RIA 57 reading ≥ 1.6 R/hr 2 RIA 58 reading ≥ 1.0 R/hr 3RIA 57/58 reading ≥ 1.0 R/hr		<u>Hours Since SD</u> — <u>RIA57/58 - R/hr</u> 0 - < 0.5 — ≥ 300/150 0.5 - < 2.0 — ≥ 80/40 2.0 - 8.0 — ≥ 32/16	<u>Hours Since SD</u> — <u>RIA57/58 - R/hr</u> 0 - < 0.5 — ≥ 1800/860 0.5 - < 2.0 — ≥ 400/195 2.0 - 8.0 — ≥ 280/130	Failure of secondary side of SG results in a direct opening to the environment with P/S leakage ≥ 10 gpm in the other SG **AND** Feeding SG with secondary side failure from the affected unit
HPI Forced Cooling	RCS pressure spike ≥ 2750 psig			Hydrogen concentration ≥ 9%	Containment isolation is incomplete and a release path to the environment exists
Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment

UNUSUAL EVENT (1-3)	ALERT (4-6)	SITE AREA EMERGENCY (7-10)	GENERAL EMERGENCY (11-13)
<u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ Any potential loss of Containment ♦ Any loss of containment	<u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ Any potential loss or loss of the Fuel Clad ♦ Any potential loss or loss of the RCS	<u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ Loss of any two barriers ♦ Loss of one barrier and potential loss of either RCS or Fuel Clad Barriers ♦ Potential loss of both the RCS and Fuel Clad Barriers	<u>OPERATING MODE:</u> 1, 2, 3, 4 ♦ Loss of any two barriers and potential loss of the third barrier ♦ Loss of all three barriers
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4

NOTE: An event with multiple events could occur which would result in the conclusion that exceeding the loss or potential loss threshold is <u>IMMINENT</u> (i.e., within 1-3 hours). In this IMMINENT LOSS situation, use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. RCS LEAKAGE (BD 14) OPERATING MODE: 1, 2, 3, 4 • Unidentified leakage ≥ 10 gpm • Pressure boundary leakage ≥ 10 gpm • Identified leakage ≥ 25 gpm 2. UNPLANNED LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/INDICATION IN CONTROL ROOM FOR > 15 MINUTES (BD 15) OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND Loss of anunciators or indicators requires additional personnel (beyond normal shift complement) to safely operate the unit 3. INABILITY TO REACH REQUIRED SHUTDOWN WITHIN LIMITS (BD 16) OPERATING MODE: 1, 2, 3, 4 • Required operating mode not reached within TS LCO action statement time (CONTINUED)	1. UNPLANNED LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/INDICATION IN CONTROL ROOM (BD 19) OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND Loss of anunciators/indicators requires additional personnel (beyond normal shift complement) to safely operate the unit AND EITHER OF THE FOLLOWING: • Significant plant transient in progress OR • Loss of the OAC and ALL PAM indications (END)	1. INABILITY TO MONITOR A SIGNIFICANT TRANSIENT IN PROGRESS (BD 21) OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND A significant transient is in progress AND Loss of the OAC and ALL PAM indications AND Inability to directly monitor any one of the following functions: 1. Subcriticality 2. Core Cooling 3. Heat Sink 4. RCS Integrity 5. Containment Integrity 6. RCS Inventory (END)	
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4. UNPLANNED LOSS OF ALL ONSITE OR OFFSITE COMMUNICATIONS (BD 17) == OPERATING MODE: All ♦ Loss of all onsite communications capability (ROLM system, PA system, Pager system, Onsite Radio system) affecting ability to perform routine operations ♦ Loss of all onsite communications capability (Selective signaling, NRC FTS lines, Offsite Radio System, AT&T line) affecting ability to communicate with offsite authorities. 5. FUEL CLAD DEGRADATION (BD 18) == OPERATING MODE: All: ♦ DEI - >5μCi/ml (END)			
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4			

Enclosure 4.3
Abnormal Rad Levels/Radiological Effluent

UNUSUAL EVENT

1. ANY UNPLANNED RELEASE OF GASEOUS OR LIQUID RADIOACTIVITY TO THE ENVIRONMENT THAT EXCEEDS TWO TIMES THE SLC LIMITS FOR 60 MINUTES OR LONGER (BD 23)

===

<u>OPERATING MODE:</u> All

- Valid indication on radiation monitor RIA 33 of ≥ 4.06E+06 cpm for > 60 minutes (See Note 1)
- Valid indication on radiation monitor RIA 45 of ≥ 1.33E+06 cpm for > 60 minutes (See Note 1)
- Liquid effluent being released exceeds two times SLC 16.11.1 for > 60 minutes as determined by Chemistry Procedure
- Gaseous effluent being released exceeds two times SLC 16.11.2 for > 60 minutes as determined by RP Procedure

> NOTE 1: If monitor reading is sustained for the time period indicated in the EAL <u>AND</u> the required assessments (procedure calculations) cannot be completed within this period, declaration must be made on the valid Radiation Monitor reading.

(CONTINUED)

INITIAL NOTIFICATION REQUIREMENTS:
SEE EMERGENCY TELEPHONE DIRECTORY

NOTIFY 1, 2, 3, 4

ALERT

1. ANY UNPLANNED RELEASE OF GASEOUS OR LIQUID RADIOACTIVITY TO THE ENVIRONMENT THAT EXCEEDS 200 TIMES RADIOLOGICAL TECHNICAL SPECIFICATIONS FOR 15 MINUTES OR LONGER (BD 28)

===

<u>OPERATING MODE:</u> All

- Valid indication on RIA 46 of ≥ 2.98E+04 cpm for >15 minutes (See Note 1)
- RIA 33 HIGH Alarm
  <u>AND</u>
  Liquid effluent being released exceeds 200 times the level of SLC 16.11.1 for > 15 minutes as determined by Chemistry Procedure
- Gaseous effluent being released exceeds 200 times the level of SLC 16.11.2 for >15 minutes as determined by RP Procedure

2. RELEASE OF RADIOACTIVE MATERIAL OR INCREASES IN RADIATION LEVELS THAT IMPEDES OPERATION OF SYSTEMS REQUIRED TO MAINTAIN SAFE OPERATION OR TO ESTABLISH OR MAINTAIN COLD SHUTDOWN (BD 30)

===

<u>OPERATING MODE:</u> All

- Valid radiation reading ≥ 15 mRad/hr in CR, CAS <u>OR</u> Radwaste CR
- Unplanned/unexpected valid area monitor readings exceed limits stated in Enclosure 4.9

(CONTINUED)

INITIAL NOTIFICATION REQUIREMENTS:
SEE EMERGENCY TELEPHONE DIRECTORY

NOTIFY 1, 2, 3, 4

SITE AREA EMERGENCY

1. BOUNDARY DOSE RESULTING FROM ACTUAL/ IMMINENT RELEASE OF GASEOUS ACTIVITY (BD 32)

===

<u>OPERATING MODE:</u> All

- Valid reading on RIA 46 of ≥ 2.98E+05 cpm for >15 minutes (See Note 2)
- Valid reading on RIA 57 or 58 as shown on Enclosure 4.8 (See Note 2)
- Dose calculations result in a dose projection at the site boundary of:
  ≥ 100 mRem TEDE or 500 mRem CDE adult thyroid
- Field survey results indicate site boundary dose rates exceeding ≥100 mRad/hr expected to continue for more than one hour
  <u>OR</u>
  Analyses of field survey samples indicate adult thyroid dose commitment of ≥ 500 mRem CDE (3.84 E^{-7} μCi/ml) for one hour of inhalation

> NOTE 2: If actual Dose Assessment cannot be completed within 15 minutes, then the valid radiation monitor reading should be used for emergency classification.

(CONTINUED)

INITIAL NOTIFICATION REQUIREMENTS:
SEE EMERGENCY TELEPHONE DIRECTORY

NOTIFY 1, 2, 3, 4

GENERAL EMERGENCY

1. BOUNDARY DOSE RESULTING FROM ACTUAL/ IMMINENT RELEASE OF GASEOUS ACTIVITY (BD 36)

===

<u>OPERATING MODE:</u> All

- Valid reading on RIA 46 of ≥ 2.98E+06 cpm for ≥15 minutes (See Note 3)
- Valid reading on RIA 57 or 58 as shown on Enclosure 4.8 (See Note 3)
- Dose calculations result in a dose projection at the site boundary of:
  ≥ 1000 mRem TEDE
  <u>OR</u>
  ≥ 5000 mRem CDE adult thyroid
- Field survey results indicate site boundary dose rates exceeding ≥1000 mRad/hr expected to continue for more than one hour
  <u>OR</u>
  Analyses of field survey samples indicate adult thyroid dose commitment of ≥ 5000 mRem CDE for one hour of inhalation

> NOTE 3: If actual Dose Assessment cannot be completed within 15 minutes, then the valid radiation monitor reading should be used for emergency classification.

(END)

INITIAL NOTIFICATION REQUIREMENTS:
SEE EMERGENCY TELEPHONE DIRECTORY

NOTIFY 1, 2, 3, 4

Assumptions used for calculation of vent monitors RIA 45 & 46:

1. Average annual meteorology (1.672 E-6 sec/m³), semi-elevated
2. Vent flow rate 65,000 cfm (average daily flow rate)
3. No credit is taken for vent filtration
4. One hour release duration for Unusual Event, 15 minute duration for Alert, Site Area Emergency, General Emergency
5. General Emergency PAGs are 1 rem TEDE and 5 rem CDE; Site Area Emergency determination is based on 10% of the General Emergency PAGs
6. Calculations for monitor readings are based on whole body dose
7. Standard ODCM guidance together with NUMARC guidance indicates that effluent releases are based on Technical Specification releases

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2. UNEXPECTED INCREASE IN PLANT RADIATION OR AIRBORNE CONCENTRATION (BD 25) OPERATING MODE: All ♦ LT 5 reading 14" and decreasing with makeup not keeping up with leakage WITH fuel in the core ♦ Uncontrolled water level decrease in the SFP and fuel transfer canal with all irradiated fuel assemblies remaining covered by water ♦ 1 R/hr radiation reading at one foot away from a damaged storage cask located at the ISFSI ♦ Valid area monitor readings exceeds limits stated in Enclosure 4.9. (END)	3. MAJOR DAMAGE TO IRRADIATED FUEL OR LOSS OF WATER LEVEL THAT HAS OR WILL RESULT IN THE UNCOVERING OF IRRADIATED FUEL OUTSIDE THE REACTOR VESSEL (BD 31) OPERATING MODE: All ♦ Valid RIA 3, 6, 41, OR 49 HIGH Alarm ♦ HIGH Alarm for portable area monitors on the main bridge or auxiliary bridge or SFP bridge ♦ Report of visual observation of irradiated fuel uncovered ♦ Operators determine water level drop in either the SFP or fuel transfer canal will exceed makeup capacity such that irradiated fuel will be uncovered (END)	2. LOSS OF WATER LEVEL IN THE REACTOR VESSEL THAT HAS OR WILL UNCOVER FUEL IN THE REACTOR VESSEL (BD 35) OPERATING MODE: 5, 6 ♦ Failure of heat sink causes loss of Cold Shutdown condition AND LT 5 indicates 0 inches after initiation of RCS makeup ♦ Failure of heat sink causes loss of Cold Shutdown condition AND Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup NOTE: This Initiating Condition is also located in Enclosure 4.4, (Loss of Shutdown Functions). High radiation levels will also be seen with this condition. (END)	
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	1. FAILURE OF RPS TO COMPLETE OR INITIATE A Rx SCRAM (BD 39) OPERATING MODE 1, 2, 3 ♦ Valid reactor trip signal received or required WITHOUT automatic scram AND ONE OF THE FOLLOWING DSS has inserted Control Rod Groups 5, 6, 7 OR Manual trip from the Control Room is successful and reactor power is less than 5% and decreasing 2. INABILITY TO MAINTAIN PLANT IN COLD SHUTDOWN (BD 41) OPERATING MODE: 5, 6 ♦ Loss of LPI and/or LPSW AND Inability to maintain RCS temperature below 200° F as indicated by either of the following: RCS temperature at the LPI Pump Suction OR visual observation (END)	1. FAILURE OF RPS TO COMPLETE OR INITIATE A Rx SCRAM (BD 42) OPERATING MODE: 1, 2 ♦ Valid reactor trip signal received or required WITHOUT automatic scram AND DSS has NOT inserted Control Rod Groups 5, 6, 7 AND Manual trip from the Control Room was NOT successful in reducing reactor power to less than 5% and decreasing 2. COMPLETE LOSS OF FUNCTION NEEDED TO ACHIEVE OR MAINTAIN HOT SHUTDOWN (BD 43) OPERATING MODE: 1, 2, 3, 4 ♦ Average of the 5 highest CETCs ≥1200° F shown on ICCM ♦ Unable to maintain reactor subcritical ♦ SSF feeding SG per EOP (CONTINUED)	1. FAILURE OF RPS TO COMPLETE AUTOMATIC SCRAM AND MANUAL SCRAM NOT SUCCESSFUL WITH INDICATION OF CORE DAMAGE (BD 45) OPERATING MODE: 1, 2 ♦ Valid Rx trip signal received or required WITHOUT automatic scram AND Manual trip from the Control Room was NOT successful in reducing reactor power to < 5% and decreasing AND Average of the 5 highest CETCs ≥1200° F on ICCM (END)
	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
		3. LOSS OF WATER LEVEL IN THE REACTOR VESSEL THAT HAS OR WILL UNCOVER FUEL IN THE REACTOR VESSEL (BD 44) OPERATING MODE: 5, 6 • Failure of heat sink causes loss of Cold Shutdown conditions AND LT-5 indicates 0 inches after initiation of RCS makeup • Failure of heat sink causes loss of Cold Shutdown conditions AND Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup (END)	
		INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. LOSS OF ALL OFFSITE POWER TO ESSENTIAL BUSSES FOR GREATER THAN 15 MINUTES (BD 47) OPERATING MODE: All ♦ Loss of all offsite AC power to both the Red and Yellow Busses for > 15 minutes AND Unit auxiliaries are being supplied from Keowee or CT5 2. UNPLANNED LOSS OF REQUIRED DC POWER FOR GREATER THAN 15 MINUTES (BD 48) OPERATING MODE: 5, 6 ♦ Unplanned loss of vital DC power to required DC busses as indicated by bus voltage less than 110 VDC AND Failure to restore power to at least one required DC bus within 15 minutes from the time of loss (END)	1. LOSS OF ALL OFFSITE AC POWER AND LOSS OF ALL ONSITE AC POWER TO ESSENTIAL BUSSES (BD 49) OPERATING MODE: 5, 6 Defueled ♦ MFB 1 and 2 de-energized AND Failure to restore power to at least one MFB within 15 minutes from the time of loss of both offsite and onsite AC power 2. AC POWER CAPABILITY TO ESSENTIAL BUSSES REDUCED TO A SINGLE SOURCE FOR GREATER THAN 15 MINUTES (BD 50) OPERATING MODE: 1, 2, 3, 4 ♦ AC power capability has been degraded to a single power source for > 15 minutes due to the loss of all but one of: Unit Normal Transformer Unit SU Transformer Another Unit SU Transformer CT4 CT5 (END)	1. LOSS OF ALL OFFSITE AC POWER AND LOSS OF ALL ONSITE AC POWER TO ESSENTIAL BUSSES (BD 51) OPERATING MODE: 1, 2, 3, 4 ♦ MFB 1 and 2 de-energized AND Failure to restore power to at least one MFB within 15 minutes from the time of loss of both offsite and onsite AC power 2. LOSS OF ALL VITAL DC POWER (BD 52) OPERATING MODE: 1, 2, 3, 4 ♦ Unplanned loss of vital DC power to required DC busses as indicated by bus voltage less than 110 VDC AND Failure to restore power to at least one required DC bus within 15 minutes from the time of loss (END)	1. PROLONGED LOSS OF ALL OFFSITE POWER AND ONSITE AC POWER (BD 54) OPERATING MODE: 1, 2, 3, 4 ♦ MFB 1 and 2 de-energized AND SSF fails to maintain Hot Shutdown AND At least one of the following conditions exist: Restoration of power to at least one MFB within 4 hours is NOT likely OR Indications of continuing degradation of core cooling based on Fission Product Barrier monitoring (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. FIRES/EXPLOSIONS WITHIN THE PLANT (BD 57) ========== OPERATING MODE: All NOTE: Within the plant means Turbine Building, Auxiliary Building, Reactor Building, Keowee Hydro ♦ Fire within the plant not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm ♦ Unanticipated explosion within the plant resulting in visible damage to permanent structures/equipment 2. CONFIRMED SECURITY THREAT INDICATES POTENTIAL DEGRADATION IN THE LEVEL OF SAFETY OF PLANT (BD 58) ========== OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications ♦ Discovery of bomb within plant protected area and outside security vital areas ♦ Hostage/Extortion situation ♦ Violent civil disturbance within the owner controlled area (END)	1. FIRE/EXPLOSION AFFECTING OPERABILITY OF PLANT SAFETY SYSTEMS REQUIRED TO ESTABLISH/MAINTAIN SAFE SHUTDOWN (BD 59) ========== OPERATING MODE: All NOTE: Only one train of a system needs to be affected or damaged in order to satisfy this condition. ♦ Fire/explosions AND ONE OF THE FOLLOWING: Affected safety-related system parameter indications show degraded performance OR Plant personnel report visible damage to permanent structures or equipment required for safe shutdown 2. SECURITY EVENT IN A PLANT PROTECTED AREA (BD 60) ========== OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications ♦ Intrusion into plant protected area by a hostile force ♦ Bomb discovered in an area containing safety related equipment (END)	1. SECURITY EVENT IN A PLANT VITAL AREA (BD 61) ========== OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications ♦ Intrusion into any of the following plant areas by a hostile force: Reactor Building Auxiliary Building Keowee Hydro ♦ Bomb detonated in the following areas: • Keowee Hydro • Keowee Dam • ISFSI • Reactor Building • Auxiliary Building • SSF (END)	1. SECURITY EVENT RESULTING IN LOSS OF ABILITY TO REACH AND MAINTAIN COLD SHUTDOWN (BD 62) ========== OPERATING MODE: All NOTE: RP/0/B/1000/007, (Security Event) shall be used in conjunction with all security related emergency classifications ♦ Loss of physical control of the control room due to security event ♦ Loss of physical control of the Aux Shutdown panel and the SSF due to a Security Event (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING THE PROTECTED AREA (BD 64) ========== OPERATING MODE: All ♦ Tremor felt and valid alarm on the strong motion accelerograph ♦ Tornado striking within Protected Area Boundary ♦ Vehicle crash into plant structures/systems within the Protected Area Boundary ♦ Turbine failure resulting in casing penetration or damage to turbine or generator seals (CONTINUED)	1. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING THE PLANT VITAL AREA (BD 69) ========== OPERATING MODE: All ♦ Tremor felt and seismic trigger actuates (0.05g) ♦ Tornado, high winds, missiles resulting from turbine failure, vehicle crashes, or other catastrophic event AND ONE OF THE FOLLOWING: NOTE: Only one train of a safety-related system needs to be affected or damaged in order to satisfy these conditions. Visible damage to permanent structures or equipment required for safe shutdown of the unit OR Affected safety system parameter indications show degraded performance 2. RELEASE OF TOXIC/FLAMMABLE GASES JEOPARDIZING SYSTEMS REQUIRED TO MAINTAIN SAFE OPERATION OR ESTABLISH/ MAINTAIN COLD SHUTDOWN (BD 71) ========== OPERATING MODE: All ♦ Report/detection of toxic gases in concentrations that will be life-threatening to plant personnel ♦ Report/detection of flammable gases in concentrations that will affect the safe operation of the plant: • Reactor Building • Auxiliary Building • Turbine Building • Control Room (CONTINUED)	1. CONTROL ROOM EVACUATION AND PLANT CONTROL CANNOT BE ESTABLISHED (BD 75) ========== OPERATING MODE: All ♦ Control Room evacuation has been initiated AND Control of the plant cannot be established from the Aux Shutdown Panel or the SSF within 15 minutes 2. KEOWEE HYDRO DAM FAILURE (BD 76) ========== OPERATING MODE: All ♦ Imminent/actual dam failure (includes any of the following: • Keowee Hydro Dam • Little River Dam • Dikes A, B, C, or D • Intake Canal Dike 3. OTHER CONDITIONS WARRANT DECLARATION OF SITE AREA EMERGENCY (BD 77) ========== OPERATING MODE: All ♦ Emergency Coordinator/EOF Director judgment (END)	1. OTHER CONDITIONS WARRANT DECLARATION OF GENERAL EMERGENCY (BD 78) ========== OPERATING MODE: All ♦ Emergency Coordinator/EOF Director judgment indicates: Actual/imminent substantial core degradation with potential for loss of containment OR Potential for uncontrolled radionuclide releases that would result in a dose projection at the site boundary greater than 1000 mRem TEDE or 5000 mRem CDE Adult Thyroid (END)
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING KEOWEE HYDRO (BD 66) == OPERATING MODE: All ♦ Reservoir elevation ≥ 807 feet with all spillway gates open and the lake elevation continues to rise ♦ Seepage readings increase or decrease greatly or seepage water is carrying a significant amount of soil particles ♦ New area of seepage or wetness, with large amounts of seepage water observed on dam, dam toe, or the abutments ♦ Slide or other movement of the dam or abutments which could develop into a failure ♦ Developing failure involving the powerhouse or appurtenant structures and the operator believes the safety of the structure is questionable 3. RELEASE OF TOXIC OR FLAMMABLE GASES DEEMED DETRIMENTAL TO SAFE OPERATION OF THE PLANT (BD 67) == OPERATING MODE: All ♦ Report/detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant ♦ Report by local, county, state officials for potential evacuation of site personnel based on offsite event (CONTINUED)	3. TURBINE BUILDING FLOOD (BD 72) == OPERATING MODE: All ♦ Turbine Building flood requiring use of AP/1,2,3/A/1700/10, (Uncontrolled Flooding Of Turbine Building) 4. CONTROL ROOM EVACUATION HAS BEEN INITIATED (BD 73) == OPERATING MODE: All ♦ Evacuation of Control Room AND ONE OF THE FOLLOWING: Plant control IS established form the Aux Shutdown Panel or the SSF OR Plant control IS BEING established from the Aux Shutdown Panel or SSF 5. OTHER CONDITIONS WARRANT CLASSIFICATION OF AN ALERT (BD 74) == OPERATING MODE: All ♦ Emergency Coordinator judgment indicates that: Plant safety may be degraded AND Increased monitoring of plant functions is warranted (END)		
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4. OTHER CONDITIONS EXIST WHICH WARRANT DECLARATION OF AN UNUSUAL EVENT (BD 68) == OPERATING MODE: All • Emergency Coordinator determines potential degradation of level of safety has occurred (END)			
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4			

NOTE: IF Actual Dose Assessment **cannot** be completed within 15 minutes.
THEN The valid monitor reading should be used for Emergency Classification.

All RIA values are considered GREATER THAN or EQUAL TO

HOURS SINCE REACTOR TRIPPED	RIA 57 R/hr		RIA 58 R/hr*	
	Site Area Emergency	General Emergency	Site Area Emergency	General Emergency
0.0 - < 0.5	5.9E+003	5.9E+004	2.6E+003	2.6E+004
0.5 - < 1.0	2.6E+003	2.6E+004	1.1E+003	1.1E+004
1.0 - < 1.5	1.9E+003	1.9E+004	8.6E+002	8.6E+003
1.5 - < 2.0	1.9E+003	1.9E+004	8.5E+002	8.5E+003
2.0 - < 2.5	1.4E+003	1.4E+004	6.3E+002	6.3E+003
2.5 - < 3.0	1.2E+003	1.2E+004	5.7E+002	5.7E+003
3.0 - < 3.5	1.1E+003	1.1E+004	5.2E+002	5.2E+003
3.5 - < 4.0	1.0E+003	1.0E+004	4.8E+002	4.8E+003
4.0 - < 8.0	1.0E+003	1.0E+004	4.4E+002	4.4E+003

*** RIA 58 is partially shielded**

Assumptions used for calculation of high range in-containment monitors RIA 57 and 58:

1. Average annual meteorology (7.308 E^{-6} sec/m^3)
2. Design basis leakage (5.6 E^6 ml/hr)
3. One hour release duration
4. General Emergency PAGs are 1 rem TEDE and 5 rem CDE; Site Area Emergency determination is based on 10% of the General Emergency PAGs
5. Calculations for monitor readings are based on CDE because thyroid dose is limiting
6. No credit is taken for filtration
7. LOCA conditions are limiting and provide the more conservative reading

NOTE: This Initiating Condition is not intended to apply to anticipated temporary increases due to planned events (e.g.; incore detector movement, radwaste container movement, depleted resin transfers, etc.).

MONITOR NUMBER	UNITS 1, 2, 3	
	UNUSUAL EVENT 1000x NORMAL LEVELS mRAD/HR	ALERT mRAD/HR
RIA 7, Hot Machine Shop Elevation 796	150	≥ 5000
RIA 8, Hot Chemistry Lab Elevation 796	4200	≥ 5000
RIA 10, Primary Sample Hood Elevation 796	830	≥ 5000
RIA 11, Change Room Elevation 796	210	≥ 5000
RIA 12, Chem Mix Tank Elevation 783	800	≥ 5000
RIA 13, Waste Disposal Sink Elevation 771	650	≥ 5000
RIA 15, HPI Room Elevation 758	NOTE*	≥ 5000

NOTE: RIA 15 normal readings are approximately 9 mRad/hr on a daily basis. Applying 1000x normal readings would put this monitor greater than 5000 mRad/hr just for an Unusual Event. For this reason, an Unusual Event will <u>NOT</u> be declared for a reading less than 5000 mRad/hr.

1. List of Definitions and Acronymns

1.1 ALERT - Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

1.2 BOMB - A fused explosive device

1.3 CONDITION A - Failure is Imminent or Has Occurred - A failure at the dam has occurred or is about to occur and minutes to days may be allowed to respond dependent upon the proximity to the dam.

1.4 CONDITION B - Potentially Hazardous Situation is Developing - A situation where failure may develop, but preplanned actions taken during certain events (such as major floods, earthquakes, evidence of piping) may prevent or mitigate failure.

1.5 CIVIL DISTURBANCE - A group of ten (10) or more people violently protesting station operations or activities at the site.

1.6 EXPLOSION - A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. A sudden failure of a pressurized pipe/line could fit this definition.

1.7 EXTORTION - An attempt to cause an action at the station by threat of force.

1.8 FIRE - Combustion characterized by heat and light. Sources of smoke, such as slipping drive belts or overheated electrical equipment, do NOT constitute fires. Observation of flames is preferred but is NOT required if large quantities of smoke and heat are observed.

1.9 GENERAL EMERGENCY - Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guidelines exposure levels outside the Exclusion Area Boundary.

1.10 HOSTAGE - A person or object held as leverage against the station to ensure demands will be met by the station.

1.11 INTRUSION/INTRUDER - Suspected hostile individual present in a Protected Area without authorization.

1.12 INABILITY TO DIRECTLY MONITOR - Operational Aid Computer data points are unavailable or gauges/panel indications are NOT readily available to the operator.

1.13 PROTECTED AREA - Encompasses all Owner Controlled Areas within the security perimeter fence.

1.14 RUPTURED (As relates to Steam Generator) - Existence of Primary to Secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

1.15 SABOTAGE - Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment unavailable.

1.16 SAFETY-RELATED SYSTEMS AREA - Any area within the Protected Area which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

1.17 SIGNIFICANT TRANSIENT - An unplanned event involving one or more of the following:

(1) Automatic turbine runback > 25% thermal reactor power
(2) Electrical load rejection > 25% full electrical load
(3) Reactor Trip
(4) Safety Injection System Activation

1.18 SITE AREA EMERGENCY - Events are in process or have occurred which involve actual or likely major failures of plant functions needed for the protection of the public. Any releases are NOT expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels outside the Exclusion Area Boundary.

1.19 SELECTED LICENSEE COMMITMENT (SLC) - Chapter 16 of the FSAR

1.20 SITE BOUNDARY - That area, including the Protected Area, in which DPC has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius from the center of Unit 2).

1.21 TOXIC GAS - A gas that is dangerous to life or health by reason of inhalation or skin contact (e.g.; Chlorine)

1.22 UNCONTROLLED - Event is not the result of planned actions by the plant staff

1.23 UNPLANNED - An event or action is UNPLANNED if it is not the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

1.24 UNUSUAL EVENT - Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

1.25 **VALID** - An indication or report or condition is considered to be **VALID** when it is conclusively verified by: (1) an instrument channel check; or, (2) indications on related or redundant instrumentation; or, (3) by direct observation by plant personnel such that doubt related to the instrument's operability, the condition's existence, or the report's accuracy is removed. Implicit with this definition is the need for timely assessment.

1.26 **VIOLENT** - Force has been used in an attempt to injure site personnel or damage plant property.

1.27 **VISIBLE DAMAGE** - Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage: deformation due to heat or impact, denting, penetration, rupture,

Operating Modes Defined In Improved Technical Specifications

MODES

MODE	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER (a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 250
4	Hot Shutdown (b)	< 0.99	NA	$250 > T > 200$
5	Cold Shutdown (b)	< 0.99	NA	≤ 200
6	Refueling (c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

6.

INITIATE HPI COOLING
(all except Davis-Besse)

Fulfillment of this CT requires the following:

RV P-T/PTS not limiting:

One HPI pump operating at full flow and the PORV maintained open

RV P-T and/or PTS limiting:

One HPI pump operating, the PORV maintained open, and flow throttled as necessary to prevent exceeding the applicable limit

1.0 PLANT CONDITIONS

For the purposes of maintaining adequate core cooling when primary to secondary heat transfer is not available, the GEOG prescribes performance of this CT for 2 specific cases. They are, 1) feedwater not available and 2) feedwater available.

Feedwater not available:

HPI cooling must be initiated when:

the PORV automatically lifts

<u>or</u>

RC pressure reaches the PORV setpoint

<u>or</u>

RC pressure reaches the PTS limit (if PTS limit is invoked)

<u>or</u>

RC pressure reaches the RV P-T limit

Feedwater available:

HPI cooling must be initiated when:

RC initially subcooled and adequate SCM is lost

<u>or</u>

RC saturated and attempts to establish adequate primary to secondary heat transfer have failed (includes establishing the SG(s) as heat sinks and bumping RCPs if possible)

2.0 ASSOCIATED GEOG BASES

HPI flow supplied to the core and subsequently released from the pressurizer (i.e., through the PORV and/or pressurizer safety valves) can provide adequate core cooling if all feedwater is lost. This has been determined by an analysis that used 10CFR50 Appendix K assumptions for all inputs except decay heat; the input assumption for decay heat was: 1.0 x (ANS 1971 decay heat value). This analysis showed that one HPI pump started at full flow within 20 minutes of the loss of feedwater, in conjunction with mass and energy removal through only the pressurizer safety valves (i.e., no PORV flow), was sufficient to cool the core (criteria of 10CFR50.46 were not violated). This situation necessarily describes RC pressure near the pressurizer safety valve setpoint.

If all feedwater is lost, the action to establish HPI cooling must be made expeditiously in order to establish core cooling before too much RC is lost. For this reason, if all feedwater is lost, HPI cooling should be established when or before the RCS pressure reaches the PORV open setpoint (i.e., the first automatic PORV lift following loss of all feedwater). This initiation criteria is appropriate, since it represents the point when RC inventory will commence being lost. Further, keying initiation of HPI cooling to RC pressure avoids the use of time as an operational criteria.

Whenever HPI cooling is initiated and only one HPI pump is operable then the PORV must be maintained open. While analysis indicated that one HPI pump operating in conjunction with pressurizer safety valves can adequately cool the core, it also indicated a small margin of collapsed liquid level to core uncovery (this margin was greater for two HPI pump operation). Promptly opening the PORV reduces the rate at which the net RCS inventory decreases, hence, the margin to core uncovery <u>for this limited makeup condition</u> is increased. This means that RCS inventory loss throughout the transient will not be as great as it would be if the PORV were not opened or if the PORV opening was delayed. Due to the minimum margins resulting from HPI cooling with <u>only one HPI pump operating</u>, it has been determined that the PORV must be opened for this situation.

If the RC is saturated and RCPs are unavailable, then HPI cooling should be established if primary to secondary heat transfer does not occur after establishing the SG(s) as a heat sink. If RCPs are available, then after establishing the SG(s) as a heat sink, RCP bumps should be continued until primary to secondary heat transfer is restored or a decision is made to establish HPI cooling.

The RC pressure must not exceed the RV P-T limit. Therefore, if the RC pressure increases to the RV P-T limit following a loss of primary to secondary heat transfer, the

PORV should be opened (i.e., to limit RC pressure increase) and HPI pumps started. After the pressurizer fills with RC, the HPI flow should be throttled as necessary to try and keep the RC pressure below the RV P-T limit.

The PTS limit must not be violated while it is in effect. If the PTS limit is in effect or will be in effect due to HPI initiation, then RC pressure must be controlled to prevent exceeding the PTS limit. This may require opening the PORV to reduce RC pressure below the PTS limit before starting HPI. After initiating HPI and opening the PORV, to limit pressure, throttling HPI may be necessary to prevent exceeding the PTS limit.

3.0 GEOG SECTION AND STEP REFERENCE

GEOG Section	Applicable Steps
LHT	3.2.a, 3.2.b.1 and 3.3.d.1
	15.1.a, 15.1.c.1 and 15.2.b.1
	19.0
EHT	5.2.a, 5.2.b.1 and 5.3.d.1
SGTR	10.2.c, 10.2.d.1 and 10.2.d.8.a

4.0 CUES

- Incore Thermocouple Temperature
- SCM meter and associated alarms
- SPDS displays and associated alarms
- SG level
- SG pressure
- RC temperature
- MFW flow
- EFW flow
- Verbal alert by plant staff that indications of inadequate primary to secondary heat transfer are occurring.
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of HPI pump controls
- Operation of HPI valve controls
- Operation of PORV and PORV block valve controls
- [plant specific performance indicators]

6.0 FEEDBACK

- HPI pump status indication
- HPI valves status indication
- PORV and PORV block valve status indication
- HPI flow
- Verbal notification by plant staff of HPI flow status
- Incore thermocouple temperatures
- [plant specific feedback indicators]

2.

FW FLOW CONTROL

Fulfillment of this CT requires the following:

Properly controlling EFW and MFW to mitigate excessive primary to secondary heat transfer and tripping all FW pumps supplying FW flow to affected SG(s) if any SG operating level > [SG high level]

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT for certain situations where too much FW flow exists. Such situations can cause excessive primary to secondary heat transfer and/or damage to plant equipment.

2.0 ASSOCIATED GEOG BASES

If excessive MFW flow exists, such that steam line flooding is imminent, then it is necessary to trip both of the running MFW pumps. Also, if EFW pumps are running, and steam line flooding is imminent, then they should also be tripped. If the FW overfill is not as severe then it should be adequate to close the appropriate FW valves. Maintaining SGs operable enhance the transient mitigation capability of the plant as normal heat removal systems remain available.

3.0 GEOG SECTION AND STEP REFERENCE

GEOG Section	Applicable Steps
VSSV	5.a and 5.b
	13.a

EHT 3.1 and 3.3

4.0 CUES

- [SG high level] alarm
- SPDS displays and associated alarms
- [secondary plant protection system] alarms
- Verbal alert by plant staff that FW flow rates are excessive
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of MFW/EFW pump controls
- Operation of associated MFW/EFW valve controls
- Operation of [secondary plant protection system] controls
- [plant specific performance indicators]

6.0 FEEDBACK

- RC temperature and pressure
- SG level and pressure
- Verbal notification by plant staff of FW flow rates
- [secondary plant protection system] status indication
- MFW/EFW pump status indications
- Associated MFW/EFW valve status indications
- [plant specific feedback]

3.

ISOLATE OVERCOOLING SG(s)

Fulfillment of this CT requires the following:

Operation of [secondary plant protection system] to isolate affected (overcooling) SG(s)

and/or

manual isolation of overcooling SG(s) by closing all steam and FW valves to the affected SG(s).

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT when excessive primary to secondary heat transfer occurs and mitigation requires isolation of affected SG(s).

2.0 ASSOCIATED GEOG BASES

If the overcooling SG has been identified then that SG should be isolated, otherwise both SGs should be isolated. Isolating a SG means to stop all FW flow (MFW and AFW) and steam flow (e.g., close TBVs, ADVs, steam supply to FW pumps, MSIVs etc.). FW flow should be maintained to the unaffected SG and cooling stabilized using the unaffected SG.

Isolation of a SG or both SGs should always follow a logical progression of increasingly more drastic attempts to isolate the SG. For example, if the overcooling is not severe it may be possible to close both the TBVs and ADVs as well as the auxiliary steam valves thus isolating the SG. If this does not work, then for those plants which have main steam isolation valves, the main steam isolation valve should then be closed. For severe overcooling situations, [secondary plant protection system] will likely actuate.

Failing to mitigate excessive primary to secondary heat transfer can lead to degradation of the transient mitigation capability of the plant if SGs become inoperable.

4.0 CUES

- SPDS displays and associated alarms
- [secondary plant protection system] alarms
- Verbal alert by plant staff that primary to secondary heat transfer is excessive
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of associated FW pump and valve controls
- Operation of associated steam valve (included TBVs/ADVs) controls
- Operation of [secondary plant protection system] controls
- [plant specific performance indicators]

6.0 FEEDBACK

- RC temperature and pressure
- SG level and pressure
- Verbal notification by plant staff that primary to secondary heat transfer is appropriate for given plant conditions
- [secondary plant protection system] status indication
- MF pump and valve status indications
- [plant specific feedback]

Facility: Oconee Scenario No.: 3 Op-Test No.: 1

Examiners: ______________________ Operators: ______________________

Objectives: The candidates will operate the simulator during all events described in the scenario as if it is actually Oconee Unit 1. During the exam the candidates will demonstrate appropriate licensed operator knowledge and abilities that will ensure safe operation of the facility during all aspects of operation. During the exam the candidates will use the following operating techniques to ensure safe plant operations and ensure health and safety of the general public is maintained at all times: proper procedure usage, communications, conservative decision making, reactivity management, equipment control and manipulation, and team skills.

Initial Conditions:
Unit 1 is at 25% power (EOL), Unit 2 is at 100%, Unit 3 is at 100%

Turnover:

- Operation at 25% power following a restart from a turbine/reactor trip due to a loss of Stator Coolant. Holding at 25% as Engineering is evaluating Generator stator parameters.
- OP/1/A/1102/04, Operation at Power, Enclosure 3.1, Power Escalation in progress – holding at step 2.5 (Auxiliaries remain on CT-1) until Engineering evaluation is complete.
- The TD EFDWP will be taken OOS for 8 hours during this shift. WCC will call when the TD should be placed in "Pull to Lock" (Water in the oil - changing oil)
- "1B" OTSG SGTL = 20 gpd (OP/1106/31 conditions have been evaluated)
- PCB-21 Gen Output Breaker open (low gas alarm occurred last shift – Transmission should add gas this shift).
- GWD Vent header cross-connected – Unit 1 has the GWD header

Event No.	Malf. No.	Event Type*	Event Description
1. Pre-Insert	Override	C, BOP	Block MSLB Circuitry
2. Pre-insert	MPS140	C,OATC	1 HP-26 fails as is
1	MSS330	TS/SRO	Unit 1 TD EFDW taken OOS
2		1/BOP	GWR
~~2~~ 3		N, ALL	De-Lithiation with the deborating Demineralizer
~~3~~ 4	Override	C,BOP	1HP-14 fails in the "bleed" position (IPE – PIP O-99-05270)

~~4~~5	Override FDW03	C,OATC	ICS STAR module failure (FDW)
~~5~~6		R,OATC	Unit/reactor shutdown
~~6~~7	MPI171, MPI500	I, OATC	RC T-Hot “A” (1) fails LOW (median select with MPI 500) RC T-Hot “A” (2) fails LOW
~~7~~	~~Override~~	~~I, BOP~~	~~1RIA-40 fails HIGH~~
8	MPS010	M, All	Steam Generator tube leak (OTSG “A”) (200 gpm) (CT D.1, D.2, D.3)
9	Override		RIAs fail: 1RIA-16 – Low, 1RIA-17 – High
10	MSS380	M, All	Main Steam line leak (OTSG “A”) (3%) out of Containment (CT B.2.1, B.2.3)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Op-Test No.:	01	Scenario No.:	03	Event No.:	1
Event Description: TD EFDWP taken OOS.					

Time	Position	Applicant's Actions or Behavior
	SRO	When called by the WCC to place the TD EFDWP to "Pull to Lock" Refer to TS 3.7.5.B restore the TD EFDWP to operable within 72 hours
	OATC	Places the TD EFDWP to "Pull to Lock"

Op-Test No.:	01	Scenario No.:	03	Event No.:	2

Event Description:
De-Lithiation with the deborating Demineralizer.

Time	Position	Applicant's Actions or Behavior
	BOP	**CUE:** Chemistry requests that RCS be de-lithiated for 5 minutes with the Unit 1 Deborating Demineralizer. Refer to OP/1/A/1103/004, Soluble Poison Control, Enclosure 3.31 (Step 2.7) to begin de-lithiation.
	BOP / SRO	Perform OP/1/A/1103/004, Soluble Poison Control, Enclosure 3.31 • Place Deborating IX in service: • Close 1CS-26 (Letdown to RC Bleed) • Open 1CS-27 (Debor IX Inlet) • Open 1HP-16 (LDST Makeup Isolation) • Verify 1HP-15 (LDST Makeup Control) in manual and open • Position 1HP-14 (LDST Bypass) to "BLEED" • Record letdown pressure (call NLO, answer 90 psig) • Start 5 minute timer
	BOP / SRO	Restore system per OP/1/A/1103/004, Soluble Poison Control, Enclosure 3.31 • Place 1HP-14 (LDST Bypass) in "NORMAL" • Close 1HP-16 (LDST Makeup Isolation) • Reset 1HP-15 (LDST Makeup Control) • Open 1CS-26 (Letdown to RC Bleed) • Close 1CS-27 (Debor IX Inlet) • Complete OP/1/A/1103/004, Soluble Poison Control, Enclosure 3.31

Op-Test No.:	01	Scenario No.:	03	Event No.:	3

Event Description:

1HP-14 FAILS in the "Bleed" position.
LDST level will decrease. Closing 1HP-5 can stop the "leak".

Time	Position	Applicant's Actions or Behavior
	BOP/ OATC	Diagnose that 1HP-14(LDST Bypass) has failed in the Bleed position. • OAC alarm – 1HP-14 (LDST Bypass) to BLEED • LDST level decreasing • Increased rate of decrease in the LDST • Increase in 1A BHUT level (This is not apparent as the tank volume is very large approx. 140,000 gallons) • No RB, Aux. Building, MS lines or CSAE off-gas RIAs in alarm • No increase in HAWT and LAWT levels (waste tanks) • Notify the CR SRO Note: The crew may attempt to ensure that no other leaks are in progress. All cues from the personnel outside the control room will not support other leakage at this time.
	SRO / BOP	Refer to AP/1/A/1700/002, Excessive RCS Leakage
	OATC / BOP / SRO	Perform AP/1/A/1700/002, Excessive RCS Leakage • Determine leak size (40 – 70 gpm) • Notify appropriate personnel • Monitor LDST level • Monitor PZR level and makeup flow • Close 1HP-5, (Letdown Isolation) This places the crew into AP/1700/14, Loss of letdown or Makeup.

Op-Test No.:	01	Scenario No.:	03	Event No.:	3

Event Description:

1HP-14 FAILS in the "Bleed" position.
LDST level will decrease. Closing 1HP-5 can stop the "leak".

Time	Position	Applicant's Actions or Behavior
	SRO / OATC / BOP	Refer to AP/1700/14, Loss of Letdown or Makeup, CASE B Loss of Letdown • Monitor PZR level increase and If PZR level exceeds 375 gpm then manually trip the reactor. • Duty Operations personnel will direct the crew to shutdown the unit. • Refer to OP/1102/04, Operations at Power and commence unit shutdown.

Op-Test No.: 01	Scenario No.: 03	Event No.: 4
Event Description: ICS FDW Star Module failure.		

Time	Position	Applicant's Actions or Behavior
	OATC	1SA-2/A12 ICS Track statalarm (Both FDW Masters in Manual) OAC alarms for Both FDW Masters and ΔTc in manual
	OATC / SRO	Continue unit shutdown with FDW in manual • Determine the cause of the Unit being in Track • Monitor plant parameters and control plant power decrease

Op-Test No.: 01	Scenario No.: 03	Event No.: 5
Event Description:		
Unit/reactor shutdown.		

Time	Position	Applicant's Actions or Behavior
	OATC / BOP / SRO	Perform OP/1/A/1102/004, Enclosure 3.2, Power Reduction with ICS FDW in Manual • Refer to OP/1/A/1106/001, Turbine Generator • Refer to OP/1/A/1106/014, Moisture Separator Reheater • Notify SOC (System Operations Center) • Ensure FDWP seal injection pumps are operating • Begin power reduction

Op-Test No.:	01	Scenario No.:	03	Event No.:	6

Event Description:
RC T-Hot "A" (1) FAILS LOW (median select with MPI 500)
RC T-Hot "A" (2) FAILS LOW
Control Rods begin to withdraw, Tave and PZR level increase. Crew should take the Diamond, FDW Masters (already in manual) to hand and adjust as required to stabilize the plant.

Time	Position	Applicant's Actions or Behavior
	OATC	Statalarm 1SA-2 / A12, ICS Tracking Statalarm 1SA-2 / B4, RC Average Temp Hi/Low • Refer to Alarm response guides
	OATC	Statalarm 1SA-2B4, RC Average Temp Hi/Low • Diagnose failed instrument and ICS is causing rod withdrawal • Place Reactor control station (Diamond) in Hand and insert CRDs and stabilize the unit • Compare Loop "A" Tave with Loop "B" Tave and use "B" Loop Tave for correct indication.

Op-Test No.:	01	Scenario No.:	03	Event No.:	7
Event Description: 1RIA-40 (CSAE Off-Gas) fails high					

Time	Position	Applicant's Actions or Behavior
	BOP	Statalarm 1SA8 / B9 RM Process Monitor High
	BOP / SRO	Monitor 1RIA-40 indication (1E+8) • Determine that the monitor has failed high and declare it OOS • Determine that a SGTL has not occurred

Op-Test No.: 01 Scenario No.: 03 Event No.: 8/9

Event Description:

Steam Generator tube LEAK OTSG "A" (200 gpm)
RIAs FAIL. 1RIA-16 – LOW ("as-is") and 1RIA –17 – HIGH;
PZR level will decrease. Crew should attempt to control PZR level with 1HP-26. 1HP-26 will not operate and the crew should use 1HP-410 to control PZR level.

Two paths are possible here:

- **FIRST**: The team does not diagnose the leak as a SG tube leak.
- In this case they should determine that an RCS leak greater than TS is occurring.
- The team should refer to AP/1/A/1700/002, Excessive RCS Leakage, and begin a unit shutdown.
- **SECOND:** Crew diagnose a tube leak:
- Loss of RCS inventory with no other indications i.e. leak in RB or Aux building.
- The Crew should request RP and Chemistry assistance in identifying leak.
- After the tube leak is identified as being greater than the TS limit the EOP should be entered (section 504) and power should be decreased in manual.

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	Statalarm 1SA-2/C3, RC PZR Level Hi/Low
	OATC / BOP	Perform Statalarm 1SA-2/C3, RC PZR Level Hi/Low • Check alternate PZR level indications • Check for proper makeup / letdown flow (letdown is isolated in event 2) • Refer to AP/1/A/1700/002, Excessive RCS Leakage

Op-Test No.:	01	Scenario No.:	03	Event No.:	8/9

Event Description:

Steam Generator tube LEAK OTSG "A" (200 gpm)
RIAs FAIL. 1RIA-16 – LOW ("as-is") and 1RIA –17 – HIGH;
PZR level will decrease. Crew should attempt to control PZR level with 1HP-26. 1HP-26 will not operate and the crew should use 1HP-410 to control PZR level.

Two paths are possible here:

- **FIRST**: The team does not diagnose the leak as a SG tube leak.
- In this case they should determine that an RCS leak greater than TS is occurring.
- The team should refer to AP/1/A/1700/002, Excessive RCS Leakage, and begin a unit shutdown.
- **SECOND:** Crew diagnose a tube leak:
- Loss of RCS inventory with no other indications i.e. leak in RB or Aux building.
- The Crew should request RP and Chemistry assistance in identifying leak.
- After the tube leak is identified as being greater than the TS limit the EOP should be entered (section 504) and power should be decreased in manual.

Time	Position	Applicant's Actions or Behavior
	OATC / BOP / SRO	Perform AP/1/A/1700/002, Excessive RCS Leakage • Determine leak size • Refer to ITS • Diagnose leak size exceeds ITS requirements • Notify personnel • Monitor LDST level • Monitor PZR RV flow detectors • Monitor PZR level and makeup flow • If 1HP-120 is full open, manually trip the reactor • Monitor for CC system • Request RP and Chemistry assistance in identifying leak.

Op-Test No.:	01	Scenario No.:	03	Event No.:	8/9

Event Description:

Steam Generator tube LEAK OTSG "A" (200 gpm)
RIAs FAIL. 1RIA-16 – LOW ("as-is") and 1RIA –17 – HIGH;
PZR level will decrease. Crew should attempt to control PZR level with 1HP-26. 1HP-26 will not operate and the crew should use 1HP-410 to control PZR level.

Two paths are possible here:

- **FIRST**: The team does not diagnose the leak as a SG tube leak.
- In this case they should determine that an RCS leak greater than TS is occurring.
- The team should refer to AP/1/A/1700/002, Excessive RCS Leakage, and begin a unit shutdown.
- **SECOND:** Crew diagnose a tube leak:
- Loss of RCS inventory with no other indications i.e. leak in RB or Aux building.
- The Crew should request RP and Chemistry assistance in identifying leak.
- After the tube leak is identified as being greater than the TS limit the EOP should be entered (section 504) and power should be decreased in manual.

Time	Position	Applicant's Actions or Behavior
	SRO / OATC / BOP	EP/1/A/1800/01, Emergency Operating Procedure, Immediate Actions • All Power Range NI's < 5% and decreasing • Transfer to Section 504, SG Tube Leak

Op-Test No.:	01	Scenario No.:	03	Event No.:	8/9

Event Description:

Steam Generator tube LEAK OTSG "A" (200 gpm)
RIAs FAIL. 1RIA-16 – LOW ("as-is") and 1RIA –17 – HIGH;
PZR level will decrease. Crew should attempt to control PZR level with 1HP-26. 1HP-26 will not operate and the crew should use 1HP-410 to control PZR level.

Two paths are possible here:

- **FIRST**: The team does not diagnose the leak as a SG tube leak.
- In this case they should determine that an RCS leak greater than TS is occurring.
- The team should refer to AP/1/A/1700/002, Excessive RCS Leakage, and begin a unit shutdown.
- **SECOND:** Crew diagnose a tube leak:
- Loss of RCS inventory with no other indications i.e. leak in RB or Aux building.
- The Crew should request RP and Chemistry assistance in identifying leak.
- After the tube leak is identified as being greater than the TS limit the EOP should be entered (section 504) and power should be decreased in manual.

Time	Position	Applicant's Actions or Behavior
	SRO / OATC / BOP	Calculate SGTL to be > than TS (150 gpd) and enter the EOP Section 504, SGTL • Maintain PZR level via IT #4, SGTL PZR Level Control • Identify the SG with the tube leak (verified via RP sampling the MS lines and/or FDW mismatch since RIA are failed) • Start the Outside Booster Fans • Evaluate for off-site release – control TBS operation • Estimate the leak size • Continue the unit shutdown (Do not trip the reactor unless needed to prevent opening the MSRV) • Depressurize the RCS to lower SCM • Cooldown the RCS to 532°F • Prepare to isolate the "A" SG

Op-Test No.: 01	Scenario No.: 03	Event No.: 8/9

Event Description:

Steam Generator tube LEAK OTSG "A" (200 gpm)
RIAs FAIL. 1RIA-16 – LOW ("as-is") and 1RIA –17 – HIGH;
PZR level will decrease. Crew should attempt to control PZR level with 1HP-26. 1HP-26 will not operate and the crew should use 1HP-410 to control PZR level.

Two paths are possible here:

- **FIRST**: The team does not diagnose the leak as a SG tube leak.
- In this case they should determine that an RCS leak greater than TS is occurring.
- The team should refer to AP/1/A/1700/002, Excessive RCS Leakage, and begin a unit shutdown.
- **SECOND:** Crew diagnose a tube leak:
- Loss of RCS inventory with no other indications i.e. leak in RB or Aux building.
- The Crew should request RP and Chemistry assistance in identifying leak.
- After the tube leak is identified as being greater than the TS limit the EOP should be entered (section 504) and power should be decreased in manual.

Time	Position	Applicant's Actions or Behavior
	SRO	Refer to ITS 3.4.13, RCS Operational Leakage • Diagnose excessive RCS leakage (SGTL) requires unit shutdown • Direct unit shutdown as required.

Op-Test No.:	01	Scenario No.:	03	Event No.:	10

Event Description:

Main Steam line BREAK (OTSG "A") outside of containment.

The team should diagnose the leak and isolate the 1A OTSG. The "A" OTSG will be being "fed" from the RCS via the tube leak. This will provide a direct release path to the environment.

Time	Position	Applicant's Actions or Behavior
	SRO / OATC / BOP	Diagnose Main Steam line break Parallel Actions transfer in the EOP to Section 503, Excessive Heat Transfer
	SRO / OATC / BOP	Perform Section 503, Excessive Heat Transfer • Refer to Rule #6, Main Steam Line Break Actions • Isolate the affected SG • Check PZR level • Ensure HPI operating • If SG level(s) > 96% trip both FDWPT's • If 1A SG isolated throttle TBV's to control cooldown. • Throttle HPI header flow to control subcooling margin
	SRO / OATC / BOP	Perform Rule #6, Main Steam Line Break Actions • Secure MDEFDWP's feeding the 1A SG • Initiate both trains of MSLB Isolation Circuit • Ensure both FDWPT's tripped • Close EFDW to 1A SG • Close main and SU FDW block valves • Throttle HPI flow if subcooling $\geq 5°$ F • Adjust TBV's to maintain CETC's constant

Examination # 3 Overview:

Initial Conditions:
Unit 1 is at 25% power (EOL), Unit 2 is at 100%, Unit 3 is at 100%
Turnover:

- The Turbine Driven EFW pump (TDEFDWP) will be taken OOS for 8 hours during this shift. WCC will call when the TD should be placed in "Pull to Lock" (Water in the oil - changing oil)
- "1B" OTSG SGTL = 20 gpd (OP/1106/31 conditions has been evaluated)
- PCB-21 Gen Output Breaker open (low gas alarm occurred last shift – Transmission should add gas this shift).

Events

1. WCC calls the CR and has the crew place the TD EFDW in the "Pull to Lock: positions. This takes the TD OOS (ITS 3.7.5 – 72 hour/10 days LCO) (SRO/TS)

2. De-Lithiation with the deborating Demineralizer: (ALL/N)

   Chemistry requests that the RCS be de-lithiated for 5 minutes with the Unit 1 Deborating Demineralizer. The BOP should use Enclosure 3.31 of OP/1103/04, Soluble Poison Control, to begin de-lithiation.

When the deborating demineralizer is place in service, run for 5minute, then returned to normal this event is complete.

   TIME = 5-10 minutes

3. 1HP-14 fails to the "bleed" position: (BOP/C)

   After 1HP-14 is returned to normal it fails to the "BLEED" position. (IPE Unit 3 event when 3HP-14 failed to the BLEED position. (PIP #99-05270)

   LDST level will decrease. Closing 1HP-5 (Letdown Isolation) will stop the "leak". When HP-5 is closed the BOP will refer to AP/1700/14, Case B Loss of Letdown. PZR level will increase requiring the unit to be shutdown.

After 1HP-14 failure has been discovered and actions taken to shut down, this event is complete.

   TIME= 5-10 minutes TOTAL 10-20 min.

4. ICS STAR MODULE fails (OATC/R)

   ICS Star Module fails placing the FDW Masters and the Delta Tc controller to HAND.

This will require the OATC to decrease power with the ICS FDW in MANUAL.

NOTE: This reactivity change is required here because the reactor may trip on the Th failure and a reactivity change will not occur.

When the crew diagnoses a star module failure and continues the shutdown this event is complete

TIME = 5 minutes TOTAL = 15-20 minutes

5. Unit shutdown per OP/1102/04 Operation at Power with the ICS FDW in manual

6. Thot fails low: (OATC/I)

   Statalarm 1SA-2/A12 (ICS Tracking)
   Statalarm 1SA-2/B4 (RC Average Temp Hi/Low)

   Control Rods begin to withdraw. Tave and PZR level increase. OATC should take the Diamond to Manual and stablize Tave. The FDW Masters are already in hand and may be adjusted as required to stabilize the plant

   When crew has stabilized the unit, this event is completed
   TIME = 10 minutes TOTAL 20 min.

7. RIA-40 fails high

   Statalarm 1SA8 / B9 RM Process Monitor High
   Monitor 1RIA-40 indication (1E+8)
   Determine that the monitor has failed high and declare it OOS
   Determine that a SGTL has not occurred

When the crew determines that RIA-40 has failed, this event is completed

TIME = 10 minutes TOTAL 30 min.

8. OTSG tube leak (≈ 200gpm) with faulted RIAs (ALL/M) and 1HP-26 failed as is (OATC/C):

Statalarm 1SA-2/C3, RC PZR Level Hi/Low

PZR level will decrease. OATC should attempt to control PZR level with 1HP-26. 1HP-26 will not operate and the crew should use 1HP-410 to control PZR level.

Two paths are possible here:
First: Crew diagnose a tube leak:
Loss of RCS inventory with no other indications i.e. leak in RB or Aux building. Should request RP and Chemisrty assistance in identifing leak.
After the tube leak is identified as being greater than the TS limit the EOP should be entered (section 504) and power should be decreased in manual.

Second: The team does not diagnose the leak as a tube leak. In this case they should determine that an RCS leak greater than TS is occuring. The team should refer to AP/1/A/1700/002, Excessive RCS Leakage, and begin a unit shutdown.

When power is decreased~5% in manual this event is completed
TIME = 15 minutes TOTAL 50 min.

9. "A" Main Steam line leak (ALL/M)

When the Procedure Director reaches step 23 in Section 504 a Main Steam line leak will occur in the "A" Main Steam line. The team should diagnose the leak and isolate the 1A OTSG. The "A" OTSG will be being "fed" from the RCS via the tube leak. This will provide a direct release path to the environment.

When the crew has determined to use the "B" OTSG to cool down or at the Examiners request the event and the exam is completed

TIME = 15-20 minutes TOTAL 75-85 min.

3.

ISOLATE OVERCOOLING SG(s)

Fulfillment of this CT requires the following:

Operation of [secondary plant protection system] to isolate affected (overcooling) SG(s)

and/or

manual isolation of overcooling SG(s) by closing all steam and FW valves to the affected SG(s).

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT when excessive primary to secondary heat transfer occurs and mitigation requires isolation of affected SG(s).

2.0 ASSOCIATED GEOG BASES

If the overcooling SG has been identified then that SG should be isolated, otherwise both SGs should be isolated. Isolating a SG means to stop all FW flow (MFW and AFW) and steam flow (e.g., close TBVs, ADVs, steam supply to FW pumps, MSIVs etc.). FW flow should be maintained to the unaffected SG and cooling stabilized using the unaffected SG.

Isolation of a SG or both SGs should always follow a logical progression of increasingly more drastic attempts to isolate the SG. For example, if the overcooling is not severe it may be possible to close both the TBVs and ADVs as well as the auxiliary steam valves thus isolating the SG. If this does not work, then for those plants which have main steam isolation valves, the main steam isolation valve should then be closed. For severe overcooling situations, [secondary plant protection system] will likely actuate.

Failing to mitigate excessive primary to secondary heat transfer can lead to degradation of the transient mitigation capability of the plant if SGs become inoperable.

4.0 CUES

- SPDS displays and associated alarms
- [secondary plant protection system] alarms
- Verbal alert by plant staff that primary to secondary heat transfer is excessive
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of associated FW pump and valve controls
- Operation of associated steam valve (included TBVs/ADVs) controls
- Operation of [secondary plant protection system] controls
- [plant specific performance indicators]

6.0 FEEDBACK

- RC temperature and pressure
- SG level and pressure
- Verbal notification by plant staff that primary to secondary heat transfer is appropriate for given plant conditions
- [secondary plant protection system] status indication
- MF pump and valve status indications
- [plant specific feedback]

2.

MINIMIZE SCM

Fulfillment of this CT requires the following:

RCS TEMPERATURE > 500°F:

Depressurize RCS as much a possible without violating the SCM curve

RCS TEMPERATURE ≤ 500°F:

Maintain RC pressure as low as possible without violating the SCM or RCP NPSH curves

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT when a SGTR occurs.

2.0 ASSOCIATED GEOG BASES

Except when RCP NPSH limits are applicable and are more restrictive, RCS pressure should be maintained close to, but above, the minimum SCM to minimize RCS-SG differential pressure. The reason for minimizing RCS-SG differential pressure is to reduce the leak flowrate from primary to secondary to as low as possible. Therefore, this procedure (minimizing SCM) is desirable whenever possible during a cooldown with a SGTR.

Reducing the leak flowrate from the RCS to the secondary side of a SG with an impaired steam system (e.g., weeping MSSV, MSL leak, etc.) is expected to lead to lower integrated radiation releases from the impaired system. Also, if the level of the leaking SG can be maintained within normal operating limits, then the SG will

remain available for continued use during the cooldown, thus enhancing the transient mitigation capability of the plant.

3.0 GEOG SECTION AND STEP REFERENCE

GEOG Section	Applicable Steps
SGTR	7.6
	8.0

4.0 CUES

- SCM meter and associated alarms
- SPDS displays and associated alarms
- Verbal alert by plant staff that SCM is excessive
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of MU/HPI pump and valve controls
- Operation of normal or auxiliary spray valve controls
- operation of PORV and/or pressurizer vent valve controls
- [plant specific performance indicators]

6.0 FEEDBACK

- SCM meter and/or plant SPDS
- RCS pressure and temperature
- MU/HPI pump and valve status indications
- Normal and auxiliary spray valve status indications
- PORV and pressurizer vent valve status indications
- Verbal indication by plant staff that SCM is being controlled at a minimum value
- [plant specific feedback]

3.

REDUCE STEAMING/ISOLATE AFFECTED SG (USE SG DRAINS IF AVAILABLE)

Fulfillment of this CT requires the following:

If only the radiation TRACC limit is approached, then open SG drains, if available, and reduce steaming of the affected SG to minimum required to prevent SG level > [plant specific SG level].

If BWST TRACC limit is approached or a combination of TRACC limits exists or alternate actions will not prevent TRACC, then the affected SG(s) should be isolated by:

> Closing [plant specific list of valves] FW valves and all flow paths from the affected SG except for TBVs/ADVs and SG drains, if available. Use drains, if available, to maintain SG level < [high SG level that prevents water entering steam annulus] while RC pressure > 1000 psig.
>
> Maintain RCS pressure < 1000 psig by using pressurizer spray, pressurizer and high point vents, PORV, letdown and, if necessary, TBVs/ADVs. When RCS pressure is maintained < 1000 psig, then fully isolate the affected SG by closing remaining isolation valves.

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT when TRACC limits are approached.

2.0 ASSOCIATED GEOG BASES

When the radiation TRACC limit is approached, reducing the steaming rate will reduce the radiation release rate from the affected SG(s). Use of drains, if available, will allow for reducing steaming rates by removing inventory by other means than

steaming. Actions to use drains (in combination with SG pressure control) are intended to prevent lifting of MSSVs (and reduce the possibility of MSSVs failing to reseat or sticking open). Reducing the steaming rate and preventing MSSV lifts/failures is expected to reduce radiation releases.

If the SG drains are unavailable, are not used, or are otherwise incapable of preventing violation of a TRACC, isolate the affected, or most affected, SG by closing all steam, feed, and drain lines to that SG. This is expected to reduce radiation releases due to steaming.

3.0 GEOG SECTION AND STEP REFERENCE

GEOG Section	Applicable Steps
SGTR	3.3
	7.2
	9.1, 9.3.a-e
	10.3, 10.4, 10.5 and 10.6

4.0 CUES

- Alert by plant staff that integrated radiation releases are approaching [plant specific limit]
- [SG high level] alarm
- Alert by plant staff that the BWST is approaching [plant specific low level]
- SPDS displays and associated alarms
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of SG drain valve controls
- Operation of affected SG(s) steam and FW isolation valve controls
- Operation of TBV/ADV controls
- Operation of pressurizer spray and vent valve controls
- Operation of PORV controls
- Operation of high point vent controls
- Operation of letdown valve controls
- [plant specific performance indicators]

6.0 FEEDBACK

- SG(s) level and pressure
- RCS pressure
- MFW/EFW flow
- MFW/EFW pump and valve status indication
- Affected SG(s) steam valve status indication
- TBV/ADV status indication
- SG drain valve status indication
- Pressurizer spray and vent valve status indications
- PORV and high point vent valve status indication
- Letdown valve status indication
- Verbal indication from plant staff of affected SG(s) steaming rate
- [plant specific feedback]

(R06-97) JMB/WHC

Duke Power Company \RNS (1) ID No AP/1/A/1700/002

PROCEDURE PROCESS RECORD

Revision No 003

LAN Location: SAROS

REPARATION

2) Station OCONEE NUCLEAR STATION

(3) Procedure Title Excessive RCS Leakage

(4) Prepared By ERIC LAMPE Date 2/11/99

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)

☐ No (Revision with minor changes)

☐ No (To incorporate previously approved changes)

(6) Reviewed By Dennis Sordan (QR) Date 2-16-99

Cross-Disciplinary Review By ______ (QR)NA ______ Date ______

Reactivity Mgmt. Review By Dennis Sordan (QR)NA ______ Date 2-16-99

(7) Additional Reviews

Reviewed By ______ Date ______

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

9) Approved By ______ Date 3/4/99

PERFORMANCE ***(Compare with control copy every 14 calendar days while work is being performed.)***

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?

☐ Yes ☐ NA Listed enclosures attached?

☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?

☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?

☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

14) Remarks *(Attach additional pages, if necessary)*

Duke Power Company Oconee Nuclear Station **Excessive RCS Leakage**	Procedure No. AP/1/A/1700/002 Revision No. 003
Continuous Use **Reactivity Management Related**	Electronic Reference No. OX002RGF

Excessive RCS Leakage

Reactivity Management Related

Table Of Contents

OCONEE NUCLEAR STATION

Excessive RCS Leakage

1. Purpose

This procedure provides the actions to be taken to maintain the plant in a safe condition in the event of a RCS leak greater than the ITS limit but within the capacity of HPI Normal Makeup. This procedure should **NOT** be used if it is determined, that the cause of RCS leakage is due entirely to OTSG leakage.

2. Symptoms

- "RM PROCESS MONITOR RADIATION HIGH" Statalarm (1SA-08/B-9)
- "RM REACTOR BLDG NORMAL SUMP ISOLATE" Statalarm (1SA-08/E-9)
- RCS Tave constant and PZR level decreasing
- RCS Tave constant and LDST level decreasing more than normal
- RCS leakage calculation indicates a RCS leak.
- "CC COMP COOLING SURGE TANK LEVEL HIGH/LOW" Statalarm (1SA-09/D-1) with 1RIA-50 in alarm.

3. Automatic Systems Actions

- 1HP-120 (RC VOLUME CONTROL) opens to maintain PZR level at 220".

Excessive RCS Leakage

4. Immediate Manual Actions

_____ 4.1 Determine leak size:

_____ 4.1.1 **REFER TO** PT/1/A/0600/010 (Reactor Coolant Leakage).

NOTE 4.1.2:	This formula may not detect a leak on the Makeup or Letdown line beyond the flow transmitter and outside the reactor building.

_____ 4.1.2 The following formula may be used to estimate leakage:

Leak Size = (RC MAKEUP FLOW) + (SEAL INLET HDR FLOW)

- (LETDOWN FLOW) - (TOTAL SEAL LEAKOFF)

_____ 4.1.3 A leak on the Makeup or Letdown line beyond the flow transmitter and outside the reactor building can be identified by monitoring the LDST and/or HAWT level using one of the following formulas:

$$\text{Leak Size} = \frac{\text{(Change in LDST Level) x (31 Gal/inch)}}{\text{Time (minutes)}}$$

$$\text{Leak Size} = \frac{\text{(Change in HAWT Level) x (18 Gal/inch)}}{\text{Time (minutes)}}$$

_____ 4.1.4 The size of a leak inside the reactor building may be estimated using the following formula:

$$\text{Leak Size} = \frac{\text{(Change in RBNS LEVEL) x (15 Gal/inch)}}{\text{Time (minutes)}}$$

Excessive RCS Leakage

5. Subsequent Actions

_____ 5.1 Based on leak size, take appropriate action:

- **REFER TO** ITS
- **REFER TO** OP/0/B/1106/033 (Primary System Leak Identification).

_____ 5.2 Notify appropriate personnel:

- Unit Supervisor
- Area Dispatcher
- Radiation Protection
- **REFER TO** RP/0/B/1000/001 (Emergency Classification).

_____ 5.3 Monitor LDST level:

- **IF** AT ANY TIME LDST level < 40",

  THEN open 1HP-24 (1A HPI BWST SUCTION).

_____ 5.4 Monitor PZR RV Flow Detectors:

- **IF** flow is indicated through 1RC-66 (PORV),

  THEN close 1RC-4 (PZR RELIEF BLOCK).

Excessive RCS Leakage

_____ 5.5 Monitor PZR level and makeup flow:

_____ 5.5.1 **IF** AT ANY TIME RC makeup flow > 100 gpm,

AND PZR level is decreasing,

THEN close 1HP-5 (LETDOWN ISOLATION).

CAUTION 5.5.2:	The Reactor should **NOT** be tripped if EP/1/A/1800/001 (Emergency Operating Procedure) is in concurrent use due to an OTSG tube leak.

_____ 5.5.2 **IF** AT ANY TIME PZR level continues to decrease with 1HP-120 (RC VOLUME CONTROL) fully open,

THEN manually trip the Reactor,

and

REFER TO EP/1/A/1800/001 (Emergency Operating Procedure).

Excessive RCS Leakage

_____ 5.6 Monitor the CC System for in-leakage.

_____ 5.6.1 **IF** the CC Surge Tank level is increasing,

AND 1RIA-50 is in alarm,

THEN isolate the affected component:

_____ 5.6.1.1 **IF** the 1A Letdown Cooler is in service,

AND the 1B Letdown Cooler is available,

THEN swap letdown coolers:

_____ A. Start the Standby CC Pump

_____ B. Verify open 1HP-4 (1B LETDOWN COOLER OUTLET)

_____ C. Open 1CC-2/1HP-2 (1B LETDOWN COOLER INLET)

_____ D. Close 1CC-1/1HP-1 (1A LETDOWN COOLER INLET)

_____ E. Close 1HP-3 (1A LETDOWN COOLER OUTLET)

_____ F. Stop the Standby CC Pump and return switch to "AUTO".

Excessive RCS Leakage

_____ 5.6.1.2 **IF** the 1B Letdown Cooler is in service,

AND the 1A Letdown Cooler is available,

THEN swap letdown coolers:

_____ A. Start the Standby CC Pump

_____ B. Verify open 1HP-3 (1A LETDOWN COOLER OUTLET)

_____ C. Open 1CC-1/1HP-1 (1A LETDOWN COOLER INLET)

_____ D. Close 1CC-2/1HP-2 (1B LETDOWN COOLER INLET)

_____ E. Close 1HP-4 (1B LETDOWN COOLER OUTLET)

_____ F. Stop the Standby CC Pump and return switch to "AUTO".

Excessive RCS Leakage

NOTE 5.6.1.3:	RCP Cooler Outlet Temperatures are available on the OAC with the following computer points: • O1A0044 RCP 1A1 COOLER CC OUTLET TEMP • O1A0045 RCP 1A2 COOLER CC OUTLET TEMP • O1A0046 RCP 1B1 COOLER CC OUTLET TEMP • O1A0047 RCP 1B2 COOLER CC OUTLET TEMP

_____ 5.6.1.3 **IF** any RCP Cooler Outlet temp is abnormally high,

THEN close the respective RCP Seal Cooler Outlet Valve:

_____ 1CC-5 (1A1 RCP SEAL CLR OUTLET)

_____ 1CC-6 (1A2 RCP SEAL CLR OUTLET)

_____ 1CC-3 (1B1 RCP SEAL CLR OUTLET)

_____ 1CC-4 (1B2 RCP SEAL CLR OUTLET).

- **REFER TO** AP/1/A/1700/016 (Abnormal Reactor Coolant Pump Operation) for continued operation of the affected RCPs.

_____ 5.6.1.4 **IF** RCS leakage into the CC System cannot be isolated,

OR the CC Surge Tank level is off-scale high,

THEN close the following valves:

_____ 1CC-7 (CC RETURN PENT INSIDE BLOCK)

_____ 1CC-8 (CC RETURN PENT OUTSIDE BLOCK)

AND **REFER TO** AP/1/A/1700/020 (Loss Of Component Cooling).

Excessive RCS Leakage

_____ 5.7 Close the following RB isolation valves:

ES Channel 1Valves:

_____ 1GWD-12 (QUENCH TANK VENT (INSIDE RB))

_____ 1LWD-1 (RB NORMAL SUMP ISOLATION)

_____ 1CS-5 (COMPONENT DRN PUMP SUCTION)

_____ 1PR-1 (RB PURGE OUTLET)

_____ 1PR-6 (RB PURGE INLET)

_____ 1RC-5 (PZR STEAM SAMPLE)

_____ 1RC-6 (PZR WATER SAMPLE)

_____ 1FDW-105 (1A SG SAMPLE ISOLATION)

_____ 1FDW-107 (1B SG SAMPLE ISOLATION)

ES Channel 2 Valves:

_____ 1GWD-13 (QUENCH TANK VENT (OUTSIDE RB))

_____ 1LWD-2 (RB NORMAL SUMP ISOLATION)

_____ 1CS-6 (COMPONENT DRN PUMP SUCTION)

_____ 1PR-2 (RB PURGE OUTLET)

_____ 1PR-3 (RB PURGE CONTROL)

_____ 1PR-4 (RB PURGE INLET)

_____ 1PR-5 (RB PURGE INLET)

(Continued)

Excessive RCS Leakage

ES Channel 2 Valves: (Continued)

_____ 1RC-7 (PZR SAMPLE)

_____ 1FDW-106 (1A SG SAMPLE ISOLATION)

_____ 1FDW-108 (1B SG SAMPLE ISOLATION)

_____ 1FDW-103 (1A SG SHELL DRAIN BLOCK)

_____ 1FDW-104 (1B SG SHELL DRAIN BLOCK).

_____ 5.8 **IF** 1HP-5 (LETDOWN ISOLATION) is closed,

THEN close the following:

_____ 1CC-1/1HP-1 (1A LETDOWN COOLER INLET)

_____ 1CC-2/1HP-2 (1B LETDOWN COOLER INLET)

_____ 1HP-3 (1A LETDOWN COOLER OUTLET)

_____ 1HP-4 (1B LETDOWN COOLER OUTLET)

Excessive RCS Leakage

_____ 5.9 **IF** Reactor Coolant Pumps are NOT operating, and RCS pressure < 550 psig,

THEN close the following RB isolation valves:

ES Channel 1 Valves:

_____ 1HP-20 (RCP SEAL RETURN)

ES Channel 2 Valves:

_____ 1HP-21 (RCP SEAL RETURN BLOCK)

ES Channel 5 Valves:

_____ 1CC-7 (CC RETURN PENETRATION INSIDE BLOCK)

_____ 1LPSW-6 (UNIT 1 RCP MOTOR COOLERS INLET)

_____ 1LPSW-15 (UNIT 1 RCP MOTOR COOLERS OUTLET)

ES Channel 6 Valves:

_____ 1CC-8 (CC RETURN PENETRATION OUTSIDE BLOCK)

_____ 1LPSW-6 (UNIT 1 RCP MOTOR COOLERS INLET)

_____ LPSW-15 (UNIT 1 RCP MOTOR COOLERS OUTLET).

_____ 5.10 Continue with unit shutdown and cooldown:

- **REFER TO** OP/1/A/1102/004 (Operation At Power)
- **REFER TO** OP/1/A/1102/010 (Controlling Procedure For Unit Shutdown).

_____ 5.10.1 Ensure adequate shutdown margin during cooldown:

- **REFER TO** OP/1/A/1103/004 (Soluble Poison Concentration Control).

END

Excessive RCS Leakage
Appendix

1. None

END

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 The EFW System shall be OPERABLE as follows:

a. Three EFW pumps shall be OPERABLE, and

b. Two EFW flow paths shall be OPERABLE.

---NOTE---
Only one motor driven emergency feedwater (MDEFW) pump and one EFW flow path are required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MDEFW pump inoperable in MODE 1, 2, or 3.	A.1 Restore MDEFW pump to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. Turbine driven EFW pump inoperable in MODE 1, 2, or 3. <u>OR</u> One EFW flow path inoperable in MODE 1, 2, or 3.	B.1 Restore turbine driven EFW pump and EFW flow path to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two MDEFW pumps inoperable in MODE 1, 2, or 3.	C.1 Restore one MDEFW pump to OPERABLE status.	12 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met. <u>OR</u> Turbine driven EFW pump and one EFW flow path inoperable in MODE 1, 2, or 3.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	12 hours 24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Three EFW pumps inoperable in MODE 1, 2, or 3. <u>OR</u> Two EFW flow path inoperable in MODE 1, 2, or 3.	E.1	----------NOTE------------ LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW pump and one EFW flow path are restored to OPERABLE status. ------------------------------ Initiate action to restore one EFW pump and one EFW flow path to OPERABLE status.	Immediately
F. Required MDEFW pump inoperable in MODE 4. <u>OR</u> Required EFW flow path inoperable in MODE 4.	F.1	Initiate action to restore required MDEFW pump and required EFW flow path to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, and non-automatic power operated valve in each water flow path and in the steam supply flow path to the turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.3	-------------------------NOTE------------------------- Not required to be met in MODES 3 and 4. -- Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.5.4	-------------------------NOTE------------------------- Not required to be met in MODES 3 and 4. -- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying valve alignment from the upper surge tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5 or 6 for > 30 days

(R06-97)

OK
Sim
NRC
106
115
JPP
JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No OP/1/A/1102/004

Revision No 085

EPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title OPERATION AT POWER

(4) Prepared By Ridings, R. (Signature) [signature] Date 2/17/00

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)

☐ No (Revision with minor changes)

☐ No (To incorporate previously approved changes)

(6) Reviewed By [signature] (QR) Date 2/17/00

Cross-Disciplinary Review By (I&E) Doug Repper (QR)NA____ Date 2-21-00

Reactivity Mgmt. Review By [signature] (QR)NA____ Date 2/17/00

(7) Additional Reviews

Reviewed By (Rx Eng) JE Sanders Date 2/21/00

Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*

By ______ (SRO/QR) Date ______

By ______ (QR) Date ______

(9) Approved By [signature] Date 2/21/00

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______

Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification:

☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?

☐ Yes ☐ NA Listed enclosures attached?

☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?

☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?

☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

Received
Operator
Training

Operation At Power

1. Purpose

This procedure provides the following:

- Procedural guidance for operation of Unit 1 to escalate and reduce Rx power with the Turbine Generator on line.
- Maneuvering restrictions for Rx power ramp rates and CRD/APSR movement rates.
- Special instructions for power operation with < four RCPs.

2. Limits and Precautions

2.1 Use of this procedure will affect core reactivity.

2.2 Unit 1 should be operated within the limits of Technical Specifications (TS) at all times.

- When operating under LCO 3.0.3 of TS, shutdown rate should be determined by Operations such that the required actions can be achieved in a controlled manner.
- IF conditions require a faster shutdown rate up to and including a Rx Trip, refer to OMP 1-4 (Actions To Be Taken In The Case Of Exceeding Limits).

2.3 Reactor Power maneuvering restrictions are found in PT/0/A/1103/020 (Power Maneuvering Guidelines).

2.4 Power Imbalance should be maintained per PT/1/A/600/001 (Periodic Instrument Surveillance).

2.5 IF reactor power will be increased ≥ 10%,

OR

IF reactor power will be reduced ≥ 40% with reactor remaining critical, Reactor Engineering should be notified to develop Maneuvering Plan.

- When possible, notification should occur at least 24 hours prior to planned changes in power. {6}

2.6 IF an unscheduled power reduction occurs, an investigation should be conducted and corrective action taken prior to power level increase.

2.7 CRD group positions should be maintained at same level to minimize power tilts.

2.8 Condenser ΔT shall be ≤ 22°F with CCW temperature > 68°F.

2.9 CCW discharge effluent temperature shall **NOT** exceed 100°F for more than two hours.

NOTE:	Non-conservative NIs is Core Thermal Power (CTP) > NIs.

2.10 **IF** any two of four power range NIs > 2% non-conservative, calibration is required to prevent exceeding safety limits.

- Prior to initiating power changes > 5% CTP or CRD changes > 15% Rod Index, NI calibration should be checked.
- 15 minutes after reaching steady conditions, NI calibration should be checked.
- In no case, should ≥ 4% in the non-conservative direction be exceeded.
- **IF** any two of the four NIs are **NOT** within ±2% of CTP during power level increases, CTP increase should be stopped and NIs calibrated. {10}

2.11 Pressurizer Heaters should be in "AUTO" during system transients.

2.12 Pressurizer spray should be in "AUTO" during system transients.

- 1RC-1 should be in AUTO and 1RC-3 should be open.
- Anytime 1RC-1 is **NOT** in AUTO and 1RC-3 is throttled, an R&R should be completed per OMP 2-18 (Tagout Removal and Restoration Procedure).

2.13 **IF** Pressurizer level decreases to 200" with Rx > 15% CTP immediate action should be taken to return Pressurizer level to normal. Pressurizer level should be limited to ≤ 260".

2.14 Primary and Secondary Chemistry limits should be maintained as established by the Chemistry Manual.

2.15 **IF** Purification or Deborating IX is exhausted the applicable inlet valves should be closed and appropriate out of service stickers and R&R sheets issued.

2.16 Powdex should be 100% in even if a load reduction or three HWPs operation is required. Operation < 100% Powdex flow is acceptable for short time intervals.

2.17 Both MSRH Drain Tanks should be routed to the Hotwell for power changes > 6% FP to minimize feeding increased Secondary contaminants to Steam Generators.

2.18 Reactor power should be reduced ≈ 4% below allowable power when performing any operation that could cause a power swing.

2.19 The following UST temperature limits apply when EFDW is required: {1}

2.19.1 UST ≤ 125°F with Rx > 30% power

2.19.2 UST ≤ 125°F two hours after a Rx trip

2.19.3 UST ≤ 125°F two hours after reducing Rx < 30% power.

2.19.4 UST ≤ 145°F during Rx startup below 30% power.

2.20 IF load is reduced > 100 MW when ≥ 600 MW, VARs should be maintained within envelope specified in OP/1/A/1106/001 (Turbine Generator) until cold gas temperature has stabilized.

2.21 Operate individual coolers of the Second Cooler Group on the Main Transformer as needed to keep oil temperature from exceeding 75°C.

- When Main Transformer is energized and its oil temperature is < 50°C, only one cooler group should be operated (9 pumps/fans).

2.22 RCS boron may affect SSF RC M/U system operability per TS 3.10.1.

- RCS boron should be maintained > minimum required for SSF operability per PT/1/A/1103/015 (Reactivity Balance).

2.23 RATE SET (%/MIN / %HR) should be 0.0 when NOT actually changing power. This could prevent unanticipated Rx power changes if ICS goes into TRACK.

2.24 FDW flow to either SG should NOT exceed 5.7 x 10^6 lbm/hr.

- This limit applies to both 3 and 4 RCP operation.
- Computer points O1P0840 and O1P0841 most accurately indicate feedwater flow.
- Other computer points and flow gauges are NOT compensated and will indicate higher than actual FDW flow if FDW Fouling Coefficient is < one. {4}

2.25 At Rx power levels in the range of 35-85%, small (< 4.5%) power swings may occur.

- These swings are caused by adjustment of SG orifice plates causing steam flow oscillations in SG downcomer region.
- These swings have been evaluated and are considered normal.
- ICS is tuned to dampen the effects on CRD motion; however, if cyclic CRD motion is observed appropriate steps should be taken to correct problems as necessary. {8}

3. Enclosures

3.1 Power Escalation

3.2 Power Reduction

3.3 Special Instructions For < 4 RCP Operation

3.4 End Of Cycle APSR Withdrawal

3.5 Special Instructions For Operation With Reduced T_{AVE}

3.6 Operation Of CRD Group 8 To Control Rx Power Imbalance

Appendix

1. Initial Conditions

✓ 1.1 IF changing reactor power ≥ 10%, verify Maneuvering Plan is available for use. {6}

✓ 1.2 Review Limits and Precautions.

2. Procedure

2.1 IF > 6% CTP change is planned, refer to OP/1/A/1106/014 (Moisture Separator Reheater) to ensure Moisture Separator drains are routed to Hotwell.

CAUTION:	Special attention should be given to the selection of rate of power escalation. Before changing RATE SET the rate %/MIN or %/HR should be selected.

2.2 Limit rate of Rx power increase per PT/0/A/1103/020 (Power Maneuvering Guidelines).

✓ 2.2.1 Refer to Maneuvering Plan to view Reactor Engineering guidelines for power escalation. {6}

CAUTION:	If any CRD Groups are in the restricted region, boration of the RCS must begin within 15 minutes per TS 3.2.1. Operation in the restricted region is limited to 2 Hours.

2.3 Maintain CRD Groups 5-8 within required position limits per PT/1/A/0600/001 (Periodic Instrument Surveillance).

2.4 Maintain Core Power Imbalance AND Quadrant Power Tilt per PT/1/A/0600/001 (Periodic Instrument Surveillance).

_____ 2.5 At ≈ 225 MWe, transfer auxiliaries from CT-1 to 1T per OP/1/A/1107/002 (Normal Power).

_____ 2.6 Prior to exceeding 30% Reactor power, verify OR reduce UST temperature to ≤ 125°F. {1}

_____ 2.7 At ≈ 30% CTP, maintain steady state conditions **AND** perform the following:

_____ _____ • **IF** any two of the four NIs are **NOT** within ±2% of CTP, have I&E calibrate all NIs to ±2% CTP

AND resume power increase after calibration. {10}

• Reset "OUTPUT MEMORY" on "MAIN TURBINE TRIP BYPASS BISTABLE" for each of the four RPS Channels:

_____ _____ 'A' RPS Channel

_____ _____ 'B' RPS Channel

_____ _____ 'C' RPS Channel

_____ _____ 'D' RPS Channel.

_____ 2.8 Start second HWP.

_____ 2.8.1 Verify **OR** place standby HWP switch in "AUTO".

_____ 2.9 At ≈ 300 MWe, start 'D' Heater Drain Pumps per OP/1/A/1106/002 (Condensate and Feedwater System).

_____ 2.10 Begin second FDWPT startup per OP/1/A/1106/002 (Condensate and Feedwater System).

NOTE:	Rx power increase may continue while performing this step, sequence is NOT required.

_____ 2.11 Monitor computer point O1P3317 (CBP DELTA PRESSURE). {9}

_____ 2.11.1 At ≤ 400 psig Δp start second CBP.

_____ 2.11.2 Ensure standby CBP switch in "AUTO".

_____ 2.12 Prior to exceeding 60% CTP place second FDWPT in service.

NOTE:	Rx power increase may continue while performing this step, sequence is NOT required.

2.13 Verify the following valves indicate "CLOSED" at Heater Panel: (T-5-J18)

_____ 1HD-298 (HTR 1F1 DRAIN LVL CONTROL BYP)

_____ 1HD-303 (HTR 1F2 DRAIN LVL CONTROL BYP)

_____ 1HD-308 (HTR 1F3 DRAIN LVL CONTROL BYP).

1. **IF** any valve **DOES NOT** indicate "CLOSED", perform the following:

    a. Verify Emergency High Level Statalarm does **NOT** exist for associated 'F' Heater with Open valve(s).

    b. Close valve(s) as necessary to accomplish Step 2.13 above.

CAUTION:	• Never depress the upper toggle switch (Contact Buffer will Trip). • The lower toggle switch is always used to reset Contact Buffers.

2.14 Reset **OR** verify reset, the following:

- 'A' RPS Channel

_____ _____ "1A FDWPT" Contact Buffer

_____ _____ "1B FDWPT" Contact Buffer

_____ _____ "MAIN FEED PUMP TRIP BYPASS BISTABLE" output memory.

- 'B' RPS Channel

_____ _____ "1A FDWPT" Contact Buffer

_____ _____ "1B FDWPT" Contact Buffer

_____ _____ "MAIN FEED PUMP TRIP BYPASS BISTABLE" output memory.

- 'C' RPS Channel

_____ _____ "1A FDWPT" Contact Buffer

_____ _____ "1B FDWPT" Contact Buffer

_____ _____ "(MAIN FEED PUMP TRIP BYPASS BISTABLE" output memory.

- 'D' RPS Channel

_____ _____ "1A FDWPT" Contact Buffer

_____ _____ "1B FDWPT" Contact Buffer

_____ _____ "MAIN FEED PUMP TRIP BYPASS BISTABLE" output memory.

2.15 After both FDWPT are operating with suction flow > 2300 gpm, place the following in "AUTO":

_____ • 1FDW-53 (1A FDWP RECIRC CONTROL)

_____ • 1FDW-65 (1B FDWP RECIRC CONTROL).

_____ 2.16 **IF** required, start additional CCW pumps per OP/1/A/1104/012 (CCW System).

NOTE:	Rx power increase may continue to meet conditions of this PT; step sequence is **NOT** required.

_____ 2.17 **WHEN** conditions are met, perform "Control Rod Movement (Startup)" enclosure of PT/1/A/0600/015 (Control Rod Movement).

CAUTION:	• Rx power increase may continue while machine gas pressure is being increased. • Refer to Capability Curve of OP/1/A/1106/001 (Turbine Generator) to ensure Generator operating limits are **NOT** exceeded while increasing pressure.

_____ 2.18 **WHEN** the load reaches 500 MWe, begin raising Machine Gas pressure to 60 psig per OP/1/A/1106/017 (Hydrogen System).

NOTE:	1. Power escalation may continue while Generator Cold Gas temperature is being adjusted.
	2. Adjustments to 1C-58 (H_2 COOLER CONTROLLER) setpoint should be made in small increments to avoid Condensate System swings.

_____ 2.19 Adjust "AUTO" setpoint for 1C-58 as needed to maintain Generator Cold Gas temperature at ≈ 40°C as indicated by computer points O1A1148-O1A1151.

_____ 2.20 **IF** required, at ≈ 500 MWe notify Reactor Engineering to complete PT/0/A/0302/006 (Review and Control of Incore Instrumentation Signals). {2}

_____ _____ 2.21 At 65% CTP, maintain steady state conditions for ≈ 15 minutes for NI calibration check.

_____ 2.22 At ≈ 70% CTP, start third Hotwell Pump.

_____ 2.23 At ≈ 70% power, perform "Steam Extraction Check Valve Test" enclosure of PT/1/A/0290/005 (Secondary System Protection Test).

2.24 **IF** operating with < 4 RCPs, perform the following prior to exceeding 70% FP:

- _____ Have I&E reset high flux limiter for applicable overpower trip setpoints per Enclosure "Special Instructions for < 4 RCP Operation".
- _____ Adjust the high flux alarm setpoints as required. (The alarm setpoints are adjusted on the NI Recorder).
- _____ Perform pre-job brief stressing parameters to be monitored during power increase **AND** other factors to consider per Enclosure "Special Instructions for < 4 RC Pump Operation".

NOTE:	The following step does **NOT** have to be done in sequence.

_____ 2.25 **WHEN** both FDWPTs hydraulic oil pressure is > 190 psig, verify FDWPTs Aux Oil Pumps off **AND** switches are in "AUTO". (Continue)

NOTE:	• The following valves will be signed off in OP/1/A/1106/014 (Moisture Separator Reheater). • The following step does **NOT** have to be done in sequence.

_____ 2.26 **WHEN** High Pressure Turbine Exhaust pressure is ≈ 118 psig, complete OP/1/A/1106/014 (Moisture Separator Reheater) startup of Moisture Separator Reheater by verifying:

- 1MS-77 (MS TO 1A1 SSRH) Open
- 1MS-78 (MS TO 1A2 SSRH) Open
- 1MS-80 (MS TO 1B1 SSRH) Open
- 1MS-81 (MS TO 1B2 SSRH) Open

_____ 2.27 **WHEN** ≥ 85% CTP, begin startup of 'E' Heater Drain Pumps per OP/1/A/1106/002 (Condensate and Feedwater System). {7}

2.28 **IF** required, stop power increase at ≈ 90% CTP (indicated by Thermal Power Best) **AND** calibrate NIs to Thermal Power Best.

NOTE:	Remaining steps in this enclosure may be performed in any sequence.

_____ _____ 2.29 Increase CORE THERMAL POWER DEMAND (CTPD) SET to final core power desired.

_____ 2.30 **Prior** to calibrating NIs **OR** exceeding 95% CTP, notify Reactor Engineer to evaluate Feedwater Fouling Coefficient of Thermal Power Secondary calculation. {3}

_____ 2.31 Place the "1A FDWP SEAL INJ PUMP" in "NORM".

_____ 2.32 Place the "1B FDWP SEAL INJECTION PUMP" in "NORM".

2.33 **WHEN** Rx power is stable:

_____ 2.33.1 Notify Secondary Chemistry of intent to feed Moisture Separator Drains forward.

_____ 2.33.2 Align Moisture Separator Drain Tanks to feed forward per OP/1/A/1106/014 (Moisture Separator Reheater).

_____ 2.34 At ≈ 100% CTP, maintain steady state conditions for ≈ 15 minutes for NI calibration check.

1. Initial Conditions

_____ 1.1 **IF** reactor power will be reduced ≥ 40% with reactor remaining on line, notify Reactor Engineering to develop Maneuvering Plan.

_____ 1.2 Auxiliary Steam Header pressurized **OR** contingencies in place.

_____ 1.3 **IF** required, NRC notified per the requirements of OMP 1-10 (Usage and Testing the Emergency Notification System (Red Phone)).

_____ 1.4 Review Limits and Precautions.

2. Procedure

_____ 2.1 Refer to OP/1/A/1106/001 (Turbine Generator) to ensure operating limits are **NOT** exceeded during shutdown.

_____ 2.2 **IF** reducing reactor power > 6%, then refer to OP/1/A/1106/014 (Moisture Separator Reheater) to ensure Moisture Separator drains are routed to hotwell.

_____ 2.3 **IF** available, refer to Maneuvering Plan to view Reactor Engineering guidelines for power decrease. {6}.

_____ 2.4 Notify System Operations Center (SOC) of load reduction.

_____ 2.5 **IF** required, advise plant personnel of load reduction.

2.6 Start the following:

- _____ Place "1A FDWP SEAL INJ PUMP" switch to "START".
- _____ Place "1B FDWP SEAL INJECTION PUMP" switch to "START".

NOTE:	If Unit 1 will be taken off-line, target CTP should be ≈ 25% CTP

2.7 Begin CTP reduction to desired power level per the following:

_____ 2.7.1 Select "HOLD".

2.7.2 Select desired shutdown rate (one of the following):

- _____ RATE %/MIN
- _____ RATE %/HR.

_____ 2.7.3 Select desired rate of power reduction on RATE SET.

_____ 2.7.4 Select CTPD SET power level.

_____ 2.7.5 Release "HOLD".

2.8 <u>**WHEN**</u> ≤ 80% CTP: {7}

_____ • Stop 1E1 Heater Drain Pump.

_____ A. Verify <u>**OR**</u> place in AUTO 1HD-254 (1E1 HD PUMP RECIRC).

_____ • Stop 1E2 Heater Drain Pump.

_____ B. Verify <u>**OR**</u> place in AUTO 1HD-276 (1E2 HD PUMP RECIRC).

NOTE:	1B FDWPT should be taken out of service first. This is due to differences in High Pressure Discharge Trip Setpoints.

_____ 2.9 Start Aux Oil Pump on both FDWPTs.

_____ 2.10 At ≈ 550 MWe, shut down one FDWPT per OP/1/A/1106/002 (Condensate and Feedwater System).

NOTE:	1C CBP should remain running, if possible. {5}

_____ 2.11 At ≈ 450 Mwe, stop all but one CBP. {9}

_____ • Ensure control switch for one of the secured CBPs is in "AUTO".

_____ 2.12 At ≈ 400 MWe, stop 'D' HD Pumps per OP/1/A/1106/002 (Condensate and Feedwater System).

_____ 2.13 At ≈ 325 MWe, stop all but two HWPs.

_____ • Place control switch for the secured HWP in "AUTO".

_____ 2.14 At ≈ 225 MWe, transfer Auxiliaries from 1T to CT-1 per OP/1/A/1107/002 (Normal Power).

_____ 2.15 Stop all but one HWP.

_____ • Verify <u>**OR**</u> place control switch for one of the secured HWPs in "AUTO".

_____ 2.16 Verify SSF Operability per Enclosure "Minimum RCS Boron Concentration to Maintain " of PT/1/A/1103/015 (Reactivity Balance).

_____ 2.17 **IF** Turbine **OR** Reactor shutdown is required, **GO TO** OP/1/A/1102/010 (Controlling Procedure For Unit Shutdown).

1. Procedure

1.1 If conditions permit, log the current quadrant power tilt and the position of the ΔT_C controller prior to securing a RC Pump during power operations.

> **NOTE:** Instructions for performing OAC trends are located in "Working With Trends" enclosure of OP/0/A/1103/020A (Operator Aid Computer Use).

1.2 Digitally trend the following data at one minute intervals:

Point ID		Description
1.0	O1P0889	CORE THERMAL POWER BEST
2.0	O1P0877	INCORE IMBALANCE
3.0	O1A0059	CONTROL ROD GROUP 7 POSN
4.0	O1A0060	CONTROL ROD GROUP 8 POSN
5.0	O1P0737	INCORE TILT QUADRANT W-X
6.0	O1P0738	INCORE TILT QUADRANT X-Y
7.0	O1P0739	INCORE TILT QUADRANT Y-Z
8.0	O1P0740	INCORE TILT QUADRANT Z-W
9.0	O1P0828	RC COLD LEG A1 TEMP (1 MIN AVG)
10.0	O1P0829	RC COLD LEG A2 TEMP (1 MIN AVG)
11.0	O1P0830	RC COLD LEG B1 TEMP (1 MIN AVG)
12.0	O1P0831	RC COLD LEG B2 TEMP (1 MIN AVG)

_____ 1.3 Notify the Reactor Engineer to reset the Feedwater Venturi Fouling Coefficient of the Thermal Power Secondary Calculation to 1.0 prior to calibrating the NIs.

_____ 1.3.1 Calibrate NIs to Thermal Power Best.

1.4 Follow PT/1/A/0600/001 (Periodic Instrument Surveillance) limits on control rod position and Power Imbalance. The 100% Power Imbalance curves also apply for runs at reduced power.

NOTE:	If Quadrant Power Tilt problems do **NOT** exist, it is **NOT** necessary to reset the high flux RPS trip setpoint for three RC Pump operation.

_____ 1.5 Notify I&E of the potential for severe tilt problems **AND** the need to adjust flux/flow imbalance trip setpoints per requirements of TS 3.2.3.

1.6 Perform the following:

_____ 1.6.1 Adjust the ICS high flux limiter to 72%. This provides control protection to minimize a trip on flux/Flow/Imb **OR** high flux in the event of an operating transient.

_____ 1.6.2 **WHEN** the ICS high flux limiter is reduced, adjust the associated alarm setpoint to ≈ 2% less than the high flux limiter. (The alarm setpoint is adjusted on the NI Recorder).

_____ 1.6.3 Anytime the ICS high flux limiter is reduced, note on Turnover Sheet.

1.7 Keep Auxiliary Steam available to the FDWP turbines. 'D' bleed pressure may **NOT** be high enough to run the FDWP turbines.

1.8 Operating with three RC Pumps may cause a quadrant power tilt. The Steam Generator Load Ratio (ΔT_C) Controller can be used to minimize the magnitude of the tilt. If the tilt is on the 'A' side, quadrant W-X **OR** X-Y, adjust the ΔT_C Controller clockwise to make 'A' hot. If the tilt is on the 'B' side, quadrants Y-Z **OR** Z-W, adjust the ΔT_C controller counter clockwise to make 'B' hot.

The ΔT_C controller should be adjusted in small increments such as 1/2 degree F changes. All adjustments to the controller should be logged so that the recorded data can be accurately analyzed by the Reactor Group to determine the causes and results of unusual core power distributions.

1.9 If 1SSH-9 (SSH DISCH. CTRL BYPASS) is being used to maintain Steam Seal Header pressure, throttle the valve during the load reduction to secure an RC Pump.

NOTE:	RCS pressure decrease in the loop with two RC Pumps running is expected.

1.10 RCS pressure decrease in the loop with two RC Pumps running may cause acceptance criteria of PT/1/A/0600/001 (Periodic Instrument Surveillance) **NOT** to be met. Note this on PT/1/A/0600/001 (Periodic Instrument Surveillance). Be aware of the effect of the indicated pressure on the margin to trip setpoint for the Reactor Protective System on the following:

1. Pressure/Temperature Trip

2. Low Pressure Trip

3. High Pressure Trip

(R10/97) GR SIM NRC 106 115 JPP JMB

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. OP/1/A/1103/004
Revision No. 50

REPARATIONSS

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title SOLUBLE POISON CONTROL

(4) Prepared By Bluford Jones Date 10/21/99

(5) Requires 10CFR50.59 evaluation?
☑ Yes (New procedure or revision with major changes)
☐ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By Edmund K Kyle (QR) Date 10/27/99
Cross-Disciplinary Review By ______ (QR)NA EKK Date 10/27/99
Reactivity Mgmt. Review By Edmund K Kyle (QR)NA ______ Date 10/27/99

(7) Additional Reviews
Reviewed By ______ Date ______
Reviewed By ______ Date ______

(8) Temporary Approval *(if necessary)*
By ______ (SRO/QR) Date ______
By ______ (QR) Date ______

(9) Approved By D B Coyle Date 10/27/99

PERFORMANCE ***(Compare with control copy every 14 calendar days while work is being performed.)***

(10) Compared with Control Copy ______ Date ______
Compared with Control Copy ______ Date ______
Compared with Control Copy ______ Date ______

(11) Date(s) Performed ______
Work Order Number (WO#) ______

COMPLETION

(12) Procedure Completion Verification:
☐ Unit 0 ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 Procedure performed on what unit?
☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ______ Date ______

(13) Procedure Completion Approved ______ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31
1999
Received
Operator
Training

Duke Power Company Oconee Nuclear Station	Procedure No. OP/1/A/1103/004
Soluble Poison Control	Revision No. 050
Multiple Use	Electronic Reference No. OX002VLL

Soluble Poison Control

1. Purpose

This procedure provides procedural guidance for primary IX operation and RCS boron and lithium concentration change.

2. Limits and Precautions

2.1 This procedure controls activities that have potential to affect core reactivity. {4}

2.2 Positive Reactivity additions should be made by only one method at a time. {4}

2.3 CBAST requirements per SLC 16.5.13 are met when: {7}

- Volume and boron are within limits of Core Operating Limits Report (COLR)

  and

- Temperature is at least 10 °F above crystallization temperature. {6}

2.4 When either 1A or 1B Bleed Transfer pump are used to pump CBAST to LDST a minimum of 60 gal must be flushed to LDST. {6}

2.5 Operating guidelines for CBAST:

- CBAST level ≥ 75 inches.
- CBAST boron > 10,000 and < 14,000 ppm.
- CBAST level within permissible region of Enclosure "CBAST Level Vs. Concentration" OP/0/A/1108/001 (Curves and General Information).
- CBAST temperature >125°F.
- When 1A or 1B Bleed Transfer Pump are used as available flow path, Heat tracing circuits must be in service. {6} {7}

2.6 Operating 1A CBAST pump in Auto allows 1HP-15 Moore Controller to stop 1A CBAST pump when the desired batch is reached.

2.7 When feed and bleed from CBAST to LDST is in progress, LDST temperature should be monitored and maintained < 130°F.

NOTE:	Filling RCS with < 5 wt% boron protects RCP seals from having boron crystal damage.

2.8 During initial RCS fill, boron additions to RCS should __NOT__ exceed 5 wt.% (8750 ppm).

2.9 With no additions to LDST in progress, 1HP-15 controller should be setup for Normal Operation:

- Mode selector to "MANUAL".
- Display selector to "P".
- Valve position: 100% open.
- Start-stop to "START".

2.10 When Pzr and RCS boron are > 100 ppm difference, Pzr spray should be initiated per OP/1/A/1103/005 (Pzr Operation).

2.11 If all RCPs are shutdown, RCS makeup sources should be premixed to a boron > 1% Δk/k SDM prior to RCS makeup. {3}

2.12 Makeup to RCS with a source < 1% Δk/k SDM boron should only be performed with at least one RCP operating. {3}

3. Enclosures

3.1 RCS Boron Change Calculation

3.2 Aligning 1B Bleed Transfer Pump Suction To 1A BHUT

<u>CBAST Transfer</u>

3.3 Unit 1 CBAST Transfer To Unit 2 CBAST

3.4 Unit 1 CBAST Transfer To Unit 3 CBAST

<u>RCS Make-Up</u>

3.5 RCS Normal Make-Up

3.6 RCS Make-Up From Unit 2 (2A or 2B BHUT)

3.7 RCS Make-Up From Unit 3 (3A or 3B BHUT)

3.8 RCS Make-Up From BHUT(s) During BHUT Purification (HPI Operating)

3.9 RCS Deboration To ECB

1. Initial Conditions

✓ 1.1 HPI System in normal operation.

✓ 1.2 Spare Deborating IX **NOT** in service.

✓ 1.3 Notify Chemist of Unit 1 Deborating IX use.

✓ 1.4 CRD Group 6 > 95% withdrawn.

✓ 1.5 Letdown flow 50 - 70 gpm.

✓ 1.6 Review Limits and Precautions.

2. Procedure

NOTE:	• This procedure may affect core reactivity by changing RCS boron. • Anytime an IX is placed in service CRD movement may result.

✓ 2.1 Evaluate for acceptable boron per Enclosure "IX Use Determination".

✓ 2.2 SRO review of evaluation for acceptable boron per Enclosure "IX Use Determination".
SRO

✓ 2.3 **IF** required, perform Enclosure "Unit 1 Deborating IX Rinse" to equalize IX boron with RCS.

✓ 2.4 Chemist has requested Unit 1 Deborating IX to be in service to de-lithiate for 5 minutes.

NOTE:	• L/D pressure should be monitored. • Letdown pressure should **NOT** exceed 140 psig to prevent lifting relief valves.

✓ 2.5 Record Letdown pressure: 90 psig.

2.6 **IF** Letdown pressure is > 95 psig, place both Letdown filters in service:

_____ 2.6.1 Open **OR** verify open 1HP-17 (1A LETDOWN FILTER INLET).

_____ 2.6.2 Open **OR** verify open 1HP-18 (1B LETDOWN FILTER INLET).

2.7 Place Unit 1 Deborating IX in service per the following:

_____ 2.7.1 Verify 1CS-32 & 37 (SPARE DEBOR IX INLET & OUTLET) closed.

_____ 2.7.2 Close 1CS-26 (LETDOWN TO RC BHUT)

_____ 2.7.3 Open 1CS-27 (DEBOR IX INLET).

_____ 2.7.4 Open 1HP-16 (LDST MAKEUP ISOLATION)

_____ 2.7.5 Verify 1HP-15 (LDST MAKEUP CONTROL) controller in "MANUAL".

_____ 2.7.6 Verify 1HP-15 (LDST MAKEUP CONTROL) open.

NOTE: CR Operator should monitor neutron error and run time to ensure proper reactivity management.

_____ 2.8 Position 1HP-14 (LDST BYPASS) to "BLEED".

_____ 2.9 Record Letdown pressure: _________ psig.

_____ 2.10 Do **NOT** become distracted.

2.11 **IF** RCS makeup is required while Unit 1 Deborating IX is in service, perform Section 3 "RCS Volume Control with Unit 1 Deborating IX in service".

NOTE: Opening 1CS-26 will remove Unit 1 Deborating IX from service.

2.12 To reduce RCS inventory (BLEED):

2.12.1 Open 1CS-26 (LETDOWN TO RC BHUT)

2.12.2 **WHEN** RCS BLEED is complete, close 1CS-26 (LETDOWN TO RC BHUT).

NOTE: IX effluent sample requires a 15 minute flush.

_____ 2.13 **IF** required, notify Chemist to sample IX effluent.

2.14 **WHEN** IX run time is complete, remove Unit 1 Deborating IX from service:

_____ 2.14.1 Position 1HP-14 (LDST BYPASS) to "NORMAL".

_____ 2.14.2 Close 1HP-16 (LDST MAKEUP ISOLATION).

_____ 2.14.3 Reset 1HP-15 (LDST MAKEUP CONTROL) controller for Normal Operation.

_____ 2.14.4 Close 1CS-27 (DEBOR IX INLET).

_____ 2.14.5 Open 1CS-26 (LETDOWN TO RC BHUT)

2.15 **IF** second Letdown Filter was required, isolate one letdown filter:

_____ • Close 1HP-17 (1A LETDOWN FILTER INLET),

OR

_____ • Close 1HP-18 (1B LETDOWN FILTER INLET).

_____ 2.16 Log IX service status in "IX Status Log".

_____ 2.17 Log IX use in Unit Log.

3. RCS Volume Control With Unit 1 Deborating IX In Service.

NOTE: The following system configuration will be in place with 1HP-14 in "NORMAL":

- Letdown flow will be directed to LDST.
- Letdown flow will **NOT** be through Unit 1 Debor IX.
- Only make-up volume will be counted on 1HP-15 controller.
- An RCS leakage may be performed, as required.
- Letdown pressure should be lower.

3.1 Position 1HP-14 (LDST BYPASS) to "NORMAL".

3.2 Reset 1HP-15 volume by pressing "STOP" / "START".

3.3 Verify 1HP-15 controller in "MANUAL".

3.4 **IF** required, adjust Letdown flow by throttling 1HP-7 (Letdown Control).

3.5 Start appropriate RC Bleed Transfer pump.

3.6 Open RC Bleed Transfer pump discharge valve.

3.7 **WHEN** RCS makeup is complete, stop RC Bleed Transfer pump **AND** close discharge valve.

3.8 **IF** required after makeup, perform RCS leakage.

3.9 Verify 1HP-15 (MAKEUP CONTROL) in "MANUAL".

3.10 Verify 1HP-15 (MAKEUP CONTROL) open.

3.11 Verify 1HP-16 (MAKEUP ISOLATION) open.

3.12 Position 1HP-14 (LDST BYPASS) to "BLEED".

3.13 Verify 1CS-26 (LETDOWN TO RC BHUT) closed.

3.14 **IF** required, adjust Letdown flow by throttling 1HP-7 (Letdown Control).

B-9

RM

PROCESS MONITOR RADIATION HIGH

1. Alarm Setpoint

1.1 Set by operator per PT/0/A/0230/001 (Radiation Monitor Check). Alarm occures when the ALERT setpoint is reached.

2. Automatic Action

2.1 1RIA-45 closes the purge isolation valves 1PR-2, -3, -4, and -5 and trips the main or mini purge fan if the high alarm setpoint is reached.

2.2 1RIA-37 and/or RIA-38 will close valves 1GWD-4, -5, -6, and -7 and GWD-206, 207 and stop the W. G. exhauster if the high setpoint is received.

2.3 1RIA-49 closes 1LWD-2 and actuates RB evacuation alarm if the high alarm setpoint is reached.

2.4 1&2 RIA-54 will terminate the release of the Unit 1&2 Turbine Building Sump if the high setpoint alarm is received.

3. Manual Action

3.1 Determine which radiation monitor is high.

3.1.1 If the VIEW node or either SCADA node is not in service, refer to OP/1/A/1103/026, (Loss of Sorrento Radiation Monitor).

3.2 Ensure that the automatic actions, if required, have taken place; if not, perform actions manually.

3.3 IF 1RIA-16 or 1RIA-17 is in alarm:

- Refer to EP/1/A/1800/001 (Emergency Operating Procedure)
- Refer to OP/0/A/1106/031 (Control Of Secondary Contamination)

3.4 Refer to AP/1/A/1700/018 (Abnormal Release of Radioactivity) to determine what action is required.

3.5 If 1RIA-39 alarms, Refer to AP/1/A/1700/018 (Abnormal Release of Radioactivity) to start the Outside Air Booster Fans.

3.6 If 1RIA-47 alarms, check the Reactor Coolant System for increase in RCS leakage by checking RB Sump rate and performing PT/1/A/0600/010 (Reactor Coolant Leakage).

3.7 If 1RIA-35 alarms, start batch of Turbine Building Sumps and isolate the leaking component (i.e., decay heat cooler, etc.) as practicable. Secure any LWRs in progress until evaluated. An immediate safety evaluation of the consequences of radioactive release must be made for continued operation of the affected system.

3.8 If RIA-32 alarms, contact RP to sample affected areas.

3.9 Dispatch operator to determine the cause of the alarm and initiate corrective action.

4. Alarm Sources and References

4.1 OEE-118-17 & 18.

4.2 IP/0/A/360/4C (Process Radiation Monitoring System RIA-37 Waste Gas Disposal Monitor [Norm])

4.3 IP/0/A/360/4D (Process Radiation Monitoring System RIA-38 Waste Gas Disposal Monitor [High])

4.4 IP/0/A/360/033 (Sorrento Process Radiation Monitor Low Range Gas Detector Calibration).

4.5 IP/0/A/360/20 (Process Radiation Monitors RIA-54 Turbine Building Sump).

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 300 gallon per day total primary to secondary LEAKAGE through all steam generators (SGs); and

e. 150 gallon per day primary to secondary LEAKAGE through any one SG.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	-------------------------NOTE------------------------- Not required to be performed until 12 hours after establishment of steady state operation. -- Evaluate RCS Operational LEAKAGE.	 72 hours
SR 3.4.13.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

Problem Investigation Process
Oconee Nuclear Station

PIP Serial No:	Action Category:	LER No:	Other Report:
O-99-05270	2		

Problem Identification

Discovered Time/Date: 20:00 12/27/1999 **Occurred Time/Date:** 19:00 12/27/1999

Unit(s) Affected:

Unit	Mode	%Power	Unit Status Remarks
3	1	100	Normal Full Power Operations

System(s) Affected:
HP High Pressure Injection (Vlv only)

Affected Equipment

(No Equipment Affected)

Location of Problem:
Bldg: Column Line: Elev:

Location Remarks:

Method Used to Discover Problem:

Brief Problem Description:
3HP-14 Valve Stem/Motor Coupling fails.

Detail Problem Description:
On 12-27-99 at ~ 19:00, a low LDST pressure alarm was recieved. Upon investigation, it was determined the LDST level was decreasing. It was noticed the 3A BHUT level was increasing. Although 3HP-14 indicated normal, it appeared to be bleeding ~ 44 gpm to the 3A BHUT. TS 3.4.13, SLC 16.5.10, and PT/3/A/0600/10, RCS Leakage was referenced. TS 3.4.13 was considered not to apply since the leakage was recoverable. SLC 16.5.10 was entered since the recoverable leakage was >30 gpm. The Unit 3 SSF RCMU pump was declared OOS due to total combine leakage being >24.7 gpm. Additionally, the E-Plan was referenced and an Unusual Event declared at 20:40. The initial notification both declared and terminated the event. Termination time was 20:42. The event was considered terminated once 3HP-5 was closed, isolating the leak. Engineering, Maintenance, and Chemistry support was called in. The problem was a broken coupling between the motor and valve.

At 20:54, a power decrease at 2.0 %FP/min was initiated per AP/3/A/1700/14, Loss of Letdown, since repairs would not be possible in the 1 R/hr field at the LDST. At ~50%FP, dose rates at the LDST was ~350 mR/hr. The power decrease was suspended at 35 %FP due to LDST area dose rates and the system being placed into a continuous delithiation lineup. This lineup allows the inventory being bled to be recovered to the LDST via the 3B Deborating Demineralizer.

12-27-99 19:03 Received OAC alarm for LDST pressure low. Investigation revealed that the LDST level/pressure was decreasing. A board scan indicated that 3HP-14 was in the "NORMAL" position, but 3A BHUT level was increasing which would be an indication that 3HP-14 was in the "BLEED" position. Calculation for batch additons to the RCS were made in preparation for addition to the LDST. eferred to AP/3/A/1700/14, Loss of HPI Makeup and Letdown.

12-27-99 20:00 Entered SLC 16.5.10, Condition A, RCS leakage > 30 gpm (actual RCS leakage ~45 gpm); Req. Action A.1- Be in MODE 3 in 12 hours AND A.2 - Be in MODE 5 in 36 hours. Referenced ITS 3.4.13, RCS Operational Leakage which does not apply

Problem Investigation Process
Oconee Nuclear Station

because leakage is recoverable.

12-27-99 20:00 Entered ITS 3.10, Condition C, U-3 SSF RCMU System Inoperable per PT/3/A/0600/010 (Reactor Coolant Leakage), total combined RCS leakage > 24.7 gpm. Required Actions C.1, restore within 7 days. SSF is out of service for U-3 only.

12-27-99 20:40 Declared Unusual Event due to leakage > 25 gpm

12-27-99 20:42 Unusual Event Terminated

12-27-99 20:42 Closed HP-5 to isolate letdown per AP/3/A/1700/14.

12-27-99 21:01 Entered ITS 3.4.9, Condition A, PZR level > 285" (due to isolating letdown per AP/3/A/1700/14); Req. Action A.1 - restore level within 1 hour.

12-27-99 21:19 Went to "HOLD" on Diamond to stop Rx power reduction at ~ 50% FP in order to limit PZR insurge. Pressurizer level ~ 350" at this time.

12-27-99 21:20 Re-established letdown to lower Pressurizer level; no entry into Emergency Plan made because letdown flow to the 3A BHUT was desired at this time for plant control.

12-27-99 22:01 Entered ITS 3.4.9, Condition B; Req. Action B.1 - Be in MODE 3 in 12 hours, and B.2 - Be in MODE 3 with RCS temp. </= 325. This is due to not meeting the required 1 hour time limit for having PZR level < 285" (Req. Actions for Condition A).

12-27-99 22:14 Exited ITS 3.4.9, Conditions A and B.

12-27-99 23:00 Closed 3HP-5, began power reduction to 30% CTP at 3% / hr because Pressurizer level was at desired level (~ [illegible]0").

12-28-99 04:00 Went into Delith Lineup.

Originated By: TPG4205: GILLESPIE JR, T P Team: NEC3262 Group: OPS Date: 12/28/1999

Other Units/Components/Systems/Areas Affected(Y,N,U): U

Industry Plants Affected(Y,N,U): U

Immediate Corrective Actions:

The following corrective actions have been taken to address the 3HP-14 failure(s) to this point (12/30/99, 2300):

1.) Root Cause Investigations performed on initial and subsequent failure of 3HP-14;
2.) Repair, mock-up validation and testing performed on 3HP-14 (WO# 98231549);
3.) In-service inspections performed on 1HP-14 and 2HP-14 (WO# 98231839 - 1HP-14 and WO# 98231847 - 2HP-14);
4.) Work requests initiated for detailed disassembly inspections of 1HP-14 and 2HP-14 (WR# 98110414 - 1HP-14 and WR# 98110415 - 2HP-14). A lotus note was sent to Dan Greene and Dave Patterson (OPS Staff) to have these work orders planned and added to each units' Hot List.

Details of corrective actions:

1.) 3HP-14 Root Cause Investigation Findings and Generic Implications / Transportability

Problem Investigation Process
Oconee Nuclear Station

Initial failure: The initial failure resulted from installation of a shaft coupling which did not have the proper bore diameter or thread pitch to accommodate the actuator shaft. The actuator shaft is 3/4 inches in diameter with 20 threads per inch. The corresponding coupling bore was found to be 13/16 inches in diameter with 16 threads per inch. This mismatch created a substandard joint that failed after innumerable valve strokes. The actuator shaft was also discovered bent, with a runout of approximately .030 inches. It is suspected that this bend occurred at some point in the past when the coupling anti-rotation guide slot became disengaged from the yoke guide creating a binding situation. This bend may have actually strengthened the joint (making up for the oversized coupling bore) preventing earlier failure.

Subsequent failure: The failed coupling was replaced with one made in the Oconee machine shop. The thread pitch and bore discrepancy noted above had been corrected. Nonetheless, failure occurred after only 7 valve strokes. This failure occurred as a result of inadequate clamping force on the valve stem (point of separation) created by the bent actuator shaft.

Generic Implications / Transportability: Work history review shows that similar events have occurred on 1HP-14 (2/11/93) and on 2HP-14 (10/03/91). Therefore, work orders were generated to perform in-service inspections on these valves (Ref. Item 3 above).

2.) Repair, mock-up validation and testing of 3HP-14

Another replacement coupling was fabricated in the Oconee machine shop. The actuator shaft was removed and a detailed inspection performed. The bent portion of the shaft was removed to allow coupling to the unaffected portion of the shaft. A mock-up was constructed using the repaired actuator shaft, a new valve stem and the second replacement coupling. An inspection was then performed on the mockup to ensure proper fit-up and to establish expected field geometry. No discrepancies were noted. The repaired actuator shaft and replacement coupling were then installed in 3HP-14. Assembly inspections revealed no anomalies; the coupling halves were verified to be parallel within .005 inches.

ɩe coupling was re-installed on the valve. After limit switch set-up, testing will be performed during which the valve will be stroked 20 ʌhes. One half of the coupling will then be removed to inspect for thread distortion or damage.

3.) In-service inspections of 1HP-14 and 2HP-14

Both valves were inspected for coupling anomalies. Specifically, the coupling halves were inspected for parallelism, the exposed portions of threaded shaft (actuator and valve) were inspected for thread distortion and the bolting material was verified tight. No discrepancies were noted on either valve.

4.) Work orders initiated for detailed inspection of 1HP-14 and 2HP-14

Although no discrepancies were noted during the in-service inspections of these valves, further inspection is warranted based on the noted failure history. Work orders were generated to perform detailed disassembly inspections on both valves during the next outage of sufficient duration.

Last Updated By: PVF1489: FISK, PAUL V Team: SDC3511 Group: MSE Date: 12/30/1999

Last Updated By: SDC3511: CAPPS, STEVEN D Team: SDC3511 Group: MSE Date: 12/31/1999

See Detailed Problem Description

Originated By: TPG4205: GILLESPIE JR, T P Team: NEC3262 Group: OPS Date: 12/28/1999

ɩmediate Corrective Action Documents / Work Orders:

	Indiv	Team	Group	Date
Problem Identified By:	TPG4205	NEC3262	OPS	12/28/1999

Problem Investigation Process
Oconee Nuclear Station

Problem Entered By: TPG4205 NEC3262 OPS 12/28/1999

Screening

Is the Problem Significant? Yes Action Category: 2

Significance Codes:

15	Emergency Plan Declaration of Alert or H
18	Severe or Unusual Plant Transients not c
3	Any Unit unplanned outage, oper at sign.
9a	Unanticipated loss of water from primary

OEP No:

Other Report Nos:

Event Codes:

F1b	Unplaned entry into T/S LCO
F2a	Generic Valve Failure
F6a	Internal to Component/System
L	Power Reduction - Unscheduled
N2	Reactivity Management Event

reening Remarks:

his event has been reviewed by the CST and found to meet the MSE significance criteria.

Screening members present for this review: Sandy Severance (ENG), RH Ledford (MNT & WCG), and Mike Pruitt (OPS)

Originated By: EHD8302: DUMMEYER, EDWARD H Team: RTB7310 Group: SRG Date: 12/28/1999

Assignments:

Responsible Groups(s) for Problem Evaluation:	MSE	Mech. Sys/Equip
Responsible Group for Present Operability:	N/A	
Responsible Group for Past Operability:	N/A	
Responsible Group for Reportability:	RGC	Regulatory Compliance
Responsible Group for Overall PIP Approval:	OPS	Operations

Signature Type	Indiv	Team	Group	Date
Screened By:	EHD8302	RTB7310	SRG	12/29/1999

Present Operability

esponsible Group: RGC Status: NotRequired

Sys/Comp Operable? (Y,N,C,E,T):

Problem Investigation Process
Oconee Nuclear Station

Required Mode:

Comments:

No Current Signatures For This Section

Past Operability:

Responsible Group: RGC Status: NotRequired

Sys/Comp Operable?(Y,N,C,E,T):

Required Mode:

Comments:

No Current Signatures For This Section

Reportability

Responsible Group: RGC Status: Closed

Problem Reportable(Y,N,E): N

;portable Per:

Comments:

This event is not reportable. There were no conditions or operations prohibited by any Technical Specifications and this event did not meet any of the 10CFR50.73 reporting criteria as stated in NUREG-1022, Rev. 1. Even with the one hour NRC notification because of the unusal event, section 3.1.1 of NUREG-1022, Rev. 1 requires that you must meet one or more of the 10CFR50.73 reporting criteria for the event to be reportable.

Originated By: JVW2844: WEAST,JAMES V Team: LEN2127 Group: RGC Date: 12/28/1999

Signature Type	Indiv	Team	Group	Date
Assigned To:			RGC	12/28/1999
Ready For Approval:	JVW2844	LEN2127	RGC	12/28/1999
Approval Assigned To:	LEN2127	LEN2127	RGC	12/28/1999
Approved By:	LEN2127	LEN2127	RGC	12/31/1999

Investigation Report:

Responsible Group: Act Date:

vestigator: Group:

Due Date:
Date Due to VP or Sta. Mgr:

Problem Investigation Process
Oconee Nuclear Station

Date Regulatory or Agency Rpt Due:
Date Investigation Report Approved:

NRC Cause Codes:

Problem Evaluation

Event	Cause Code	Cause Description	Primary	Causing Groups
F2a	B4c	Omission of relevant information	Yes	MSE
F2a	O2c	Improper reassembly of component	Yes	MNT

Problem Evaluation From: Resp. Group: MSE Status: Closed OEDB Checked: Yes

Edit to clear up problems where fractions and table data did not correctly import from source document.
Last Updated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/24/2000

Root Cause Failure Analysis Report

3HP-14 Failed in Bleed Position on 12/27/99 and then again on 12/29/99

:vision # 0
IP# 99-5270

Prepared By: Milton Addis Date 01/14/2000

Reviewed By: Tommy Mills Date 01/17/2000

Approved By: Tom Curtis Date 01/19/2000

Root Cause Executive Summary

Title of Event or Issue: 3HP-14 Failed in Bleed Position on 12/27/99 and then again on 12/29/99

PIP Number: 99-5270 (PIP 99-5278 was initiated for second failure but both failures are being addressed in 99-5270)

Date of Event or Identification of Issue: 12/27/99 and 12/29/99

Date Report Completed: 1/14/00

Root Cause Investigator: Root Cause Team Investigation with Tommy Mills as Team Leader and Milton Addis as Root Cause Evaluator

Description of Event/Issue:
On 12/27/99 and again on 12/29/99, 3HP-14 (Fisher 657-YY valve with a Limitorque SMB-000

actuator) failed in the bleed position causing inability to control makeup and letdown of the RCS.

Summary of Root Cause:

Root cause determination for the December 27 event;
The actuator stem for 3HP-14 pulled from the coupling that connects it to the valve stem due to a mismatch in bore diameter and thread size. The actuator stem outside diameter was 3/4 inch with 20 threads per inch and the coupling inside diameter was 13/16 inch with 16 threads per inch.

Root cause determination for the December 29, 1999 event;
The valve stem for 3HP-14 pulled from the coupling that connects it to the actuator stem due to inadequate clamping force caused by a bent actuator stem.

Planned Corrective Actions:

Remedial/Immediate Corrective Actions:
> Remove the run-out from the actuator stem by removing approximately 1 inch of the threaded portion (coupling end). Then, rethread the stem end to match the diameter and threads per inch of the coupling. (This item has been completed. It should be noted that the length of the actuator stem was verified to be of sufficient length to facilitate as much as 1-7/8 inches without interfering with the proper operation of the valve/actuator).
> Fabricate (and install) a new coupling that has the correct diameter and threads per inch to match the actuator stem and valve stem. (This item has been completed)
> Test valve to ensure coupling is adequately connected and verify no internal binding of the valve/actuator exists. (Test criteria: Stroke the valve every 6 minutes until 10 full cycles are mpleted {20 strokes}, perform feeler gauge checks and record measurements, remove bolts /coupling and remove one half of coupling, inspect threads on coupling for signs of deformation, reinstall the coupling half, perform feeler gauge check of coupling on all four corners and verify halves are parallel, cycle valve two more times, monitor amp readings during stroking (MPM) to verify no binding is present. (This has been completed. The valve stroked successfully approximately 20 times and there was no degradation observed on the coupling/stem threads. In addition, MPM data taken during the strokes indicated no evidence of internal binding.)

Interim Corrective Actions:
> Perform visual inspection of the condition of the couplings for 1HP-14 and 2HP-14 to ensure they are connected and there are no external signs of degradation. (This has been completed with no problems being discovered. It should be noted that this was a limited scope inspection, with the coupling assembled, due to radiation levels in the area at full power and the valve being in service).

Corrective Actions to Prevent Recurrence:
1. Perform detailed inspection of the coupling for 1HP-14 during unit shutdown or low power operation (power level equal to or less than 30 %). Inspection to include:
> Disassemble and removal of coupling
> Verify there are no bends in the valve and actuator stem in the coupling area by checking straightness (i.e. measure stem run-out)
> Verify diameter and threads on stems match those of the coupling
> Verify there is no thread damage to either the stems or coupling
> Reinstall the coupling

?. Perform detailed inspection of the coupling for 2HP-14 during unit shutdown or low power peration (power level equal to or less than 30 %). Inspection to include:
> Disassemble and removal of coupling
> Verify there are no bends in the valve and actuator stem in the coupling area by checking straightness (i.e. measure stem run-out)

> Verify diameter and threads on stems match those of the coupling
> Verify there is no thread damage to either the stems or coupling
> Reinstall the coupling

3. Revise MP/0/A/1200/027A (HP-14 - Valve - Fisher - YY Type - Disassembly and Reassembly) to include installation details for the coupling to ensure acceptable configuration and assembly.

4. Once detailed inspections are complete for 1HP-14 and 2HP-14 , ensure proper stock codes are set up for couplings for all three units.
****

Problem Identification

On December 27, 1999, while investigating a low LDST pressure alarm, it was discovered that inventory from the Unit 3 LDST was bleeding to the 3A Bleed Holdup Tank although 3HP-14 indicated being in the NORMAL (FEED) position.

On December 29, 1999, after flushing the CBAST line with approximately 100 gallons of demin water, 3HP-14 was placed in the BLEED position to facilitate lowering the level of the LDST. When the bleeding was complete, 3HP-14 was placed back in the NORMAL position but the bleeding was observed to continue.
****

Data Collection

3HP-14 Operation

3HP-14 is 0.333HP, 208VAC, 3phase, 60Hz, three-way motor operated valve that determines the flow path for the purification demineralizer effluent. It is a Fisher 657-YY type valve with a Limitorque SMB-000 actuator. In the NORMAL position, 3HP-14 directs the dimineralizer discharge to the letdown storage tank; while in the BLEED position, flow is directed to either the 3A RC Bleed Hold-up Tank or the deborating demineralizer. For normal operation, a three position, spring returned to neutral, NORMAL-BLEED control switch is provided in the control room on the unit board 3UB1. 3HP-14 is placed in the normal position by momentarily turning the control switch to NORMAL. The valve will continue to move after the switch is released, until it is de-energized by a valve position limit switch. 3HP-14 is placed in the bleed position in a similar manner by momentarily turning the control switch to BLEED. Torque Switch contacts are provided in the valve control circuits to de-energize the valve motor in the event of excessive operating torque such as that caused by an obstruction or valve failure.

3HP-14 is positioned to NORMAL automatically on receipt of a letdown storage tank low level signal (via 3HPILT0033P1 or P2) or by a signal from the feed controls indicating that the desired batch has been delivered from the coolant storage system to refill the letdown storage tank.

3HP-14 is QA Condition 1 but the valve actuator is non-QA.

Sequence of events

On 12/27/99, at approximately 1900, a low LDST pressure alarm was received in the Unit 3 Control Room. Operations personnel began investigating and discovered the level of the LDST to be decreasing. It was estimated that approximately 44 gallons per minute of RCS was bleeding through 3HP-14 to the 3A Bleed Holdup Tank in spite of the fact that 3HP-14 indicated being in the normal (feed) position.

Operations personnel referenced TS 3.4.13 and SLC 16.5.10 and made the determination that 3.4.13 did not apply since the leakage was recoverable. SLC 16.5.10 was entered since the recoverable leakage was greater than 30 gpm. The SSF RC Makeup Pump was declared inoperable due to leakage exceeding 24.7 gpm.
In addition, the Emergency Plan was referenced and an unusual event declared at 2040. The unusual event was terminated at 2042 when 3HP-5 was closed, isolating the leak.

At approximately 2300 on 12/27/99, following the closing of 3HP-5, Operations personnel began reducing unit power at 3% per hour. Power was reduced to 30% and the HPI letdown and purification system was placed in the delithiation mode to maintain primary tank levels.

A recovery team made up of Technicians and Engineers was assembled and began an apparent cause investigation to determine the cause of 3HP-14 to fail in the bleed mode. It was soon discovered that the actuator stem had pulled from the coupling that connects it to the valve stem. A replacement coupling could not be located so measurements of the actuator stem and valve stem were made and a new coupling was fabricated. The new coupling was installed on 12/28/99. Motor power monitoring following the coupling replacement did not indicate any problems with internal binding so the team made the assumption that the coupling had failed from cycle fatigue. Soon after repairs were complete, Operations personnel began power increase to return Unit 3 to full power.

At 0549 on 12/29/99, while unit 3 was at 90% power, 3HP-14 was placed in the BLEED position to lower the LDST level following a flush of the CBAST line with demin water. When inventory in the LDST reached the desired level, 3HP-14 was placed in the NORMAL (FEED) position but RCS continued to bleed. Operation personnel began taking compensatory actions as described ove for the 12/27/99 event.

According to interviews on 12/29/99 with Operations personnel on duty during the second failure of 3HP-14, the valve was cycled several times during the shift for normal feed and bleed operation with no problems being experienced. The Control Room personnel were aware of the problems that had taken place on 12/27/99 and were very focused on LDST level. At approximately 0549, 3HP-14 was switched from the BLEED position to the NORMAL (FEED) position and the LDST level was observed to continue decreasing. They began compensatory actions per their procedures.

Root Cause Team Organization

A Root Cause Team was organized on December 29, 1999 and given responsibility for determining the root cause for inability to control makeup and letdown of RCS with 3HP-14. (This team was a sub-team of a FIP/Recovery Team) In addition, the team responsibilities were to determine the corrective actions necessary to prevent recurrence.
The Team members were:
Tommy Mills - Leader
Milton Addis - RC Evaluator
Steve Capps - Night Lead
Paul Fisk - HPI Systems Engineer
Jason Patterson - Systems Engineer
Kevin Matthews - Valve Engineer
Jim Kiser - Valve Engineer
Carleton Burell - Valve Engineer
Eddie Welch - CEN Engineer
(Reference Attachment 1 for FIP Team Organization and Pre-Job Briefing)

The Root Cause Team convened on 12/29/99 at approximately 0900 and began their root cause investigation into the two failures of 3HP-14. The facts (based on field observations) known at that time were:

Problem Investigation Process
Oconee Nuclear Station

1) Yoke guide has missing part at top (approximately 1/4 inch angled)
2) Motor is running and coupling is turning
3) Coupling bolts are intact
4) Coupling is detached from valve stem. The initial failure (12/27/99 @ 2030 was at top of coupling (actuator stem)
5) Some metal particles on valve stem
6) Appears to be a gap between block halves

Addition facts discovered during the course of the investigation on 12/29 - 12/31/99 were:

7) Per the vendor, the original actuator stem threads were 13/16 inch diameter x 16 threads/inch
8) As found actuator stem threads were 3/4 inch diameter x 20 threads/inch
9) First failed coupling was original coupling or OEM spare part (determination made due to cadmium plating)
10) OEM has modified part twice; now supplied as carbon steel
11) Tribal knowledge (Phillip Bowers) says lower disc assembly backing off has been a previous failure. Procedure(s) changed to require staking. This was noted as being done during valve repair 2/15/99 per WMS documentation
12) For 12/29/99 failure, visual inspection of coupling and valve stem revealed the valve stem had been pulled out of the coupling. Stem threads were in good condition (cleaned up with thread chaser) but coupling threads were damaged
13) 3HP-14 in bleed position to lower LDST level then placed in normal position. Indication lights switched position in approximately 12 seconds. Low pressure alarm on LDST ~pproximately 2 minutes later. OPS demanded valve to normal position again. Estimated proximately 43 gallons bleed.
14) Coupling installed by skill of craft, not by procedure (proper thread engagement of stems into block was reported to be obtained

Work History for 3HP-14

8/29/92
Work Order 92065469-01
Stem found pulled from coupling with coupling threads destroyed
Installed new stem coupling

2/21/94
Work Order 94015026-01
Valve diagnosed with seat leak
Stem thrust increased to prevent seat leakage
Task included loosening and retightening stem coupling

2/16/97
Work Order 97014041-01, 08, 09, 11, 15
Valve diagnosed with seat leak
Cut valve out, refurbed valve, reinstalled valve
Task included removal and reinstallation of stem coupling (task details say the coupling was retapped but does not say why this was done or how)

Root Cause Team Notes

2/30/99

The following activities were reported by Phillip Bowers to have been performed / observed last night by Maintenance Personnel

1. Made sure valve stem was up (normal position)
2. Backed out actuator stem (force was required, used channel locks)
3. Removed indicator plate and reinstalled plate in proper position
4. Positioned coupling down approximately 4 threads from end on actuator end (to ensure that if the actuator stem had tapered threads they would not cause a problem)
5. Installed coupling. Observed approximately 1/8 inch gap on actuator end of coupling and approximately 1/16 inch gap on the valve end of coupling. Immediately recognized this to be an incorrect fit of the coupling and removed the coupling.

12/30/99

Root Cause Team Turnover from Day Shift to Night Shift
1. At approximately 06:45 this morning the FIP Team was notified that the new coupling would not fit correctly. Specifically, there was approximately 1/8 inch gap between the halves at the actuator end and approximately 1/16 inch gap at the valve end.
2. Decision was made to remove the actuator stem
3. Maintenance personnel removed the actuator stem and carried it to the machine shop for close inspection. Inspection results:
> approximately .032" run-out in the shaft on the threaded portion of the coupling end
> Machinist attempted to run a 3/4 inch x 20 threads/inch die onto the actuator shaft with no success
> Root Cause Team personnel attempted to install the coupling in the shop using a spare valve stem, the actual actuator stem, and the fabricated coupling with each attempt ending with a wide variance in the gap between the halves at the top and bottom.
4. Maintenance personnel performed field measurements and discovered that the actuator stem uld be shortened by as much as 1 7/8 inches without affecting adequate thread engagement side the actuator. Machinist cut 1 inch off the shaft to remove the run-out portion and cut new threads with a 3/4 x 20 TPI die.
5. Root Cause Team personnel coupled the spare valve stem to the actuator stem and observed very uniform gap all the way around the coupling. Maintenance personnel observed this proper fit in order to have a mental reference when installing in the field. Maintenance was instructed to install reconditioned actuator stem and new coupling.

Functional Testing Recommended by the Root Cause Team following repair of 3HP-14 to verify root cause has been corrected.

1. Perform 20 strokes of valve.
2. Perform feeler gauge check of coupling on all four corners and record measurements
3. Remove bolts in coupling and remove one half of coupling (CAUTION: Leave one half in place. This is important to prevent another set up of limit switch)
4. Inspect threads on coupling half removed for any signs of deformation due to an overload condition. Report any damage
5. Reinstall the coupling half that was removed.
6. Perform feeler gauge check of coupling on all four corners and record measurements (should be within plus or minus .003")
7. Cycle valve two times

****

Failure Modes and Failure Scenario

Using facts derived from field observations, Vendor input, work history, PIP Database, OEDB arches, mockups, and work experience, the FIP Team developed a list of potential failure odes. Each failure mode was evaluated for credibility.

Potential Failure Modes

Problem Investigation Process
Oconee Nuclear Station

{NOTE: The following potential failure mode data was in table format in the hard copy root cause report. When the report was imported to this PIP Problem Evaluation, the table columns were lost. In the paragraphs below each line represents a column: the first line "List Credible, Potential Failure Modes", the second line "List Information Needed to Evaluate", the third line "Does Evaluation Support or Refute Failure Mode? Explain", the forth line "Actual Failure Mode (yes/no)", the fifth line "Contributing Factor (yes/no)".}

Electrical Failure

E1-Torque switch
Functional test
Refute: Torque switch verified functional
NO
NO

E2-Power Supply
As found condition
Refute: Found motor running
NO
NO

E3-Control
As found condition
Refute: Valve position will change
NO
NO

'echanical Failure

M1-Actuator

M1.1 Stem nut not staked
As found condition
Refute: Actuator is driving stem
NO
NO

M2-Valve

M2.1 Internal binding
Stroke by hand
Refute: Could be easily stroked by hand
NO
NO

M2.2-Stem

M2.2.1-Threads failed on end connected to coupling
Visual inspection
Refute: Threads were intact
NO
NO

'M2.2.2-Stem height changed due to separation from disc assembly
.lve travel check
.efute: Normal travel of 1 1/8 inch
NO
NO

Problem Investigation Process
Oconee Nuclear Station

M2.2.3-Stem height changed due to lower plug retaining nut loosening
Valve travel check
Refute: Normal travel of 1 1/8 inch
NO
NO

M2.3-Anti rotation device (guide)

M2.3.1-Piece broken off
Visual inspection
Refute: No significant amount of metal missing
NO
NO

M2.4-Coupling

M2.4.1.A-Thread insertion depth (coupling #1)
Visual inspection
Refute: Thread damage on actuator stem indicated adequate insertion
NO
NO

M2.4.1.B-Thread insertion depth (coupling #2)
Visual inspection
efute: Thread damage on valve stem indicated adequate insertion
O
NO

M2.4.2.A-Thread engagement (coupling #1)
Visual inspection
Supports: Actuator stem threads did not match those of coupling
YES
NO

M2.4.2.B-Thread engagement (coupling #2)
Visual inspection
Supports: Valve stem threads did not mate (see M2.4.4.B)
YES
NO

M2.4.3.A-Wrong material (coupling #1)
Work history
Refute: Vendor supplied material w/ years of service
NO
NO

M2.4.3.B-Wrong Material (coupling #2)
Stock code / material spec for replacement material
Refute: Per stock code information , material is 416 SS as specified on drawing
NO
NO

M2.4.4.A-Not clamped tight enough (coupling #1)
As-found condition
Indeterminate: Extensive damage to threads prevented inspection

Indeterminate
Indeterminate

M2.4.4.B-Not clamped tight enough (coupling #2)
Visual inspection of new coupling fit
Supports: Lack of parallelism between coupling halves
YES
NO

M2.4.5.A-Inadequate engagement with anti rotation device (coupling #1)
As found condition
Refutes: Anti rotation device found engaged
NO
NO

M2.4.5.B-Inadequate engagement with anti rotation device (coupling #2)
As found condition
Refutes: Anti rotation device found disengaged but attributed to valve stem separating from coupling
NO
NO

After careful evaluation, the FIP Team developed the following failure scenarios.

Failure Scenario for the 12/27/99 event:

he actuator stem for 3HP-14 pulled from the coupling that connects it to the valve stem due to a mismatch in bore diameter and thread size. The actuator stem outside diameter was 3/4 inch with 20 threads per inch and the coupling inside diameter was 13/16 inch with 16 threads per inch.
It could not be determined exactly when this mismatch first occurred. The vendor (Limitorque) reports the actuator stem supplied for 3HP-14 was 13/16 in diameter with 16 threads per inch. It is possible the actuator stem was modified at some point but no details were discovered from work history that would prove this to be the case. The valve has been worked on several times over the years with the most recent work being performed on 2/16/97 per Work Order 97014041. According to details in task 09 of the work order, the stem coupling was retapped at that time. However, the details in the work order were not specific enough to determine how or why the stem coupling was retapped or if it was a smaller size.

Failure Scenario for the 12/29/99 event:

The valve stem for 3HP-14 pulled from the coupling that connects it to the actuator stem due to inadequate clamping force caused by a bent actuator stem.
The actuator stem had approximately .030 inches of run-out on the first inch of the threaded portion. This bend in the stem prevented the coupling halves from clamping in a parallel fashion and created minimal thread engagement on the valve stem side of the coupling.
It could not be determined when or how the actuator stem was bent but it was speculated to have existed for a long period of time and occurred during an event when the coupling guide slot became disengaged from the yoke guide (anti rotation device). During this scenario, the valve stem separated from the coupling causing the actuator stem to retract enough to cause the anti rotation device to disengage. When disengagement occurred, the coupling spun around making contact with the valve yoke guide thus bending the stem.
e bent stem was not discovered following the 12/27/99 failure because a measurement of stem un-out was not taken and coupling fit was dependent on the skill of the craft.
This failure mechanism did not exist prior to the 12/27/99 event due to the oversized bore on the actuator end of the coupling (13/16 inch coupling ID Vs 3/4 inch shaft OD). When the new

coupling was installed on 12/27/99, with the correct ID of 3/4 inch, the bent actuator stem prevented the coupling halves from mating properly thus producing a weak connection at the valve stem.

****

Cause Determination

The root cause for the failure of 3HP-14 on 12/27/99 was determined to be:

The actuator stem for 3HP-14 pulled from the coupling that connects it to the valve stem due to a mismatch in bore diameter and thread size. The actuator stem outside diameter was 3/4 inch with 20 threads per inch and the coupling inside diameter was 13/16 inch with 16 threads per inch.

The root cause for the failure of 3HP-14 on 12/29/99 was determined to be:

The valve stem for 3HP-14 pulled from the coupling that connects it to the actuator stem due to inadequate clamping force caused by a bent actuator stem.

****

Benchmarking

A search was conducted of OEDB using search criteria "valve" and "anti" and "rotation" and "stem". This resulted in 73 hits. The brief descriptions for these events were reviewed with two items being identified as somewhat related.
These two items were (listed below) were reviewed in detail.

Item No: 90-002840
OEP No: OE 3801
RHR/LPCI Injection Valve Failure

Item No: 84-021538
OEP No: O & MR 197
Failure of Anchor/Darling Globe Valve anti rotation devices

A search was conducted of OEDB using search criteria "valve" and "stem" and `failure". This resulted in 69 hits. The brief descriptions for these events were reviewed and none of the items were determined to be relevant to the failures of 3HP-14.

(Reference Attachment 5 for Benchmarking Information)
****

Operability, Generic Applicability, and Transportability

An operability was not required for this event.

There are two other valves in the plant that utilize this same coupling design employed by 3HP-14 and are therefore considered susceptible to the same failures. These two valves are 1HP-14 and 2HP-14.
Visual inspections of the couplings for both of these valves were performed on 12/31/99. These inspections revealed both couplings to be connected with no visible evidence of degradation. It should be noted that these inspections were very brief. High radiation levels in the area of the valves prevented detailed inspections. The proposed corrective actions will address future detailed inspections to be performed during unit shutdown or low power operation.

Problem Investigation Process
Oconee Nuclear Station

****

Planned Corrective Actions

Remedial/Immediate Corrective Actions:
> Remove the run-out from the actuator stem by removing approximately 1 inch of the threaded portion (coupling end). Then, rethread the stem end to match the diameter and threads per inch of the coupling. (This item has been completed. It should be noted that the length of the actuator stem was verified to be of sufficient length to facilitate as much as 1-7/8 inches without interfering with the proper operation of the valve/actuator).
> Fabricate (and install) a new coupling that has the correct diameter and threads per inch to match the actuator stem and valve stem. (This item has been completed)
> Test valve to ensure coupling is adequately connected and verify no internal binding of the valve/actuator exists. (Test criteria: Stroke the valve every 6 minutes until 10 full cycles are completed {20 strokes}, perform feeler gauge checks and record measurements, remove bolts in coupling and remove one half of coupling, inspect threads on coupling for signs of deformation, reinstall the coupling half, perform feeler gauge check of coupling on all four corners and verify halves are parallel, cycle valve two more times, monitor amp readings during stroking (MPM) to verify no binding is present. (This has been completed. The valve stroked successfully approximately 20 times and there was no degradation observed on the coupling/stem threads. In addition MPM data taken during the strokes indicated no evidence of internal binding.)

Interim Corrective Actions:
> Perform visual inspection of the condition of the couplings for 1HP-14 and 2HP-14 to ensure ey are connected and there are no external signs of degradation. (This has been completed ıth no problems being discovered. It should be noted that this was a limited scope inspection, with the coupling assembled, due to radiation levels in the area at full power and the valve being in service).

Corrective Actions to Prevent Recurrence:
1. Perform detailed inspection of the coupling for 1HP-14 during unit shutdown or low power operation (power level equal to or less than 30 %). Inspection to include:
> Disassemble and removal of coupling
> Verify there are no bends in the valve and actuator stem in the coupling area by checking straightness (i.e. measure stem run-out)
> Verify diameter and threads on stems match those of the coupling
> Verify there is no thread damage to either the stems or coupling
> Reinstall the coupling

2. Perform detailed inspection of the coupling for 2HP-14 during unit shutdown or low power operation (power level equal to or less than 30 %). Inspection to include:
> Disassemble and removal of coupling
> Verify there are no bends in the valve and actuator stem in the coupling area by checking straightness (i.e. measure stem run-out)
> Verify diameter and threads on stems match those of the coupling
> Verify there is no thread damage to either the stems or coupling
> Reinstall the coupling

3. Revise MP/0/A/1200/027A (HP-14 - Valve - Fisher - YY Type - Disassembly and Reassembly) to include installation details for the coupling to ensure acceptable configuration and assembly.

4. Once detailed inspections are complete for 1HP-14 and 2HP-14 , ensure proper stock codes are set up for couplings for all three units.
****

Problem Investigation Process
Oconee Nuclear Station

**

List of Attachments

1. Fault Tree
2. Miscellaneous Photos of Coupling Assembly
3. Root Cause Team Organization, Pre-job Briefing
4. Root Cause Team Notes
5. Interviews
6. Test Data and Coupling Installation Instructions
7. Work History
8. OEDB Information
9. Miscellaneous Data (Copies of PIPs 99-5270 and 99-5278, Schedule Information, DBD Description for 3HP-14, Drawings, Copy of MP/0/A/1200/027A, rev 8)

Duke Power Company Oconee Nuclear Station

Page 3

Originated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

OEDB Comments:

The results of the OEDB search are detailed in the Failure Analysis Report in the Problem Evaluation section of this PIP.

Originated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Remarks Comments:

Signature Type	Indiv	Team	Group	Date
Due Date:	01/26/2000			
Accepted By:	BGD7309	BGD7309	MSE	12/30/1999
Assigned To:	MWA7315	TDM8270	MSE	01/20/2000
Ready For Approval:	MWA7315	TDM8270	MSE	01/24/2000
Approval Assigned To:	TDM8270	TDM8270	MSE	01/24/2000
Approved By:	TDM8270	LJA2713	MSE	01/24/2000
Concurrence Assigned To:	MWA7315	RTB7310	SRG	01/24/2000
Concurrence By:	TEC7318	RTB7310	SRG	02/21/2000

Corrective Actions

CA Seq. No: 1

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
MSE	Closed	MSE	F2a	B9	B4c

Proposed Corrective Action:

Last Updated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Work Order 98232044 has been initiated to perform detailed coupling inspection as described below. This Proposed CA is for MSE Valve Engineering to evaluate and document inspection results.

Problem Investigation Process
Oconee Nuclear Station

1. Perform detailed inspection of the coupling for 1HP-14 during unit shutdown or low power operation (power level equal to or less than 30 %). Inspection to include:
> Disassemble and removal of coupling
> Verify there are no bends in the valve and actuator stem in the coupling area by checking straightness (i.e. measure stem run-out)
> Verify diameter and threads on stems match those of the coupling
> Verify there is no thread damage to either the stems or coupling
> Reinstall the coupling

Originated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	MWA7315	TDM8270	MSE	01/19/2000
Approval Assigned To:	TDM8270	TDM8270	MSE	01/19/2000
Approved By:	TDM8270	LJA2713	MSE	01/24/2000

General: Outage: 1EOC19 Mode:

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: B9Status: Closed Due Date: 06/24/2000
Minor Modification ONOE-14730 was implemented on 2/25/2000. This minor modification performed a complete inspection of the stems and the stem connector on 1HP-14. All components were found to be in satisfactory condition. However, for design improvement, the two bolt stem connector was replaced with a four bolt stem connector. The actuator stem O.D is 0.817" with 16 threads per inch. Their is a reduction in the stem diameter immediately above the threads on the smooth portion to 0.750". The new stem connector had a very good fit to the actuator and the valve stem. The torque was applied to the stem connector bolting in an uniform manner and the gap at each end of the coupling was within acceptable limits. There was a 0.005" gap at the end of the coupling where the anti-rotation groove is contained and there was an 0.008" gap at the opposite end. The implementation of the minor mod was successful and no further work is required for 1HP-14.

Last Updated By: DDK8372: KING, DAVID D Team: DAK7363 Group: MSE Date: 02/28/2000

This PIP corrective action must be classified as a management exception since completion requires either a forced or scheduled unit outage. Unit 1 is currently in a forced outage and work is scheduled to be completed by 3/1/2000. The current corrective action schedule completion date is satisfactory

Originated By: DDK8372: KING, DAVID D Team: DAK7363 Group: MSE Date: 02/21/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	06/24/2000			
Accepted By:	DAK7363	DAK7363	MSE	02/01/2000
Assigned To:	DDK8372	DAK7363	MSE	02/01/2000

Problem Investigation Process
Oconee Nuclear Station

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	DDK8372	DAK7363	MSE	02/28/2000
Approval Assigned To:	DAK7363	DAK7363	MSE	02/28/2000
Approved By:	DAK7363	DAK7363	MSE	02/28/2000
Concurrence By:	RWVASSEY	RTB7310	SRG	02/29/2000

CA Seq. No: 2

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
MSE	Closed	MSE	F2a	B9	B4c

Proposed Corrective Action:
Last Updated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Work Order 98232042 has been initiated to perform detailed coupling inspection as described below. This Proposed CA is for MSE Valve Engineering to evaluate and document inspection results.

Perform detailed inspection of the coupling for 2HP-14 during unit shutdown or low power operation (power level equal to or less than 30 %). Inspection to include:
Disassemble and removal of coupling
Verify there are no bends in the valve and actuator stem in the coupling area by checking straightness (i.e. measure stem run-out)
> Verify diameter and threads on stems match those of the coupling
> Verify there is no thread damage to either the stems or coupling
> Reinstall the coupling

Originated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	MWA7315	TDM8270	MSE	01/19/2000
Approval Assigned To:	TDM8270	TDM8270	MSE	01/19/2000
Approved By:	TDM8270	LJA2713	MSE	01/24/2000

General: Outage: 2EOC18 Mode:

Other Tracking Processes
Type Number TextM1 #1

Actual Corrective Action:
Actual CAC: B9Status: Open Due Date: 06/24/2000
This PIP corrective action needs to be classified as a management exception since completion requires either a forced or scheduled unit ıtage. Unit 2 is scheduled for a refueling outage 2EOC-18 in June of 2001, therefore this PIP correction action should be scheduled to ė completed in September of 2001.

Problem Investigation Process
Oconee Nuclear Station

Originated By: DDK8372: KING, DAVID D Team: DAK7363 Group: MSE Date: 02/21/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	06/24/2000			
Accepted By:	DAK7363	DAK7363	MSE	02/01/2000
Assigned To:	DDK8372	DAK7363	MSE	02/01/2000
Mgt Excepted By:	RWVASSEY	RTB7310	SRG	03/06/2000

CA Seq. No: 3

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
MSE	Closed	MSE	F2a	A3	B4c

Proposed Corrective Action:
Revise MP/0/A/1200/027A (HP-14 - Valve - Fisher - YY Type - Disassembly and Reassembly) to include installation details for the coupling to ensure acceptable configuration and assembly.

Originated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	MWA7315	TDM8270	MSE	01/19/2000
Approval Assigned To:	TDM8270	TDM8270	MSE	01/19/2000
Approved By:	TDM8270	LJA2713	MSE	01/24/2000

General: Outage: N/A Mode:

Other Tracking Processes
Type Number TextM1 #1

Actual Corrective Action:
Actual CAC: A3Status: Closed Due Date: 06/24/2000
MP/)/A/1200/27A - Valve - Fisher Model YY procedure has been changed to incorporate the operator/valve coupling inspection and installation guidelines. This procedure change incorporated all the lessons learned from the FIP process.

Originated By: JMA7313: ALEXANDER, JOHN M Team: DAK7363 Group: MSE Date: 03/27/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	06/24/2000			
Accepted By:	DAK7363	DAK7363	MSE	02/01/2000
Assigned To:	JMA7313	DAK7363	MSE	02/01/2000
Mgt Excepted By:	RWVASSEY	RTB7310	SRG	03/06/2000
Ready For Approval:	JMA7313	DAK7363	MSE	03/27/2000

Problem Investigation Process
Oconee Nuclear Station

Signature Type	Indiv	Team	Group	Date
Approval Assigned To:	DAK7363	DAK7363	MSE	03/27/2000
Approved By:	DAK7363	DAK7363	MSE	03/28/2000
Concurrence Assigned To:	MWA7315	RTB7310	SRG	03/28/2000
Concurrence By:	MWA7315	RTB7310	SRG	03/29/2000

CA Seq. No: 4

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
MSE	Closed	MSE	F2a	J	O2c

Proposed Corrective Action:

Once detailed inspections are complete for 1HP-14 and 2HP-14, ensure proper stock codes are set up for couplings for all three units.

Originated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	MWA7315	TDM8270	MSE	01/19/2000
Approval Assigned To:	TDM8270	TDM8270	MSE	01/19/2000
Approved By:	TDM8270	LJA2713	MSE	01/24/2000

General: Outage: N/A Mode:

Other Tracking Processes

Type Number Text

Actual Corrective Action:

Actual CAC: B3Status: Open Due Date: 06/24/2000

This PIP corrective action needs to be classified as a management exception since completion requires corrective actions numbers 1 & 2 to be completed. The unit 2 inspection will be the last unit to be completed in June of 2001, therefore this PIP correction action should be scheduled to be completed in December of 2001.

Originated By: DDK8372: KING, DAVID D Team: DAK7363 Group: MSE Date: 02/21/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	06/24/2000			
Accepted By:	DAK7363	DAK7363	MSE	02/01/2000
Assigned To:	DDK8372	DAK7363	MSE	02/01/2000

Problem Investigation Process
Oconee Nuclear Station

CA Seq. No: 5

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	MSE	F2a	A3	B4c

Proposed Corrective Action:
The following Proposed CA did not come about as a result of the Root Cause Investigation of 3HP-14. It is being added, at the request of OPS (Jean Miller), to incorporate lessons learned during recovery from 3HP-14 failure.

OP/x/A/1900/14, Loss of HPI Makeup or Letdown, could be more detailed for the mitigation of this event. The Operations crew had to make decisions that were not adequately covered in the AP. Example: The shutdown rate of 2%/minute was not adequate to prevent pressurizer level from exceeding 375 inches. The crew had the ability to re-open 3HP-5 and that action is what prevented exceeding 375" in the pressurizer. There are other compensatory actions that are not captured in the procedure that the shift used for mitigation such as constant makeup to the LDST, placing the delithiating demineralizer in service, etc. Preston Gillespie and Barry Honeycutt can provide more insight for this CA. Jean Miller has agreed to accept this CA.

Last Updated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Originated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	MWA7315	TDM8270	MSE	01/19/2000
Approval Assigned To:	TDM8270	TDM8270	MSE	01/19/2000
Approved By:	TDM8270	LJA2713	MSE	01/24/2000

General: Outage: Mode:

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: Status: Open Due Date: 06/24/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	06/24/2000			
Accepted By:	MAP7314	HRL7353	OPS	01/24/2000
Assigned To:	DLG7312	KCM2861	OPS	02/01/2000

CA Seq. No: 6

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
RGC	Closed	MSE	F2a	J	B4c

Proposed Corrective Action:

Problem Investigation Process
Oconee Nuclear Station

The following Proposed CA did not come about as a result of the Root Cause Investigation of 3HP-14. It is being added, at the request of OPS (Jean Miller), to provide clarification on RCS leakage in respect to SLC 16.5.10.

SLC 16.5.10 says: "Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system, shall be <= 30gpm when added to RCS LEAKAGE.
Does this SLC apply to RCP seal leakage only? If so, please clarify the part of the statement that says "and system valves". If the SLC does apply to any system valves, which valves? Please explain in the bases.
Regulatory Compliance (Larry Nicholson) has agreed to accept this CA.

Last Updated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Originated By: MWA7315: ADDIS, MILTON W Team: TDM8270 Group: MSE Date: 01/19/2000

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	MWA7315	TDM8270	MSE	01/19/2000
Approval Assigned To:	TDM8270	TDM8270	MSE	01/19/2000
Approved By:	TDM8270	LJA2713	MSE	01/24/2000

General: Outage: Mode:

Other Tracking Processes
<u>Type</u> <u>Number</u> <u>Text</u>

Actual Corrective Action:
Actual CAC: Status: Open Due Date: 06/24/2000

Signature Type	Indiv	Team	Group	Date
Due Date:	06/24/2000			
Accepted By:	RVGAMBRE	LEN2127	RGC	01/27/2000
Assigned To:	NTC7312	LEN2127	RGC	01/27/2000

Final and Overall PIP Approval

Responsible Group: OPS Status: Screened

Signature Type	Indiv	Team	Group	Date
Assigned To:			OPS	12/28/1999

Problem Investigation Process
Oconee Nuclear Station

Any Supplemental Concurrence Signatures Above Do Not Affect PIP Closure.

Microfilm Roll / Frame: /

Closure Document Type **Closure Document No**

Attachments

Generic Applicability

Responsible Group: OEA Status: Closed

GO PIP No:

Assessment Remarks:

No GA required, this issue is specific to ONS HP-14.

Originated By: LAH7368: HENTZ, LEE A Team: JWP7322 Group: OEA Date: 02/24/2000

Signature Type	Indiv	Team	Group	Date
Assigned To:			OEA	12/28/1999
Ready For Approval:	LAH7368	JWP7322	OEA	02/24/2000
Approval Assigned To:	JWP7322	JWP7322	OEA	02/24/2000
Approved By:	LAH7368	JWP7322	OEA	02/24/2000

Failure Prevention Investigation

Quality of CA: Quality of Cause: Resp Group: SRG Status: Closed

Special Codes:

N5

Comments

Signature Type	Indiv	Team	Group	Date
Assigned To:			SRG	12/28/1999
Ready For Approval:	RWVASSEY	RTB7310	SRG	02/21/2000
Approval Assigned To:	RTB7310	RTB7310	SRG	02/21/2000
Approved By:	RWVASSEY	RTB7310	SRG	02/21/2000

Remarks

No Remarks for this PIP.

Maintenance Rule

Responsible Group: MSE Status: Closed

Problem Investigation Process
Oconee Nuclear Station

Maintenance Rule SSC

SSC	Description	Risk Significant	Primary System
HPI	High Pressure Injection System	Yes	Yes

Equipment Group: M02
Applicable Unit: Unit 3
Functional Failure: Yes MPFF: Yes Repetitive MPFF: No

Functional Failure Comments:

The failure of valve 3HP-14 (separation of actuator and valve stems due to coupling failure) resulted in the loss of HPI MR function HPI.06, "Provide RCS inventory control, provide RCS make-up water to make up for contraction in RCS volume and provide letdown capability when RCS volume increases". Refer to Problem Identification and Problem Evaluation sections for details.

Originated By: PVF1489: FISK, PAUL V Team: SDC3511 Group: MSE Date: 02/11/2000

MPFF Comments:

he Problem Evaluation section of this PIP identifies the following root cause:

"Root cause determination for the December 27 event;

The actuator stem for 3HP-14 pulled from the coupling that connects it to the valve stem due to a mismatch in bore diameter and thread size. The actuator stem outside diameter was 3/4 inch with 20 threads per inch and the coupling inside diameter was 13/16 inch with 16 threads per inch."

This is considered maintenance preventable in that an incorrect size (thread diameter and pitch) coupling had been installed and subsequently failed.

Originated By: PVF1489: FISK, PAUL V Team: SDC3511 Group: MSE Date: 02/11/2000

Repetitive MPFF Comments:

Reactor Trip: No Safety System Actuation: No Loss of Heat Decay Removal: No
Force Outage Rate or Plant Transient: Yes Loss Of Spent Fuel: No

Comments:

Signature Type	Indiv	Team	Group	Date
Assigned To:	PVF1489	SDC3511	MSE	01/09/2000
Ready For Approval:	PVF1489	SDC3511	MSE	02/11/2000

Problem Investigation Process
Oconee Nuclear Station

Signature Type	Indiv	Team	Group	Date
Approval Assigned To:	SDC3511	SDC3511	MSE	02/11/2000
Approved By:	SDC3511	SDC3511	MSE	02/14/2000

End of the Document for PIP No:	O-99-5270
The status of this PIP is:	Screened
The duration of this PIP was:	3 days

D CRITICAL TASK DESCRIPTION OF CTs BASED UPON *MAINTAIN ACCEPTABLE LIMITS OF RADIATION RELEASES DUE TO SGTR INDUCED RB BYPASS*

1.

MAINTAIN PRESSURIZER LEVEL

Fulfillment of this CT requires the following:

Increasing MU flow and minimizing letdown in an attempt to maintain pressurizer level ≥ [normal pressurizer level]. If this is not possible, then initiate HPI as necessary in an attempt to maintain pressurizer level ≥ [plant specific low pressurizer level]

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT when a SGTR has occurred and the reactor has not tripped (trip setpoint not reached; plant being controlled at power).

2.0 ASSOCIATED GEOG BASES

If possible, reduce power as quickly as possible, but in a controlled manner, to well within the turbine bypass system capacity before tripping the reactor to prevent lifting of the MSSVs. This includes cases where maximum MU or HPI flow and letdown isolation are required to keep up with the tube leak and maintain pressurizer level. Power reduction is intended to minimize atmospheric radiation releases due to SG safety valve operation. Also, if a reactor trip can be averted through controlled operations, then ability to mitigate the transient is expected to be enhanced as normal transition from power operations to a controlled cooldown occurs.

3.0 GEOG SECTION AND STEP REFERENCE

GEOG Section	Applicable Steps
SGTR	2.1 and 2.2

4.0 CUES

- Main steam line radiation alarm
- Pressurizer low level alarm
- Verbal alert by plant staff that a SGTR is occurring and the reactor has not tripped
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of MU/HPI pump controls
- Operation of MU/HPI valve controls
- Operation of letdown valve controls
- [plant specific performance indicators]

6.0 FEEDBACK

- Pressurizer level
- Letdown flow
- MU/HPI flow
- Verbal alert by plant staff of pressurizer level status
- [plant specific feedback]

2.

MINIMIZE SCM

Fulfillment of this CT requires the following:

RCS TEMPERATURE > 500°F:

Depressurize RCS as much a possible without violating the SCM curve

RCS TEMPERATURE ≤ 500°F:

Maintain RC pressure as low as possible without violating the SCM or RCP NPSH curves

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT when a SGTR occurs.

2.0 ASSOCIATED GEOG BASES

Except when RCP NPSH limits are applicable and are more restrictive, RCS pressure should be maintained close to, but above, the minimum SCM to minimize RCS-SG differential pressure. The reason for minimizing RCS-SG differential pressure is to reduce the leak flowrate from primary to secondary to as low as possible. Therefore, this procedure (minimizing SCM) is desirable whenever possible during a cooldown with a SGTR.

Reducing the leak flowrate from the RCS to the secondary side of a SG with an impaired steam system (e.g., weeping MSSV, MSL leak, etc.) is expected to lead to lower integrated radiation releases from the impaired system. Also, if the level of the leaking SG can be maintained within normal operating limits, then the SG will

remain available for continued use during the cooldown, thus enhancing the transient mitigation capability of the plant.

3.0 GEOG SECTION AND STEP REFERENCE

GEOG Section	Applicable Steps
SGTR	7.6
	8.0

4.0 CUES

- SCM meter and associated alarms
- SPDS displays and associated alarms
- Verbal alert by plant staff that SCM is excessive
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of MU/HPI pump and valve controls
- Operation of normal or auxiliary spray valve controls
- operation of PORV and/or pressurizer vent valve controls
- [plant specific performance indicators]

6.0 FEEDBACK

- SCM meter and/or plant SPDS
- RCS pressure and temperature
- MU/HPI pump and valve status indications
- Normal and auxiliary spray valve status indications
- PORV and pressurizer vent valve status indications
- Verbal indication by plant staff that SCM is being controlled at a minimum value
- [plant specific feedback]

3.

REDUCE STEAMING/ISOLATE AFFECTED SG (USE SG DRAINS IF AVAILABLE)

Fulfillment of this CT requires the following:

If only the radiation TRACC limit is approached, then open SG drains, if available, and reduce steaming of the affected SG to minimum required to prevent SG level > [plant specific SG level].

If BWST TRACC limit is approached <u>or</u> a combination of TRACC limits exists <u>or</u> alternate actions will not prevent TRACC, then the affected SG(s) should be isolated by:

> Closing [plant specific list of valves] FW valves and all flow paths from the affected SG <u>except</u> for TBVs/ADVs and SG drains, if available. Use drains, if available, to maintain SG level < [high SG level that prevents water entering steam annulus] while RC pressure > 1000 psig.
>
> Maintain RCS pressure < 1000 psig by using pressurizer spray, pressurizer and high point vents, PORV, letdown and, if necessary, TBVs/ADVs. When RCS pressure is maintained < 1000 psig, then fully isolate the affected SG by closing remaining isolation valves.

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT when TRACC limits are approached.

2.0 ASSOCIATED GEOG BASES

When the radiation TRACC limit is approached, reducing the steaming rate will reduce the radiation release rate from the affected SG(s). Use of drains, if available, will allow for reducing steaming rates by removing inventory by other means than

steaming. Actions to use drains (in combination with SG pressure control) are intended to prevent lifting of MSSVs (and reduce the possibility of MSSVs failing to reseat or sticking open). Reducing the steaming rate and preventing MSSV lifts/failures is expected to reduce radiation releases.

If the SG drains are unavailable, are not used, or are otherwise incapable of preventing violation of a TRACC, isolate the affected, or most affected, SG by closing all steam, feed, and drain lines to that SG. This is expected to reduce radiation releases due to steaming.

3.0 GEOG SECTION AND STEP REFERENCE

GEOG Section	Applicable Steps
SGTR	3.3
	7.2
	9.1, 9.3.a-e
	10.3, 10.4, 10.5 and 10.6

4.0 CUES

- Alert by plant staff that integrated radiation releases are approaching [plant specific limit]
- [SG high level] alarm
- Alert by plant staff that the BWST is approaching [plant specific low level]
- SPDS displays and associated alarms
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of SG drain valve controls
- Operation of affected SG(s) steam and FW isolation valve controls
- Operation of TBV/ADV controls
- Operation of pressurizer spray and vent valve controls
- Operation of PORV controls
- Operation of high point vent controls
- Operation of letdown valve controls
- [plant specific performance indicators]

6.0 FEEDBACK

- SG(s) level and pressure
- RCS pressure
- MFW/EFW flow
- MFW/EFW pump and valve status indication
- Affected SG(s) steam valve status indication
- TBV/ADV status indication
- SG drain valve status indication
- Pressurizer spray and vent valve status indications
- PORV and high point vent valve status indication
- Letdown valve status indication
- Verbal indication from plant staff of affected SG(s) steaming rate
- [plant specific feedback]

B.2 CRITICAL TASK DESCRIPTION OF CTs BASED *UPON ESTABLISH/MAINTAIN ADEQUATE DECAY HEAT FLOW FROM CORE TO AVAILABLE HEAT SINKS, CATEGORY 2: "MITIGATE EXCESSIVE HEAT FLOW TO AVAILABLE HEAT SINKS"*

1.

CONTROL SG PRESSURE TO:
MAINTAIN RC TEMPERATURE CONSTANT

OR TO:
MAINTAIN APPROPRIATE COOLDOWN RATE

Fulfillment of this CT requires the following:

Control of turbine bypass system using TBVs/ADVs manually to maintain desired header pressure if header pressure is not properly controlled following a reactor trip.

In the event of a station black out, RC temperature is to be maintained constant (do not initiate cooldown with TBVs/ADVs) and non-essential steam loads are to be isolated.

If following initiation of the [secondary plant protection system] automatic isolation valves are not in proper alignment, then align valves manually.

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT for situations where excessive RCS cooling may be occurring due to poor steam pressure control immediately following reactor trip.

2.0 ASSOCIATED GEOG BASES

Following a reactor trip, SG pressures should be controlled using the TBVs/ADVs to prevent initial RCS cooldown. In the event that [secondary plant protection system] actuation is necessary the operator provides appropriate operation of this system for its control and plant stabilization. Proper control of SG pressures leads to enhanced transient mitigation capability of the plant as normal heat removal systems remain available.

If a station blackout has occurred, then secondary steam pressure should be controlled to prevent RCS cooldown and contraction. A predetermined list of potential secondary steam flow paths from the SGs should be used to identify and isolate non-essential open steam flow paths.

3.0 GEOG SECTION AND STEP REFERENCE

GEOG Section	Applicable Steps
VSSV	7.0
	10.b.2 and 10.b.3
	13.b

4.0 CUES

- SPDS displays and associated alarms
- [secondary plant protection system] alarms
- Verbal alert by plant staff that cooldown rate is excessive
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of TBV/ADV controls
- Operation of [secondary plant protection system] controls
- Operation of associated steam valve controls
- [plant specific performance indicators]

6.0 FEEDBACK

- RC temperature and pressure
- SG pressure
- Verbal notification by plant staff of cooldown rates
- [secondary plant protection system] status indication
- Associated steam valve status indications
- [plant specific feedback]

3.

ISOLATE OVERCOOLING SG(s)

Fulfillment of this CT requires the following:

Operation of [secondary plant protection system] to isolate affected (overcooling) SG(s)

and/or

manual isolation of overcooling SG(s) by closing all steam and FW valves to the affected SG(s).

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT when excessive primary to secondary heat transfer occurs and mitigation requires isolation of affected SG(s).

2.0 ASSOCIATED GEOG BASES

If the overcooling SG has been identified then that SG should be isolated, otherwise both SGs should be isolated. Isolating a SG means to stop all FW flow (MFW and AFW) and steam flow (e.g., close TBVs, ADVs, steam supply to FW pumps, MSIVs etc.). FW flow should be maintained to the unaffected SG and cooling stabilized using the unaffected SG.

Isolation of a SG or both SGs should always follow a logical progression of increasingly more drastic attempts to isolate the SG. For example, if the overcooling is not severe it may be possible to close both the TBVs and ADVs as well as the auxiliary steam valves thus isolating the SG. If this does not work, then for those plants which have main steam isolation valves, the main steam isolation valve should then be closed. For severe overcooling situations, [secondary plant protection system] will likely actuate.

Failing to mitigate excessive primary to secondary heat transfer can lead to degradation of the transient mitigation capability of the plant if SGs become inoperable.

4.0 CUES

- SPDS displays and associated alarms
- [secondary plant protection system] alarms
- Verbal alert by plant staff that primary to secondary heat transfer is excessive
- [plant specific cues]

5.0 PERFORMANCE INDICATORS

- Operation of associated FW pump and valve controls
- Operation of associated steam valve (included TBVs/ADVs) controls
- Operation of [secondary plant protection system] controls
- [plant specific performance indicators]

6.0 FEEDBACK

- RC temperature and pressure
- SG level and pressure
- Verbal notification by plant staff that primary to secondary heat transfer is appropriate for given plant conditions
- [secondary plant protection system] status indication
- MF pump and valve status indications
- [plant specific feedback]

INITIAL SUBMITTAL

OCONEE EXAM 2000-301
50-269, 270, AND 287/2000-301

JULY 10 - 14, 18, 19, AND 20, 2000

INITIAL SUBMITTAL

SRO ONLY
WRITTEN EXAM

SRO ONLY

Oconee
2000

NRC Copy

QUESTION # 76

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	___	**1**
	Group #	_____	**1**
	K/A #	**000001**	**G2.4.21**
	Importance Rating	_____	**4.3**

Technical Reference(s): **IC-RPS, TS Bases 3.3.1.3**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-RPS #4 & #26**

Question Source:
Bank # ________
Modified Bank # **IC-84**
New ________

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __**X**__

10 CFR Part 55 Content:
55.41 _____
55.43 __X__

Comments:

1 POINT

QUESTION # 76

SRO ONLY

Unit 1 plant conditions:

- Power escalation in progress
- Power = 95% increasing
- Neutron Error = (-) 5%
- Group 7 control rods at 88%
- Diamond CRD out command light "ON"
- SPDS – CSF Subcriticality Alarm actuated why?

ASSUME NO operator action

Which ONE of the following is correct?

The High _____________ RPS trip is designed to terminate the above condition.

A. Flux

B. RCS Pressure

C. RCS Temperature

D. Flux / Flow / Imbalance

000 1 6-24.21

KA D

2.4.2 3.9 | 4.1

Knowledge of system set points associated with EOPs

1 POINT

QUESTION # 76

000001 G 2.4.21 (3.7/4.3) CFR 43.5, 45.12 SRO ONLY PRA 3-23-00

A. In correct – High flux prevents damage to fuel and clad from reactivity excursions to rapid for RCS pressure and temperature instrumentation for the RPS to detect.

B. Correct – Normally the High Flux trip would preclude significant RCS pressure increases however a rod ejection or slow rod withdrawal accident will cause RCS pressure to increase faster than flux/reactor power increases.

C. Incorrect – High reactor outlet temperatures (Th) trips are designed to prevent excessive operating temperatures and prevent exceeding RCS design temperature limits (650°F) thus preventing core temperatures from exceeding the safety analysis during a DBA.

D. Incorrect – Flux/Flow/Imbalance trips are designed to prevent exceeding DNBR limits (1.18) during mismatches between reactor power and RCS flow such as a loss of flow accident at high power due to an electrical failure.

OBJECTIVES

TERMINAL OBJECTIVES

1. Describe the function and correct operation of the Reactor Protective System during all modes of plant operation, including specific limits or precautions associated with the particular component or operation. (T1)

2. Discuss all inputs and signals used in the RPS that affect the specific operation-related functions associated with the RPS. (T2)

3. Describe or identify the purpose of each indicator, control, and indicating light in the RPS that provides operator-related information. (T3)

4. Evaluate normal and off normal system operation and predict RPS and plant response to specific degraded components within the RPS, or components or power supplies that feed the RPS. Analyze the status of the RPS and the plant and develop a plan to return the system to normal. (T4)

ENABLING OBJECTIVES

1. List the two general components that the RPS is designed to protect. (R1)

2. List the two basic methods employed by the RPS to protect the fuel rod clad and RCS. (R2)

3. List all eleven variables in RPS that will initiate a reactor trip, and where applicable, the maximum or minimum allowable setpoint at which trip will occur. (R3)

4. For each trip, choose which of the two general components (fuel clad or RCS) each of the tripping parameters in RPS is designed to protect, and be able to describe the basis of the protection afforded by each. (R4)

5. Explain the following concerning the Shutdown Bypass function in RPS: (R5)

    5.1 The three plant operations for which the SD Bypass function provides the capability to perform.

    5.2 The four normal tripping parameters in RPS that are bypassed when SD Bypass is active.

    5.3 The new tripping parameter that is automatically inserted into RPS when SD Bypass is active, the trip set point set, and the basis for the new trip variable.

19.1 Explain how AC power is delivered to the UV coils and ST relays.

19.2 Describe how each component indicated on the diagram relates to the normal operation of the RPS channel.

20. Given a one-line diagram of the CRD groups power supplies: (R22)

    20.1 Describe how power can be delivered to each CRD Group.

    20.2 List the combinations of CRD breakers that, when tripped, will initiate a reactor trip.

21. Given a one-line diagram of the RPS Logic Relays and contacts, predict the system response to a trip of one or more RPS channel/channels. (R23)

22. Describe the basic procedure for resetting a tripped RPS channel. (R25)

23. Describe how an individual RPS channel can be manually tripped. (R26)

24. Evaluate a failure of the RPS to properly initiate a reactor trip and develop alternate methods to de-energize the CRDs. (R27)

25. Given a copy of PT/1,2,or3/A/600/01 (Periodic Instrument Surveillance) describe the corrective action for an "out of tolerance" condition existing with an RPS channel input. (R28)

26. Given a copy of ITS/SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's. (R29)

27. Apply all ITS/SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R30)

28. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITS/SLC's. (R31)

1) 100% x 1.07 = 107%(Unit 1,2,& 3)

e) With only 3 RCPs operating, design flow rate is 74.7%.

f) For 3 RCP combinations, the trip set point is derived by:

1) 74.7% x 1.07 = 79.9% Full Power(Unit 1,2,& 3)

g) In addition to preventing DNBR going below 1.18 if loss of RCS flow occurs, trip set points are modified if Core Axial Imbalance becomes excessive, to prevent core thermal limits (fuel centerline temperature) from being exceeded.

h) Imbalance considerations will automatically reduce the high flux trip set point based on RCS flow, once imbalance exceeds an acceptable value for normal operation. **(OC-IC-RPS-6)**

3. Power Level Based on the Number of RCPs Running (Flux/Pump) Trip

a) Provides redundant trip protection to prevent DNBR from decreasing below 1.18, by tripping reactor due to loss of reactor coolant pumps.

b) Redundant since flux/pump trip signal is diverse from the flux/flow/imbalance trip signal.

c) This trip signal will also restrict the power level obtainable, based upon the number of operating RCPs.

d) Per Improved Tech Specs, if reactor power is above 2% Full Power, RPS will trip the reactor if two RCPs are lost. The actual setpoint is 1.5%.

4. High RCS Pressure Trip at ≤2355 psig. **(OC-IC-RPS-7)**

a) Improved Tech Specs requires reactor trip before 2355 psig to prevent the RCS from exceeding the safety limit of 2750 psig (110% of design pressure of 2500 psig), for any design transient.

b) The safety limit of 2750 psig is established to maintain the integrity of the RCS and to prevent the release of significant amounts of fission product activity.

c) Normally, high flux trip would preclude significant RCS pressures - however, a rod ejection during startup or a slow rod withdrawal accident at high power, would cause RCS pressure to increase faster than reactor power.

d) Actual RPS trip set point at 2345 psig is for conservatism due to instrument and calibration errors, so that Improved Tech Specs will not be exceeded.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Nuclear Overpower

a. Nuclear Overpower – High Setpoint

The Nuclear Overpower – High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core neutron leakage flux.

The Nuclear Overpower – High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to prevent exceeding acceptable fuel damage limits.

Thus, the Nuclear Overpower – High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower Flux/Flow Imbalance, provide more direct protection. The role of the Nuclear Overpower – High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

The Nuclear Overpower – High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident and the rod ejection accident. By providing a trip during these events, the Nuclear Overpower – High Setpoint trip protects the unit from excessive power levels and also serves to limit reactor power to prevent violation of the RCS pressure SL.

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower – High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Nuclear Overpower – Low Setpoint

Prior to initiating shutdown bypass, the Nuclear Overpower – Low Setpoint trip must be reduced to ≤ 5% RTP. The low power setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that the unit is protected from excessive power conditions when other RPS trips are bypassed.

The setpoint Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. RCS High Outlet Temperature

The RCS High Outlet Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor vessel outlet temperature approaches the conditions necessary for DNB. Portions of each RCS High Outlet Temperature trip channel are common with the RCS Variable Low Pressure trip. The RCS High Outlet Temperature trip provides steady state protection for the DNBR SL.

The RCS High Outlet Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to ensure that a trip occurs before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip is not required to mitigate accidents that create harsh conditions in the RB.

3. RCS High Pressure

The RCS High Pressure trip works in conjunction with the pressurizer and main steam relief valves to prevent RCS overpressurization, thereby protecting the RCS High Pressure SL.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. RCS HIGH PRESSURE (continued)

The RCS High Pressure trip has been credited in the transient analysis calculations for slow positive reactivity insertion transients (rod withdrawal transients and moderator dilution). The rod withdrawal transient covers a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower – High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

The setpoint Allowable Value is selected to ensure that the RCS High Pressure SL is not challenged during steady state operation or slow power increasing transients. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions because the equipment is not required to mitigate accidents that create harsh conditions in the RB.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). Harsh RB conditions created by small break LOCAs cannot affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination.

Exam Question Report

27-Jan-99

Question ID:	IC084	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC084				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-RPS - Reactor Protective System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 4; SRO = 4	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

The RPS High RCS Pressure is designed to protect the reactor against which ONE of the following? (.25)

A) Loss of all FDW.

B) Loss of all RCS flow.

C) Fast rod ejection accident at power.

D) Slow rod withdrawal accident at power.

Answer

D

Lessons

ID	Description
IC-RPS	Reactor Protective System IC-RPS

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

QUESTION # 77

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	___	1
Group #	___	1
K/A #	**00003 A2.01**	
Importance Rating	___	3.9

Technical Reference(s): **AP/1700/15**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **ADM-APG #4**

Question Source:
Bank # ______
Modified Bank # ______
New ___X___

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ______
55.43 __X__

Comments:

1 POINT

QUESTION # 77

SRO ONLY

Unit 3 plant conditions:

INITIAL CONDITIONS:

- Reactor startup in progress
- ECP calculated for Group 6 rods 1-9 at 53%
- Actual CRD position = Group 6 rods 1-9 at 50%
- SUR = 0 dpm
- Critical Data taken by OATC

CURRENT CONDITIONS:

- Group 6 rods 1-7 at 50%
- Group 6 rod 9 at 45%
- Group 6 rod 8 at 30%
- SUR = -.20 dpm

Which ONE of the following describes your directions to the OATC?

A. Insert Group 5 and 6 to 0%.

B. Insert Group 6 rod 1-7 to 45%.

C. Insert all control and safety rods to Group 1 at 50%.

D. Insert ALL control and safety rods to 0% withdrawn.

1 POINT

QUESTION # 77

000003 A2.01 3.7/3.9 (CFR 43.5/45.13) SRO ONLY 4-14-00 NEW

Question setup:

In the current conditions the group 6 average position is 47%.

A. Incorrect – This would be correct if the crew decided to insert all Regulating group rods and leave the Safety Group 1-4 at 100% ("cocked").

B. Incorrect - Rod 9 at 45% is misaligned with the group average but it is not asymmetric (< 6% from the group average). If Group 6 rods were positioned to 45% Rod 8 would still be asymmetric.

C. Correct – Per AP/1700/15, If a control rod is dropped on the approach to criticality or results in the reactor returning to a subcritical state from a critical condition then insert CRD to Group 1 @ 50%.

D. Incorrect – This does not match guidance in the AP and does not position the CRs to be capable of adding negative reactivity if needed. (Gr 1 @ 50%).

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

At the conclusion of both this training and the student's self-study of the Abnormal Procedures, the student will be able to demonstrate a working knowledge of the Abnormal Procedures (AP).

ENABLING OBJECTIVES

1. State the purpose of the Abnormal Procedures. (R1)

   1-18 p&t 2-23-03 &

2. Describe all Abnormal Procedures listed in OMP ~~2-1~~, for referral from memory, include the conditions that require this referral (R2)

3. Briefly describe how to properly use an Abnormal Procedure including: (R3)

   3.1 Describe the sequence the procedure is formatted to.

   3.2 Describe the correct use of logical statements including, IF/THEN, AND/OR, IF AT ANY TIME, and WHEN statements.

   3.3 Describe the proper use of Place Keeping Aids.

   3.4 Describe the purpose for NOTES and CAUTION statements and how they are used.

4. Given a copy of the AP, perform the following: (R4)

   4.1 Walk-through and discuss each procedure step.

   4.2 Locate all instrumentation and controls referred to in the AP including those devices outside the Control Room that would require manual operations should any automatic action fail.

   4.3 Briefly summarize the actions to be taken in the Subsequent Actions section of the AP.

4. Immediate Manual Actions

_____ 4.1 **IF** more than one Control Rod has dropped,

THEN manually trip the Reactor:

- GO TO EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 4.2 **IF** more than one Control Rod is misaligned > 9" (6%),

THEN manually trip the Reactor:

- GO TO EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.3 **IF** due to a malfunction, a Control Rod Group is misaligned > 9" (6%) from the acceptable region of PT/1/A/0600/001, (Periodic Instrument Surveillance),

THEN manually trip the Reactor:

- GO TO EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.4 **IF** a Control Rod has dropped on an approach to criticality,

OR a dropped Control Rod results in a return to subcriticality from a critical condition,

THEN manually insert all Control Rods to Group 1 at 50% WD.

QUESTION # 78

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	___	**1**
Group #	___	**1**
K/A #	**000011 A2.10**	
Importance Rating	___	**4.7**

Technical Reference(s): **EOP Encl. 7.11**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E40 OBJ. #5**

Question Source:
Bank # ______
Modified Bank # **EAP-543**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __**X**__

10 CFR Part 55 Content:
55.41 ______
55.43 __**X**__

Comments:

1 POINT

QUESTION # 78

SRO ONLY

Unit 1 plant conditions:

- A LBLOCA has occurred
- RCS pressure = 50 psig
- RB pressure = 35 psig
- BWST level = 6.5 ft
- RB level = 3.6 ft
- CFT A and B = 1.5 ft
- 1LP-19 and 20 (1A and 1B Rx Bldg Suction) are open
- 1LP- 21 and 22 (1A and 1B LPI BWST Suction) are open
- The BOP is performing EOP Enclosure 7.11, ECCS Suction Swap with Both LPI Flows $\geq$ 1000 gpm

Which ONE of the following describes the status of the ECCS?

A. HPI and LPI pumps are operating with suction provided from the BWST and CFTs have dumped.

B. HPI and LPI pumps are operating with suction provided from the RBES and CFTs are dumping.

C. HPI pumps are secured, LPI pumps are operating with suction provided from the BWST and CFTs are dumping.

D. HPI pumps are secured, LPI pumps are operating with suction provided from the RBES and CFTs have dumped.

1 POINT

QUESTION # 78

000011A2.10 4.5/4.7 (CFR 43.5/45.13) SRO ONLY PRA 4-14-00 NEW

Question setup:
HPIPs are secured during the first step of Enclosure 7.11. This step is completed first because the HPIPs are not needed during a LBLOCA if LPIPs are providing core heat removal. This step also prevents possible damage from air entrapment to the HPIPs suction when shifting suctions from the BWST to the RBES (when both sets of valves are open). Suction is being provided to the LPIPs via the RBES as indicated by elevated RB pressure. If RB pressure is ≥ 10 psig the LPIP suction will be provided via the RBES. CFTs have dumped and are reading their low scale indication of 1.5 ft., these transmitters do not indicate 0 when the CFTs are empty.

A. Incorrect – HPI pumps are secured. The second part would be correct if RB pressure were less than 10 psig.

B. Incorrect – HPI pumps are secured. The second part would be correct if CFT indicated 0" when empty.

C. Incorrect – HPI pumps are secured. The second part would be correct if RB pressure were less than 10 psig.

D. Correct - HPI pumps are secured. Suction is from RBES due to high RB pressure. CFTs are empty (reads 1.5 feet).

ECCS Suction Swap With Both LPI Header Flows ≥ 1000 gpm

1. Secure all HPI Pumps:

_____ 1A HPI Pump

_____ 1B HPI Pump

_____ 1C HPI Pump.

2. Throttle RBS flow in all headers with an operating pump to 900 - 1000 gpm per header:

_____ 1BS-1 (1A HDR RB ISOLATION)

_____ 1BS-2 (1B HDR RB ISOLATION).

_____ 2.1 **IF** RBS flow **CANNOT** be throttled ≤ 1000 gpm in either header,

THEN secure the associated RBS Pump:

_____ 1A RBS Pump

_____ 1B RBS Pump.

NOTE 3: A RB level of ≥ 2 feet is expected when BWST level reaches 9 feet.

_____ 3. **WHEN** BWST level reaches 9 feet,

AND RB level is rising,

THEN perform the following to swap LPI suction to RBES: {3}

3.1 Simultaneously open the following valves:

_____ 1LP-19 (1A RX BLDG SUCTION)

_____ 1LP-20 (1B RX BLDG SUCTION).

ECCS Suction Swap With Both
LPI Header Flows ≥ 1000 gpm

_____ 3.2 **IF** 1LP-19 (1A RX BLDG SUCTION) fails to open,

THEN perform the following:

_____ 3.2.1 **IF** BWST level is continuing to decrease,

THEN wait until BWST level is ≤ 6 feet before proceeding.

_____ 3.2.2 **IF** AT ANY TIME BWST level is ≤ 6 feet,

THEN immediately align the following valves:

_____ Close 1LP-22 (1B LPI BWST SUCTION)

_____ Open 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)

_____ Open 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

3.2.2.1 Stop the following pumps:

_____ 1A LPI Pump

_____ 1A RBS Pump.

3.2.2.2 Throttle total LPI flow per the following:

_____ A. **IF** 1LP-12 (1A LPI COOLER OUTLET) has been locally throttled,

THEN throttle 1LP-14 (1B LPI COOLER OUTLET) to maximize 'B' LPI Header flow ≤ 1100 gpm.

_____ B. **IF** 1LP-12 (1A LPI COOLER OUTLET) has NOT been locally throttled,

THEN maximize flow in each LPI Header ≤ 1100 gpm.

_____ 3.2.2.3 **GO TO** Step 7.

ECCS Suction Swap With Both LPI Header Flows ≥ 1000 gpm

_____ 3.3 **IF** 1LP-20 (1B RX BLDG SUCTION) fails to open,

THEN perform the following:

_____ 3.3.1 **IF** BWST level is continuing to decrease,

THEN wait until BWST level is ≤ 6 feet before proceeding.

_____ 3.3.2 **IF** AT ANY TIME BWST level is ≤ 6 feet,

THEN immediately align the following valves:

- _____ Close 1LP-21 (1A LPI BWST SUCTION)
- _____ Open 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)
- _____ Open 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

3.3.2.1 Stop the following pumps:

- _____ 1B LPI Pump
- _____ 1B RBS Pump.

3.3.2.2 Throttle total LPI flow per the following:

_____ A. **IF** 1LP-14 (1B LPI COOLER OUTLET) has been locally throttled,

THEN throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize 'A' LPI Header flow ≤ 1100 gpm.

_____ B. **IF** 1LP-14 (1B LPI COOLER OUTLET) has NOT been locally throttled,

THEN maximize flow in each LPI Header ≤ 1100 gpm.

_____ 3.3.2.3 **GO TO** Step 7.

ECCS Suction Swap With Both
LPI Header Flows ≥ 1000 gpm

_____ 4. IF BWST level is continuing to decrease,

THEN wait until BWST level is ≤ 6 feet before proceeding.

_____ 5. IF AT ANY TIME BWST level is ≤ 6 feet,

THEN immediately perform the following:

5.1 Simultaneously close the following valves:

_____ 1LP-21 (1A LPI BWST SUCTION)

_____ 1LP-22 (1B LPI BWST SUCTION).

ECCS Suction Swap With Both LPI Header Flows ≥ 1000 gpm

_____ 5.2 **IF** 1LP-21 (1A LPI BWST SUCTION) fails to close,

THEN perform the following:

5.2.1 Simultaneously open the following valves:

_____ 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)

_____ 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

5.2.2 Stop the following pumps:

_____ 1A LPI Pump

_____ 1A RBS Pump.

5.2.3 Throttle total LPI flow per the following:

_____ 5.2.3.1 **IF** 1LP-12 (1A LPI COOLER OUTLET) has been locally throttled,

THEN throttle 1LP-14 (1B LPI COOLER OUTLET) to maximize 'B' LPI Header flow ≤ 1100 gpm.

_____ 5.2.3.2 **IF** 1LP-12 (1A LPI COOLER OUTLET) has NOT been locally throttled,

THEN maximize flow in each LPI Header ≤ 1100 gpm.

ECCS Suction Swap With <u>Both</u>
LPI Header Flows ≥ 1000 gpm

_____ 5.3 <u>IF</u> 1LP-22 (1B LPI BWST SUCTION) fails to close,

<u>THEN</u> perform the following:

5.3.1 Simultaneously open the following valves:

_____ 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)

_____ 1LP-10 (1C LPIP DISCH TO 1B LPI HDR).

5.3.2 Stop the following pumps:

_____ 1B LPI Pump

_____ 1B RBS Pump.

5.3.3 Throttle <u>total</u> LPI flow per the following:

_____ 5.3.3.1 <u>IF</u> 1LP-14 (1B LPI COOLER OUTLET) has been locally throttled,

<u>THEN</u> throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize 'A' LPI Header flow ≤ 1100 gpm.

_____ 5.3.3.2 <u>IF</u> 1LP-14 (1B LPI COOLER OUTLET) has <u>NOT</u> been locally throttled,

<u>THEN</u> maximize flow in <u>each</u> LPI Header ≤ 1100 gpm.

2. CFT Level: Each tank is provided with two differential pressure detectors. Both of these detectors indicate in the Control Room. The range of each indicator is 1.5-14 feet.

B. Setpoints:

1.	CFT Pressure	Required Pressure	585-615 psig
2.	CFT Level -	Required Level	13 ± .44 feet

C. Alarms:

1. SA-8/A-11and 12 CFT Pressure High/Low 615 psig increasing High/ 585 psig decreasing/Low
2. SA-8/B-11 and 12 CFT Level High/Low Alarm - 13.3 feet increasing High,12.7 feet decreasing Low
3. SA-8/C-11 CFT "A" Outlet Valve - "Not Open"
4. SA-8/C-12 CFT "B" Outlet Valve - "Not Open"

2.6 Limits and Precautions

A. LTOP must be established on the CF system prior to the RC System going below 325°F.

B. When filling the Core Flood Tanks after they have been drained, use 'A" BHUT with a boron concentration of 2500 ppm.

C. Verify the CFT boron concentration is 2500 - 3750 ppm following a change in inventory. If in-leakage exists, a sampling program should be established.

D. If CF Tank inleakage exists makeup to the CF Tank should have a boron concentration ≥ 2700 ppm.

E. Pressure increase in the CF Tanks should not exceed 100 psig in 15 minutes (6.6 psig/min) unless the Nitrogen Heater is placed in service.

F. Do not increase the nitrogen header pressure while adding nitrogen to CF tanks.

2.7 ITS/SLC's

A. ITS 3.5.1 Core Flood Tanks

1. Review ITS 3.5.1 with the students.
2. Work an example with the student using Conditions and Required Actions of the Specification.

B. ITS 3.4.12

1. Review ITS 3.4.12 with the students.
2. Work an example with the students using the Conditions and Required Actions associated with the Core Flood Tanks for LTOP.

OBJECTIVES

TERMINAL OBJECTIVE

Following this lecture, the Operator should be able to correctly align the LPI system and successfully make the swap from the BWST to the RBES. The Operator will be capable of making the swap taking all applicable contingency actions warranted and complete the swap before the depletion of the BWST.

ENABLING OBJECTIVES

1. State the purpose of Enclosures 7.11 (ECCS Suction Swap to RBES with Both LPI Header flows ≥1000 gpm) and 7.12 (ECCS Suction Swap to RBES with Either LPI Header flow <1000 gpm). (R1)
2. Evaluate unit conditions and determine which circumstances require the use of Enclosure 7.11 (ECCS Suction Swap to RBES with Both LPI Header flows ≥1000 gpm) and which circumstances require the use of 7.12 (ECCS Suction Swap to RBES with Either LPI Header flow <1000 gpm). (R2)
3. Compare Enclosure 7.11 (ECCS Suction Swap to RBES with Both LPI Header flows ≥1000 gpm) to enclosure 7.12 (ECCS Suction Swap to RBES with Either LPI Header flow <1000 gpm) and describe the differences in the two enclosures. (R3)
4. Explain why it is desirable to operate valves, which can't be operated remotely, early in the swap from the BWST to the RBES. (R4)
5. Predict the results of having LP-19, LP-20, LP-21, and LP-22 open at the same time for the following conditions: (R5)
   - 5.1 High (>10 psig) Reactor Building pressure conditions.
   - 5.2 Low (<10 psig) Reactor Building pressure conditions.
6. Given different LPI system alignments and predict whether the LPIP suctions will be overpressurized for each alignment. (R6)
7. Discuss the actions taken to prevent overfilling the LDST. (R7)
8. Compare different LPI flow conditions and determine which ones meet the LPIP minimum flow requirements. (R8)
9. Explain why BS-1 and BS-2 are opened before cross-connecting the LPIP discharge. (R9)
10. Discuss the alignment of LPSW to the LPI coolers including the three possible conditions the enclosures cover. (R10)

2. PRESENTATION

2.1 A review of the EOP was performed in an attempt to determine the susceptibility of the EOP to single failures. Several issues with EOP guidance were identified. These issues affected both large break LOCAs (LBLOCAs) and small break LOCAs (SBLOCAs). A review of various equipment failures within the design basis (loss of TC,TD, LP-19, LP-20, LP-12 or LP-14, etc.) identified the following concerns with EOP BWST to RBES swap guidance:

A. Certain failure scenarios resulted in a loss of LPI throttling capability from the control room. For example, a failure of LP-12 to throttle on demand or LP-17 to open. Once LPI suction is swapped to the RBES, access to the valves locally will likely be lost due to radiation levels.

B. For some LOCA scenarios, including LBLOCAs, the RB depressurizes below 10 psig before the LPI suction is swapped to the RBES. During the time when LP-19/20 are open and LP-21/22 have not been closed, gravity draining will occur from the BWST to the RBES. If BWST level gets low enough before LP-21/22 are closed, air will be entrained in the LPIP's suction. LPI pumps may continue to operate under these conditions but if piggyback is aligned, HPI pumps would probably fail within a matter of seconds. Since the HPI pump suctions are all cross-connected, this would impact all HPI pumps. **(OC-EAP-E40-3)**

1. Even if piggyback was not aligned, air entrainment could have caused a problem. If LPI pump A was unavailable, the EOP relied on LPI pump C and cross-connected the LPI pump suction header (opened LP-6 and 7). These procedural steps created a system configuration that exposed both LPI trains to air entrainment if a BWST outlet valve (LP-21 or 22) failed to close. The single failure of TC would result in a loss of the A LPI pump and a failure of LP-21 to close.

C. The LPI discharge header is cross-connected (LP-9 and 10 opened) in certain scenarios. Caution must be exercised to ensure that the suction side of an idle LPI pump is not over pressurized. For example, assume LPI pump A is off and LPI pump B is cross-connected to the A header. Back flow through the LPI pump A minimum flow line can pressurize the suction header on the A LPI train above the relief valve setpoint if LP-19, LP-6, LP-7, and BS-1 are closed. Again, a failure of TC would set up these conditions. **(OC-EAP-E40-4)**

D. EOP guidance had prevented overfilling the LDST by opening HP-363 and routing HPI pump recirc. flow to the suction of LPI pump A. However, the procedure did not directly address actions in the event that HP-363 could not be opened. In addition, even if HP-363 was opened, a pressurized LPI pump suction header could prevent flow through the HP-363 flow path and would have filled up the LDST. **(OC-EAP-E40-5)**

d) When both LPI headers are capable of being throttled, flow is still set at 1100 gpm.

1) This gives a total flow of 2200 gpm.

(a) High enough that sufficient flow is delivered to the core.

(b) Lower than the maximum limit of 3000 gpm.

2) Gives us only one number to remember.

e) When throttling flows, guidance is given to "maximize" header flow, limited to ≤ to some high limit; this is new terminology.

1) This means to place the flow as close to the maximum limit as possible without exceeding the limit.

2) There is a possibility that LPI flow will cycle rather than flow at a constant value.

3) If the LPI flow does cycle, the flow should not exceed the maximum limit at the high point of the cycle.

2.5 Enclosure 7.11 (ECCS Suction Swap to RBES With BOTH LPI Header Flows ≥1000 gpm) gives the following guidance:

A. Secure all HPIPs

1. This enclosure is only used with LBLOCAs, LPI is already injecting and HPI injection is not needed.

B. Throttle RBS flow between 900 to 1000 gpm/hdr.

1. If flows can't be throttled <1000 gpm, secure the RBS pump in that header.

a) The water that is in the RBES is much hotter than the water in the BWST. While we may be able to handle RBS flows of up to 1500 gpm with the colder BWST water, the hotter RBES water changes the NPSH requirements for the pump. This is why we must reduce the RBS flow before the swap.

C. Hold step to wait for a BWST level of 9 feet and a rising Reactor Building level.

1. FSAR guidance says that we begin the swap when the BWST low Level alarm is received.

2. The alarm actuates at 9 feet.

3. Guidance is given to comply with the FSAR.

4. The requirement for the RB level increasing ensures that the water is going to the Reactor Building and is available for taking a suction.

D. Guidance is given to simultaneously open LP-19 and 20 (RBES Suction Valves).

1. Valve stroke time for these valves is > 1 minute.
2. Stroke time must be taken into consideration when making the swap from the BWST to the RBES.
3. Opening these valves simultaneously means that the stroke times are in parallel rather than series.
4. If these valves are stroked and one appears to have a burned out light bulb do not take the time to change the light bulb.
   a) Monitor the progress of the valve with the good position indication.
   b) When the valve with good position indication opens fully, allow a few more seconds for the valve in question to open.
   c) If the valve in question does not indicate open, take the contingency actions for a suction valve failure.
5. The intent of this step is to attempt to open both valves at the same time to save time in the swap to the RBES.
   a) Remember that if either of these valves has a loss of power or will not operate for some other reason the contingency steps must be performed.

E. The next guidance looks at a failure of the RBES suction valves (LP-19 or 20). The enclosure will swap the LPIP suction to the RBES on the suction header aligned to the sump and will isolate the other as follows: **(OC-EAP-E40-9)**

1. If BWST level is still decreasing, hold until BWST level is ≤6 feet.
   a) Get as much inventory out of the BWST as possible before swapping to the sump. SBLOCA analysis assumes 40 feet of water transferred from the BWST into containment. Assuming ITS minimum of 46 feet at beginning, must be ≤6 feet.
   b) If there is a slight difference in BWST indications, use the lowest value.
   c) If BWST level is not continuing to decrease, the operator will continue to the step that aligns LPSW.
      1) The condition of the BWST level at this time is dependent upon reactor building pressure.
         (a) If Reactor Building pressure is greater than about 10 psig, the head from the RBES will be higher than the head from the BWST, at this low level, so the LPIPs and RBSPs will take suction from the RBES. When this happens, the BWST level will stop decreasing and begin to draw a straight line.

(b) When building pressure gets within the 10 psig range the NPSH from the BWST and the RBES will begin to equalize and the pumps will start taking a suction off of the BWST again. When this occurs BWST level will begin to decrease.

(1) The BWST level can begin to drop at a relatively fast rate because not only will the LPI and RBS pumps be taking a suction but there will also be some gravity feed from the BWST to the RBES.

2. IF AT ANY TIME BWST level is ≤6 ft the enclosure will direct the following:

   a) Immediately align suction by closing the BWST suction valve on the suction header with the RBES suction valve that did open and opening the LPIP discharge cross-connect valves.

   b) The term immediately is used to indicate that we need to close the suction and open the two discharge cross-connects at the same time; as close together as possible.

3. Secure the LPIP and RBSP receiving suction from the header with the failed RBES suction valve.

   a) This header will not be able to take a suction from the RBES and the BWST is being depleted at a fast rate.

   b) The pumps must be secured to prevent sucking air in from the BWST as it empties.

   c) With the LPI suction cross-connect valves closed, each header must be swapped to the sump individually.

4. If any cooler outlet valve has been manually throttled, limit flow in the header that can be throttled from the control room to 1100 gpm.

F. If both RBES suction valves open properly we will perform the following: **(OC-EAP-E40-10)**

1. If BWST level is continuing to decrease, wait for BWST level to decrease to ≤6 ft before proceeding.

   a) It looks like this guidance is given twice but if both RBES valves (LP-19 & 20) open, this is the first time that the reader gets this guidance.

   b) The reason for this guidance is the same as previously discussed.

2. IF AT ANY TIME BWST level is ≤6 feet then immediately perform the following:

   a) Simultaneously close LP-21 and 22 (stroke time again).

Exam Question Report

27-Jan-99

Question ID:	EAP543	Revision No:	0`	Revision Date	02/23/2000
Question Description:	EAP-E31				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-E31 - Cooldown Following a Large LOCA
Last Used Date:	01/28/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	Reference: 1998 FARLEY NRC EXAM	Max. Point Value:	0.25
		Passing Point Value:	0.25

27-Jan-99

Question

Given the following plant conditions:

- A large break LOCA has occurred on Unit 1.
- Actions required by Section 501, Loss of Subcooling, have been completed.
- A transfer to CP-601, Cooldown Following Large LOCA, has been made.
- BWST level is twenty-one (21) feet.
- All ECCS systems are operating properly.
- RCS pressure is stable at 60 psig.

Which ONE of the following mechanisms describes the PRIMARY method of decay heat removal? (.25)

A) Condensation of reflux boiling in the Steam Generators.

B) Natural circulation cooling between the RCS and the Steam Generators.

C) Forced circulation between the RCS and the Steam Generators.

D) Forced circulation cooling from the BWST through the core and out the break.

Answer

D

A. Incorrect, heat transfer is not occurring in the SGs because of the lack of RCS inventory.

B. Incorrect, voids in the hot leg will prevent natural circulation form taking place.

C. Incorrect, RCPs will not be operating.

D, Correct, LPI is taking a suction from the BWST and out the break, this is removing decay heat.

Lessons

Exam Question Report

27-Jan-99

ID	Description
EAP-E31	Cooldown Following Large LOCA (EAP-E31)

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag
1998 FARLEY NRC	Reference created by conversion		

KA'S

ID	Description

QUESTION # 79

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	____	**1**
	Group #	____	**1**
	K/A #	**000015/17 A2.01**	
	Importance Rating	____	**3.5***

Technical Reference(s): **AP/1700/16 (all 3 units)**
PNS-RCP seals

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-CPS #6**

Question Source:
Bank # ____
Modified Bank # ____
New __X__

Question History:
Previous NRC Exam ____
Previous Quiz / Test ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ____
55.43 ____

Comments:

1 POINT

QUESTION # 79

SRO ONLY

Plant Conditions:
Unit 1:

- MODE 1, Power = 100%
- 1B1 RCP seal leak off flow = 4.5 gpm
- 1A2 RCP radial bearing temperature = 195°F
- OSTG levels:
  - 1A = 65% OR
  - 1B = 68% OR

Unit 2:

- MODE 1, Power = 65%
- 2A2 RCP seal return flow = 4.5 gpm
- 2A1 RCP lower seal cavity pressure = 2000 psig
- OSTG levels:
  - 1A = 45% OR
  - 1B = 40% OR

Unit 3:

- MODE 1, Power = 100%
- 3A1 RCP upper seal temperature = 195°F
- 3B1 RCP motor shaft vibration = 15 mils
- OSTG levels:
  - 1A = 78% OR
  - 1B = 81% OR

~~BOP~~ ROs
the ~~ATC~~
Which ONE of the following directions should the SROs provide to ~~prospected crews~~?

SEE ATTACHMENTS

Trip ____, then trip _____.

A. Unit 1 / RCPs 1A2 and 1B1

B. Unit 2 / RCP 2A2

C. Unit 3 / RCP 3B1

D. no Units / 2A2 RCP

ABNORMAL REACTOR COOLANT PUMP OPERATION
AP/1/A/1700/016

CASE A

Reactor Coolant Pump Evaluation

4.0 Immediate Manual Actions

_____4.1 Monitor the Reactor Coolant Pump(s) for Immediate Trip Criteria.

Parameter	Trip Limit
RCP #1 Seal Leakoff Flow As read on "RC Pump Seal Leakoff Flow" recorder	**≥ 5 gpm AND RCP Radial Bearing and/or Seal Leakoff Temp Increasing** Computer Points: O1A1249 - O1A1256
RCP #1 Seal Leakoff Flow As read on "RC Pump Seal Leakoff Flow" recorder	**< 0.8 gpm AND RCP Radial Bearing and/or Seal Leakoff Temp Increasing** Computer Points: O1A1249 - O1A1256
RCP Upper Motor Bearing Temp Computer Points: O1A1588, O1A1590, O1A1592, O1A1594	**190°F**
RCP Lower Motor Bearing Temp Computer Points: O1A1589, O1A1591, O1A1593, O1A1595	**190°F**
RCP Thrust Bearing Temp Computer points: O1A1572 - O1A1574, O1A1576 - O1A1578, O1A1580 - O1A1582, O1A1584 - O1A1586	**190°F**
RCP Stator Temp Computer points: O1A0904 - O1A0911	**295°F**
RCP Radial Bearing Temp Computer points: O1A1249 - O1A1252	**225°F**
RCP Seal Leakoff Temp Computer points: O1A1253 - O1A1256	**225°F**
RCP Vibration	**Sustained actual Emergency High Vibration as verified by Alarm Response Guide for (1SA9/E-2) "RCP VIBRATION EMERG. HIGH".**
RCP Low oil pot level Computer points: O1A2032 - O1A2039	**AND any Motor or Thrust Bearing Temp Increasing** Computer points: O1A1572 - O1A1574, O1A1576 - O1A1578, O1A1580 - O1A1582, O1A1584 - O1A1586, O1A1588 - O1A1595
Loss of HPI Seal Injection	**AND Component Cooling has been lost**

Abnormal Reactor Coolant Pump Operation
AP/2/A/1700/016

CASE A

Reactor Coolant Pump Evaluation

4.0 Immediate Manual Actions

_____ 4.1 Monitor the Reactor Coolant Pump(s) for Immediate Trip criteria:

Parameter	Trip Limit
RCP Seal Return Flow Actual (computer points; A1648, A1649, A1650, A1651) plus Seal Leakage Flow	> 4.1 gpm
RCP Upper Seal Temperature (2TE-1707, 1709, 1711, 1713)	>200 °F
RCP Control Bld Off TE (computer points; A1272, A1273, A1274, A1275)	>200 °F
RCP Seal Integrity (2A1, 2B1, 2A2, 2B2) • RCP UPPER SEAL CAVITY PRESSURE (2PT-205, 206, 207, 208) • RCP LOWER SEAL CAVITY PRESSURE (2PT-219, 220, 221, 222)	Two of three RCP seal stages fail as evidenced by d/p across the remaining stage approximately equal to RCS pressure (with Seal Return established). *
RCP Vibration	Sustained actual Emergency High Vibration as verified by Alarm Response Guide for "RC PUMP VIBRATION EMERG HIGH" statalarm (2SA-9/ E-2).
Low oil pot level	AND any RCP Motor Brg Temp Increasing
Loss of HPI Seal Injection	AND Component Cooling has been lost

Chg # 009B

* RCP seal d/p is determined as follows:

- 1st stage d/p = system pressure - RCP Lower Seal Cavity Pressure.
- 2nd stage d/p = RCP Lower Seal Cavity Pressure - RCP Upper Seal Cavity Pressure.
- 3rd stage d/p = RCP Upper Seal Cavity Pressure - RB atmospheric pressure.

ABNORMAL REACTOR COOLANT PUMP OPERATION
AP/3/A/1700/16

CASE A

Reactor Coolant Pump Evaluation

4.0 Immediate Manual Actions

_____4.1 Monitor the Reactor Coolant Pump(s) for Immediate Trip criteria.

Parameter	Trip Limit
RCPSeal Return Flow Actual (computer points; A1656, A1661, A1667, A1668) plus Seal Leakage Flow	> 4.1 gpm.
RCP Upper Seal Temperature (3TE-1707, 1709,1711, 1713)	> 200°F.
RCP Control Bld Off TE (computer points; A1272, A1273, A1274, A1275)	> 200°F.
Two of the three RCP seals stages fail as evidenced by d/p across the remaining stage.** • RCP UPPER SEAL CAVITY PRESSURE (3PT-205, 206, 207, 208) • RCP LOWER SEAL CAVITY PRESSURE (3PT-219, 220, 221, 222)	approximately equal to RCS pressure, with Seal Return established.
RCP Motor Frame Vibration	5 mils (.3 in/s)
RCP Shaft Vibration	20 mils
RCP Shaft Vibration while operating with < 4 RCP's	30 mils
Low oil pot level	AND any RCP Motor Brg Temp Increasing
Loss of HPI Seal Injection	AND Component Cooling has been lost

** RCP seal d/p is determined as follows:

- 1st stage d/p = system pressure - RCP Lower Seal Cavity Pressure.
- 2nd stage d/p = RCP Lower Seal Cavity Pressure - RCP Upper Seal Cavity Pressure.
- 3rd stage d/p = RCP Upper Seal Cavity Pressure - RB atmospheric pressure.

1 POINT

QUESTION # 79

000015/17 A2.01 3.0/3.5 CFR43.5/45.13 SRO ONLY - PRA 3-23-00 (GTH/PMS)

A. Incorrect - Unit 1 1A2 radial bearing temperature immediate trip criteria is > 225°F at 195°F this does not require a trip of the unit or the RCP 1A2. 1B1 RC seal leak off flow = 4.5 gpm, it's immediate trip criteria is > 5 gpm with seal return temperatures increasing. Unit 1 reactor and RCPs should remain operating.

B. Incorrect - Unit 2 is operating < 70% with 4 RCPs operating based on FDW OTSG level data so Unit 2 can secure a RCP without tripping the reactor from its present power level. RCP 2A2 requires to be tripped due to seal return flow exceeding the immediate trip criteria of 4.1 gpm. 2A1 RCP lower seal cavity pressure = 2000 psig indicates that 1 of the three seals are degraded and immediate trip criteria is if 2 of the 3 seals are degraded.

C. Incorrect – 3A1 RCP upper seal temperature = 195°F immediate trip criteria is 200°F this does not require a trip of the reactor or the 3A1 RCP. 3B1 RCP motor shaft vibration = 15 mils. Motor frame vibration limits for immediate trip criteria is 5 mils and this information is not provided in the question.

D. Correct – Unit 2 is operating < 70% with 4 RCPs operating based on FDW OTSG level data so Unit 2 can secure a RCP without tripping the reactor from its present power level. RCP 2A2 requires to be tripped due to seal return flow exceeding the immediate trip criteria of 4.1 gpm. 2A1 RCP lower seal cavity pressure = 2000 psig indicates that 1 of the three seals are degraded and immediate trip criteria is if 2 of the 3 seals are degraded.

4. When given a set of unit conditions analyze the operating status of the Bingham RCP seals during all modes of operation. (R20)

5. Draw a simplified drawing of the Unit's Reactor Coolant Pump including the following: (R22)

   5.1 Seal Package

   5.2 HPI seal flow path and approximate flows expected

   5.3 #1 Seal Bypass valve

   5.4 Standpipe fill valve

   5.5 Radial Bearing

   5.6 Recirc Impeller

   5.7 Heat exchangers and coolers

   5.8 Thermal barrier, restriction bushing

6. Given a set of conditions, evaluate the total RCP seal leakage and determine the appropriate operator action(s) utilizing the information and guidance provided in PT/1,2,3/A/600/10, Reactor Coolant Leakage, and OP/1,2,3/A/1103/06, Reactor Coolant Pump Operation. (R23)

7. Given a copy of ITS/SLCs and associated Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's. (R21)

8. Apply all ITS/SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R21)

9. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITS/SLC's. (R21)

QUESTION # 80

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	____	**1**
	Group #	____	**1**
	K/A #	**000024**	**G2.2.22**
	Importance Rating		**4.1**

Technical Reference(s): **OMP 1-18**
IC-RPS

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **ADM-OMP OBJ. #2 & 3**
ICS-RPS

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 __X__

Comments:

1 POINT

QUESTION # 80

SRO ONLY

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 70%
- 2B2 RCP secured

CURRENT CONDITIONS:

- T-hot instrumentation to ICS fails
- The OATC places the ICS in manual to mitigate the event.
- Reactor power = 90%

Which ONE of the following is correct?

The SRO should ensure the OATC...

Perform immediate manual Actions and
A. emergency borate per RULE #1, ATWS Actions.

B. continues to stabilize the unit with ICS in manual.

C. decreases power and stabilizes at the pre-transient power level.

D. contacts SPOC to repair the failed ICS instrumentation and return the ICS to automatic per AP/1700/28, ICS Instrument Failure.

1 POINT

QUESTION # 80

000024 G2.2.22 3.4/4.1 (CFR 43.2/45.2) SRO ONLY PRA 4-14-00 NEW

A. Correct – Reactor power is > than the RPS limits (≈75% w/ 3 RCPs) and RPS did not automatically trip the reactor, Manually tripping the reactor is required. If the reactor does not trip then emergency boration is required.

B. Incorrect – If 4 RCP were operating instead of 3 or post transient power level was < 75% this would be correct answer.

C. Incorrect - If 4 RCP were operating instead of 3 or post transient power level was < 75% this would be correct answer.

D. Incorrect - If 4 RCP were operating instead of 3 or post transient power level was < 75% this would be correct answer. AP/1700/28, ICS Instrument Failure is a new AP to assist the crew in returning ICS to automatic following an instrumentation failure.

OBJECTIVES

TERMINAL OBJECTIVES

1. Describe the function and correct operation of the Reactor Protective System during all modes of plant operation, including specific limits or precautions associated with the particular component or operation. (T1)
2. Discuss all inputs and signals used in the RPS that affect the specific operation-related functions associated with the RPS. (T2)
3. Describe or identify the purpose of each indicator, control, and indicating light in the RPS that provides operator-related information. (T3)
4. Evaluate normal and off normal system operation and predict RPS and plant response to specific degraded components within the RPS, or components or power supplies that feed the RPS. Analyze the status of the RPS and the plant and develop a plan to return the system to normal. (T4)

ENABLING OBJECTIVES

1. List the two general components that the RPS is designed to protect. (R1)
2. List the two basic methods employed by the RPS to protect the fuel rod clad and RCS. (R2)
3. List all eleven variables in RPS that will initiate a reactor trip, and where applicable, the maximum or minimum allowable setpoint at which trip will occur. (R3)
4. For each trip, choose which of the two general components (fuel clad or RCS) each of the tripping parameters in RPS is designed to protect, and be able to describe the basis of the protection afforded by each. (R4)
5. Explain the following concerning the Shutdown Bypass function in RPS: (R5)
    - 5.1 The three plant operations for which the SD Bypass function provides the capability to perform.
    - 5.2 The four normal tripping parameters in RPS that are bypassed when SD Bypass is active.
    - 5.3 The new tripping parameter that is automatically inserted into RPS when SD Bypass is active, the trip set point set, and the basis for the new trip variable.

1.16 OMP 2-14, Operations Test Group Use of Blue Tags (R29)

1.17 OMP 2-16, Shift Turnover (R33)

1.18 OMP 3-1, Operations Training (R21)

1.19 OMP 3-9, New RO/SRO Mentoring Guides (R34)

2. Be able to recite, from memory, any required procedure or administrative items as detailed in OMP 1-18, Licensed Operator Memory Items Attachment: (R1)

   2.1 The student is not required to be able to list each item in the attachment from memory.

   2.2 The student is expected to be able to recall from memory those actions or statements listed in the attachment as they relate to the specific task or evolution being performed.

3. When given a copy of the Operations Manual, or portions thereof, be able to demonstrate an understanding of the guidance or rules within specific OMP's by locating the answer to or interpreting required responses for a given situation. (R2)

4. The operator will become <u>well versed</u> in the requirements set forth in the following OMP's, in order to meet the expectations of Operations Management and conduct safe reliable operations of all Oconee units at all times. The operator will comprehend and exercise the OMP as it relates to the following conditions:

   4.1 OMP 1-2, Rules of Practice (R3)

      A. Acceptable means of operator conduct and operational practices.

      B. Limits for acceptable work schedules.

      C. Minimum shift staffing requirements.

   4.2 OMP 1-9, Use of Procedures (R4)

      A. Provide guidance to the operator in the following areas concerning procedures:

         - establish consistent methods for using procedures
         - control of approved procedures
         - use of approved procedures
         - completion of procedures
         - control of procedure changes
         - deviation from approved procedures

2. Reactor Trip Requirements:

2.1. Initiate a manual reactor trip if any RPS trip setpoints are exceeded:

- Reactor power is > 105.5%
- RCS Thot is > 618 °F
- RCS pressure is > 2355 psig
- RCS pressure is < 1800 psig
- RCS pressure is below the variable pressure/temperature curve
- RB pressure is > 4 psig
- < 3 RCPs are operating with NI power > 2%
- Reactor power and imbalance are outside the limits of the axial imbalance RPS trip curve
- Turbine trip at > 30% FP

2.2. Initiate a manual reactor trip if any of the following conditions exist:

- Two or more CRDM stator temperatures ≥ 180 °F
- Two or more asymmetric control rods
- Pressurizer level ≥ 375"
- If all WR and PR NI channels fail at power
- RCS temperature ≤ 527 °F
- RCS leakage is greater than NORMAL MAKEUP CAPABILITY (except OTSG tube leakage)
- *Either SG ≤ 15" and control rods are greater than 50% withdrawn on GP 1
- *Both Main FDW Pumps trip and control rods are greater than 50% withdrawn on GP 1

*Manual Reactor trip is not required if Group 1 is positioned ≤ 50% to provide available Shutdown Margin.

5. Summer Amplifiers
6. Test Modules
7. Bistables
8. Contact Buffers
9. Auxiliary Relays
10. Shutdown Bypass Key Switch
11. Manual Bypass Key Switch
12. Contact Monitor

2.3 Component Description

A. Protective Functions

1. High Flux Trip at ≤105.5%

   a) Improved Technical Specifications requires high flux trip at no greater than 105.5%; Actual RPS set points are set at 104.75% for conservatism.

   b) The high flux trip is provided to prevent damage to the fuel and fuel clad from reactivity excursions too rapid to be detected by pressure or temperature measurements of the RPS.

   c) The Improved Tech Spec set point of 105.5% is based on not exceeding the maximum power levels reached in the Safety Analysis Reports. When 105.5% RPS set point is used, adding to this the variations that could result due to calibration and instrument errors and the delay time from trip set point reached until the power excursion is terminated, the maximum power level reached could be 112% Full Power.

2. High flux trip based on RCS Flow and Axial Imbalance (Flux/Flow/Imbalance)

   a) An example of Core Operating Limits Report Maximum Allowable trip set points for various combinations of reactor power and RCS flow are shown in Handout **OC-IC-RPS-5**.

   b) Specified power/flow set points prevent a DNBR of less than 1.18 from occurring should a loss of RCS flow occur at high power, due to an electrical malfunction.

   c) As a typical example (actual numbers will differ slightly), set points are based on 100% design flow - since actual RCS flow is greater than design flow (~ 107% for Unit 1,2,& 3) this credit for additional flow is taken when establishing trip set points based on RCS flow.

   d) For 4 RCPs (100% flow) maximum set point based on RCS flow is:

1) 100% x 1.07 = 107%(Unit 1,2,& 3)

e) With only 3 RCPs operating, design flow rate is 74.7%.

f) For 3 RCP combinations, the trip set point is derived by:

1) 74.7% x 1.07 = 79.9% Full Power(Unit 1,2,& 3)

g) In addition to preventing DNBR going below 1.18 if loss of RCS flow occurs, trip set points are modified if Core Axial Imbalance becomes excessive, to prevent core thermal limits (fuel centerline temperature) from being exceeded.

h) Imbalance considerations will automatically reduce the high flux trip set point based on RCS flow, once imbalance exceeds an acceptable value for normal operation. **(OC-IC-RPS-6)**

3. Power Level Based on the Number of RCPs Running (Flux/Pump) Trip

a) Provides redundant trip protection to prevent DNBR from decreasing below 1.18, by tripping reactor due to loss of reactor coolant pumps.

b) Redundant since flux/pump trip signal is diverse from the flux/flow/imbalance trip signal.

c) This trip signal will also restrict the power level obtainable, based upon the number of operating RCPs.

d) Per Improved Tech Specs, if reactor power is above 2% Full Power, RPS will trip the reactor if two RCPs are lost. The actual setpoint is 1.5%.

4. High RCS Pressure Trip at ≤2355 psig. **(OC-IC-RPS-7)**

a) Improved Tech Specs requires reactor trip before 2355 psig to prevent the RCS from exceeding the safety limit of 2750 psig (110% of design pressure of 2500 psig), for any design transient.

b) The safety limit of 2750 psig is established to maintain the integrity of the RCS and to prevent the release of significant amounts of fission product activity.

c) Normally, high flux trip would preclude significant RCS pressures - however, a rod ejection during startup or a slow rod withdrawal accident at high power, would cause RCS pressure to increase faster than reactor power.

d) Actual RPS trip set point at 2345 psig is for conservatism due to instrument and calibration errors, so that Improved Tech Specs will not be exceeded.

QUESTION # 81

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	**1**
	Group #	_____	**1**
	K/A #	**000026 A2.02**	
	Importance Rating	_____	**3.6**

Technical Reference(s): **AP/1700/18 Encl. 6.11**
OSFD-107A-1, 144A-1, PNS-CC

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-CC #17**

Question Source:
Bank # **NRC-070**
Modified Bank # _______
New _______

Question History:
Previous NRC Exam **(ONS 1997)**
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 _____
55.43 __X__

Comments:

1 POINT

QUESTION # 81

SRO ONLY

Given the following on Unit 1:

Initial conditions

- 100% Power
- Quench Tank in recirc
- 1A Letdown Cooler isolated for a suspected tube leak

Current conditions

- CC Surge Tank Level Hi/Low Statalarm actuates
- 1RIA-50, Component Cooling alarm is in alert

Which ONE of the following is the required operator action?

A. Secure Quench Tank from recirc

B. Close 1HP-5 (Letdown Isolation)

C. Close RCP Seal Cooler Outlet valves

D. Stop CC pumps and Close 1CC-7 and 8 (CC Return Penetration Blocks)

1 POINT

QUESTION # 81

000026 A2.02 2.9/3.6 SRO ONLY (CFR 43.5 / 45.13) GCW

A. Correct, - For the conditions given the operator is required to isolate the affected component e.g. Quench- Tank per AP/1/A/1700/002. Take the QT out of recirc per AP/1/A/1700/018.

B. Incorrect, - This action is unnecessary and will not isolate the intersystem leak. ~~CC pressure is higher than letdown system pressure at the Letdown cooler. Leakage would be from CC to Letdown.~~ 1RIA-50 would not be in alert. WRONG

C. Incorrect, - CC pressure is higher than RCP seal return pressure at the cooler. Leakage would be from CC to RCP seal return. 1RIA-50 would not be in alert.

D. Incorrect, - This would isolate the leak however, this action is extreme and will result in a loss of letdown, subject system components to unnecessary high temperatures.

1RIA-50 (CC) Alarm Guidelines

_____ 1. Attempt to locate source of inleakage.

- Swap Letdown Coolers
- Monitor RCP parameters affected by CC
- Secure recirc of QT.

_____ 2. Notify other units NOT to reclaim water from the CC Drain Tank.

_____ 3. Notify Chemistry to increase survey frequency of the CC system and CC Drain Tank.

END

Excessive RCS Leakage

_____ 5.6 Monitor the CC System for in-leakage.

_____ 5.6.1 **IF** the CC Surge Tank level is increasing,

AND 1RIA-50 is in alarm,

THEN isolate the affected component:

_____ 5.6.1.1 **IF** the 1A Letdown Cooler is in service,

AND the 1B Letdown Cooler is available,

THEN swap letdown coolers:

_____ A. Start the Standby CC Pump

_____ B. Verify open 1HP-4 (1B LETDOWN COOLER OUTLET)

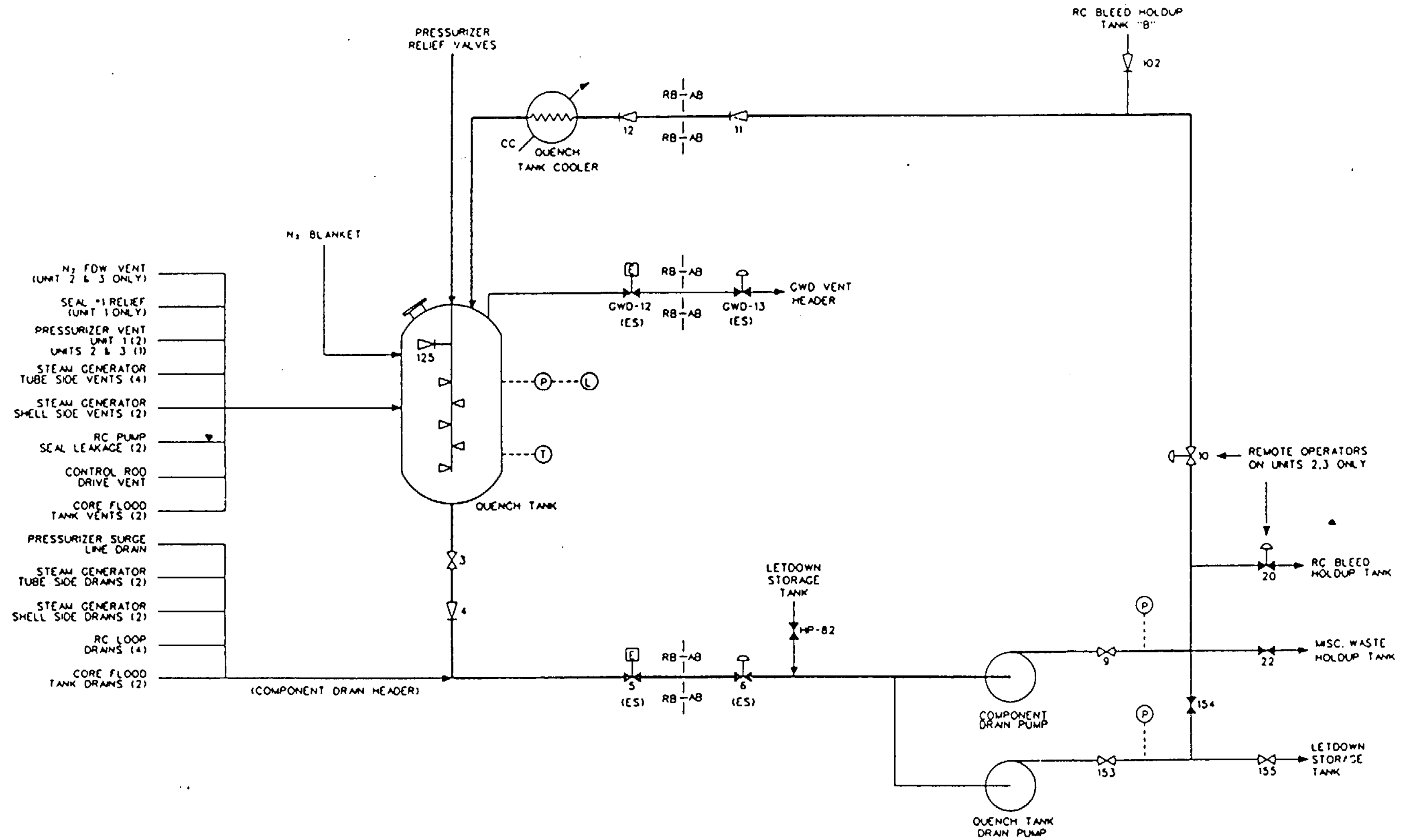
_____ C. Open 1CC-2/1HP-2 (1B LETDOWN COOLER INLET)

_____ D. Close 1CC-1/1HP-1 (1A LETDOWN COOLER INLET)

_____ E. Close 1HP-3 (1A LETDOWN COOLER OUTLET)

_____ F. Stop the Standby CC Pump and return switch to "AUTO".

DWG. NO. OSFD-107A



TYPICAL FOR UNITS 1, 2 & 3

ALL VALVES "CS" EXCEPT AS NOTED

ERN: 0X001184

LEGEND

- SHUTOFF VALVE
- FLOW CONTROL VALVE
- CHECK VALVE (ALL TYPES)
- RELIEF VALVE
- NORMALLY OPEN
- NORMALLY CLOSED
- NORMALLY THROTTLED
- F-FLOW, L-LEVEL, P-PRESSURE, T-TEMPERATURE, R-RADIATION MONITOR
- E-ELECTRIC, H-HYDRAULIC, P-PISTON (PNEUMATIC), S-SOLENOID
- DIAPHRAGM (PNEUMATIC)
- (ES)-RECEIVES ENGINEERED SAFEGUARD SIGNAL

THIS DRAWING IS A SUMMARY FLOW DIAGRAM. FOR COMPLETE SYSTEM DESIGN INFORMATION REFER TO FLOW DIAGRAMS LISTED BELOW

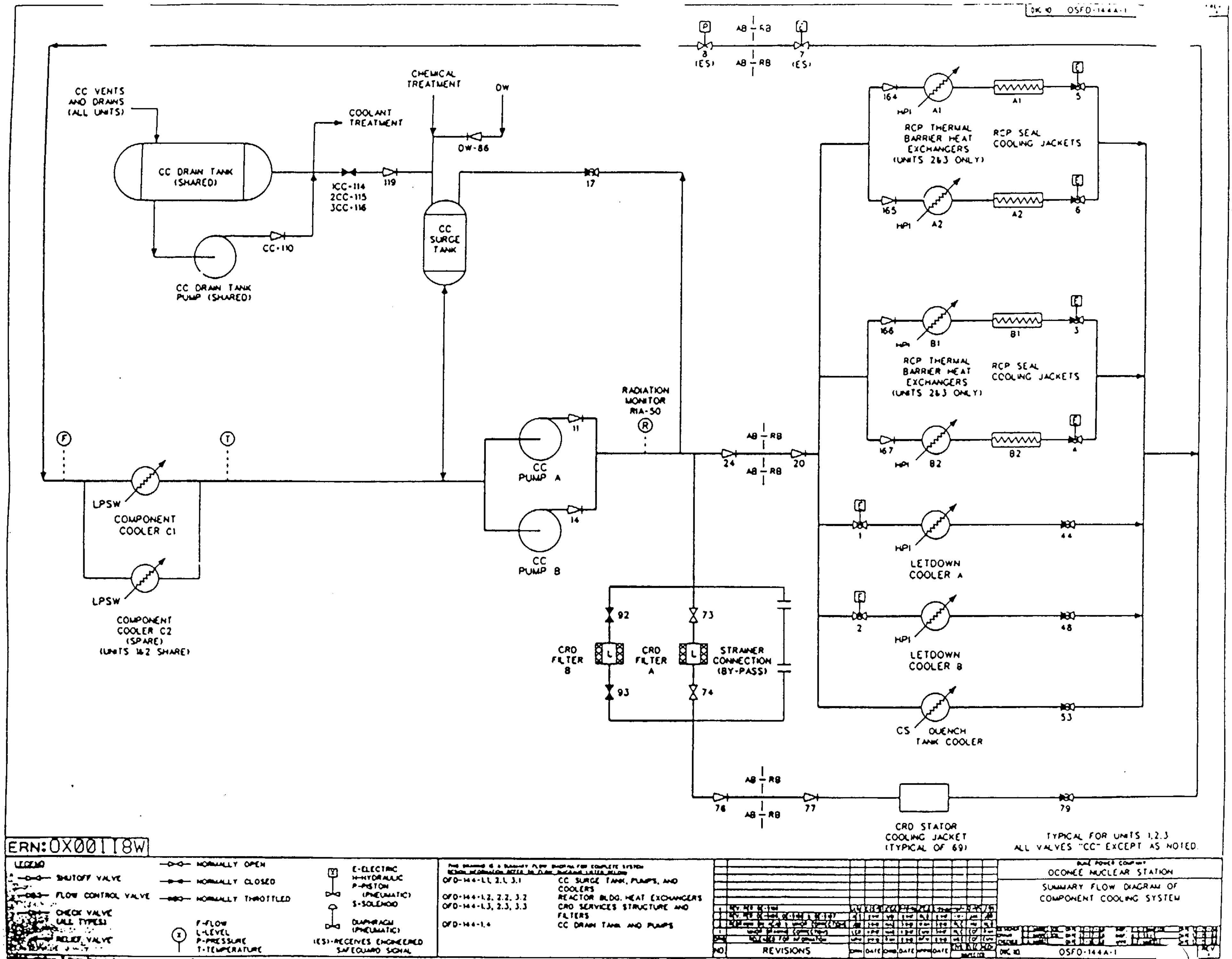
OFD-100A-1.2, 2.2, 3.2	QUENCH TANK AND COOLER
OFD-101A-1.2, 2.2, 3.2	COMPONENT DRAIN PUMP AND
OFD-107A-1.1, 2.1, 3.1	QUENCH TANK DRAIN PUMP
OFD-107A-1.2, 2.2, 3.2	LETDOWN STORAGE TANK PRESSURIZER

NO.	REVISIONS
6	REV. PER EQ. EX. M 16
4	REV. PER EQ. EX. M 16
3	REV. PER NSM ON-12907, 22907 & 32907
2	REDRAWN ON ACAD & MINOR CONNECTIONS
1	MINOR DRAWING CORRECTIONS
—	RELEASED FOR INFORMATION

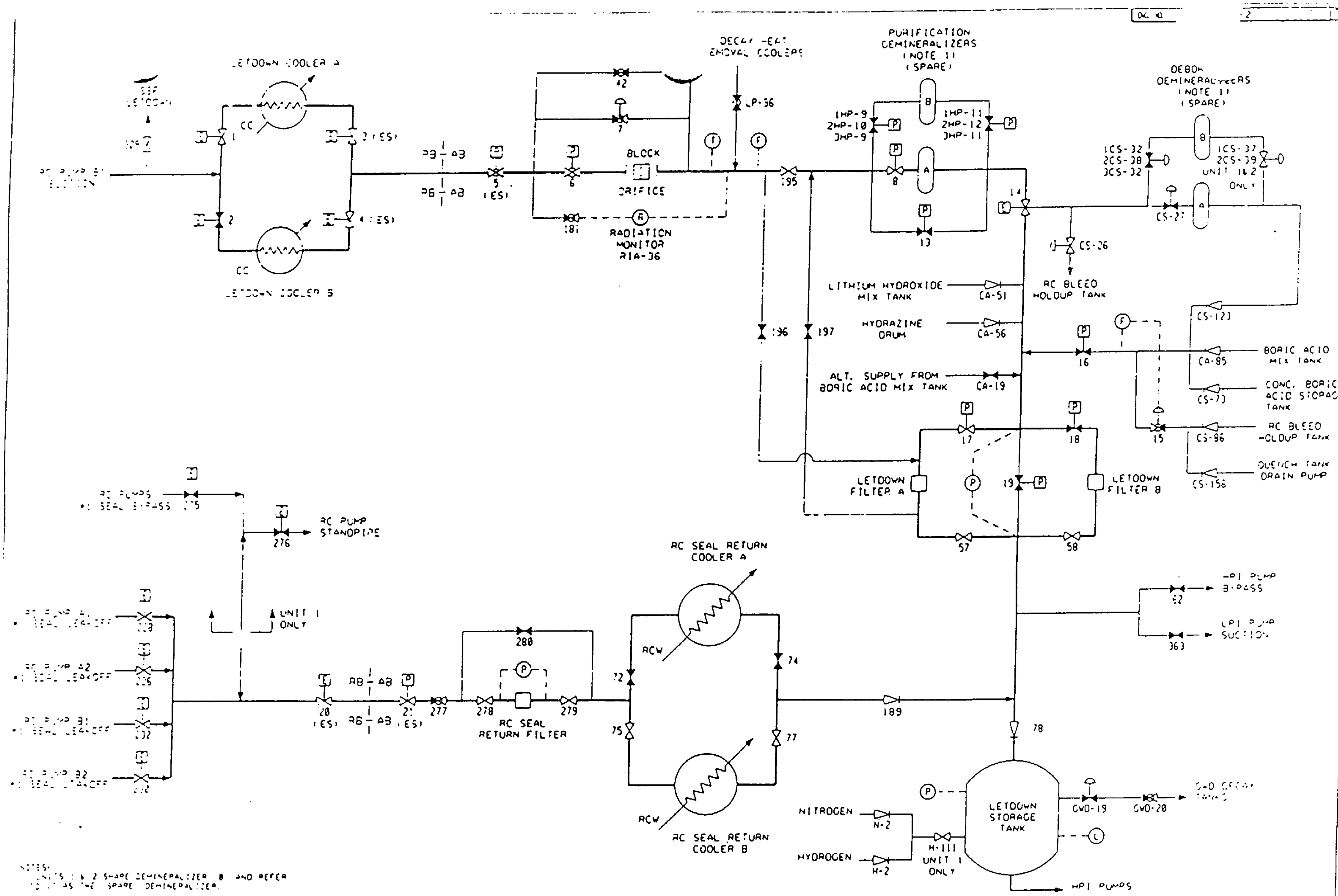
DUKE POWER COMPANY
OCONEE NUCLEAR STATION

SUMMARY FLOW DIAGRAM OF COOLANT STORAGE SYSTEM QUENCH TANK PORTION

DWG. NO. OSFD-107A-1 REV. 6



OSFD-144A-1
CC VENTS AND DRAINS (ALL UNITS)
CC DRAIN TANK (SHARED)
CC DRAIN TANK PUMP (SHARED)
CC-110
COOLANT TREATMENT
1CC-114
2CC-115
3CC-116
119
CHEMICAL TREATMENT
DW
DW-86
CC SURGE TANK
17
AB RB
8 (ES)
7 (ES)
164
165
166
167
HPI
A1
A2
B1
B2
RCP THERMAL BARRIER HEAT EXCHANGERS (UNITS 2&3 ONLY)
RCP SEAL COOLING JACKETS
RADIATION MONITOR RIA-50
CC PUMP A
CC PUMP B
11
14
24
20
LPSW
COMPONENT COOLER C1
COMPONENT COOLER C2 (SPARE) (UNITS 1&2 SHARE)
LETDOWN COOLER A
LETDOWN COOLER B
44
48
53
CS
QUENCH TANK COOLER
92
93
73
74
CRD FILTER B
CRD FILTER A
STRAINER CONNECTION (BY-PASS)
76
77
79
CRD STATOR COOLING JACKET (TYPICAL OF 69)
TYPICAL FOR UNITS 1,2,3
ALL VALVES "CC" EXCEPT AS NOTED.
ERN:0X00118W
LEGEND
SHUTOFF VALVE
FLOW CONTROL VALVE
CHECK VALVE (ALL TYPES)
RELIEF VALVE
NORMALLY OPEN
NORMALLY CLOSED
NORMALLY THROTTLED
F-FLOW
L-LEVEL
P-PRESSURE
T-TEMPERATURE
E-ELECTRIC
H-HYDRAULIC
P-PISTON (PNEUMATIC)
S-SOLENOID
DIAPHRAGM (PNEUMATIC)
(ES)-RECEIVES ENGINEERED SAFEGUARD SIGNAL
OFD-144-1.1, 2.1, 3.1
CC SURGE TANK, PUMPS, AND COOLERS
OFD-144-1.2, 2.2, 3.2
REACTOR BLDG. HEAT EXCHANGERS
OFD-144-1.3, 2.3, 3.3
CRD SERVICES STRUCTURE AND FILTERS
OFD-144-1.4
CC DRAIN TANK AND PUMPS
REVISIONS
DUKE POWER COMPANY
OCONEE NUCLEAR STATION
SUMMARY FLOW DIAGRAM OF COMPONENT COOLING SYSTEM
OSFD-144A-1



NOTES:
1. UNITS 1 & 2 SHARE DEMINERALIZER B AND REFER TO IT AS THE SPARE DEMINERALIZER.

RN: OX-221172

TYPICAL FOR UNITS 1, 2, 3
ALL VALVES HP EXCEPT AS NOTED

LEGEND
- SHUTOFF VALVE
- FLOW CONTROL VALVE
- CHECK VALVE (ALL TYPES)
- RELIEF VALVE
- NORMALLY OPEN
- NORMALLY CLOSED
- NORMALLY THROTTLED
- E-ELECTRIC, H-HYDRAULIC, P-PISTON (PNEUMATIC), S-SOLENOID
- DIAPHRAGM (PNEUMATIC)
- F-FLOW, L-LEVEL, P-PRESSURE
- (ES)-RECEIVES ENGINEERED SAFEGUARDS SIGNAL

THIS DRAWING IS A SUMMARY FLOW DIAGRAM. FOR COMPLETE SYSTEM DESIGN INFORMATION REFER TO FLOW DIAGRAMS LISTED BELOW.

OFD-100A-1.1, 2.1, 3.1	REACTOR COOLANT SYSTEM
OFD-100A-1.3, 2.3, 3.3	REACTOR COOLANT PUMPS
OFD-101A-1.1, 2.1, 3.1	LETDOWN & RC SEAL RETURN COOLERS
OFD-101A-1.2, 2.2, 3.2	LETDOWN FILTERS & STORAGE TANK
OFD-101A-1.3, 2.3, 3.3	HPI PUMPS
OFD-101A-1.5, 2.5, 3.5	SSF PORTION
OFD-105A-1.1, 2.1, 3.1	PURIFICATION DEMINERALIZERS
OFD-105A-1.2, 2.2, 3.2	DEBORATING DEMINERALIZERS

DUKE POWER COMPANY
OCONEE NUCLEAR STATION
SUMMARY FLOW DIAGRAM OF HIGH PRESSURE INJECTION SYSTEM (LETDOWN & RETURN)

OBJECTIVES

TERMINAL OBJECTIVE

Upon completion of this lesson, the student will be able to describe the purpose, operation, and response of the Component Cooling System during normal and abnormal plant conditions

ENABLING OBJECTIVES

1. Draw the Component Cooling System, showing the pumps, coolers, and major valves. (R1)
2. List the four (4) major categories of components cooled by the CC System. (R2)
3. State the cooling medium for the CC coolers. (R3)
4. Explain how the CC System acts as a barrier to prevent the release of radioactive liquid to the environment. (R4)
5. List the purpose(s) of the following CC System components: (R5)
   - 5.1 Coolers
   - 5.2 Pumps
   - 5.3 Surge tank
   - 5.4 Control Rod Drive Filters
   - 5.5 Return Penetration Block Valves, CC-7 and CC-8.
   - 5.6 Drain Tank and Pump
   - 5.7 G. RIA-50
6. Describe the corrosion inhibitor used in the CC System, how it protects the system, and its associated hazard to personnel. (R6)
7. Explain how repositioning of system valves can adversely affect CC System performance once the system has been set up for proper operation. (R7)
8. List the CC System controls and indications available to the operator in the control room. (R8)
9. Describe briefly the steps involved in startup of the CC System. (R9)
10. Describe the sequence and precautions necessary while valving in the spare CC cooler. (R10)

11. Explain the reason for draining the CRD service structure prior to pulling the reactor vessel head prior to refueling. (R11)

12. Describe the two methods of draining the CRD service structure. (R12)

13. Explain how CC-8 failing closed at power affects plant operation. (R13)

14. Describe briefly the steps involved in reopening CC-8 after the valve has failed closed because of a loss of Instrument Air. (R14)

15. Describe the six (6) interlocks and/or automatic actions associated with the CC System. (R15)

16. Explain why the CC System must be in operation: (R16)

    16.1 before letdown is established if RCS temperature is > 120° F

    16.2 if RCS temperature is > 190° F

17. Given a set of plant conditions, diagnose the cause of a CC System problem and/or determine the required corrective action. (R17)

18. Evaluate the overall affect on other plant systems based on the normal and/or abnormal operation of the CC system. (R18, R19)

E. Component cooling water vents and drains from all three units are collected in a common component cooling drain tank that is located on the first floor of the auxiliary building. A component cooling drain tank pump is provided to return the contents of the drain tank to each unit's CC surge tank.

F. CRD cooling water filters are provided to remove iron oxide and other contaminants that may be attracted by the magnetic field of the stator and subsequently block cooling water flow through the stator coils. Stainless steel piping is used between the filter discharge and the copper CRD cooling coils. The CRD cooling filters are located in the auxiliary building, just outside the reactor building. One filter is normally maintained in service with the second as a spare.

G. Radiation monitor RIA-50 monitors CC system activity for indications of an RCS leak into the system. The radiation monitor is located in the vicinity of the CC pumps

H. During normal operation, one CC pump and cooler re-circulates cooling water to accommodate the system requirements with the remaining pump and cooler acting as spares. The CC pumps are manually started and stopped from the control room. The standby pump will automatically start on low system flow. Flow in the CC system is balanced during unit startup by adjusting the electric and manually operated system throttle valves. Caution must be exercised to prevent repositioning any of the flow control valves once a balanced flow condition has been obtained.

I. Normal system component flows are:

Quench Tank Cooler	25 gpm
Letdown Cooler	400 gpm (each)
RCP Seal Coolers	50 - 60 gpm (each)
CRD Cooling Coils	145 - 150 gpm (total)

The CC system has a high flow limit of 1200 gpm (i.e., minimum system throttling or excessive leakage) and a low flow limit of 550 gpm. The standby pump automatically starts at 575 gpm decreasing system flow.

2.3 Component Description

A. Component Coolers

1. Purpose

The purpose of the component coolers is to transfer heat from the CC system to the LPSW system. Each component cooler is designed to remove the total CC system heat load for a unit.

2. Description

a) Component cooling return penetration outside block valve, CC-8, is a pneumatically operated valve that serves as the outside containment isolation valve for the component cooling return header.

b) A 125VDC solenoid valve controls the supply of air to position CC-8. When the solenoid valve is energized, CC-8 is closed.

c) In the event of an Engineered Safeguards channel 6 actuation, (actuated by high reactor building pressure) CC-8 automatically closes and is blocked from being remotely operated.

d) CC-8 limit switch contacts trip the CC pumps and prevents their restart when CC-8 is closed.

e) The solenoid valve for CC-8 is powered from 125VDC Power Panelboard DIB.

G. Component Cooling Drain Tank and Pump

1. One component cooling drain tank for all three units is located on the first floor of the auxiliary building.

   a) Provides a place to drain the system from which the CC water can be recovered.

   b) The CC System can also be drained directly to the low activity waste tank and/or the reactor building normal sump.

2. The CC drain tank pump is used for the following functions:

   a) To re-circulate the water in the CC drain tank.

   b) To pump from the CC drain tank to the CC surge tank for any of the three units.

   c) To pump from the CC drain tank to MWHUT or LHST "A"

H. RIA-50

1. A process radiation monitor which indicates leakage of primary coolant into the component cooling system. A likely location may be leaking letdown coolers.

2. Flow from the discharge of the CC pumps re-circulates through RIA-50 back to the suction of the CC pumps through a 1.25" line.

3. AP/1,2,3/A/1700/18, Abnormal Release Of Radioactivity, provides guidance in locating the source of leakage and minimizing the spread of contamination.

Exam Question Report

27-Jan-99

Question ID:	NRC070	Revision No:	0	Revision Date	10/29/1999
Question Description:	NRC070				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	Reference: 1997 ONS-NRC	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Given the following on Unit 1:

Initial conditions

- 100% Power
- Quench Tank in recirc
- 1A Letdown Cooler isolated for a suspected tube leak

Current conditions

- CC Surge Tank Level Hi/Low Statalarm actuates
- 1RIA-50, Component Cooling alarm is in alert

Which ONE of the following is the required operator action?

A) Secure Quench Tank from recirc

B) Close 1HP-5 (Letdown Isolation)

C) Close RCP Seal Cooler Outlet valves

D) Stop CC pumps and Close 1CC-7 and 8 (CC Return Penetration Blocks)

Answer

A

Lessons

ID	Description

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag
1997 ONS-NRC	Reference created by conversion		

KA'S

ID	Description

000026A2.02
NRC070

Question

Given the following on Unit 1:

Initial conditions

- 100% Power
- Quench Tank in recirc
- 1A Letdown Cooler isolated for a suspected tube leak

Current conditions

- CC Surge Tank Level Hi/Low Statalarm actuates
- 1RIA-50, Component Cooling alarm is in alert

Which ONE of the following is the required operator action?

A) Secure Quench Tank from recirc

B) Close 1HP-5 (Letdown Isolation)

C) Close RCP Seal Cooler Outlet valves

D) Stop CC pumps and Close 1CC-7 and 8 (CC Return Penetration Blocks)

Answer

A

QUESTION # 82

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	_____	1
Group #	_____	1
K/A #	**000029 G 2.4.8**	
Importance Rating	_____	3.7

Technical Reference(s): **EOP Section 506 p. #11**
EAP-E26 p. #13

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E26 OBJ. #4**

Question Source:
Bank # _______
Modified Bank # _______
New ___X___

Question History:
Previous NRC Exam _______
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 _____
55.43 __X__

Comments:

1 POINT

QUESTION # 82

SRO ONLY

Unit 3 plant conditions:

- RCS pressure = 2420 psig
- Reactor power = 13% and decreasing
- OAC Subcooling Margin Monitors are "flashing" 0000's
- 3HP-31 (RCP Seal Flow Control) is failed shut
- 3CC-8 (CC Return Outside Block) is failed shut
- The OATC is performing IMA's
- The BOP is performing RULE #2 (Loss of SCM Actions)

Which ONE of the following is correct?

A. Trip all RCPs per IMAs.

B. Trip all RCPs per RULE #2, Loss of SCM Actions.

C. Operate all RCPs until further directions are obtained from Section 506, UNPP

D. Operate all RCPs until further directions are obtained from AP/1700/16, Abnormal RCP Operation.

1 POINT

QUESTION # 82

000029 G2.4.8 (3.0/3.7) CFR 41.10/43.5/45.13 – SRO ONLY PRA 4-5-00 (GTH)

A. Incorrect – RCPs should remain operating in attempt to regain control of subcooling. The last step of IMAs checks for the operation of RCP seals and CC, if both are unavailable then the RCPs will be tripped. In the above case the last step of IMA will not be performed as RULE #1 transfers the EOP to Section 506 after the completion of the RULE #1, ATWS Actions

B. Incorrect - RCPs are normally tripped when subcooling margins = 0°F per RULE #2, Loss of SCM Actions but with an ATWS in progress this action will cause core damage as forced flow is stopped with the reactor operating at power. RCPs should remain operating in attempt to regain control of subcooling and provide forced flow conditions.

C. Correct - per EP/1/A/1800/001, Section 506, RCPs should remain running until both the loss of RCP Seal Injection (HP-31 failed shut) AND Component Cooling is addressed. This flow path through the EOP will insure reactor power is low enough to prevent core damage if RCPs are tripped.

D. Incorrect - RCPs should remain operating in attempt to regain control of subcooling although loss of seal injection and CC is an immediate RCP trip criteria. This action is not taken during an ATWS.

_____ 4. **WHEN** all Wide Range NIs are ≤ 1% power and decreasing,

THEN perform the following:

_____ 4.1 **IF** **AT ANY TIME** core subcooling margin ≥ 5°F,

THEN HPI may be throttled.

_____ 4.1.1 Reposition the following components as required:

- 1HP-26 (1A HP INJECTION)
- 1HP-27 (1B HP INJECTION)
- 1HP-410 (1HP-26 BYPASS)
- 1HP-409 (1HP-27 BYPASS)
- 1A, 1B, and 1C HPI Pumps.

_____ 4.2 **IF** **NEITHER** of the following is available,

- RCP Seal Injection
- Component Cooling,

THEN perform the following:

_____ 4.2.1 Trip all RCPs.

_____ 4.2.2 **REFER TO** AP/0/A/1700/025 (Standby Shutdown Facility Emergency Operating Procedure).

OBJECTIVES

TERMINAL OBJECTIVE:

1. Describe the use of Section 506, Unanticipated Nuclear Power Production, of the Emergency Operating Procedure in order to perform the required actions of a Nuclear Control Operator during an UNPP event.

ENABLING OBJECTIVES:

1. State when Section 506 of the EOP should be implemented. (R1)

2. Explain the basis for the entry conditions of this section. (R2)

3. Recognize that actual industry events have occurred where CRD breakers have failed to trip on demand and that prompt operator action was necessary to insure reactor shutdown. (R4)

4. Briefly explain why the reactor coolant pumps should remain in operation, even if RCS subcooling margin is lost. (R11)

5. State the three major operator actions that should be performed during an UNPP event based on Rule #1 (ATWS Actions). (R3)

6. Briefly explain why it is important that the turbine be tripped if NO Main FDW is available during an UNPP event.(R5)

7. State the two primary reasons why RCS expansion will occur during an UNPP event, requiring RCS letdown to be re-established.(R6)

8. Explain why the operator should control feedwater to match Rx power production during an UNPP event until RCS temperature stabilizes, i.e. no heatup or overcooling. (R7)

During an ATWS event, RCS pressure must increase to 2450 psig before DSS actuates. Shortly after operators begin to take action per Rule #1, DSS actuates and sometimes the operators don't understand what actions to take. The operators should understand the following:

- ***NEVER delay Rule #1 actions in anticipation of a DSS actuation. What if DSS doesn't actuate as it should?***
- ***Once Rule #1 actions have been started, complete them even if the regulating rods trip. The UNPP section of the EOP will give direction for securing from emergency boration.***

C. If the CRD system is NOT de-energized, dispatch an operator to open the CRD breakers (in the Cable Room and at the Load Centers).

The following section gives guidance for securing from emergency boration and checking for a SG tube leak and its subsequent release to the environment:

2.3 When all Wide Range NIs are ≤ 1% power and decreasing, then perform the following: (3)

A. IAAT core SCM ≥ 5°F, then HPI may be throttled.

This step gives clear guidance on when to secure from emergency boration to limit RCS inventory increases.

B. If RCP seal injection AND CC are NOT available, then trip all RCPs and refer to SSF EOP.

P&I 3-12-00 JMB

2.4 IF CETCs indicate superheated conditions, then go to Section 505 (Inadequate Core Cooling). (4)

ICC falls just behind UNPP based on EOP heirarchy. After actions are taken to shutdown the reactor, ICC conditions (superheat) must be addressed.

2.5 IF a SG Tube leak is indicated, then reduce Main Steam pressure to ≈ 950 psig to reseat MSRVs. (5)

This is prompt action to limit a release to the environment.

2.6 Go to Step 5.2 of Subsequent Actions. (6)

Since reactor shutdown has been accomplished, the remaining vital systems status verifications (Subsequent Actions) should be completed.

Going to step 5.2 skips the part in Subsequent actions where the operator checks to see if the reactor is shutdown and rods are on the bottom. If section 506 has been completed, the reactor has been successfully shutdown. Also by skipping the step about all rods being on the bottom we avoid getting in a "do-loop" that sends us back to 506 then to SA and then back to 506 over and over.

QUESTION # 83

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	1
	K/A #	**00004**	**A2.05**
	Importance Rating	_____	4.5

Technical Reference(s): **EAP-E23**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E23 #8 & #9**

Question Source:
- Bank # _______
- Modified Bank # **EAP 536**
- New _______

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 __X__

Comments:

1 POINT

QUESTION # 83

SRO ONLY

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Power = 50%

CURRENT CONDITIONS:

- 2A OTSG is isolated with pressure = 0 psig and steady
- 2B OTSG pressure = 650 psig and steady
- PZR level = 200" and increasing
- RCS pressure = 1500 psig and increasing
- Tc = 494° F
- All ES Channels 1 and 2 components in automatic

Which ONE of the following is correct?

SEE ATTACHMENT

If HPI ____ throttled the ________.

A. is not / TSOR curve may be violated causing concerns to reactor vessel integrity.

B. is not / NDT curve may be violated causing concerns to reactor vessel integrity.

C. is / RCP seals could be damaged due to loss of HPI and CC flow.

D. is / core could be damaged due to inadequate inventory.

1 POINT

QUESTION # 83

000040 A2.05 4.1/4.5 (CFR 45.3/45.13) SRO ONLY PRA 4-15-00 Bank modified EAP536

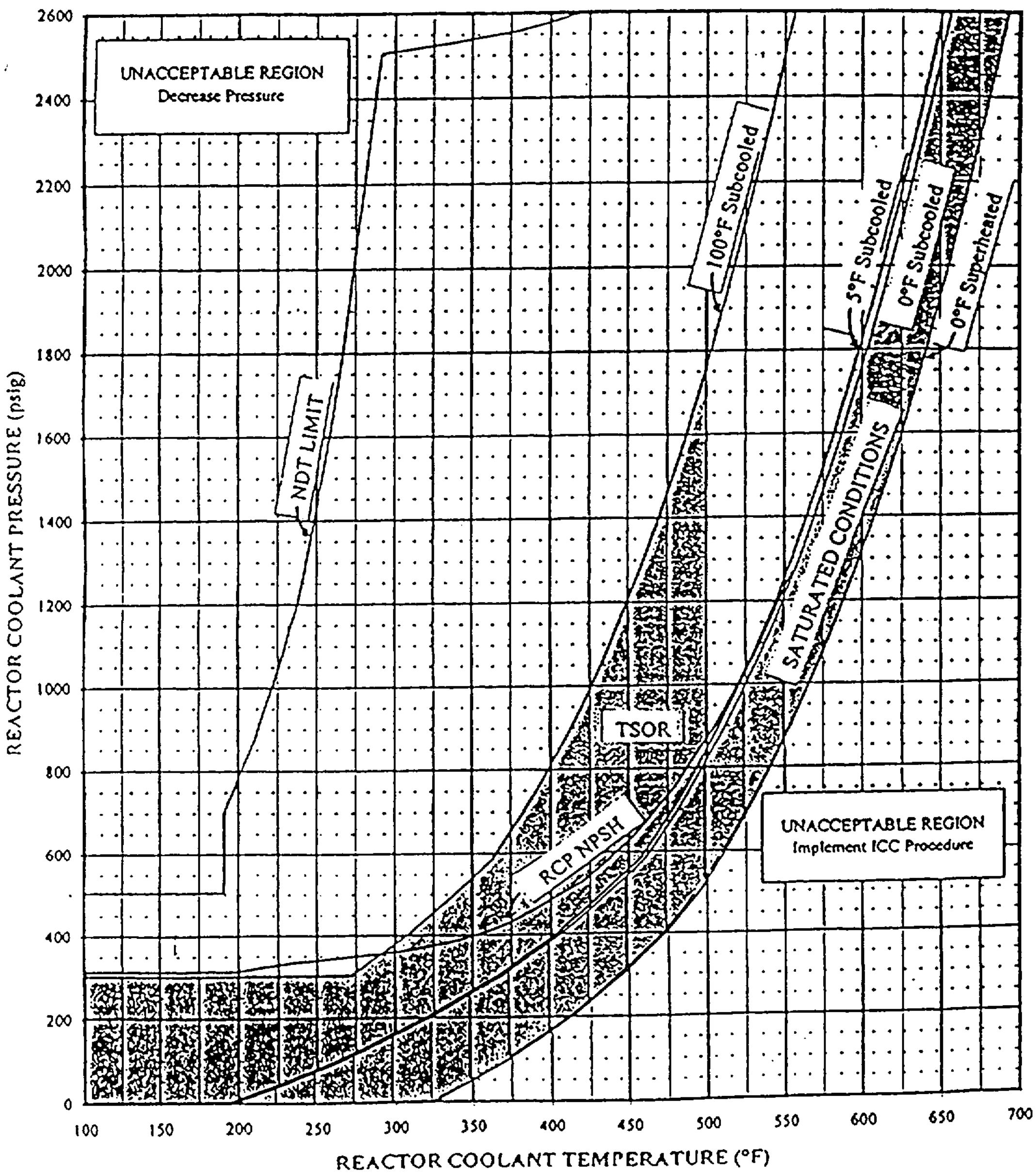
Question setup: A MS line break has occurred and an excessive cooldown has occurred. ES 1 and 2 has actuated but has not been throttled. If HPI is not throttled then RCS pressure will increase. If HPI is properly throttled then RCS pressure will be maintained constant.

A. Correct – When HPI is properly throttled HPI pressurization and cooling will be terminated since the Pri to Sec heat transfer matched (RC P/T should stablize). The TSOR is the most limiting curve to prevent PTS at this time. The TSOR curve is the 100°F subcooled curve.

B. Incorrect – If HPI is not throttled then high RCS pressure conditions will occur. RCS pressure would increase to the PORV setpoint which will prevent exceeding the design RCS pressure limit and reaching the NDT limit.

C. Incorrect – This could be true if ES 3 and 4 had actuated which would have isolated CC to the RCPs.

D. Incorrect – This could be true if the RCS was saturated and a LOCA was in progress rather than a MSLB.


UNACCEPTABLE REGION
Decrease Pressure
NDT LIMIT
100°F Subcooled
5°F Subcooled
0°F Subcooled
0°F Superheated
SATURATED CONDITIONS
TSOR
RCP NPSH
UNACCEPTABLE REGION
Implement ICC Procedure
REACTOR COOLANT PRESSURE (psig)
REACTOR COOLANT TEMPERATURE (°F)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
100
150
200
250
300
350
400
450
500
550
600
650
700

OBJECTIVES

TERMINAL OBJECTIVE

1. Describe the use of Section 503 (Excessive Heat Transfer) of the Emergency Operating Procedure in order to perform the required actions of a Control Room Operating crew during an event involving an excessive heat transfer transient and provide the operators guidance to properly use and understand the steps within the Excessive Heat Transfer section of the EOP.

ENABLING OBJECTIVES

1. Describe the conditions that would require entry into Section 503, Excessive Heat Transfer. (R1)
2. Recognize that pressurized thermal shock conditions may develop if HPI flow is not appropriately throttled during an overcooling event. (R2)
3. Explain the possible personnel safety hazard involved with reestablishing feedwater to a SG with a MS line leak; discuss the precautions that should be taken before feeding the SG. (R4)
4. Explain the concern involved with reestablishing feedwater to an intact SG that is dry. (R5)
5. Recognize that a transfer to Section 501, Loss of Subcooling, following an excessive heat transfer event, is made per the parallel actions step if any SCM is reading 0°F. (R6)
6. Describe the bases for securing the RBS System when RB pressure <10 psig if the Reactor Building radiation levels are normal. (R7)
7. Discuss the significance of maintaining SG tube to shell dT within limits following an excessive heat transfer event. (R9)
8. Define the term "Pressurized Thermal Shock". (R10)
9. Explain why the NDT curve does not have to be violated to run the risk of brittle fracture to the reactor vessel. (R11)
10. Demonstrate the ability to determine if operation in the Thermal Shock Operating Region is required, given plant conditions and the appropriate EOP enclosure. (R12)

32. **If** SG Tube Leak is indicated, **THEN GO TO** Section 504, SGTL
    - It is possible that a SGTL has occurred in addition to the overcooling transient, or that as a result of the excessive cooldown a tensile stress on the steam generator tubes has resulted and caused cracked, flawed or thinned tubes to fail. Since all excessive heat transfer symptoms have been mitigated, a transfer to the lower priority tube rupture guidance is appropriate if any indications of a SG tube leak exist.

33. Notify Chemistry to initiate sampling for indications of SG tube leakage.

Unit Status

(If no transfers out have been made)

- An overcooling event occurred.
- The RCS is subcooled with SG(s) steaming to the condenser, atmosphere, or RB.

34. Determine If:
    A. RCS must be depressurized to get into the TSOR
    B. RCS must be cooled down within the TSOR.
    - If an RCS cooldown occurs which results in a temperature transient where RCS temperature is < 500°F, determine if operation in the Thermal Shock Operating Region (TSOR) is required per the appropriate Enclosure 7.1 (P/T Curves).
    - Pressurized Thermal Shock (PTS) is the name given to a sequence of events whereby a reactor vessel experiences a severe thermal cycle (overcooling) followed by a repressurization of the system. If the overcooling is severe enough, the risk of brittle fracture of the reactor vessel exists.
    - PTS is of particular concern to Oconee since it is a safety concern primarily in older plants that have a larger degree of radiation induced embrittlement. Newer plants that have lower copper and phosphorus content in the reactor vessel welds are less susceptible to such embrittlement.

a) It is important to note that the NDT cooldown curve does not have to be violated to run the risk of brittle fracture to the reactor vessel. The NDT cooldown curve is based on a 50°F/halfhour cooldown, which can be exceeded during an excessive heat transfer event. Cooldown rates in excess of 50°F/halfhour can greatly increase the stress felt by the RV inner wall. This is why the TSOR region requirements should be followed (if required) following an overcooling event.

- If at any time during the transient either set of conditions has existed:

a) When RCPs are OFF, and RCS Temp (Tc) < 500°F, and HPI operation in injection mode (HP-26 or HP-27 used),

OR

b) When RCPs are ON, and cooldown rate exceeds 50°F/½ Hr, and 100°F temperature change in Tc occurs,

The 1 Hour hold in the TSOR is required (RC pressure and temperature stabilized) in order to allow the temperature gradient that exists across the RV wall to equalize and a cooldown within the TSOR is required.

- The cooldown rate required for operation in the TSOR when all RCPs are off is not as great as when at least one RCP remains in operation due to the lack of mixing of the HPI flow from the cold leg to the downcomer region. This reduced mixing (due to no RC flow) results in colder injection water being exposed to the vessel wall.
- The "one-hour hold" does not apply to a SBLOCA or a SGTR, (but operation in the TSOR does apply).
- Following a normal unit trip, typical RCS temperature response shows that Th, Tave, and Tc all approach a temperature of approx. 550°F.
  1) With forced circulation (RCPs operating) and during natural circulation, RCS cold leg temperature (Tc) is used for calculating cooldown rate.
  2) This indication of coolant temperature entering the vessel downcomer is the best approximation of the reactor vessel inner wall temperature.
  3) Thus, the rate of change of Tc is utilized to control reactor vessel cooldown rate.
- When calculating a cooldown rate during transients following a normal unit trip, a value of 550°F is to be utilized as the starting temperature for Tc. From this point the maximum cooldown rate limit is then 50°F in any 1/2 hour period.

- EOP guidance during SGTR provides for cooldown to 532°F limited only by ability to maintain Pressurizer level.

  4) Once 532°F is reached, a controlled cooldown rate must be established.

  5) The rapid 18°F cooldown must be accounted for in the cooldown rate calculation.

  6) In all cases, control cooldown rate so that the cooldown below 500°F does not happen until 1/2 hr has elapsed from the time cooldown below 550°F was initiated.

- If a transient occurs at a time other than following a unit trip, for example, from hot shutdown, then the starting temperature will be the last stable Tc indication.

- 550°F is the starting temperature for determining operation in the TSOR also. Following a trip, if RCS Tc drops below 450°F and at a rate > 50°F per half-hour, operation in the TSOR is required.

  7) If at any time during the one hour hold all subcooled margins $\geq$ 50°F then one RCP/loop should be started to enhance the equalization of temperature across the vessel wall. This, however, does not eliminate the need for operation in the TSOR for the remainder of the one hour period.

  8) If depressurization is required, then priority for depressurization is:

     (a) Use normal PZR spray (RCPs on)

     (b) Turn PZR heaters off and lower level

     (c) If ΔT < 410°F, Aux PZR Spray

     (d) Open RC-66 (PORV)

     (e) If ΔT > 410°F, Aux PZR spray (Emergency Coordinator authorization to violate 410°F dT).

        (1) CAUTION: Core subcooling margin should be maintained $\geq$ 5°F while depressurizing into TSOR.

  9) If the RCS does heatup after a SG is isolated due to excessive heat transfer, do <u>not</u> re-cooldown to get within the TSOR region of Enclosure 7.1.

- Although no PTS event, to date, has caused pre-existing flaws to propagate through a reactor pressure vessel, transients have occurred that demonstrate the potential for overcooling at pressure. Two such events occurred at B&W plants similar in design to Oconee; another occurred on Oconee Unit 1:

10) At Rancho Seco, the RCS was cooled from 582°F to about 285°F in slightly more than one hour, while RCS pressure was about 2000 psig. This was caused by a loss of most control room indications due to a loss of ICS and NNI power.

11) At Crystal River, again a loss of NNI power initiated a transient that caused the PORV to open, the HPI pumps to start on low RCS pressure and repressurize the RCS after the primary had cooled approximately 90°F in 30 minutes time.

12) At Oconee Unit 1, fire in 6900 VAC switch gear caused RCPs to be stopped, FDW system controls were lost, and RCPs off hampered RCS pressure control. Resulted in minimum Tc of 398°F just prior to RCP restart, and RCS pressure maximum spike of 2395 psig. 50°F/halfhour cooldown rate was exceeded.

35. IF the Main Steam or Feedwater pressure boundaries are ruptured in either SG, **THEN** Go to CP-604, Solid Plant Cooldown and establish or verify the existence of a PZR steam bubble.

36. **IF** MS and FDW boundaries are intact for both SGs,

THEN Further action is at the discretion of station management including when to exit this procedure

END

Exam Question Report

27-Jan-99

Question ID:	EAP536	Revision No:	0	Revision Date	10/29/1999
Question Description:	EAP536				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-E23 - Excessive Heat Transfer
Last Used Date:	03/03/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:		Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

A main steam line rupture which caused a rapid, excessive cooldown and loss of pressurizer level has occurred

Which ONE of the following describes the most limiting curve due to the possibility of reactor vessel brittle fracture?
(.25)

A) Nil Ductility Temperature (NDT) Limit

B) 100°F Subcooled Margin

C) 50°F Subcooled Margin

D) 20°F Subcooled Margin

Answer

B - correct - The TSOR is defined by the 100°F Subcooled Margin curve below 500°F. To the left of the TSOR (left of the 100°F SCM curve) the possibility of Brittle Fracture increases following an excessive cooldown situation until the stresses from the cooldown are relieved by maintaining P/T within the TSOR for 1 hour.
A - incorrect - The NDT curve does not define the region of possible brittle fracture if CDR have exceeded the Tech Spec cooldown rate limits. Brittle fracture can occur at much higher temperatures if Tech Spec CDRs are not adhered to.
B & C - incorrect - see A

Lessons

ID	Description
EAP-E23	Excessive Heat Transfer (EAP-E23)

Enabling Objectives

Exam Question Report

27-Jan-99

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

QUESTION # 84

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	_____	1
Group #	_____	1
K/A #	000057	A2.04
Importance Rating	_____	4.0

Technical Reference(s): **IC-ES**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-ES OBJ. #12 & #14**

Question Source:
- Bank # **IC-868**
- Modified Bank # _______
- New _______

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 __X__

Comments:

1 POINT

QUESTION # 84

SRO ONLY

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 100%

CURRENT CONDITIONS:

- 2KVIA AC Vital Power Panelboard supply breaker trips OPEN
- RB pressure = 3.8 psig and increasing

Which ONE of the following correctly describes the ES Channels that will actuate?

	ANALOG CHANNELS	/	DIGITAL CHANNELS
A.	A	/	1 and 2
B.	C	/	5 and 6
C.	A, B, and C	/	2, 4, and 6
D.	A, B, and C	/	1, 3, and 5

1 POINT

QUESTION # 84

000057 A2.04 3.7/4.0 SRO ONLY (CFR 43.5 / 45.13) rsi/gcw 04/24/00

A. Incorrect, - Analog channels will trip upon loss of Vital power and the Digital channels require power to trip. "A" ANALOG will trip. Digital channels 1 and 2 are actuated by RCS pressure < 1500 psig but, channels 3-6 should also actuate on high RB pressure >3 psig. Channel 1 will not trip due to the loss of power.

B. Incorrect, - Channel 5 digital will not trip due to loss of power. Also Channels 1-6 receive trip signal @ RB press > 3 psig. ODDs channels cannot trip due to loss of KVIA.

C. Correct, - KVIA feeds "A" ANALOG ES channels, which will trip/actuate upon loss of power. KVIA also feed the ODD (1, 3, and 5) Digital ES channels which require power to trip/actuate. Channels 1-6 receive an actuation signal from high RB pressure > 3 psig but, ONLY the EVEN (2, 4, and 6 Digital channels will trip/actuate. ANALOG channels A, B and C trip due to high RB pressure >3 psig.

D. Incorrect, - ODD Digital channels will not actuate as KVIA powers the ODD channels which require power to actuate.

12. Predict the response of ES analog and digital channels following a loss of power to: (R12)

    12.1 Analog channels

    12.2 Digital channels

    12.3 Analog and Digital channels simultaneously

13. Explain the actions necessary to manually trip and/or reset an analog or digital ESG channel. (R13)

14. Predict the emergency operation of the ESG analog and digital channels in response to a LOCA that results in RCS pressure gradually decreasing to ≈100 psig accompanied by a gradual increase in Reactor Building pressure to ≈15 psig. (R14)

15. Discuss the proper operation of all RZ Module controls and indications located on a unit's vertical control board in the Control Room. (R15)

16. Discuss and properly apply the guidance associated with repositioning ES equipment following an ES actuation. (R16)

17. Describe the actions necessary to properly return HPI pumps, Reactor Building Cooling Units and Keowee Hydro Units to normal operation following ES actuation. (R17)

2) This statalarm is also fed from the same RC pressure transmitter that feeds analog channel 'A' or 'B', whichever is selected with the amphenol connector to feed the CR recorder.

3) The same delay after alarm actuation may exist for the LPI bypasses as does for the HPI bypasses.

g) Performed in the Control Room.

1) HPI Bypass switches are on UB1 above the HPI instrumentation.

2) LPI Bypass switches are on UB2 above the LPI instrumentation.

h) Bypasses for LPI and HPI will automatically be removed whenever RC pressure is increased above 900 and 1750 psig respectively.

2. Inadvertent ESG Actuation

a) Operator must verify ESG actuation is not required by RB pressure or RC system pressure

b) Operator must take manual control of individual components and stabilize plant as soon as possible to:

1) Limit boration of the RCS

2) Limit pressurization transient

3) Insure cooling water restored to necessary components - RCPs, CRDMs

4) Limit chemical spray hazard to RB equipment

c) Operator must determine cause and correct problem with ESG system insuring required Tech Specs are met.

d) Possible causes of inadvertent actuation of ESG include:

1) Pressure Instrument failure while testing another Analog channel could result in ES actuation. (Example: while testing Channel 'A' Analog, RCS WR to Channel B fails low.)

2) Power spike on vital busses

Refer to OC-IC-ES-2

(a) Loss of power to an analog channel results in a trip signal being sent from that channel. (Only the outputs supplied from bistables - HPI, LPI and NR RB pressure; therefore, RBS will not actuate as a result of a vital power failure.)

(b) Digital channels must have power to actuate their associated safeguards action.

(c) KVIA Bkr #2 feeds Analog Ch A & Odd Dig Chnls

KVIB Bkr #2 feeds Analog Ch B & Even Dig Chnls

KVIC Bkr #2 feeds Analog Ch C

(d) Loss of KVIA and KVIC - Analog channels A & C send trip signal to digital channels 1-6. Only even channels 2, 4, and 6 actuate (KVIB powered) channels 7 and 8 do not receive a trip signal on a loss of analog channel power.

(e) Loss of KVIB and KVIC - Analog channels B and C send trip signal to digital channels 1-6. Only odd channels 1, 3, and 5 actuate (KVIA powered).

(f) Loss of KVIA and KVIB - Analog channels A and B send trip signal but no power available for digital channels - no actuation.

3. Manual Trip and Reset of ES Channels

a) Analog channels

1) Individual portions, HPI, LPI and RB pressure (4 psig), can be manually tripped by taking the rotary switch on the associated Pressure Test Module to the "Test Operate" position. There is one test module for WR RC pressure which will trip both the HPI and LPI bistables, and a separate test module for NR RB pressure which will trip the 3 psig RB pressure bistable.

2) Located in each Analog channel are the RC Pressure bistable modules (one for 1600# and one for 550#) and the Reactor building pressure trip bistable module.

If tripped, these three modules, and thus the channel, can be reset by depressing the red "output state" toggle switch (located on each bistable module) for each bistable that has tripped, when RC pressure is > 1600# (HPI), > 550# (LPI), or RB pressure < 3#.

3) As stated earlier, no action is required to reset analog outputs feeding channels 7 & 8. When RB pressure is < 10 psig the output signal will clear.

b) Digital channels

1) Can be manually tripped at the manual trip/reset panel in control room.

Refer to OC-IC-ES-10

3. When Rx. Building pressure increases to 3.0 psig (ITS≤ 4 #) the RB pressure trip bistables trip.
   a) Bistable output is fed to digital channels 5 & 6 and also through an OR gate to digital channels 1, 2, 3 & 4

Refer to OC-IC-ES-2

   b) This logic provides for actuation of RB cooling, PRV and Essential RB Isolation (Channels 5 & 6) as well as LPI & HPI on high RB pressure.
4. If Rx. Building pressure continues to increase to 10 psig (ITS ≤ 15 #)the RB spray system will actuate:
   a) Uses 6 pressure switches in a double, 2/3 logic.
   b) Three used with channel 7, and three are used with channel 8.
   c) Each pressure switch is fed to a contact buffer and from there straight to the digital channels.
   d) Channels 7 and 8 (Bldg. Spray) are the only digital channels that will not receive an analog trip signal if there is a loss of power to the analog cabinets.

2.6 RZ Module Operation

Refer to OC-IC-ES-12 thru 17

Some RZ modules provide indication only, while others allow control of components as well as indicating their status.

A. Module Indication/Control
   1. Auto lamp/pushbutton
      a) Lamp and control power for auto lamp/pushbutton:
         1) KVIA - Channels 1, 3, 5, 7
         2) KVIB - Channels 2, 4, 6, 8
         3) Control power is 120 VAC
         4) Lamp power is 120/24 VAC
      b) With no emergency signal present:
         1) Lamp is off.

Exam Question Report

27-Jan-99

Question ID:	IC868	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC868				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-ES - Engineered Safeguards
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	NLO = 12 Reference: SF2 013A1.01 (4.0/4.2)	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Unit 2 plant conditions:

- Reactor trip from 100% power
- 2KVIA AC Vital Power Panelboard supply breaker trips OPEN
- RB pressure = 3.8 psig and increasing

Which ONE of the following correctly describes the ES Channels that will actuate? (.25)

ANALOG CHANNELS / DIGITAL CHANNELS

A) A / 1 and 2

B) C / 5 and 6

C) A, B, and C / 2, 4, and 6

D) A, B, and C / 1, 3, and 5

Answer

C

A. Incorrect, KVIA Analog and Digital actuation channels of ES. Analog channels will trip upon loss of power and the Digital channels require power to trip. "A" ANALOG will trip. Digtals channels 1 and 2 are actuated by RCS pressure < 1500 psig but, channels 3-6 also actuate on high RB pressure >3 psig. Channel 1 will not trip. See "C".
B. Incorrect, Channels 3 and 4 are actuated by high RB pressure > 3 psig OR RCS pressure < 550 psig, but channels 1, 2, 5, and 6 also actuate. Channel 5 will not trip. See "C".
C. Correct, KVIA feeds "A" ANALOG ES channels which will trip/actuate upon loss of power. KVIA also feed the ODD (1, 3, and 5) Digital ES channels which require power to trip/actuate. Channels 1-6 receive an actuation signal from high RB pressure > 3 psig but, ONLY the EVEN (2, 4, and 6 Digital channels will trip/actuate. ANALOG channels A, B and C trip due to high RB pressure >3 psig.
D. Incorrect, See "C" ODD Digital channels will not actuate as KVIA powers the ODD channels which require power to actuate.

Lessons

ID	Description
IC-ES	Engineered Safeguards (IC-ES)

Enabling Objectives

ID	Description
ICESR12	12. Predict the response of ES analog and digital channels

QUESTION # 85

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	_____	1
Group #	_____	1
K/A #	000067 A2.15	
Importance Rating	_____	3.9

Technical Reference(s): **SLC 16.9**

Proposed references to be provided to applicants during examination: **SLC-16.9.2 & 4 & 6**

Learning Objective: **ADM-SRG #5**

Question Source:
- Bank # **ADM 946**
- Modified Bank # _______
- New _______

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 __X__

Comments:

1 POINT

QUESTION # 85

SRO ONLY

Unit 3 plant conditions:

- Reactor power = 100%
- The East Penetration Room fire hose station is isolated for hose replacement
  - 3HPSW-444, Hose Station Block is red tagged CLOSED
- Engineering notified the OSM that seven smoke detectors in the East Penetration Room are out of service

Which ONE of the following is the correct OSM direction to the NLO?

A fire watch shall be established for the East Pent. Room conducting a tour at least once every ____ minutes and backup suppression _______ required.

SEE ATTACHMENT

A. 15 / IS NOT

B. 60 / IS

C. 15 / IS

D. 60 / IS NOT

1 POINT

QUESTION # 85

000067A2.15 2.9/3.9 (CFR 43.5/45.13) SRO ONLY PRA 4-14-00 BANK-SRG17

A. Incorrect – hourly fire watch is required along with back-up suppression.

B. Correct – hourly fire watch meets the SLC requirements. 7 of 10 detectors are not operable for the area affected (>50%). Back-up suppression is required, as the fire hose station is isolated.

C. Incorrect – hourly fire watch is required since > 50% of the detectors are inoperable. Back-up suppression is required.

D. Incorrect – hourly fire watch is required with back-up suppression.

4. When given the applicable data be able to make correct parameter computations. (R4)

5. When given a set of plant conditions and/or senior reactor operator actions be able to predict plant/system/component response, or the effect on the same or other systems or components. (R5)

6. Demonstrate an understanding of the guidance or rules in procedures by locating the answer to specific SRO related questions. (R6)

7. Be able to recite, from memory, required procedural or administrative items detailed in Operations Management Procedure 2-1 (OMP 2-1, Encl. 4.9). (R7)

8. For APs, OMPs, SDs, Tech Specs, SLCs, and the EOP, become familiar with the content of each so as to be able to answer, from memory, questions relating to general systems alignments, available operator controls and instrumentation, bases for specific actions, and in the case of the EOP, the order of priority assigned for mitigating simultaneous casualties. (R8)

16.9 AUXILIARY SYSTEMS

16.9.2 Sprinkler and Spray Systems

COMMITMENT Sprinkler and Spray Systems in safety related areas listed in Table 16.9.2-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Sprinkler or Spray Systems inoperable. <u>AND</u> Affected Area(s) has no OPERABLE fire detection.	A.1 Establish continuous fire watch with backup fire suppression equipment in the area.	1 hour
B. One or more required Sprinkler or Spray Systems inoperable. <u>AND</u> Affected Area(s) has OPERABLE fire detection.	B.1 Establish hourly fire watch with backup fire suppression equipment in the area.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.2.1	------------------------NOTE------------------------ Not required to be performed for systems in the cable spreading room, equipment rooms and cable shafts. -- Functionally test each required Sprinkler or Spray System.	12 months
SR 16.9.2.2	Inspect each required Sprinkler System's spray headers and nozzles.	12 months
SR 16.9.2.3	Verify by visual inspection each nozzle's spray area to ensure spray pattern is not obstructed.	18 months

Table 16.9.2-1
Sprinkler and Spray Systems

a. Oconee Nuclear Station		
i.	Turbine Driven Emergency FDW Pump	Units 1, 2, and 3
ii.	Transformers[1]	CT-I, CT-2, CT-3, CT-4, and CT-5
iii.	Cable Room	Units 1, 2, and 3
iv.	Equipment Room	Units 1, 2, and 3
v.	Cable Shaft (3rd Level)	Units 1, 2, and 3
vi.	Cable Shaft (4th & 5th Level)	Units 1, 2, and 3
b. Keowee Hydro Station		
i.	Main Lube Oil Storage Room	
ii.	Main Transformer	

1. The transformers do not have fire detection devices. They have Activation devices that actuate the deluge valve of the fire suppression systems only.

BASES

The OPERABILITY of the NRC committed Fire Suppression System ensures that adequate fire suppression capability is available to confine and extinguish fires occurring at the Oconee or Keowee facilities. The regulatory requirement is to have NRC committed Sprinkler and Spray Systems OPERABLE only when the equipment it is protecting is required OPERABLE for plant safety. However, to protect the equipment for property conservation and minimize equipment loss due to fire; the Oconee and Keowee NRC committed Sprinkler and Spray Systems will be required to be OPERABLE at all times.

The Oconee CT-1, 2, 3, 4, and 5 transformers do not have fire detection devices. They have fire actuation devices that actuate the deluge valve of the fire suppression systems. These actuation devices do not directly annuciate to the Control Rooms. When the deluge valve trips, the flow pressure switch is the sensor that activiates the Control Room alarms. With HPSW deactivated for maintenance or testing, there is no form of annucation of a fire in the Control Room.

During periods of time when the Sprinkler or Spray System is not OPERABLE and detection instrumentation is OPERABLE, a hourly fire watch patrol will be required to inspect the affected area frequently as a precaution. If the Sprinkler or Spray System in the area is not OPERABLE and no detection instrumentation is OPERABLE, a continuous fire watch is required to be maintained in the vicinity of the affected Sprinkler or Spray System until the system is restored to OPERABLE status.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service.

The test requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met.

This Selected Licensee Commitment is part of the Oconee Fire Protection Program and therefore subject to the provisions of Oconee Facility Operating License conditions.

REFERENCES

1. Oconee UFSAR, Chapter 9.5-1.
2. Oconee Fire Protection SER dated August 11, 1978.
3. Oconee Fire Protection Review, (currently contained in the Fire Protection DBD), as revised.
4. Oconee Plant Design Basis Specification for Fire Protection, as revised.

16.9 AUXILIARY SYSTEMS

16.9.4 Fire Hose Stations

COMMITMENT The Fire Hose Stations listed in Table 16.9.4-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Fire Hose Station outside reactor building inoperable.	A.1 Provide additional equivalent capacity fire hose of length to reach unprotected area at OPERABLE hose station.	1 hour
B. Required Fire Hose Station inside reactor building inoperable (water not available to isolation valves LPSW-563 and LPSW-564).	B.1 Ensure availability of 4 portable fire extinguishers outside the reactor building in the personnel air lock area of the auxiliary building for fire brigade use upon entering reactor building.	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.4.1	Perform visual inspection, including inspection of coupling gaskets, of the fire hose stations located outside the reactor building and inside reactor building that are accessible during power operation.	31 days
SR 16.9.4.2	Perform visual inspection, including inspection of coupling gaskets, of reactor building fire hose stations that are inaccessible during power operation.	18 months
SR 16.9.4.3	Partially stroke test Fire Hose Station Valves.	36 months
SR 16.9.4.4	Subject each fire hose to hydrostatic test at pressure ≥ 50 psig greater than the maximum pressure at the station.	36 months
SR 16.9.4.5	Perform maintenance inspection including removal and reracking the hoses and inspection of coupling gaskets.	36 months

Table 16.9.4-1
Fire Hose Stations

a. Oconee Nuclear Station

Location No.	Valve No.	Area or Component Protected
3-D-28	2HPSW-194	1&2 Blockhouse, 1 & 2 3rd Floor Switchgear
AX-35	1HPSW-436	#1 Cable Spread Room
AX-32	2HPSW-436	#2 Cable Spread Room
AX-33	2HPSW-437	1 & 2 Cable Spread Room
AX-30	3HPSW-436	#3 Cable Spread Room
AX-31	3HPSW-437	#3 Cable Spread Room
5-M-31	2HPSW-304	1 & 2 Control Room, 1 & 2 Emergency Shutdown Panels
TOH-3	3HPSW-338	#3 Control Room, #3 Emergency Shutdown Panels
1-J-28	2HPSW-242	#1 First Floor MCCs HPSW Pumps, 1 & 2 LPSW Pumps
1-J-43	3HPSW-344	#3 1st Floor Motor Control Centers
1-B-19	1HPSW-283	#1 EFWP
1-D-39	2HPSW-236	#2 EFWP
1-D-53	3HPSW-336	#3 EFWP
AX-13	1HPSW-448	1 & 2 HPI Pumps, 1 & 2 LPI Pumps
AX-14	3HPSW-449	3 HPI Pumps, 3 LPI Pumps
1-J-47	3HPSW-348	3 LPSW Pumps
AX-36	1HPSW-445	#1 West Penetration Room
AX-45	1HPSW-444	#1 East Penetration Room
AX-42	2HPSW-444	#2 East Penetration Room
AX-43	2HPSW-445	#2 West Penetration Room
AX-29	3HPSW-444	#3 East Penetration Room
AX-44	3HPSW-445	#3 West Penetration Room
AX-21	HPSW-457	1 & 2 Equipment Room
AX-19	3HPSW-458	3 Equipment Room
3-M-24	HPSW-176	1 Equipment Room
3-M-29	2HPSW-245	2 Equipment Room
3-M-43	3HPSW-339	3 Equipment Room
3-J-28	2HPSW-241	1 & 2 3rd Floor Switchgear
3-M-43	3HPSW-339	3 3rd Floor Switchgear, 600V Load Center
AX-22	1HPSW-440	1 Battery Room
AX-20	2HPSW-440	2 Battery Room
AX-18	3HPSW-440	3 Battery Room
1RBH1	1LPSW-471	Ground Floor Level - East Side
2RBH1	2LPSW-471	Basement Floor Level - East Side
3RBH1	3LPSW-471	Basement - East side
1RBH2	1LPSW-473	Intermediate Floor Level - East Side
2RBH2	2LPSW-473	Intermediate Floor Level - East Side
3RBH2	3LPSW-473	Intermediate Floor Level - East Side
1RBH3	1LPSW-475	Top of Shielding Floor Level - East Side
2RBH3	2LPSW-475	Top of Shielding Floor Level - East Side
3RBH3	3LPSW-475	Top of Shielding Floor Level - East Side
1RBH4	1LPSW-465	Top of Shielding Floor Level - West Side
2RBH4	2LPSW-465	Top of Shielding Floor Level - West Side
3RBH4	3LPSW-465	Top of Shielding Floor Level - West Side
1RBH5	1LPSW-467	Intermediate Floor Level - West Side

Table 16.9.4-1
Fire Hose Stations

Location No.	Valve No.	Area or Component Protected
2RBH5	2LPSW-467	Intermediate Floor Level - West Side
3RBH5	3LPSW-467	Intermediate Floor Level - West Side
1RBH6	1LPSW-469	Ground Floor Level - West Side
2RBH6	2LPSW-469	Basement Floor Level - West Side
3RBH6	3LPSW-469	Basement - West Side
VBH-1	HPSW-916	Essential Siphon Vacuum Building
VBH-2	HPSW-917	Essential Siphon Vacuum Building
Basement	-	EL. 777' 6"
Ground	-	EL. 797' 6"
Intermediate	-	EL. 825' 0"
Top of Shielding	-	EL. 861' 0"

b. Keowee Hydro Station

Location No.	Valve No.	Area or Component Protected
Operating Deck (NW)	KH-1	Operating Floor
Operating Deck (NE)	KH-2	Operating Floor
Operating Deck (SW)	KH-4	Operating Floor
Operating Deck (SE)	KH-3	Operating Floor
Control Room	KH-6	Control Room
Mech. Equip. Gallery	KH-5	Mech. Equip. Gallery

ACTIONS

-----NOTE-----
OPERABILITY of fire detection instrumentation for adequate equipment/location coverage may also be determined by the Site Fire Protection Engineer or designee.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. > 50% of required detectors for one or more Oconee equipment/location inoperable. OR 2 required adjacent detectors for one or more Oconee equipment/location inoperable.	A.1 -----NOTE----- An hourly firewatch is not required for inaccessible equipment/locations such as the Reactor Building at power operation. Periodic inspections using a TV camera (if available) are permitted as described in Site Directives, or, the inaccessible equipment condition may be monitored by remote indications which would provide early warning of a fire. ---------- Establish hourly fire watch patrol (or as permitted by Site Directives) to inspect the accessible area with the inoperable instrumentation.	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. > 50% of required detectors for one or more Keowee equipment/location inoperable. OR 2 required adjacent detectors for one or more Keowee equipment/location inoperable.	B.1 Establish hourly fire watch patrol to inspect the accessible area with the inoperable instrumentation.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.9.6.1	Perform CHANNEL FUNCTIONAL TEST of Oconee Fire Detection Instruments using Fire Detection Instrumentation Control Board Panel Test Switch.	31 days
SR 16.9.6.2	Visually inspect Oconee Fire Detection Instruments accessible during power operation.	184 days
SR 16.9.6.3	Visually inspect Keowee Fire Detection Instruments.	184 days
SR 16.9.6.4	Test each Oconee fire detector for sensitivity.	12 months
SR 16.9.6.5	Perform CHANNEL FUNCTIONAL TEST of Keowee Fire Detection Instruments.	12 months
SR 16.9.6.6	-------------------------NOTE------------------------- Not required to be performed for Keowee Generator Detectors. -- Test each Keowee fire detector for sensitivity.	12 months
SR 16.9.6.7	Visually inspect Oconee Fire Detection Instruments not accessible during power operation.	18 months

Table 16.9.6-1
FIRE DETECTION INSTRUMENTATION

OCONEE NUCLEAR STATION

Units 1, 2. and 3 Reactor Buildings

Equipment	Detectors Provided
Reactor Building Penetrations	8 (each unit)
Reactor Building Cooling Units	6 (each unit)
Reactor Coolant Pumps	8 (each unit)

Units 1, 2, and 3 Auxiliary Building
EL. 822' +0

Room No.	Equipment	Detectors Provided
71-Q	Unit 1 Cable Shaft	2
510	Unit 1 and 2 Control Room	10
75-Q	Unit 2 Cable Shaft	2
552	Unit 3 Control Room	8
90-Q	Unit 3 Cable Shaft	2

EL. 809' + 3"

Room No.	Equipment	Detectors Provided
400	Unit 1 Control Battery Room	5
402	Unit 1 East Penetration Room	12
403	Unit 1 Cable Room and Cable Shaft	19
404	Unit 2 Cable Room and Cable Shaft	18
407	Unit 2 East Penetration Room	20
408	Unit 2 Control Battery Room	5
409	Unit 1 West Penetration Room	5
410	Unit 2 West Penetration Room	5
450	Unit 3 Cable Room	28
452	Unit 3 East Penetration Room	10
455	Unit 3 Ventilation Equipment	2
456	Unit 3 West Penetration Room	5
458	Unit 3 Control Battery Room	2

Exam Question Report

27-Jan-99

Question ID:	ADM946	Revision No:	0	Revision Date	10/29/1999
Question Description:	ADM946				

Exam Question Status		Exam Question Criteria	
Reference Flag:	X	Topic Area:	ADM-SRG - Shift Review Guide
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	NLO = R9; LRO = SLC 16.9.7 Reference: KA-APE 0000 67A2.15 2.9/3.9	Max. Point Value:	0.25
		Passing Point Value:	0.25

27-Jan-99

Question

Unit 3 plant conditions:

• Reactor power = 100%
• The East Penetration Room fire hose station is isolated for hose replacement (3HPSW-444, Hose Station Block CLOSED).
• Engineering has notified Operations that seven (7) smoke detectors in the zone for the East Penetration Room are OOS.

Which ONE of the following describes the FIRE WATCH required? (.25)

A fire watch shall be established for the East Penetration Room conducting a tour at least, once every __________ minutes and back-up suppression __________ required.

A) 15 / IS NOT

B) 60 / IS

C) 15 / IS

D) 60 / IS NOT

Answer

C
SLC 16.9.6 - Detector operability: no more that 50% in a zone inoperable or 2 adjacent.
SLC 16.9.4 - Fire hose stations: inoperability requires additional hoses (suppression) from an operable location in place within hour
NSD 316-5 - Definitions, continuous fire watch at 15 minute intervals

A. Incorrect, continuous fire watch is required along with back-up suppression.
B. Incorrect, hourly fire watch does not meet the SLC requirements for the area affected. Back-up suppression is required.

C. Correct, continuous fire watch is required since >50% of the detectors in the zone is inoperable. Back-up suppression is required since the East Pent. Rm. hose station is isolated/inoperable.
D. Incorrect, Refer to A, B and C. continuous fire is required with backup suppression.

Lessons

ID	Description
ADM-SRG	Shift Review Guide ADM-SRG

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag
16.9.7	Keowee Lake Level		X

KA'S

ID	Description

QUESTION # 86

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	**1**
	Group #	_____	**1**
	K/A #	**000074**	**2.4.41**
	Importance Rating	_____	**4.1**

Technical Reference(s): **RP/0/B/1000/01 Encl. 4.1**

Proposed references to be provided to applicants during examination: **RP/0/B/1000/01 Encl. 4.1**

Learning Objective: **EAP-SEP OBJ. #14**

Question Source:
- Bank # ________
- Modified Bank # **EAP 523**
- New ________

Question History:
- Previous NRC Exam ________
- Previous Quiz / Test ________

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 __X__

Comments:

1 POINT

QUESTION # 86

SRO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Time = 0300 (50 minutes after a reactor trip)
- Tc = 490°F
- RCS Pressure = 650 psig
- DEI = 6 uCi/ml
- 1RIA-57 = 45 R/hr
- 1RIA-58 = 25 R/hr

CURRENT CONDITIONS:

- Time = 0330
- Tc = 482°F
- RCS Pressure = 600 psig
- DEI = 287 uCi/ml
- 1RIA-57 = 492 R/hr
- 1RIA-58 = 153 R/hr

overlap with A.4
quantitative test

i.e. Are these were clearly initial

Which ONE of the following pairs of classifications is correct?

Initial conditions classification is _____ and the current conditions classification is _____.

SEE ATTACHMENT

A. Alert / Site Area Emergency

B. Alert / General Emergency

C. Site Area Emergency / Site Area Emergency

D. Site Area Emergency / General Emergency

1 POINT

QUESTION # 86

000074 2.4.41 SRO GCW 04/15/00

Question setup:

Enclosure 4.1 of RP/0/B/1000/001 (Fission Product Barrier Matrix) is used to determine classification.

Initial Conditions:
5 points for "RCS leak rate > available makeup....". Indicated by loss of SCM.
OR 5 points for RIA-57/58 > 1 R/hr.
Total of 5 points which is an Alert.

Current conditions:
RIA-57/58 have increase substantially. This gives points from all three areas.
RCS barriers = 5
Fuel Clad barriers = 5
Containment barriers = 1
Total = 11
11 points is a General Emergency.

A. Incorrect, An Alert is correct but should upgrade to General Emergency.

B. Correct, An Alert with upgrade to General Emergency.

C. Incorrect, If misread could add points for RIA-57/58 and start with SAE.

D. Incorrect, If misread could add points for RIA-57/58 and start with SAE and the upgrade would be correct.

Enclosure 4.1
Fission Product Barrier Matrix

DETERMINE THE APPROPRIATE CLASSIFICATION USING THE TABLE BELOW: CIRCLE EALS CHOSEN. ADD POINTS TO CLASSIFY. (SEE NOTE BELOW)

RCS BARRIERS (BD 5-7)		FUEL CLAD BARRIERS (BD 8-9)		CONTAINMENT BARRIERS (BD 10-12)	
Potential Loss (4)	Loss (5)	Potential Loss (4)	Loss (5)	Potential Loss (1)	Loss (3)
RCS Leakrate > Makeup capacity of one HPI pump in normal makeup mode (approx. 160 gpm) with Letdown isolated.	RCS Leak rate > available makeup capacity as indicated by a loss of subcooling	Average of the 5 highest CETC ≥ 700° F	Average of the 5 highest CETC ≥ 1200° F	CETC ≥ 1200° F ≥ 15 minutes OR CETC ≥ 700° F ≥ 15 minutes with a valid RVLS reading 0"	Rapid unexplained containment pressure decrease after increase OR containment pressure or sump level not consistent with LOCA
SGTR > Makeup capacity of one HPI pump in normal makeup mode (approx. 160 gpm) with Letdown isolated.		Valid RVLS reading of 0"	Coolant activity ≥ 300 μCi/ml DEI	RB pressure ≥ 59 psig OR RB pressure ≥ 10 psig and no RBCU or RBS.	Failure of secondary side of SG results in a direct opening to the environment with P/S leakage ≥ 10 gpm in the same SG
Entry into the TSOR (Thermal Shock) operating range	1RIA 57/58 reading ≥ 1.0 R/hr 2 RIA 57 reading ≥ 1.6 R/hr 2 RIA 58 reading ≥ 1.0 R/hr 3RIA 57/58 reading ≥ 1.0 R/hr		Hours Since SD — RIA57/58 - R/hr 0 - < 0.5 — ≥ 300/150 0.5 - < 2.0 — ≥ 80/40 2.0 - 8.0 — ≥ 32/16	Hours Since SD — RIA57/58 - R/hr 0 - < 0.5 — ≥ 1800/860 0.5 - < 2.0 — ≥ 400/195 2.0 - 8.0 — ≥ 280/130	Failure of secondary side of SG results in a direct opening to the environment with P/S leakage ≥ 10 gpm in the other SG AND Feeding SG with secondary side failure from the affected unit
HPI Forced Cooling	RCS pressure spike ≥ 2750 psig			Hydrogen concentration ≥ 9%	Containment isolation is incomplete and a release path to the environment exists
Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment

UNUSUAL EVENT (1-3)	ALERT (4-6)	SITE AREA EMERGENCY (7-10)	GENERAL EMERGENCY (11-13)
OPERATING MODE: 1, 2, 3, 4 • Any potential loss of Containment • Any loss of containment	OPERATING MODE: 1, 2, 3, 4 • Any potential loss or loss of the Fuel Clad • Any potential loss or loss of the RCS	OPERATING MODE: 1, 2, 3, 4 • Loss of any two barriers • Loss of one barrier and potential loss of either RCS or Fuel Clad Barriers • Potential loss of both the RCS and Fuel Clad Barriers	OPERATING MODE: 1, 2, 3, 4 • Loss of any two barriers and potential loss of the third barrier • Loss of all three barriers
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1,2,3,4

NOTE: An event with multiple events could occur which would result in the conclusion that exceeding the loss or potential loss threshold is IMMINENT (i.e., within 1-3 hours). In this IMMINENT LOSS situation, use judgment and classify as if the thresholds are exceeded.

12. Briefly describe the required actions for each of the following procedures/events covered by the Site Emergency Plan:(R-10)
    12.1 Control Room Emergency Coordinator, (RP/0/B/1000/02)
    12.2 Security Event (RP/0/B/1000/07)
    12.3 Site Assembly (RP/0/B/1000/09) including how Personnel Accounting is reported, plus time limits for reporting
    12.4 Emergency Evacuation/Relocation of Site Personnel (RP/0/B/1000/10) including the different locations Personnel may be Relocated/Evacuated to per this procedure.
    12.5 Planned Emergency Exposure (RP/0/B/1000/11)
    12.6 Medical Response (RP/0/B/1000/16)
    12.7 Spill Response (RP/0/B/1000/17)
    12.8 Fire Damage assessment and Repairs (RP/0/B/1000/22)
13. Given a copy of the appropriate portions of the Site Emergency Plan, demonstrate the ability to implement the following procedures as they apply to the reactor operator: (R-11)
    13.1 Site Assembly (RP/0/B/1000/09)
    13.2 Medical Response (RP/0/B/1000/16)
14. Given a copy of the appropriate portions of the Site Emergency Plan, demonstrate the ability to implement the following procedures as they apply to the senior reactor operator:(R-12)
    14.1 Emergency Classification (RP/0/B/1000/01).
    14.2 Control Room Emergency Coordinator, (RP/0/B/1000/02)
    14.3 Security Event (RP/0/B/1000/07)
    14.4 Site Assembly (RP/0/B/1000/09)
    14.5 Emergency Evacuation/Relocation of Site Personnel (RP/0/B/1000/10)
    14.6 Planned Emergency Exposure (RP/0/B/1000/11)
    14.7 Medical Response Emergency (RP/0/B/1000/16)
    14.8 Spill Response (RP/0/B/1000/17)
    14.9 Fire Damage assessment and Repairs (RP/0/B/1000/22)

5. Procedure Use (Refer to RP/0/B/1000/01)

   a) This procedure is used by the Emergency Coordinator and/or EOF Director to determine the Emergency Classification for EVENTS that have occurred or are occurring at the Oconee Site.

      1) The Event or Events are compared to the Procedure's Classification **Matrix**.

      2) This **Matrix** is a cross-reference of different Events and/or Emergency Action Levels compared to the appropriate classification level [based on the severity of the Event or Emergency Action Level].

   b) The first step in the use of this procedure is to determine the operating MODE that existed at the time the event occurred prior to any operator action initiated in response to the event.

   c) If the unit was at Hot S/D or above, and one or more fission product barriers have been affected, refer to Encl. 4.1 and review the criteria listed to determine if the event should be classified.

      1) For each Fission Product Barrier, review the associated EALs to determine if there is a Loss or Potential Loss of that barrier. Circle any that apply.

         NOTE: An event with multiple events could occur which would result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e. within 1-3 hours). In this situation, use judgment and classify as if the thresholds are exceeded.

      2) Three possible outcomes exist for each barrier. No challenge, potential loss, or loss. Use the worst case for each barrier and the classification table at the bottom of the page to determine appropriate classification.

3) The numbers in parentheses out beside the label for each column can be used to assist in determining the classification. If no EAL is met for a given barrier, that barrier will have 0 points. The points for the columns are as follows:

Barrier	Failure	Points
RCS	Potential Loss	4
	Loss	5
Fuel Clad	Potential Loss	4
	Loss	5
Containment	Potential Loss	1
	Loss	3

4) To determine the classification, add the highest point value for each barrier to determine a total for all barriers. Compare this total point value with the numbers in parentheses beside each classification to see which one applies.

EXAMPLE: Failure to properly isolate a 'B' MS Line Rupture outside containment, results in extremely severe overcooling.

TSOR entry conditions were satisfied.

Stresses on the 'B' S/G resulted in failure of multiple S/G tubes.

RCS leakage through the S/G exceeds available makeup capacity as indicated by loss of subcooling margin.

Barrier	EAL	Failure	Points
RCS	SGTR > Makeup capacity of one HPI pump in normal makeup mode with letdown isolated	Potential Loss	4
	Entry into TSOR operating range	Potential Loss	4
	RCS leakrate > available makeup capacity as indicated by a loss of subcooling	Loss	5
Fuel Clad	No EALs met and no justification for classification on judgment	No Challenge	0
Containment	Failure of secondary side of SG results in a direct opening to the environment	Loss	3

RCS _5_ + Fuel _0_ + Containment _3_ = Total _8_

(a) Even though two Potential Loss EALs and one Loss EAL are met for the RCS barrier, credit is only taken for the worst case (highest point value) EAL, so the points from this barrier equal 5.

(b) No EAL is satisfied for the Fuel Clad Barrier so the points for this barrier equal 0.

(c) One Loss EAL is met for the Containment Barrier so the points for this barrier equal 3.

(d) When the total points are calculated the result is 8, therefore the classification would be a Site Area Emergency.

d) After referring to the Fission Product Barrier matrix, the next step is to review the listing of enclosures to determine if the event is applicable to one of the categories shown. If any of the categories might be affected, refer to those enclosures to see if any EALs are satisfied to determine the appropriate classification.

   1) To use the other classification enclosures (4.2 - 4.7), compare plant conditions with the criteria in the matrices. If the criteria are satisfied, refer to the top of the column to determine the classification.

e) If the conditions require an emergency classification, go to RP/0/B/1000/02, Control Room Emergency Coordinator Procedure.

6. Other Enclosures in RP/0/B/1000/01

   a) Enclosure 4.8, Radiation Monitor Readings for Emergency Classification. This Encl. is referenced by both Enclosure 4.3 and the Subsequent Actions of RP/0/B/1000/02, Control Room Emergency Coordinator Procedure. By comparing RIA-57, and 58 readings with values in the table, an event may need to be classified as a Site Area Emergency or a General Emergency.

   b) Enclosure 4.9, Unexpected/Unplanned Increase in Area Monitor Readings. This Encl. is also referenced by Enclosure 4.3. By comparing Area Monitor readings with values in the table, an event may need to be classified as an Unusual Event or an Alert.

   c) Enclosure 4.10, Definitions. This Encl. provides definitions for some of the terms utilized in the Emergency Plan.

Exam Question Report

27-Jan-99

Question ID:	EAP523	Revision No:	0	Revision Date	10/29/1999
Question Description:	EAP523				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-SEP - Station Emergency Plan
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	SRO = 10 Reference: RP/0/B/1000/01; RP/1000/01	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

SRO ONLY

The following Unit 1 conditions are observed fifty (50) minutesafter a unit trip caused by a lockout of 1TA and 1TBswitchgear:

- Tc is 490°F
- RCS Pressure is 1580 psig
- Pressurizer level is 25 inches and increasing
- LDST level is 40 inches and increasing
- ES Channels 1 & 2 have actuated
- DEI is 6 uCi/ml
- 1RIA-57/58 are reading 85/45 R/hr

Which ONE of the following is the correct Emergency Plan classification for these conditions? (.25)

A) Unusual Event

B) Alert

C) Site Area Emergency

D) General Emergency

Answer

C
A. Incorrect - The points accumulated if the TSOR is used will be a total of nine with the Rad monitor reading. The minimum is an SAE.
B. Incorrect - The higher of the two values for the column are added. This would be based on Rad monitor readings. The accumulated points are > 6, so it is not an alert.
C. Correct - The accumulated points are ten. Five (5) for the Rad monitor reading in the RCS Barrier column and five (5) for the Fuel Clad Barrier column. The ten points accumulated are a Site Area Emergency.
D. Incorrect - Points accumulated are less than 11 so it cannot be classified as a General Emergency.

Lessons

QUESTION # 87

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	**1**
	Group #	_____	**2**
	K/A #	**000009 A2.04**	
	Importance Rating	_____	**4.0**

Technical Reference(s): **EOP Section #604**
EAP-E35

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E35 #6 & #7**

Question Source:
- Bank # ________
- Modified Bank # ________
- New ___X___

Question History:
- Previous NRC Exam ________
- Previous Quiz / Test ________

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 __X__

Comments:

SRO ONLY

Unit 1 plant conditions:

- EOP Section 604, Solid Plant Cooldown in progress
- PZR level = 308 inches and decreasing
- A PZR steam bubble is being established PZR HTR BANKS ARE energized
- 1RC-4 (PORV Block) operable and OPEN
  1RC-3 closed
- 1RC-1 (PZR Spray) failed OPEN
- 1A1 RCP is operating
- PZR tail pipe thermocouples are indicating:
  - 1RC-66 = 205°F
  - 1RC-67 = 122°F
  - 1RC-68 = 129°F

Which ONE of the following will prevent PZR steam bubble formation?

A. 1A1 RCP trips with a RV head void.

operator deenergizes

B. PZR heater Bank's 2 and 4 de-energized.

C. Leakage path through 1RC-66 (PZR PORV).

operator opens

D. ~~Leakage path through~~ 1RC-3 (PZR Spray Block).

spray vlv failed open and no mention of 1RC3

1 POINT

QUESTION # 87

SRO ONLY

Unit 1 plant conditions:

- EOP Section 604, Solid Plant Cooldown in progress
- PZR level = 308 inches and decreasing
- A PZR steam bubble is being established
  - All PZR heaters are energized
- 1RC-4 (PORV Block) ~~operable and OPEN~~ closed
- 1RC-1 (PZR Spray) failed OPEN
- 1RC-3 (PZR Spray Block) operable and closed
- 1A1 RCP is operating

Which ONE of the following will prevent PZR steam bubble formation?

A. 1A1 RCP trips with a RV head void.

B. Operator de-energizes PZR heater bank #2.

C. ~~Leakage path through 1~~RC-66 (PZR PORV). failed open

D. Operator fully opens 1RC-3 (PZR Spray Block).

No leakage path indicated since you deleted data from original question.

1 POINT

QUESTION # 87

000009 A2.04 (3.8/4.0) CFR 43.5 / 45.13 SRO ONLY - PRA 3-23-00 (GTH)

A. Incorrect – A PZR can be established if in a natural circulation cooling mode. EOP Section 604 cautions the operator on conditions when establishing a PZR bubble with and established RV Head void. The EOP mitigation strategy being accomplished is to first establish a PZR bubble that can be controlled and then remove the RV Head void later in the cooldown.

B. Incorrect – A bubble can be established with only two PZR Heater Banks, the process will take longer but 604 step 13.2 directs the operator to energize all available PZR heaters and if no heaters are available then notify the TSC.

C. Incorrect – The leakage path can be isolated with 1RC-4 (PORV Block), Section 604 directs this action.

D. Correct – A PZR bubble cannot be established with continuous spray from 1A1 RCP. The leakage path cannot be isolated, as 1RC-1 (PZR Spray) has failed open and 1RC-3 (Spray Block) has an established leakage path.

LESSON PLAN OBJECTIVES

Terminal Objectives

1. Describe the use of CP-604 (Solid Plant Cooldown) of the Emergency Operating Procedure in order to perform the required actions of a Nuclear Control Operator during an event which involves the use of this procedure.(T1)

2. Describe the use of CP-604 (Solid Plant Cooldown) of the Emergency Operating Procedure in order to direct the required actions of a Nuclear Control Operator during an event which involves the use of this procedure. (T2)

Enabling Objectives

1. State the purpose of this procedure. (R1)

2. Recognize that during solid plant conditions manual control of HPI, PZR heaters, and heat transfer may be required to maintain RCS pressure constant. (R2)

3. Briefly explain why changes in the RCS temperature significantly affect primary system pressure when the plant is solid. (R3)

4. Objective R4 deleted.

5. Explain why the Pressurizer is assumed to be solid at 375 inches. (R5)

6. Explain why RC-4 (PORV block) and RC-3 (Spray block) are closed if steam space leaks are indicated when establishing a steam bubble in the Pressurizer. (R6)

7. Recognize that a steam bubble will begin to form when RCS pressure decreases below the saturation pressure for the existing Pressurizer temperature. (R7)

8. State the reasons for steaming a previously isolated S/G and unisolating a leaking PORV prior to restarting an RCP when Pressurizer level > 375". (R8)

9. Be able to evaluate the plant conditions for the need to restart RCPs.

10. Be able to evaluate the plant conditions for the consequences of restarting a RCP.

2. PRESENTATION

2.1 Initiate Normal Letdown (Step ~~1~~ 1&2) PIT 2/24/00

A. Since the subcooled margin is >0°F and secondary heat transfer is available, HPI Forced Cooling is no longer needed for decay heat removal and may be terminated.

1. The sequence of actions given here and in subsequent steps should dampen any abrupt RCS pressure increase, which could result from operating against a water solid system. This should result in a smooth recovery from HPI Forced Cooling.

2. Letdown through HP-5 is initiated to help dampen pressure surges and accommodate the RCS expansion as a pressurizer bubble is drawn.

B. Subcooled Margin ≥ 5°F

1. If Core Subcooling Margin ≥ 5°F, Then throttle HPI header flow to maintain RCS P/T within the proper region of the appropriate Encl. 7.1 (P/T Curves)

2. HPI flow must be throttled to maintain a 5°F subcooled margin and prevent the RCS pressure from opening the PORV and/or the pressurizer safety relief valves.

2.2 Secure RCS Bleed Pathways (Steps ~~2~~ 3 - 6)

A. Verify Closed all High Point Vent Valves. (Step ~~2~~ 3)

1. All high point vent valves should be closed, if open, as a step of the HPI Forced Cooling recovery action. This will secure these bleed pathways and help in the effort to regain pressurizer pressure control.

B. If Core Subcooling Margin ≥ 5°F, Then throttle HPI header flow to maintain RCS P/T within the proper region of the appropriate Encl. 7.1 (P/T Curves)(Step ~~3~~ 4)

C. Select required setting on RC-66 Setpoint Selector. (Step ~~4 – 5~~ 5 & 7) PIT 2/24/00

1. The PORV should be placed in HIGH or LOW, depending on RCS temperature, to allow automatic RCS pressure control during and after HPI Forced Cooling recovery.

a) If RCS > 325°F, select High on RC-66 Setpoint Selector.

b) If, at any time, RCS Press < 475 psig And < 325°F, select Low on RC-66 Setpoint Selector.

D. Ensure open RC-4. (step ~~6~~ 8)

E. These actions may result in an increase in RCS pressure since all primary system bleed paths have been isolated and the HPI pumps may be pumping against a water solid system.

1. For this reason, the operator must carefully control HPI flow and be ready to promptly throttle HPI as the PORV cycles closed. Otherwise, the operator may soon have to deal with RV thermal shock conditions.

2.3 Control and stabilize RCS pressure. (Steps 9+ ~~7 and 8~~)

A. Stabilize ~~Maintain~~ RCS P/T ~~Constant~~. (Step 9) PIT 2/24/00

1. Steam generator pressure should be controlled to maintain the existing RCS temperature and to stabilize prior to establishing a pressurizer bubble. This is especially important if the RCS is water-solid since any change in the average fluid density will directly affect the system pressure.

   a) A 1^0F change in RCS temperature will result in about a 100 psi change in RCS pressure.

B. If possible, re-establish normal makeup from LDST. (Step 10 ~~8~~) PIT 2/24/00

1. Normal RCS makeup should be re-established, if possible, to approach typical plant operations during the cooldown.

2. If a higher than normal makeup flow is required to maintain RCS inventory, then excessive leakage is indicated. Measures should be initiated to identify and isolate the leakage.

3. Adequate LDST inventory is checked prior to securing HPI suction from the BWST. (step 8.4)

C. If leakage is Indicated past RC-66, Then Close RC-4. (Step 10.5 ~~8.5~~) PIT 2/24/00

1. The PORV block valve should not be closed unless leakage is indicated and then only after the recovery sequence is completed. Isolation may be needed to draw a steam bubble later.

2.4 Check BWST Level (Step 11 ~~9~~) PIT 2/24/00

A. If BWST level < 35 feet, then initiate makeup to BWST with boron concentration > COLR limit.

1. Ensures continuous HPI pump suction source maintained, and also replenishes inventory lost during HPI cooling.

2.5 Check Pressurizer Level (Step 12 ~~10~~) PIT 2/24/00

A. If Pressurizer Level < 300 Inches, Then Go To CP-605, Subcooled Cooldown.

1. Whenever the pressurizer level is <300 inches, the operator is assured that a substantial steam bubble exists in the pressurizer. He should proceed with the cooldown by transferring to CP-605, Subcooled Cooldown.

2.6 Establish Pzr. Steam Bubble (Steps ~~11–12~~ 13)

A. CAUTION

1. If all RCPs off, RV Head or Hot Leg voids may exist.
2. If PZR level >375", the RCS may be water-solid
   a) When instrument errors are considered, the pressurizer may be water-solid when the indicated level exceeds 375 inches. A water-solid RCS and pressurizer will be extremely sensitive to perturbations in pressure since there is no steam cushion to dampen the effects.

B. NOTE ~~11~~ 13 P3I 2/24/00 WM

1. Manual control of HPI, PZR Heaters, and heat transfer may be required to maintain RCS pressure constant.
   a) This note was added to remind operators that the RCS is very sensitive to HPI inventory control (makeup/letdown) or RCS Temperature changes. An event occurred at the Salem Nuclear Station (SOER 94-1) where the operators did not properly control RCS Temperature and Inventory and allowed the RCS Pressure to reach their PORV Setpoint during Solid-Water conditions. Their PORV cycled rapidly between 100 and 200 times each before RCS Pressure and Inventory were brought under control!
   b) Recognize that in a Water-Solid condition our PORV and CODE relief valves continue to provide overpressure protection, but that our CODE relief valves may not properly reseat after passing water. This could become a small break event until the relief valve reseats, if it reseats at all.

C. If PZR steam space leaks are indicated, then close RC-4 (PORV Block) and RC-3 (Spray Block). (Step ~~11.1~~ 13.1) P3I WM 2/24/8

1. Either of these conditions should be corrected since it would be difficult to establish bubble otherwise.
   a) Steam formed by the flashing in the PZR would escape or be quenched by spray.

D. Energize <u>all</u> available PZR Heaters. (Step ~~11.2~~ 13.2) P3I WM 2/24/00

1. If Normal Power is NOT available to PZR heaters, then notify the TSC to evaluate use of the SSF PZR Heaters.
2. Pressurizer heaters should be manually energized as necessary to increase the pressurizer water temperature to the corresponding saturation temperature for the existing RCS pressure.

P3I WM 2/24/00

E. If a steam bubble cannot be established in the PZR, GO TO step ~~13~~ 15 (Step ~~11.3~~ 13.3) P3I 2/24/00 WM

F. Any increase in RCS pressure observed while heating the Pressurizer liquid should be accommodated by throttling HPI or increasing letdown. This will enable the RCS pressure to remain constant as the fluid in the pressurizer expands due to heater operation. (Step ~~11.4~~) 13.4 PJC wtd 2/24/00

G. Monitor PZR temperature. (Step ~~12~~) 14 PJC wtd 2/24/00

1. When the pressurizer finally reaches saturation, HPI flow should be throttled and/or letdown increased to allow a bubble to be drawn.

   a) As the outsurge of pressurizer liquid begins to reduce the level, saturated fluid will flash and help maintain RCS pressure as the bubble is formed.

   b) This process should continue until a level < 300 inches is obtained.

   c) HPI and letdown rate must be carefully handled while lowering PZR level to assure RCS SCMs remain ≥ 5°F.

2. **Note** ~~12~~ 14 PJC wtd 2/24/00

   a) If all RCPs are off, pressurizer level should be verified to decrease consistently with changes in makeup or letdown flow.

      1) Any inconsistencies could indicate that a void exists external to the pressurizer and that it is either developing or condensing.

      2) Potential void formations will <u>not</u> be a concern when any RCP is operating because of the excellent mixing associated with forced circulation.

      3) With no RCPs on, RV Head level can be used to verify void formation.

   b) Void indication should not halt efforts to draw a bubble.

   c) Removal of RCS voids will be handled in CP-605 once a PZR bubble has been drawn.

2.7 Check for Steam Generator Tube Leak (Step ~~13~~) 15 PJC wtd 2/24/00

A. If A SGTL Is Indicated, Then Go To Step 12 of Section 504, <u>SG Tube Leak</u>.

1. Prior to initiating a transfer to CP-605 or initiating a water-solid cooldown, a check for a SGTL is made.

2. If a SGTL exists, then a transfer to Section 504 is in order since guidance unique to cooling down with a SGTL is provided there.

~~2.8 Check Pzr. Level (Step 14)~~ 16 PJC wtd 2/24/00

~~A. <u>If at any time</u> Pressurizer Level <300 Inches, Then Go To CP-605, <u>Subcooled Cooldown</u>.~~

Delete PJC 2/24/00 wtd

See Pg 9a of B

_____ 10.5 **IF** leakage is indicated past 1RC-66 (PORV),

THEN close 1RC-4 (PZR RELIEF BLOCK).

_____ 11. **IF** BWST level < 35 feet,

THEN initiate makeup to the BWST with boron concentration > COLR limit.

- **REFER TO** OP/1/A/1104/004A (BWST Operation).

_____ 12. **IF** Pzr level < 300",

THEN **GO TO** CP-605, Subcooled Cooldown.

CAUTION 13:	• If all RCPs are off, RxV Head or Hot Leg voids may exist. • If Pzr level > 375", the RCS may be water solid.

NOTE 13: Manual control of HPI, Pzr Heaters, and heat transfer may be required to maintain RCS pressure constant.

_____ 13. Establish a steam bubble in the Pzr.

- Maximum allowable heatup or cooldown rate for the Pzr is 90°F/hr.

_____ 13.1 **IF** Pzr steam space leaks are indicated,

THEN close the following valves:

_____ 1RC-4 (PZR RELIEF BLOCK)

_____ 1RC-3 (SPRAY BLOCK).

_____ 13.2 Energize all available Pzr Heaters.

_____ 13.2.1 **IF** normal power is **NOT** available to the Pzr Heaters,

THEN notify TSC to evaluate use of the SSF Pzr Heaters.

_____ 13.3 **IF** a steam bubble **CANNOT** be established in the Pzr,

THEN **GO TO** Step 15.

_____ 13.4 **IF** RCS pressure increases with Pzr heaters on,

THEN perform the following as necessary:

_____ Throttle makeup

_____ Increase letdown to accommodate RCS expansion.

NOTE 14:

- With RCPs off, a Rx Vessel Head void may be indicated if changes in makeup and letdown flow are **NOT** consistent with Pzr level changes.
- Void indication should **NOT** halt efforts to draw a Pzr bubble.
- Void will be removed in CP-605, Subcooled Cooldown.

_____ 14. **IF** **AT ANY TIME** Pzr temperature is equal to Pzr T_{sat},

THEN perform the following:

_____ Maintain all the RCS subcooling margins ≥ 5°F.

_____ Reduce Pzr level to < 300" to draw a bubble.

QUESTION # 88

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	______	1
	Group #	______	2
	K/A #	**000027 A2.04**	
	Importance Rating	______	4.3

Technical Reference(s): **PNS-PZR**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PZR OBJ. #11**

Question Source:
- Bank # ________
- Modified Bank # ________
- New ___X___

Question History:
- Previous NRC Exam ________
- Previous Quiz / Test ________

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 __X__

Comments:

1 POINT

QUESTION # 88

SRO ONLY

Unit 1 conditions:

INITIAL CONDITIONS:

- Reactor Power = 100%

CURRENT CONDITIONS:

- 1RC-1 (PZR Spray) fails open

Which ONE of the following is correct?

they will cycle 1 time!

ASSUME NO OPERATOR ACTIONS

RCS pressure will decrease and...

A. PZR Heaters will energize and continuously cycle on and off between setpoints. it will return to normal for a moment

B. PZR Heaters will energize at setpoint and ~~return~~ stabilize RCS pressure to normal.

C. reactor will trip, ES Channels 1 and 2 will actuate with PZR spray controlling RCS pressure below PZR relief valves setpoint.

D. reactor will trip, ES Channels 1 and 2 will actuate with PZR relief valves maintaining RCS pressure below safety limit.

1 POINT

QUESTION # 88

000027 A2.04 SRO (CFR 43.5/45.13) SRO ONLY GCW 04/15/00 NEW

A. Incorrect, PZR spray will overcome the heat input from PZR heaters.

B. Incorrect, PZR spray will overcome the heat input from PZR heaters.

C. Incorrect, HPI will cause PZR to become solid and spray will have no affect.

D. Correct, HPI will fill the PZR and when solid HPI will repressurize the RCS up to the PZR relief valve setpoints.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

Upon completion of this lesson, the student will demonstrate an understanding of the components, indications, controls and operation of the Pressurizer. The student will be able to assess the status of the Pressurizer during normal, abnormal and emergency conditions and determine corrective actions for improper system operation. The student will also be able to apply any ITS/SLC Conditions and Required Actions associated with the Pressurizer.(T1)

ENABLING OBJECTIVES

1. Explain the design basis of the pressurizer. (R21)
2. Describe pressurizer response during load or RCS temperature changes. (R1)(R2)(R3)
3. Given a set of conditions, calculate the change in pressurizer level for a change in RCS temperature. (R33)
4. Explain what is meant by a "subcooled" pressurizer and how to determine if the pressurizer is in a subcooled condition.(R22)(R27)
5. Explain what is meant by a pressurizer "hard bubble" and describe the adverse effects of a "hard bubble" on plant operation, (R23)
6. Identify the source of pressurizer spray for each unit. (R4)
7. Discuss the automatic setpoints and any interlocks associated with pressurizer instrumentation. (R5)
8. Explain the operation of the ICS RC pressure signal median select function as it relates to RC pressure control including: (R28)
   - 8.1 How median select chooses the controlling signal
   - 8.2 Which pressurizer components receive a median selected RC pressure signal.
9. Given a set of conditions, determine which RC pressure signal has been selected for control by the ICS RC pressure signal median select function. (R36)
10. Discuss the reasons for bypass flow around the pressurizer spray valve during normal operation. (R6)
11. Evaluate plant response to a failed open pressurizer spray valve without operator action. (R20)
12. Explain the operation of the Pressurizer Water Space Saturation Recovery Circuit. (R29)

4. AC motor operated valve, RC-3, is used in series with the spray valve for remote spray line isolation. This is a throttle valve, so the switch must be held depressed for full valve travel, and should be held for approx. 3 seconds after closed indication is received to ensure the valve is fully closed.

5. RC-2, (Spray Control Bypass), continuously circulates reactor coolant through the spray loop bypassing RC-1. Since the pressurizer is a remotely located component connected to a hot leg, this bypass flow minimizes temperature differentials in the spray and surge lines, prevents thermal shock of the spray nozzle, and minimizes the boron concentration difference between the pressurizer and RCS.

6. The heat removal capability of pressurizer spray exceeds the heat input capability of the pressurizer heaters. If the spray valve were to fail open and no operator action was taken, RCS pressure would slowly decrease, the reactor would trip and eventually automatic engineered safeguards actuation would occur due to continually decreasing RCS pressure.

   a) In December 1991, Crystal River Unit 3 experienced a reactor trip and ECCS actuation due to low RCS pressure.

   b) The cause of this event was a failed open pressurizer spray valve with a concurrent failure of the valve position indication in the closed position.

7. Pressurizer spray is most effective if the RCP that supplies spray is operating. Pressurizer spray experiences a decrease in effectiveness if the RCP that supplies PZR spray is secured and a loss of normal pressurizer spray capability occurs if all RCPs must be secured.

   a) At a Combustion Engineering plant (Millstone 2), a transient initiated by the loss of a DC bus resulted in the plant being at hot shutdown with only two RCPs operating. The two RCPs that were operating provided no significant pressurizer spray flow. The operator, however, did not associate the ineffectiveness of spray with an incorrect RCP combination, and incorrectly diagnosed that a "hard bubble" had resulted from collection of non-condensable gases in the pressurizer.

   b) As a result, about 2 hours into the transient, RCS pressure had increased to the point where the PORV cycled to control pressure. Subsequently, it was recognized that inadequate spray flow was the cause and auxiliary spray was aligned to depressurize and control RCS pressure.

   c) From this we can see that without spray flow the pressurizer behaves much like it would if it were filled with non-condensable gases. Prompt recognition of reduced spray flow due to ineffective RCP combinations can preclude challenging the PORV by either establishing the most effective RCP combination or initiating auxiliary spray.

QUESTION # 89

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	______	1
Group #	______	2
K/A #	000037	G 2.4.18
Importance Rating	______	3.6

Technical Reference(s): **EOP Encl. 7.1A**
EAP-E23

Proposed references to be provided to applicants during examination: **EOP Encl. 7.1A**

Learning Objective: **EAP-E23 OBJ. #10**

Question Source:
Bank # ______
Modified Bank # __X__
New ______

Question History:
Previous NRC Exam **1998 Q. #34 (MODIFIED)**
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ______
55.43 __X__

Comments:

1 POINT

QUESTION # 89

SRO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS:

- MS Line "B" rupture at 100% power
- OTSG "A" tube leak = 150 gpm

CURRENT CONDITIONS:

- Plant has been stabilized with 1B OTSG isolated
- RCS "A" Loop Tc is 495°F and steady
- RCS pressure is 1300 psig and steady
- All RCPs are secured
- Reactor Building pressure = 4.1 psig

Which ONE of the following describes the action that must be taken by the operating crew?

Operation in the TSOR is…

SEE ATTACHMENT

A. required; cooldown RCS as necessary to minimize SCM.

B. required; depressurize and prevent RCS heatup.

C. NOT required; cooldown RCS as necessary to minimize SCM.

D. NOT required; depressurize and prevent RCS heatup.

1 POINT

QUESTION # 89

000037 G2.4.18 CFR41.10/43.2/45.6 SRO ONLY – PRA 4-12-00 (1998 ONS NRC Exam) (GTH/PMS)

A. Incorrect – Operation in TSOR is NOT required. It is necessary to minimize SCM but the operator must depressurize not cooldown to reduce SCM. If cooldown is established this would increase SCM.

B. Incorrect – Operation in TSOR is NOT required. It would be required if only a Main Steam line break had occurred.

C. Incorrect – Operation in TSOR is NOT required per Enclosure 7.1. It is necessary to minimize SCM but the operator must depressurize not cooldown to reduce SCM. If cooldown is established this would increase SCM.

D. Correct – Operation in TSOR is NOT required per Enclosure 7.1. It is desired to depressurize to minimize SCM.

OBJECTIVES

TERMINAL OBJECTIVE

1. Describe the use of Section 503 (Excessive Heat Transfer) of the Emergency Operating Procedure in order to perform the required actions of a Control Room Operating crew during an event involving an excessive heat transfer transient and provide the operators guidance to properly use and understand the steps within the Excessive Heat Transfer section of the EOP.

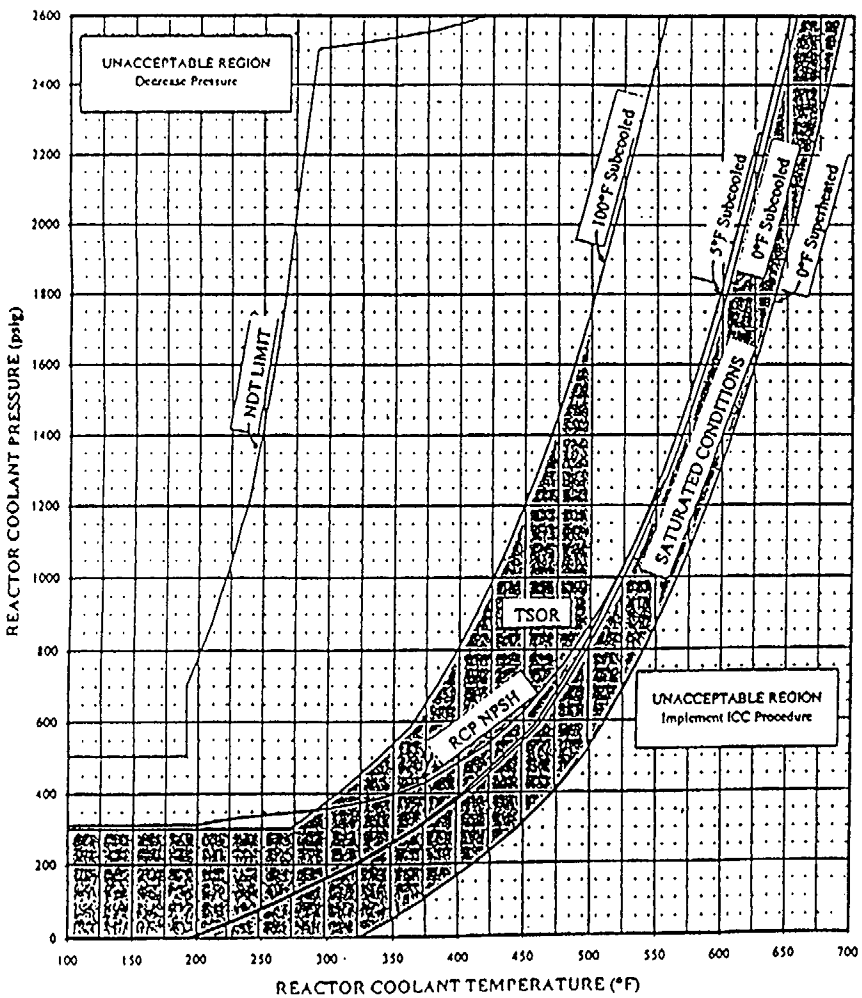
ENABLING OBJECTIVES

1. Describe the conditions that would require entry into Section 503, Excessive Heat Transfer. (R1)
2. Recognize that pressurized thermal shock conditions may develop if HPI flow is not appropriately throttled during an overcooling event. (R2)
3. Explain the possible personnel safety hazard involved with reestablishing feedwater to a SG with a MS line leak; discuss the precautions that should be taken before feeding the SG. (R4)
4. Explain the concern involved with reestablishing feedwater to an intact SG that is dry. (R5)
5. Recognize that a transfer to Section 501, Loss of Subcooling, following an excessive heat transfer event, is made per the parallel actions step if any SCM is reading 0°F. (R6)
6. Describe the bases for securing the RBS System when RB pressure <10 psig if the Reactor Building radiation levels are normal. (R7)
7. Discuss the significance of maintaining SG tube to shell dT within limits following an excessive heat transfer event. (R9)
8. Define the term "Pressurized Thermal Shock". (R10)
9. Explain why the NDT curve does not have to be violated to run the risk of brittle fracture to the reactor vessel. (R11)
10. Demonstrate the ability to determine if operation in the Thermal Shock Operating Region is required, given plant conditions and the appropriate EOP enclosure. (R12)

Enclosure 7.1A

Wide Range Pressure Transmitter Cooldown Limits

Normal Containment (P_{RB} < 3 psig): Use OAC or ICCM.
Abnormal Containment (P_{RB} > 3 psig): Use ICCM only.



CAUTION: All subcooling margins should be maintained ≥ 5°F while depressurizing into TSOR.

1. Maintain the RCS within the TSOR when either of the following conditions exist:

   1.1 **IF** AT ANY TIME all RCPs off
   AND any RCS temp (T_c) < 500°F,
   AND HPI has operated in injection mode (1HP-26, 1HP-27, 1HP-410 or 1HP-409 used),
   THEN depressurize the RCS to maintain T_c in the TSOR.

   1.2 **IF** AT ANY TIME RCPs on,
   AND Cooldown rate exceeds 50°F/½ Hr,
   AND 100°F temperature change in T_c occurs,
   THEN depressurize the RCS to maintain T_c in the TSOR.

NOTE 2: The TSOR 1 hour hold is NOT applicable for SBLOCAs or SGTRs.

2. **IF** operation in the TSOR is required,
   THEN stabilize RCS conditions in the TSOR and maintain for 1 hour.
   - Prevent any significant heatup or repressurization.

3. **IF** depressurization is required,
   THEN methods for depressurization are:

   3.1 Use normal Pzr spray (RCPs on).

   3.2 Turn Pzr heaters off and lower level.

   3.3 If ΔT < 410°F, Aux. Pzr Spray.

   3.4 Open 1RC-66 (PORV).

   3.5 If ΔT > 410°F, Aux Pzr spray (Emergency Coordinator authorization to violate 410°F ΔT).

32. **If** SG Tube Leak is indicated, **THEN GO TO** Section 504, SGTL

    - It is possible that a SGTL has occurred in addition to the overcooling transient, or that as a result of the excessive cooldown a tensile stress on the steam generator tubes has resulted and caused cracked, flawed or thinned tubes to fail. Since all excessive heat transfer symptoms have been mitigated, a transfer to the lower priority tube rupture guidance is appropriate if any indications of a SG tube leak exist.

33. Notify Chemistry to initiate sampling for indications of SG tube leakage.

Unit Status

(If no transfers out have been made)

- An overcooling event occurred.
- The RCS is subcooled with SG(s) steaming to the condenser, atmosphere, or RB.

34. Determine If:

    A. RCS must be depressurized to get into the TSOR

    B. RCS must be cooled down within the TSOR.

    - If an RCS cooldown occurs which results in a temperature transient where RCS temperature is < 500°F, determine if operation in the Thermal Shock Operating Region (TSOR) is required per the appropriate Enclosure 7.1 (P/T Curves).
    - Pressurized Thermal Shock (PTS) is the name given to a sequence of events whereby a reactor vessel experiences a severe thermal cycle (overcooling) followed by a repressurization of the system. If the overcooling is severe enough, the risk of brittle fracture of the reactor vessel exists.
    - PTS is of particular concern to Oconee since it is a safety concern primarily in older plants that have a larger degree of radiation induced embrittlement. Newer plants that have lower copper and phosphorus content in the reactor vessel welds are less susceptible to such embrittlement.

a) It is important to note that the NDT cooldown curve does <u>not</u> have to be violated to run the risk of brittle fracture to the reactor vessel. The NDT cooldown curve is based on a 50°F/halfhour cooldown, which can be exceeded during an excessive heat transfer event. Cooldown rates in excess of 50°F/halfhour can greatly increase the stress felt by the RV inner wall. This is why the TSOR region requirements should be followed (if required) following an overcooling event.

- If at any time during the transient either set of conditions has existed:

  a) When RCPs are OFF, and RCS Temp (Tc) < 500°F, and HPI operation in injection mode (HP-26 or HP-27 used),

  <u>OR</u>

  b) When RCPs are ON, and cooldown rate exceeds 50°F/½ Hr, and 100°F temperature change in Tc occurs,

  The 1 Hour hold in the TSOR is required (RC pressure and temperature stabilized) in order to allow the temperature gradient that exists across the RV wall to equalize and a cooldown within the TSOR is required.

  - The cooldown rate required for operation in the TSOR when all RCPs are off is not as great as when at least one RCP remains in operation due to the lack of mixing of the HPI flow from the cold leg to the downcomer region. This reduced mixing (due to no RC flow) results in colder injection water being exposed to the vessel wall.
  - The "one-hour hold" does <u>not</u> apply to a SBLOCA or a SGTR, (but operation in the TSOR does apply).
  - Following a normal unit trip, typical RCS temperature response shows that Th, Tave, and Tc all approach a temperature of approx. 550°F.

    1) With forced circulation (RCPs operating) and during natural circulation, RCS cold leg temperature (Tc) is used for calculating cooldown rate.

    2) This indication of coolant temperature entering the vessel downcomer is the best approximation of the reactor vessel inner wall temperature.

    3) Thus, the rate of change of Tc is utilized to control reactor vessel cooldown rate.

  - When calculating a cooldown rate during transients following a normal unit trip, a value of 550°F is to be utilized as the starting temperature for Tc. From this point the maximum cooldown rate limit is then 50°F in any 1/2 hour period.

- EOP guidance during SGTR provides for cooldown to 532°F limited only by ability to maintain Pressurizer level.

    4) Once 532°F is reached, a controlled cooldown rate must be established.

    5) The rapid 18°F cooldown must be accounted for in the cooldown rate calculation.

    6) In all cases, control cooldown rate so that the cooldown below 500°F does not happen until 1/2 hr has elapsed from the time cooldown below 550°F was initiated.

- If a transient occurs at a time other than following a unit trip, for example, from hot shutdown, then the starting temperature will be the last stable Tc indication.

- 550°F is the starting temperature for determining operation in the TSOR also. Following a trip, if RCS Tc drops below 450°F and at a rate > 50°F per half-hour, operation in the TSOR is required.

    7) If at any time during the one hour hold all subcooled margins $\geq$ 50°F then one RCP/loop should be started to enhance the equalization of temperature across the vessel wall. This, however, does not eliminate the need for operation in the TSOR for the remainder of the one hour period.

    8) If depressurization is required, then priority for depressurization is:

        (a) Use normal PZR spray (RCPs on)

        (b) Turn PZR heaters off and lower level

        (c) If ΔT < 410°F, Aux PZR Spray

        (d) Open RC-66 (PORV)

        (e) If ΔT > 410°F, Aux PZR spray (Emergency Coordinator authorization to violate 410°F dT).

            (1) CAUTION: Core subcooling margin should be maintained $\geq$ 5°F while depressurizing into TSOR.

    9) If the RCS does heatup after a SG is isolated due to excessive heat transfer, do <u>not</u> re-cooldown to get within the TSOR region of Enclosure 7.1.

- Although no PTS event, to date, has caused pre-existing flaws to propagate through a reactor pressure vessel, transients have occurred that demonstrate the potential for overcooling at pressure. Two such events occurred at B&W plants similar in design to Oconee; another occurred on Oconee Unit 1:

10) At Rancho Seco, the RCS was cooled from 582°F to about 285°F in slightly more than one hour, while RCS pressure was about 2000 psig. This was caused by a loss of most control room indications due to a loss of ICS and NNI power.

11) At Crystal River, again a loss of NNI power initiated a transient that caused the PORV to open, the HPI pumps to start on low RCS pressure and repressurize the RCS after the primary had cooled approximately 90°F in 30 minutes time.

12) At Oconee Unit 1, fire in 6900 VAC switch gear caused RCPs to be stopped, FDW system controls were lost, and RCPs off hampered RCS pressure control. Resulted in minimum Tc of 398°F just prior to RCP restart, and RCS pressure maximum spike of 2395 psig. 50°F/halfhour cooldown rate was exceeded.

35. IF the Main Steam or Feedwater pressure boundaries are ruptured in either SG, **THEN** Go to CP-604, Solid Plant Cooldown and establish or verify the existence of a PZR steam bubble.

36. **IF** MS and FDW boundaries are intact for both SGs,

THEN Further action is at the discretion of station management including when to exit this procedure

END

QVALUE 1.0

QUESTION 34 B15

Unit 1 Initial Plant Condition:

- MS Line B rupture at 100% power

Current Plant Conditions:

- Plant has been stabilized with 1B OTSG isolated.
- RCS "A" Loop Tc is 495°F and steady.
- RCS pressure is 1300 psig and steady.
- All RCPs are secured.
- Reactor Building pressure is 4.1 psig.

Which ONE of the following describes the action that must be taken by the operating crew?

Operation in TSOR is ...

(SEE ATTACHMENT)

A. ...required; cooldown RCS as necessary to establish approximately 5°F SCM.

B. ...required; prevent RCS repressurization and heatup.

C. ...NOT required; cooldown RCS as necessary to establish approximately 5°F SCM.

D. ...NOT required; prevent RCS repressurization and heatup.

B15

ANSWER	B
COGNITIVE	Analysis
REFSPECIFIC	EP Encl. 7.1.B (provide enclosure to applicant)
MODULE	Lesson Plan E23, Sect. 2.20, pg. 16 of 20
OBJECTIVE	ELO-11

ABASIS Incorrect. Operation in TSOR is required and the operator is required to prevent repressurization and heatup of RCS for 1 hour. Cooldown would prevent repressurization and heatup but would not stabilize RCS as required by EOP Enclosure 7.1B. It is necessary to establish SCM = 5°F but the operator must depressurize not cooldown to establish the required SCM. If a cooldown is established this would increase SCM further away from 5°F.

BBASIS Correct. Operation in TSOR is required per Encl. 7.1.B and the TSOR 1 hour hold applies.

CBASIS Incorrect. Operation in TSOR is required per Encl. 7.1.B . See A and B. Encl. 7.1.B requires 5°F SCM during depressurization.

DBASIS Incorrect. Operation in TSOR is required per Encl. 7.1.B. See A and B. Second half of distracter is correct. 495°F is just inside the TSOR.

QUESTION # 90

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	____	**2**
Group #	____	**1**
K/A #	**00004 A2.22**	
Importance Rating	____	**3.1**

Technical Reference(s): **PNS-HPI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **None**

Question Source:
- Bank # ____
- Modified Bank # ____
- New __X__

Question History:
- Previous NRC Exam ____
- Previous Quiz / Test ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 __X__

Comments:

1 POINT

QUESTION # 90

SRO ONLY

Unit 1 conditions:

- LDST pressure = 30 psig
- Letdown flow = 60 gpm
- Makeup to the LDST from the BHUT = 40 gpm
- The BOP is preparing to place the Deborating Demineralizer in service
- RCS DEI = 0.027 uCi/gm

Which ONE of the following is correct?

Prior to placing the IX in service _______ Letdown flow to _______.

ASSUME the ΔP across the Deborating IX = 30 psig

A. decrease / prevent lifting a relief valve.

B. decrease / prevent channeling the Demineralizer resin.

C. increase / reduce RCS DEI to within limits.

D. increase / normal letdown flow (70 gpm)

1 POINT

QUESTION # 90

004 A2.22 SRO ONLY (CFR 41.5 / 43/5 / 45/3 45/5) NEW (RSI) GCW 04/17/00

Question setup:
There is no procedure guidance concerning the effect of changing letdown and LDST makeup flow when placing deborating demineralizers in service. Lesson plan PNS-HPI contains a discussion on the effect of these three variables. There is a 1 psig increase from each gallon of letdown flow. There is a 1 psig increase for each gallon of LDST makeup flow. The backpressure seen by the letdown piping is 30 psig if going to the LDST and 0 if going to the BHUT. Placing the deborating demineralizers in service can add as much as 30 psig. The effect of letdown pressure is important because the reliefs on the letdown line and the demineralizers lift around 145 – 150 psig. In this case: Letdown flow = 60 psig, Makeup flow to the LDST = 40 psig, LDST = 30 psig, deborating demineralizers = 30 psig. Total pressure = 160 psig. This would exceed the relief valve setpoint.

A. Correct, Need to reduce letdown flow to prevent lifting relief valve.

B. Incorrect, Design flow for the deborating demineralizers is 70 gpm, channeling would not occur.

C. Incorrect, Letdown flow is increased to keep DEI < 1.0 uCi/gm per AP/A/1700/021, High Activity in RC System.

D. Incorrect, Increasing Letdown flow would make the pressure even higher.

- Another problem is the lack of understanding about control of letdown pressure. The reliefs on the letdown line and the demineralizers lift around 145-150 psig. Operators must be careful not to lift these reliefs during periods when makeup and letdown are being changed. Experience has shown that a good rule of thumb for control of letdown and makeup is : Assume one psig letdown for each gallon of letdown flow, i.e., 70 gpm equates to about 70 psig letdown pressure. It is actually a little less, but is conservative. Add to this flow the backpressure that is seen by letdown (roughly 30 psig if going to LDST, 0 if going to BHUT). The third component of this formula is makeup flow, where once again a one to one ratio is good, 1 gpm=1 psig. With this thumb rule if we say we have 70 gpm letdown flow and 30 psig on the LDST, then letdown pressure is approximately 100 psig. If we makeup during this situation, at 45 gpm, we could possibly be at 145 psig at the letdown relief. Another variable is the deborating demineralizer. They have shown to have as much as a 30 psig ΔP. So 70+30+30= 130 psig. Using this operating experience thumb rule, the operator should be able to calculate allowable letdown and makeup flows to prevent lifting any letdown or demineralizer relief.
- The computer alarm setpoint for High Pressure in the Letdown line has recently been changed on all 3 units to alarm at 130 psi vs. the previous setpoint of 145 psi. This was done to give the operator time to respond to the alarm and take action to prevent lifting the letdown line relief valves.

c) *During low pressure conditions, such as unit S/U or S/D, it may be necessary to use the manual bypass (HP-42) located in the seal supply filter room, if HP-7 is unable to pass the required flow. Key is required for lock. Do not exceed 120 psig at local gauge.*

d) Normal letdown flow is approximately 70 gpm. If increased significantly above this value, an additional CC pump may be placed in service. Procedures have been changes to run both CC pumps during heat-up. When Unit is at normal operating temperature and pressure the heat removal capabilities are checked and the second CC Pump stopped.

   1) When removing a L/D cooler from service, adjust HP-7 to maintain L/D cooler CC outlet temperature <225°F

   2) events have resulted in flashing of the CC system as a result of inadequate flow balancing of CC, at a letdown flow of 88 gpm. Procedure revisions to enhance the flow balancing instructions should alleviate the problem.

QUESTION # 91

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	______	**2**
Group #	______	**1**
K/A #	**014000 A2.04**	
Importance Rating	______	**3.9**

Technical Reference(s): **IC-CRI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-CRI #10.4**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis ___X___

10 CFR Part 55 Content:
55.41 ______
55.43 __X__

Comments:

1 POINT

QUESTION # 91

SRO ONLY

Unit 1 plant conditions:

- Reactor power is 95%.
- CRD movement test has just been completed.
- Groups 1 through 6 rods have 100% PI panel lamps illuminated.
- Group 7 rods are at 95%.
- The Diamond Control Panel will not go into "AUTO".

Which ONE of the following would prevent the Diamond Control Panel from returning to "AUTO"?

A. Neutron error failed to midscale (0).

B. Group 3 rods are NOT at the "Out Limit".

C. Group 6 rods are NOT at the "Out Limit".

D. Sequence Override selected on Diamond.

1 POINT

QUESTION # 91

014000 A2.04 (3.9/4.1) SRO ONLY T2-G1#87 (PRA/RSI 5-1-00)

A. Incorrect, - Auto inhibit will actuate if there is greater than or equal to 1% neutron error in either direction.

B. Correct, - Auto inhibit will actuate if any Safety rod group is not indicating Group out on the Diamond.

C. Incorrect, - Group 6 rods at their out limit is not one of the inputs to Auto inhibit. Group 6 position must be greater than 95% to satisfy the Continuous Boron Dilute Permissive interlock.

D. Incorrect, - If Sequence Override is selected the Diamond can still be place in Automatic.

8. Identify the conditions that will actuate an asymmetric alarm and/or fault. (R8)

9. Given a set of conditions, determine if an automatic asymmetric runback will occur including how the operator will verify proper runback response. (R9)

10. Concerning the CRD Diamond control panel, identify the conditions that will actuate and the resultant control actions for the following circuits: (R10)

    10.1Trip Confirm

    10.2Out Inhibit

    10.3Sequence Inhibit

    10.4Auto Inhibit

    10.5Out Limit

    10.6In Limit

11. Given a combination of tripped CRD breakers, determine which groups of rods will be de-energized and if a reactor trip will occur. (R11)

12. Explain the operation of the "Safety Rods Out Bypass" relays including how and when these relays are bypassed (Automatically or Manually) (R12)

13. Discuss the purpose and operational characteristics of the following pushbuttons and switches associated with the CRD Diamond Control Panel: (R13)

    13.1In Limit Bypass Pushbutton/Indicator (Latch).

    13.2Transfer Reset Pushbutton/Indicator

    13.3Fault Reset Pushbutton/Indicator

    13.4Trip Reset Pushbutton/Indicator

    13.5Manual Transfer/Sync and Transfer Confirm Pushbutton/Indicator

    13.6Sequence/Sequence Override Pushbutton Indicator

    13.7Group/Auxiliary Pushbutton/Indicator

    13.8Auto/Manual Pushbutton/Indicator.

    13.9Clamp/Clamp-Release Pushbutton/Indicator

    13.10 Speed Selector Switch (Jog/Run)

    13.11 Manual Command Switch (Insert/Withdraw)

    13.12 Single Select Switch (Off, 1-12, A11)

    13.13 Group Select Switch (Off, 1-8)

b) The three turn potentiometer is connected to a stable 5 VDC reference source which allows a 0 to 5 VDC linear signal to be fed from the potentiometer to an operational amplifier.

3. When the rod is at the bottom of the core, the relative position indicator can be reset to zero by a reset pulser located on the Position Indicator Panel (PI) to establish the zero point as a reference.
4. Every two steps of the CRD motor results in one step of the relative position stepping motor and a corresponding change in the potentiometer output.
5. The accuracy is ±2.5 inches and is dependent upon the accuracy of the three turn potentiometer.
6. This RPI 0 to 5 VDC output signal is fed to the same switch as API which will allow the operator to select which indication system will be used as an input to the **Computer** and the **PI panel**.
7. The RPI average signals for Groups 5 thru 7 are used as reference signals to supply indication to the **Sequence Monitor**.
   1) Should the rod groups fail to operate in sequence, resulting in overlap to be <20% or >30% between sequential groups, the sequence monitor will cause:
      - Sequence Fault statalarm (1SA1/E1)
      - Sequence Inhibit lamp on the Diamond panel illuminate
      - Diamond panel will trip to "hand" if in automatic.

D. Asymmetric Alarm and Fault

1. Both alarms and fault circuits are identical except for adjustment of the feedback resistors. A signal will be generated when a rod becomes misaligned with its group average by a preset amount .
   a) The **ALARM**; 1SA2/B10, CRD Position Error, is adjusted for 7 inches deviation from the group average.
   b) The **FAULT** is adjusted for 9 inches deviation from the group.
      1) The Fault logic provides input into the ICS Asymmetric Rod Runback circuitry.
2. The misaligned rod is calculated in the group average as fed by API.

c) The Sequence Monitor provides a control input for this indication.

1) The sequence monitor tracks Groups 5, 6 and 7 RPI Average Signals and provides a fault indication when the overlap between sequential groups is NOT 25% $\pm$ 5%.

(a) Group 5 < 80% and Group 6 > 5% OR

(b) Group 6 < 80% and Group 7 > 5% OR

(c) Group 7 > 5% and Group 5 < 95% OR

(d) Group 6 > 20% and Group 5 < 95% OR

(e) Group 7 > 20% and Group 6 < 95%

2) Actuation of the Sequence Inhibit will:

(a) switch the Diamond to manual

(b) illuminate the Diamond lamp

(c) actuate statalarm 1SA1/E1, CRD Sequence Fault.

5. Auto Inhibit

a) The Diamond cannot be placed in automatic.

b) Will actuate if any of the following are met:

1) ≥ 1% neutron error in either direction.

(a) Neutron error is the error signal generated between the Reactor Demand signal from ICS and the actual NI flux level.

2) All Safety rods <u>NOT</u> withdrawn to the outlimit.

3) Loss of Auto power (KI vital power supply to the ICS.)

- Once the Diamond is in Auto, the conditions 1 &2 above will <u>not</u> cause the Diamond to revert to manual

6. Out Limit (Group)

a) Actuated by the first rod in a group to reach the API out limit switch.

b) Stops out motion of all control rods in that group.

c) Out Limit switch is located between 139.75 and 140.25 inches, which is about 1.5" beyond the API switches that light the 100% rod lamps on the PI panel.

7. Control On

a) Indicates that control logic power to a group has been turned on.

b) This can be done automatically through the sequence logic or manually by the operator from the Diamond panel, if the proper sequence for selecting rod control has been followed.

QUESTION # 92

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	_____	**2**
Group #	_____	**1**
K/A #	**000026 A2.08**	
Importance Rating	_____	**3.7**

Technical Reference(s): **CP-601**
EAP-E31

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E31 OBJ. #3**

Question Source:
Bank # _______
Modified Bank # **NRC-078**
New _______

Question History:
Previous NRC Exam **ONS 1997**
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 _____
55.43 __X__

Comments:

1 POINT

QUESTION # 92

SRO ONLY

Unit 1 plant conditions:

- LOCA in progress
- RCS pressure = 1126 psig
- RB Pressure is 9 psig and slowly approaching 10 psig
- All ES systems have actuated as designed
- CP-602 (SG Cooldown with a Saturated RCS) in progress
  - CP-602 NOTE:

Continued operation of the RB spray will result in additional risk of RB equipment degradation due to the acidic spray.

Which ONE of the following is correct?

The CRSRO should direct the RO to…

A. prevent actuation of RB Spray by placing all RBCU's in high speed.

B. allow RB Spray to actuate and then ~~immediately~~ secure RB Spray to protect equipment . ~~when RB pressure < 10 psig decreasing~~

C. prevent RB Spray from actuating by placing the "A" and "B" RB Spray pumps to manual.

D. allow RB Spray to actuate and secure RB Spray when RB pressure is < 3 psig ~~or as directed by the TSC.~~

1 POINT

QUESTION # 92

000026 A2.08 (10 CFR 41.5 / 43.5 / 45.3 / 45.13) SRO ONLY GCW 04/17/00

A. Incorrect, - Placing all three RBCUs in high speed will allow greater cooling within the RB but running the RBCUs in high speed during a LOCA will damage the RBCUs due to the high density.

B. Incorrect, - Securing RB spray for these conditions is contrary to the purpose of RB spray to mitigate RB building pressure. RB pressure must be < 3 psig to secure RB Spray.

C. Incorrect, - This will not prevent RB Spray actuation because an ES component cannot be placed into manual until after ES has actuated.

D. Correct, - Mitigation of RB pressure with RB spray has greater safety significance than the degradation of RB equipment resulting from RB spray and therefore is allowed to decrease RB pressure before securing RB spray.

NOTE 46:	• Continued operation of the RB Spray system will result in additional risk of RB equipment degradation due to the acidic spray. • The RB Spray system may remain in operation to aid removal of airborne iodine at the direction of Station Management.

_____ 46. **IF** AT ANY TIME RB pressure < 3 psig,

AND elapsed time into the event < 24 hours,

THEN stop both RB Spray Pumps:

_____ 1A RB Spray Pump

_____ 1B RB Spray Pump.

_____ 47. Ensure all RB Aux Fans are operating to promote hydrogen mixing.

OBJECTIVES

TERMINAL OBJECTIVE

1. Describe the use of CP-601 (Cooldown Following Large LOCA) of the Emergency Operating Procedure; in order to perform the required actions of an Operator during a Large LOCA event. (T1)

ENABLING OBJECTIVES

1. Recognize the plant conditions that require entry into CP-601, Cooldown Following Large LOCA. (R1)
2. State the BWST level at which the operator should begin to swap LPI and RBS pump suction to the RB Emergency Sump. (R2)
3. Concerning operation of the Reactor Building Spray system after RB pressure is < 3 psig. (R3)
   - 3.1 Describe the benefits of allowing the RBS system to continue to operate.
   - 3.2 Explain why continued operation will result in additional risk of RB equipment degradation.
4. Briefly explain why the Post LOCA Boron Dilution Valves are opened following a LOCA. (R4)
   - 4.1 Discuss the bases for the Post LOCA Boron Dilution alternate flow path.

2.12 Ensure the following ES operated RB isolation valves are closed: (12.0)

A. One page list of ES operated RB isolation valves, which are verified closed.

1. This page should be taken out of the EOP and given to the Reactor Operator to complete.

2. Rationale: to prevent or attenuate the leakage of radioactive material from the Reactor Building. All Reactor Building penetrations will be verified isolated, including the systems necessary to support RCP operation, namely, CC and LPSW. These fluid systems operate at a higher pressure than containment so the only potential leakage could be into containment. Isolation should be verified since the RCPs are no longer in operation and potential leakage would only serve to add unborated water to the sump.

2.13 Isolation of OTSG's (13.0)

A. If the RCS is being cooled by LPI and RCS pressure < 100 psig, then isolate both OTSG's and stop feedwater systems.

B. If the RCS pressure is < 100 psig, LPI should be providing adequate injection flow for core cooling. The steam generators will no longer be necessary for primary to secondary heat transfer since the Reactor Building has now become the heat sink. All feedwater and main steam piping penetrations should be isolated to prevent potential radiation leakage or emergency sump boron dilution.

2.14 Securing RB Spray System (14.0)

A. Stop RBS when RB pressure < 3 psig and elapsed time into the event < 24 hours.

1. RBS may be run longer to aid in removing airborne iodine at the discretion of station management.

2. RBS operating can cause RB equipment degradation because of the acidic nature of the spray. Electrical equipment is especially susceptible.

B. If the RB pressure decreases below 3 psig within 24 hours of a LOCA, then either the break is to small to challenge any Environmental Qualification (EQ) requirements or enough RB cooling capacity exists to prevent the EQ limits from being challenged. Therefore if RBCU's are operating properly, as verified earlier, they are capable of reducing RB temperature and pressure independently of the RBS system and RBS may be secured.

2.15 Verify RB Auxiliary Fans Operating (15.0)

A. These fans operate continuously during power operation. It is desirable to run these fans after a LOCA to prevent hydrogen pocket formation and to obtain a better analyzer indication of the average hydrogen concentration in containment.

Exam Question Report

27-Jan-99

Question ID:	NRC078	Revision No:	0	Revision Date	10/29/1999
Question Description:	NRC078				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	Reference: 1997 ONS-NRC	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

KA: 026000K1.01

The following Plant conditions exist:

- A LOCA has occurred.
- The Reactor is tripped.
- The SRO has transitioned from Section 501, Loss of Subcooling to CP-601, "Cooldown Following Large LOCA".
- RB Pressure is 9 psig and slowly approaching 10 psig.
- All ESF systems have actuated as expected.

A CAUTION in CP-601 states, "Continued operation of the RB spray will result in additional risk of RB equipment degradation due to the acidic spray".

Which <u>ONE</u> of the following statements describes proper operator action?

A) The CR SRO should direct the RO to prevent actuation of RB Spray since it is <u>not</u> required to mitigate overpressurization of containment.

B) The operator should allow RB Spray to actuate and then immediately secure RB Spray with SRO concurrence to protect equipment.

C) The operator should prevent RB Spray from actuating and actuate it once he/she is satisfied that it is required to perform its safety function and the CR SRO concurs.

D) The CR SRO should allow RB Spray to actuate despite potential equipment degradation and secure RB Spray when RB pressure is < 3 psig or as directed by the TSC.

Answer

D

Lessons

ID	Description

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag
1997 ONS-NRC	Reference created by conversion		

QUESTION # 93

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	**2**
	Group #	_____	**1**
	K/A #	**059000**	**A2.03**
	Importance Rating	_____	**3.1**

Technical Reference(s): **AP/1700/15**
STG-ICS

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **STG-ICS OBJ. #3**

Question Source:
- Bank # _______
- Modified Bank # _______
- New ___X___

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _______
- Comprehension or Analysis ___X___

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 __X__

Comments:

1 POINT

QUESTION # 93

SRO ONLY

Unit 3 plant conditions:

INITIAL CONDITIONS:

- Time = 0500
- Reactor Power = 69%
- ICS in automatic

CURRENT CONDITIONS:

- Time = 0510
- 3B2 RCP secured
- Group 2 Rod 9 dropped in the core

If the dropped rod is <u>not</u> recovered, <u>predict</u> which ONE of the following will be an acceptable FDW condition at <u>**0730**</u>?

ASSUME All automatic actions and all procedural actions have been completed.

"A" Main FDW Flow ____ X 10^6 lbm/hr / "B" Main FDW Flow ____ X 10^6 lbm/hr.

A. 4.0 / 2.0

B. 2.9 / 1.5

C. 2.0 / 4.0

D. 1.5 / 2.9

possible comment on
Req'd knowledge from memory

1 POINT

QUESTION # 93

059000 A2.03 xx/3.1 SRO ONLY 04-05-2000 (GTH/PMS)

Question setup:
The dropped control rod will cause a runback to 55% power. The crew will take the unit to at least 45% power (60% of allowable reactor power (75%) for RCP combination) within 2 hours.

A. Incorrect, FDW flows for 55% power with a "B" RCP off.

B. Correct, 40% power with a "B" RCP off.

C. Incorrect, FDW flows for 55% power with a "A" RCP off.

D. Incorrect, 40% power with a "A" RCP off.

OBJECTIVES

TERMINAL OBJECTIVE

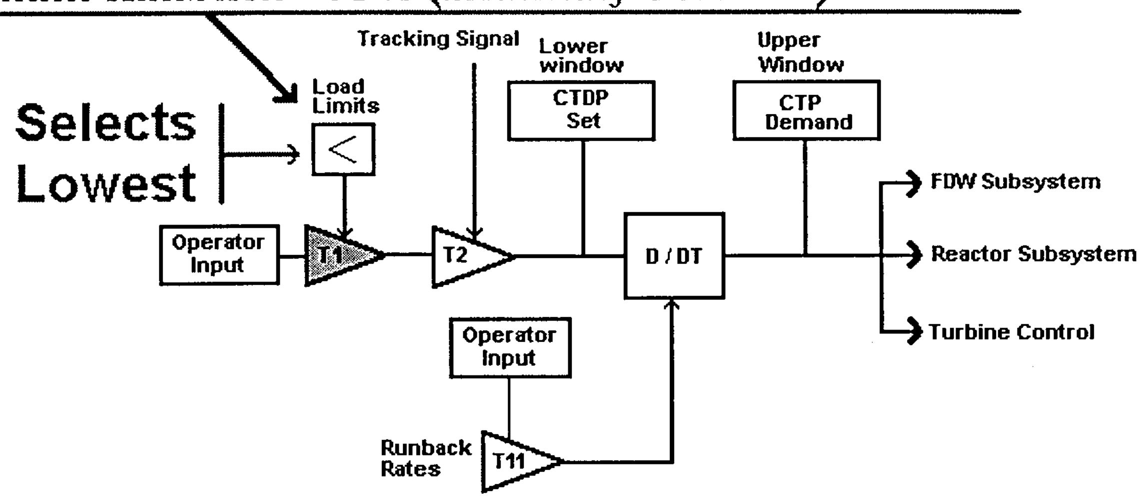
1. Summarize the operational aspects of the Integrated Control System (ICS) with respect to the coordination of plant systems and controls. (T1)
2. Predict automatic actions performed by the ICS and identify corrective actions upon failure of the automatic actions. (T2)
3. Summarize the purpose and operation of the ICS indications and controls available to the operator. (T3)

ENABLING OBJECTIVES

1. Define the functions of the Core Thermal power Demand (CTPD) subsystem. (R1)
2. Given a set of conditions, determine the method to achieve a load change using the Load Control Panel (LCP) (R2)
3. Identify the operations of automatic and manual load limits including: (R3)
   - 3.1 LCP indications
   - 3.2 Load Limit values
   - 3.3 Runback Rates
   - 3.4 Over-riding conditions
4. Given a load limit condition, assess plant runback response and determine the source of any failure. (R4)
5. Define the purpose and operation of the HOLD push-button. (R5)
6. Identify the operation of the TRACKING mode including: (R6)
   - 6.1 Initiating conditions
   - 6.2 Tracking Parameters
   - 6.3 Operator interface
7. Describe the ICS response to a load change in the Integrated mode. (R7)
8. Describe the conditions and responses of the Integrated Master in maintaining turbine header pressure control. (R8)

2. Automatic Load Limiting

Loss of RCP	**74%**
Loss of RC FLow	**Variable**
Asymmetric CR	**55%**
Loss of FWPT	**65%**
Both Gen Bkrs OPEN	**20%**
Reactor Trip	**0%**
Maximum Runback	**15% (Manually Selected)**



a) The CTPD will automatically impose demand limits under conditions that prevent the unit from producing threshold values of power. The ICS will recognize limiting conditions and maintain or reduce unit output to within these pre-designated limits.

b) The value of the load limit will be input as an ICS demand thru transfer function T1 and will block the operator input from the LCP.

c) Automatic Load limiting conditions and limits:

1) Loss of one RCP = 74% power

(a) Sensed by RCP breaker open contacts

(b) RPS will not allow power ops with < 3 RCPs therefore there is no load limit for loss of more than one RCP..

2) **Loss of one FDWPT = 65% power**

   (a) i.e. If only one main feedwater pump were operating, the unit would be limited to producing 65% power......therefore 65% is the load limit.

   (b) Sensed by a low hydraulic oil on the FWPT.

3) **Asymmetric rod = 55% power**

   (a) Sensed by a Diamond logic for asymmetric control rod from the Absolute Position Indication system.

4) **Loss of RC flow = Variable with flow degradation**

   (a) Sensed by the total of the two median selected Loop RC flow signals.

5) **Maximum Runback (when selected) = 15% power**

6) **Both Generator breakers OPEN = 20% power**

   (a) Sensed from breaker auxiliary relays in generator breakers.

7) **Reactor Trip = 0% power**

   (a) Sensed from Trip Confirm on Diamond Panel or DSS.

d) If a load limit is reached, the appropriate light on the LCP panel will be illuminated indicating the source of the limit. This light will remain on until the CTP Demand is at or below the limit value.

   - The "On High" light will also be on as long as the CTP Demand is above the limit value.

e) If more than one load limit exists, the **MOST LIMITING** (lower limit) will be selected by ICS. If that particular limit were satisfied or no longer true, the next most limiting load limit would control.

f) **Load limits can only be applied in the Integrated Mode (automatic) of operation.**

   1) Manual operation will cause Tracking, which inputs a demand signal downstream of the load limit signal input and will therefore block any load limit.

g) When a load limit is imposed to the ICS, the operator cannot adjust the ICS via the LCP.

OCONEE NUCLEAR STATION

Dropped Control Rods

1. Purpose

This procedure provides the actions necessary to maintain the plant in a safe condition following a dropped control rod(s).

2. Symptoms

- "CRD POSITION ERROR" statalarm (1SA-2, B-10)
- "CRD SAFETY RODS NOT AT UPPER LIMIT" statalarm (1SA-2, C-10)
- Control Rod "IN-LIMIT" light on (Green light on Position Indication Panel)
- Control Rod "ASYMM. RODS" light on (Yellow light on Diamond).

3. Automatic Systems Actions

3.1 **IF** ICS is in Auto,

AND an "ASYMM. RODS" (Yellow Light on Diamond) occurs,

THEN an "OUT" inhibit at 60% power is established and the Reactor will runback to 55% power.

3.1.1 **IF** the "ASYMM. RODS" (Yellow Light on Diamond) clears,

THEN runback may stop before reaching 55% power.

4. Immediate Manual Actions

_____ 4.1 **IF** more than one Control Rod has dropped,

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 4.2 **IF** more than one Control Rod is misaligned > 9" (6%),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.3 **IF** due to a malfunction, a Control Rod Group is misaligned > 9" (6%) from the acceptable region of PT/1/A/0600/001, (Periodic Instrument Surveillance),

THEN manually trip the Reactor:

- **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).{1}

_____ 4.4 **IF** a Control Rod has dropped on an approach to criticality,

OR a dropped Control Rod results in a return to subcriticality from a critical condition,

THEN manually insert all Control Rods to Group 1 at 50% WD.

5. Subsequent Actions

_____ 5.1 **IF** the Reactor has tripped,

THEN **GO TO** EP/1/A/1800/01, (Emergency Operating Procedure).

_____ 5.2 Verify Reactor runback <60% Full Power is in progress.:

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

NOTE 5.2.1:	If a control rod drops while the ICS Diamond Panel is in MANUAL, an "Auto Inhibit" (safety rod) or "Sequence Enable" (regulating rod) alarm will prevent placing the ICS in automatic.

_____ 5.2.1 **IF** the Reactor has **NOT** runback,

THEN commence manual Reactor Power reduction to < 60% of the allowable thermal power for the RCP combination.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.3 **IF** operating with only three (3)RCPs,

THEN commence manual Reactor Power reduction to < 45% Full Power.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 5.4 Notify I&E to begin investigation for the cause of the Dropped Control Rod.

Dropped Control Rods

_____ 5.5 Initiate actions to meet the Required Action of ITS 3.1.4 by performing the following:

_____ 5.5.1 Within one hour verify > 1% SDM
with allowance for the inoperable control rod(s):

- Perform PT/1/A/1103/15, (Reactivity Balance Calculation).

_____ 5.5.2 Within two hours reduce Reactor Power < 60%
of the allowable thermal power for the RCP combination.

NOTE 5.5.3:	Notification to I&E should be made as soon as possible due to the complexity of resetting the RPS trip setpoints and the short ITS allowable time limits.

_____ 5.5.3 Notify I&E to reduce RPS Flux/Flow-Imbalance trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.5.4 Notify I&E to reduce RPS High Flux trip setpoints to 65.5%
of thermal power allowable for the RCP combination.

_____ 5.6 **<u>WHEN</u>** Reactor Power is < 60%
of the allowable thermal power for the RCP combination,

<u>THEN</u> notify I&E to begin repair of the Dropped Control Rod.

_____ 5.7 **<u>WHEN</u>** I&E is ready to begin repairs on the Dropped Control Rod,

<u>THEN</u> Place the ICS Diamond control station in MANUAL,

<u>AND</u> permit I&E to repair Dropped Control Rod.

Dropped Control Rods

CAUTION:5.8	The Duty Reactor Engineer must evaluate the effects of local power distribution and the necessity for special maneuvering limits prior to the recovery of a dropped or an asymmetric Control Rod.

_____ 5.8 **WHEN** I&E has repaired the Dropped Control Rod,

THEN recover the Dropped Control Rod per OP/0/A/1105/009, (Control Rod Drive System).

END

QUESTION # 94

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	_____	**2**
Group #	_____	**1**
K/A #	**063000**	**G2.2.24**
Importance Rating	_____	**3.8**

Technical Reference(s): **SLC 16.8.3**
EL-DCD

Proposed references to be provided to applicants during examination: **SLC 16.8.3**

Learning Objective: **EL-DCD OBJ. #7.6**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ______
55.43 __X__

Comments:

1 POINT

QUESTION # 94

SRO ONLY

Plant conditions:

- Unit 1, MODE 1 at 100% power
  - 1PA Battery is planned to be removed from service during the shift
- Unit 2, MODE 1 at 100% power
- Unit 3, MODE 5
  - 3PB Battery is out of service and expected to be returned during the shift

Which ONE of the following describes the minimum conditions that must be met PRIOR to removing the 1PA Battery from service?

SEE ATTACHMENT

All three Unit's Power Batteries must be in service AND...

A. be supplied by their respective chargers, AND all three unit's PA bus tied together.

B. be supplied by their respective chargers, AND all three unit's PA and PB busses separated.

C. the 1PA Charger must be in service AND remain tied to the 1PA bus while 1PA Battery is out of service.

D. be supplied by their respective chargers, AND Unit 1 PA bus tied to Unit 3 PA bus, AND separated from Unit 2 PB bus.

1 POINT

QUESTION # 94

063000 G.2.24 3.8/XX SRO ONLY 4-13-00 (GTH)

A. Correct – 3PB must be returned to service before removing another battery (SLC requires 5 of the 6 power batteries operable at all times).

B. Incorrect – This is a normal system alignment. Busses are not tied together when a power battery is removed from service as when a control battery is removed from service.

C. Incorrect – A charger is never placed on to the buss without the battery also being tied to the bus (dampening effect). (L/P 1107/10)

D. Incorrect – When batteries are taken OOS All three units are cross-tied this ensures proper over current protection and is required for 250 volts to be generated by the system. One unit should not be separated.

16.8 ELECTRIC POWER SYSTEM

16.8.3 Power Battery Parameters

COMMITMENT Power Battery parameters shall be within specified limits.

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Electrolyte level below top of cell plates. OR Battery cell float voltage < 2.06 volts. OR Electrolyte Temperature < 60°F. OR No battery chargers are available to a battery.	A.1 Declare associated battery inoperable.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. A single battery inoperable.	B.1	-----------NOTE----------- Not required when associated buses (PA or PB) are cross-tied for all ONS units. ------------------------------	
		Declare associated distribution center (1DP, 2DP, 3DP) inoperable.	Immediately
	AND		
	B.2	-----------NOTE----------- Not required when associated buses (PA or PB) are cross-tied for all ONS units. ------------------------------	
		Declare Turbine Driven Emergency Feedwater (TDEFW) System and Anticipated Transients Without Scram (ATWS) System inoperable.	Immediately
	AND		
	B.3	Initiate action to cross-tie the associated buses (PA or PB) for all ONS Units.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two or more batteries inoperable.	C.1	Declare associated distribution center (1DP, 2DP, or 3DP) inoperable.	Immediately
	AND		
	C.2	Declare Turbine Driven Emergency Feedwater (TDEFW) System and Anticipated Transients Without Scram (ATWS) System inoperable.	Immediately
D. One battery charger inoperable.	D.1	Initiate action to connect the standby charger to the associated bus.	Immediately
E. Electrolyte level < minimum or > maximum level indication marks.	E.1	Restore electrolyte level to within limits.	90 days
F. Battery cell float voltage < 2.13 Volts and ≥ 2.06 Volts.	F.1	Restore cell float voltage to within limits.	90 days
G. Required Action and associated Completion Time not met.	G.1	Declare associated battery inoperable.	Immediately

3.8 What the "monitor test" push buttons are used for.

4. Explain why ground detection is important to ungrounded DC systems. (R14)
   4.1 Recognize that grounds on vital DC systems can render the entire system inoperable.
   4.2 Recognize that NSD 311, Nuclear Safety-Related DC Systems Ground Response, sets the standard for responses to grounds.
   4.3 Recognize that SLC 16.8.5, 125 Vdc Vital I&C System Ground Locating Policy, sets the standards for ONS response to grounds.

5. Briefly discuss the Vital DC Instrument and Control System operation including: (R4)
   5.1 Purpose of the system.
   5.2 Six typical loads.
   5.3 The way power is normally supplied to the buses.
   5.4 How power is supplied in the event of a charger failure.
   5.5 How power may be supplied from another unit.
   5.6 Reason for tying DCA and DCB buses together before removing a battery from service.
   5.7 The power supplies to Vital I&C Battery Chargers.
   5.8 How to perform a battery ground test.
   5.9 Separating buses between units for ground location.
   5.10 Location of the batteries, battery chargers, distribution centers, DC panelboards and Isolating Diode Assemblies.

6. Briefly describe the Essential DC Power System operation including: (R6)
   6.1 The normal source of power to the system.
   6.2 Two alternate sources of power to each bus.
   6.3 The loads supplied by the system.
   6.4 Location of the Isolating Diode Assemblies.

7. Briefly discuss the Power Battery System Operation including: (R7)
   7.1 Purpose of the system.
   7.2 Battery bank and distribution network.
   7.3 How 250 VDC is achieved on the system.

7.4 Ten loads supplied from the system.

7.5 The location of the battery banks and chargers.

7.6 Taking a power battery bank out of service, and the considerations involved.

8. Briefly describe the 230 KV Switchyard DC Power System, including: (R8)

   8.1 Purpose of the system.

   8.2 Batteries, chargers and distribution network.

   8.3 How redundant power feeds to the common closing coils for the PCBs are provided.

   8.4 Isolating a battery from the bus and the considerations involved.

   8.5 The power supplies to the battery chargers.

9. Briefly describe the 525 KV Switchyard DC Power System, including: (R9)

   9.1 Purpose of the system.

   9.2 Batteries, chargers and distribution network.

   9.3 Isolating a battery from the bus.

   9.4 The power supplies to the battery chargers.

10. Briefly describe the SSF DC Power System, including: (R13)

    10.1 Purpose of the system.

    10.2 Batteries, chargers and distribution network.

    10.3 Isolating a battery from the bus.

    10.4 The power supplies to the battery chargers.

11. Draw a one-line diagram of the DC Power Distribution System including the 'Vital' I&C distribution system and 'Essential' DC Power Systems. (R10)

12. Concerning restoring power to a de-energized power panel, the operator should be aware of: (R11)

    12.1 The procedures available for operating Vital and Essential power panels.

    12.2 That power to vital I&C DC system should be restored only after careful consideration of the loads supplied and consultation with knowledgeable personnel concerning the consequences of re-powering the loads in the present plant conditions.

12.3 That transferring an inverter or its buses to Regulated Power is an acceptable way to re-energize AC power panels quickly.

12.4 That tripped breakers and blown fuses should be thoroughly investigated by appropriate personnel prior to closing in/replacing.

12.5 That restoring power to de-energized panels can re-energize equipment unexpectedly, therefore "expect the unexpected" and react accordingly. Use the STAR method to energize a power panel. (STOP - THINK - ACT - REVIEW)

13. Concerning the loss of 1DIA event that occurred on 8/23/93 briefly discuss: (R12)

13.1 The root cause of 1ADA #2 Isolating Diodes being unable to supply power to 1DIA.

13.2 Why 1A1 and 1B1 RCP breakers did not trip when 1TA de-energized.

13.3 Why 1TA de-energized.

13.4 Why the Main Turbine tripped.

13.5 Why 1A1 and 1B1 RCPs started when 1DIA power was regained and the consequences of 1A1 and 1B1 RCPs starting the way they did.

13.6 Why SG EFDW Auto Level control did not control SG level after SG levels increased above 21 inches and the operator placed 1FDW-315 and 316 back to Auto after taking manual control to control EFDW flow following Auto start on 1A SG "Dry-Out".

14. Given a copy of Tech Specs/SLC and associated bases, analyze a given set of conditions for applicable LCOs. (R15)

15. Apply all Tech Specs/SLC rules to determine applicable Conditions and Required Actions to ensure compliance with Tech Specs/SLC. (R16)

16. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with Tech Specs/SLC. (R17)

E. Locations

1. Power Batteries and Chargers for Unit 1 and 2 are located in the Turbine Building on the third floor near the west wall across from the stairwell to the fifth floor north entrance to Unit 1 & 2 Control Room.
2. Power Batteries for Unit 3 are located on the Turbine Building fifth floor outside the Unit 3 south entrance elevator lobby.
3. Power Battery Chargers for Unit 3 are located on the third floor of the Turbine Building at the elevator lobby entrance.

F. Degraded Operation

1. When taking a Power Battery, either PA or PB, out of service on a unit, care must be taken to ensure that an adequate supply of power is available between all three units to supply all three units' loads, even with a battery out of service.
2. **SLC 16.8.3** allows only one of the six Power Batteries to be removed from service at a time, without having to declare the TDEFWP and ATWS circuits inoperable as long as the associated buses are cross-tied for all units.
   a) This is due to possible excessive fault currents if a PA & PB Battery were both unavailable with both PA & PB Buses cross-connected between all three units.
3. Incident Reports B-1323 and B-1279 both deal with improper isolation and restoration of a Power Battery:
   a) In both cases the bus ties were opened prior to placing the out-of-service battery back on the bus (placing it back in service).
   b) This would not seem to be a problem since there is no specific Tech Spec dealing with Power Batteries, but as the next section on Tech Specs shows, Tech Specs are a concern.

G. Technical Specifications

1. There are no direct Tech Specs dealing with the Power Battery System.
2. Incident Reports B-1323 and 1279 were written because the TDEFWP and CCW-8 were rendered inoperable when no longer supplied with 250 VDC power.
   a) The TDEFWP was rendered inoperable due to not being able to auto start when its Aux. Oil Pump lost its source of power.
   b) The TDEFWP is required by Tech Spec 3.7. The ATWS system is also supported by the power battery system (FDWPT Control circuitry) and is required by SLC 16.7.2.
3. Therefore, the Power Battery System is required to meet the Tech Spec definition of operability for TDEFWP's and the ATWS circuit.

H. Selected Licensee Commitments **(SLC 16.8.3)**

1. Sets the Power Battery requirements for:

   a) Battery operability

      1) If only one station power battery in inoperable the associated distribution centers (1DP,2DP,3DP) are considered operable provided all 3 units power battery distribution centers are cross-connected.

      2) If two (2) or more station power batteries are inoperable then the station power battery distribution centers (1DP,2DP,3DP) are considered inoperable. This renders the TDEFEWP and ATWS (TS 3.4/SLC 16.7.2) inoperable for all three (3) units until the station is returned to ≤ 1power battery inoperable.

   b) Individual Battery Cell Voltage

   c) Electrolyte Level

   d) Electrolyte Temperature.

      1) The minimum electrolyte temperature is assured through:

         (a) I&E readings

         (b) NLO rounds to enclosure that the room temperature is maintained 60°F or greater. This temperature ensures that there is sufficient battery capacity for design basis accidents. Low battery electrolyte temperatures reduce battery capacity so we maintain this temperature per the round sheet 70°F or higher.

I. Ground Detection System **(OC-EL-DCD-9)**

1. Purpose: Allow reading each leg's bus voltage to ground (125 v) and bus to bus (250 v) to be able to detect low voltages indicative of a bus ground and alert operators to a grounded condition.

NOTE: These ground detection instruments are not useable at present due to not having a reference to be compared with.

   a) Controls and voltage indication located on the Switchyard Mimic Bus back panel.

   b) Ground Indication Lights and Power Battery voltmeters located on AB1

      1) Lights go "bright" if system has a ground

      2) 3 lights - one for P leg, one for N leg and one for the PN leg.

2. Presently, there is no procedural guidance to refer to these instruments or to check the operability of the ground detection circuit.

QUESTION # 95

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	______	**2**
	Group #	______	**1**
	K/A #	**000071 A2.04**	
	Importance Rating	______	**2.7**

Technical Reference(s): **OP/1104/018**
SLC 16.11.14

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **NONE FOUND**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 __X__

Comments:

1 POINT

QUESTION # 95

SRO ONLY

Unit 1 Plant Conditions:

INITIAL CONDITIONS:

- Shutdown for refueling is in progress
- RCS degassing is in progress
- "1A" GWD tank is in service

CURRENT CONDITIONS:

- "1A" GWD tank hydrogen concentration = 3.6%.

Which ONE of the following describes the **required** action?

A. Immediately stop RCS degassing.

B. Immediately stop addition of gasses to "1A" GWD tank.

C. Reduce hydrogen concentration to less than limit within 48 hours.

D. Isolate the "1A" GWD tank and sample for hydrogen within 48 hours.

1 POINT

QUESTION # 95

000071 A2.04 (CFR 41.5 / 43.5 / 45.3 / 45.13) SRO ONLY RSI/GCW 04/17/00

A. Incorrect - At 3.6% hydrogen, hydrogen concentration must be reduced to < 3% within 48 hours. Additions to the "A" GWD tank can continue. RCS degassing could continue by placing another GWD tank in service.

B. Incorrect - Immediately stopping addition of gasses to tank would be required if concentration were > than 4%.

C. Correct - At 3.6% hydrogen, hydrogen concentration must be reduced to < 3% within 48 hours.

D. Incorrect - Gas tank is not required to be isolated. However if it were, an isolated GWD tank must be sampled for hydrogen within 24 hours of isolation. 48 hours is the time limit for reduction of hydrogen to less than 3%.

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.14 Explosive Gas Mixture

COMMITMENT The concentration of Hydrogen in the Waste Gas Holdup Tanks shall be ≤ 3% by volume.

APPLICABILITY: At all times.

ACTIONS

-----NOTE-----
Separate Condition Entry is allowed for each tank.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Concentration of Hydrogen in Waste Gas Holdup tank is > 3% and ≤ 4% by volume.	A.1	Reduce Concentration of Hydrogen to within limit.	48 hours
B. Concentration of Hydrogen in Waste Gas Holdup tank is > 4% by volume.	B.1	Suspend addition of waste gases to tank.	Immediately
	AND		
	B.2	Reduce Concentration of Hydrogen to within limit.	24 hours

GWD System

1. Purpose

1.1 Provides procedural guidance for collecting and reducing activity of radioactive gases.

1.2 To provide procedural guidance for release of GWD Tanks.

2. Limits and Precautions

2.1 The quantity of radioactivity contained in each GWD tank shall be limited to ≤ 3.8E5 curies noble gases (considered as Xe-133).

2.2 An isolated GWD tank must be sampled for hydrogen within 24 hours of isolation.

2.3 Maximum hydrogen concentration in GWD system is 3%.

2.4 If hydrogen concentration in GWD tank > 3%, but ≤ 4% volume, concentration must be reduced to ≤ 3% within 48 hours.

2.5 If hydrogen concentration in a GWD tank > 4%, all additions from vent header to GWD tank must be suspended and hydrogen reduced to ≤ 3% within 24 hours.

2.6 GWD Tank pressure shall **NOT** exceed 85 psig.

2.7 The following approval levels are required for releases:

All Releases at Station in Progress (including this one)	Required Level of Approval
1/3 Station Limit - 1 GWR in progress	SRO
1/3 Station Limit - 2 GWRs in progress	OSM
1/3 Station Limit - 3 GWRs in progress	OSM
2/3 Station Limit - 1 GWR in progress	OSM
1/3 Station Limit on 1 GWR and 2/3 Station Limit on Another GWR	OPS Superintendent

C. Alarms

1. 1 & 3SA-9 / A12, GWD Filter Differential Pressure High
    a) 2.5" H_20 pre-, absolute
    b) 0.8" H_2O Charcoal
2. 1 & 3SA-9/ B12, GWD Moisture Separator Level High
3. 1 & 3SA-9/C12, GWD Vent Header Pressure High/Low
    a) High: +2.0" H_2O
    b) Low: -8.0" H_2O
4. 1 & 3SA-9 /D12, GWD Tank A Press High - 70 psig
5. 1 & 3SA-9 /E12, GWD Tank B Press High - 70 psig
6. 1SA-8/E-10, 3SA-8/C-10 GWD Discharge Rad Inhibit
7. 1SA-16/ C-5, GW Interim Waste Gas Decay Tank 1C Pressure High - 70 psig
8. 1SA-16/ C-6, GW Interim Waste Gas Decay Tank 1D Pressure High - 70 psig
9. 3SA-18/ A-4, Interim Waste Gas Decay Tank Press. High - 70 psig

2.4 Limits & Precautions

A. Hydrogen

1. Present due to:
    a) H_2 overpressure on LDST
        1) H_2 enters the vent header by coming out of solution from bleed water or venting of the LDST gas space.
2. Must be maintained at low concentrations due to flammability.
3. On-service tank is sampled 5 times/week and after isolation.
4. If the hydrogen concentration is discovered to be in excess of 3%, the concentration must be reduced as stated in the Limits and Precautions of the GWD procedure listed below:
    a) Maximum hydrogen concentration in GWD System is 3%.
    b) An isolated GWD tank must be sampled for hydrogen within 24 hours of isolation.
    c) If the concentration of hydrogen in a Waste Gas Holdup Tank exceeds 3% by volume, but is ≤ 4% by volume, then the hydrogen concentration must be reduced to ≤ 3% within 48 hours.

d) If the concentration of hydrogen in a Waste Gas Holdup Tank exceeds 4% by volume, then all additions of waste gases to the tank must be suspended immediately and the hydrogen concentration reduced to ≤ 3% within 24 hours.

B. Activity Limit and Precaution

1. The quantity of radioactivity contained in each waste gas holdup tank shall be limited to ≤ 3.8E5 curies noble gases (considered as Xe-133).

a) Local area or process monitors would alarm long before this limit could be reached

b) Tech Spec requires the GWD tank(s) activity be determined daily, when gas is being added to the tank(s), to ensure that the limit of 3.8E5 curies is not exceeded.

1) RP is tasked with this requirement. By monitoring RCS activity (samples and instrument readings) daily, RP is ensured that if the RCS Xe-133 equivalent activity is below 50 uci/ml on all three units, that the GWD tank activity will not reach its limit. When RCS activity exceeds 50 uci/ml on any unit, RP begins to perform daily surveys of the tanks for activity.

2. Limits WB exposure of individual at site boundary to less than .5 Rem in the event of an uncontrolled tank release.

C. Samples for the RB Purge are valid for 24 hours. Samples taken for Gas Tanks Release are valid for 72 Hours.

1. This will allow Operations to HOLD a Gas Tank Release for favorable meteorological conditions to exist without having to resample. Sample results will be conservative as no additional radioactive gas would have been added to the tank, and some of the radioactive material will have decayed.

D. Maximum GWD Tank pressure shall not exceed 85 psig.

2.5 System Operation

Utilize OP/1&2 or 3/A/1104/18 "Gaseous Waste Disposal System" to perform all GWD System operations described in this section, except Depressurization of Reactor Building.

A. Normal Operation (Refer to OC-WE-GWD-2)

1. Vent Header Split (Isolation Valves Closed)

2. For each system:

a) one compressor on

b) one tank in service

QUESTION # 96

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	**2**
	Group #	_____	**3**
	K/A #	**008000 A2.07**	
	Importance Rating	_____	**2.8**

Technical Reference(s): **PNS-CC**
EAP-SSF
AP/1700/020, Loss of CC

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-CC 13, 14, 15, 17, 18**
EAP-SSF 26.1

Question Source:
Bank # _______
Modified Bank # _______
New __X__

Question History:
Previous NRC Exam _______
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 _____
55.43 __X__

Comments:

1 POINT

QUESTION # 96

SRO ONLY

Unit 1 Plant conditions:

- Reactor Power = 50%
- 1CC-8 (CC Return) indicates CLOSED
- Statalarm "CC Return Flow Low" actuated
- No HPI Pumps operating
- IA and AIA pressure = 0 psig

Which ONE of the following describes the highest priority operator immediate action?

A. Manually open 1CC-8 (CC Return).

B. Trip the reactor due to loss of CRD cooling.

C. Shutdown the reactor due to loss of HPI letdown.

D. Dispatch operators to SSF to supply seals with the RCMU Pump.

1 POINT

QUESTION # 96

008000 A2.07 (2.5/2.8) SRO ONLY T2-G3 #85 RSI/PRA 5-1-00

A. Incorrect, - The first two parts of the distracter are correct. The CC pumps do not have to be manually started when re-opening CC-8.

B. Correct, - The operator has only to be "placed" in MANUAL and the CC pumps will AUTOMATICALLY start when CC-8 is reopened.

C. Incorrect, - When CC-8 closes both CC pumps trip. The CCW low flow alarm comes in at 550 gpm. The standby CC pump starts at 575 gpm. The shutting of CC-8 is the only condition that would stop both pumps and result in a low flow alarm.

D. Incorrect, - The operator does not have to be "held" in manual position while opening the valve. Manual operation of motor operated valves require that the de-clutch lever be held in position while operating the valve.

11. Explain the reason for draining the CRD service structure prior to pulling the reactor vessel head prior to refueling. (R11)

12. Describe the two methods of draining the CRD service structure. (R12)

13. Explain how CC-8 failing closed at power affects plant operation. (R13)

14. Describe briefly the steps involved in reopening CC-8 after the valve has failed closed because of a loss of Instrument Air. (R14)

15. Describe the six (6) interlocks and/or automatic actions associated with the CC System. (R15)

16. Explain why the CC System must be in operation: (R16)

    16.1 before letdown is established if RCS temperature is > 120° F

    16.2 if RCS temperature is > 190° F

17. Given a set of plant conditions, diagnose the cause of a CC System problem and/or determine the required corrective action. (R17)

18. Evaluate the overall affect on other plant systems based on the normal and/or abnormal operation of the CC system. (R18, R19)

OBJECTIVES

TERMINAL OBJECTIVE

Upon completion of this lesson, the student will be able to describe the purpose, operation, and response of the Component Cooling System during normal and abnormal plant conditions

ENABLING OBJECTIVES

1. Draw the Component Cooling System, showing the pumps, coolers, and major valves. (R1)
2. List the four (4) major categories of components cooled by the CC System. (R2)
3. State the cooling medium for the CC coolers. (R3)
4. Explain how the CC System acts as a barrier to prevent the release of radioactive liquid to the environment. (R4)
5. List the purpose(s) of the following CC System components: (R5)
   - 5.1 Coolers
   - 5.2 Pumps
   - 5.3 Surge tank
   - 5.4 Control Rod Drive Filters
   - 5.5 Return Penetration Block Valves, CC-7 and CC-8.
   - 5.6 Drain Tank and Pump
   - 5.7 G. RIA-50
6. Describe the corrosion inhibitor used in the CC System, how it protects the system, and its associated hazard to personnel. (R6)
7. Explain how repositioning of system valves can adversely affect CC System performance once the system has been set up for proper operation. (R7)
8. List the CC System controls and indications available to the operator in the control room. (R8)
9. Describe briefly the steps involved in startup of the CC System. (R9)
10. Describe the sequence and precautions necessary while valving in the spare CC cooler. (R10)

26. Explain the basis for the following requirements: (R29)

26.1 The SSF RCMU System must be placed in service on Unit 1 <u>WITHIN</u> ten (10) minutes of a loss of HPI Seal Injection and CC to the RCPs.

26.2 Electrical breaker transfers are required to be completed <u>WITHIN</u> ten (10) minutes of a recognized event (Appendix R fire).

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26.3 The SSF RCMU System must be placed in service on Units 2 and 3 <u>WITHIN</u> twenty (20) minutes of a loss of HPI Seal Injection and CC to the RCPs.

26.4 The SSF ASW System must be placed in service on any unit <u>WITHIN</u> 14 minutes of a loss of all sources of water to the SGs if RCMU flow is required (assumes HPI is lost).

27. Given a set of parameters, determine any required actions for manning / activation of the SSF. (R30)

28. Explain why the RCP's must be secured within three (3) minutes of an event requiring the use of SSF ASW. (R31)

29. Describe the basic steps of powering the SSF from the diesel generator during an emergency. (R32)

30. Given a set of diesel generator parameters, determine the appropriate corrective action for supplying power to the SSF. (R33)

SSF EMERGENCY OPERATING PROCEDURE

AP/0/A/1700/25, Standby Shutdown Facility Emergency Operating Procedure, was developed for and provides guidance to the licensed and non-licensed operators for placing the SSF D/G, RCMU, and ASW systems in service for the worst case design scenario, a station blackout. However, based on the initiating event, not all of the systems may be required to be placed in operation.

Guidance is also provided for maintaining the affected unit(s) in MODE 3 with $T_C \approx 555°F$, as well as for replenishing the inventory in the CCW system and SFP.

AP/0/A/1700/25 requires that the SSF RCMU system be placed in operation within 10 minutes (20 minutes on Units 2 & 3) of the initiating event and that the SSF ASW system be placed in service within 14 minutes of the initiating event. Analysis has shown that if the systems are placed in service within this time frame, insufficient RCS inventory will be lost out the PORV to cause formation of steam voids in the hot leg of a magnitude that would interrupt natural circulation during the subsequent cooldown and RCS pressure will remain low enough for the SSF RCMU System to inject water into the RCP seals to prevent seal degradation.

Immediate Manual Actions:

1. If CC and HPI Seal Injection are lost to the RCP's, establish RCP seal flow with the SSF RCMUP on Unit 1 within 10 minutes and on Units 2 & 3 within 20 minutes.
   - If flow is not established within the required time, the RCP seals may be damaged. The concern for maintaining integrity of the RCP seals (with Unit 1 seals being the most critical) is that in approximately ten minutes, HOT RCS water would be at the #1 seal possibly causing seal damage/degradation (on Units 2 and 3, the time would be closer to 20 minutes before seal damage would occur).

d) After the service structure quits draining, an air hose is attached to the inlet vent valve and the remaining contents are blown to the portable drain tank.

5. The CRD service structure is filled by either:

a) Filling it in conjunction with the rest of the system by repeatedly filling the CC surge tank with demineralized water until the surge tank level stabilizes or by,

b) Filling it from the portable CC drain tank:

1) Hose is attached between the discharge of the portable drain tank pump and the service structure low point drain valve.

2) Service structure low point drain valve and inlet vent valve are opened.

3) Portable drain tank pump discharge valve is throttled open, pump is started, and contents of drain tank are transferred to the service structure.

D. Reopening CC-8 Manually Due to a Loss of Instrument Air to the Valve

1. CC-8 must be re-opened as soon as possible if it shuts during power operation.

a) CRD stator temperature will exceed 180°F within 4 minutes of a loss of CC flow.

b) The reactor must be manually tripped if two or more individual CRD stator temperatures exceed 180°F.

c) Also, a loss of cooling will result in HP-5 automatically shutting at 135°F letdown temperature.

d) CC pumps will trip when CC-8 shuts, and will automatically restart after it has been re-opened.

2. CC-8 is reopened manually after it shuts by:

a) Placing the selector lever in the MANUAL position and then rotating the handwheel in the open direction (counterclockwise). The lever does not have to be held in the manual position while operating the valve.

b) If containment integrity is required, the operator must stay with the valve while it is open in manual, and return the lever to AUTO once the situation has been corrected. This returns the valve to automatic. Otherwise, the valve will be inoperable remotely.

E. High Letdown temperature effects on the RCS

1. As letdown temperature increases, the demineralizer in service will tend to release born into the system. This will add negative reactivity to the core, resulting in an RCS temperature decrease and/or outward rod motion to maintain the same reactor power level.

2. If temperature reaches 135°F, HP-5, Letdown Isolation will automatically close to protect the downstream demineralizers.

   a) When HP-5 closes and letdown isolates, pressurizer level will begin to increase. This causes the RCS volume control valve (HP-120) to close to maintain RCS level.

   b) Although letdown and makeup are now essentially stopped, there will remain a net increase in RCS volume (pzr increases) due to the continued flow of seal injection into the RCS.

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   c) Unit 1 will see an RCS volume increase of ~20 gpm (5 gpm/pump of seal flow enters RCS) while Units 2&3 will be ~~much slower at ~4~~ ~36 gpm (9 gpm/pump enters RCS).

   d) The operator will utilize AP/*/1700/14, Loss of Normal HPI Makeup or Letdown to re-establish proper makeup flow. If this cannot be accomplished, guidance will be given to shutdown.

      1) If a shutdown is required, the rate of shutdown will have to be fast enough to reach 15% and begin cooling the RCS before the pressurizer fills, causing RCS pressure control problems and potentially challenging the PORV.

2.6 Interlocks Associated With the CC System

A. If in AUTO, the standby CC Pump starts at 575 GPM flow.

B. If de-energized, the CRDs cannot be energized if CC flow is less than 138 GPM to the CRDs.

C. A reactor coolant pump cannot be started if CC flow is less than 575 GPM. Low CC flow will not affect a running RCP.

D. Letdown cooler CC inlet valve CC-1 (CC-2) must be open before letdown cooler inlet valve HP-1 (HP-2) will open.

E. CC-7 and 8 close on actuation of ES Channels 5 and 6 (respectively)

F. If CC-7 or CC-8 goes closed, the CC pumps will trip and automatically restart when CC-7 and CC-8 are reopened.

2.7 Procedural Limits and Precautions:

A. Caution must be exercised to prevent repositioning of any component flow control valve from the balanced flow position. Any change in position of any flow control valve(s) will alter the flow balance through all other parallel flow paths.

B. Component cooling system temperatures and affected equipment (RCPs, letdown temperature, quench tank temperature, and CRD temperatures) must be monitored closely when removing or placing component coolers in service. Any change in the system will alter the flow balance and may result in temperature changes. Caution must be exercised to ensure operational limits are not exceeded.

C. The component cooling system must be in operation for any of the following conditions:

1. Control rod drives energized. There is an interlock to prevent the CRDs from being energized without component cooling water, but will not de-energize the drive upon loss of cooling water.
   - *Prevents thermal damage to the CRD stators*

   Instructor note:

   This interlock can be overridden by pressing green "CC Interlock button" located in System Logic Cabinet No. 3 in the cable room to allow I&E testing of CRDs during unit outage.

2. Prior to operating any RC Pump.
   - *RCP starting interlock which ensures that the seal coolers are operational prior to pump start.*

3. Prior to establishing RC letdown flow if RC temperature is above 120° F.
   - *Prevent isolation of letdown on high temperature interlock.*

4. When RC temperature is greater than 190° F.
   - *Prevents thermal damage to the CRD stators*

D. Component cooling contains a corrosion inhibitor. Avoid getting CC water on skin and particularly in the eyes by utilizing proper safety equipment.

E. Any ES valve that has been operated manually must be cycled electrically to assure operability.

F. When aligning the component cooling system for operation, an independent verification of valve (mainline) positions is required by a separate individual.

G. If CC-7 or CC-8 goes closed, then both A and B CC pumps will trip. CC pumps cannot be restarted if either CC-7 or CC-8 is in the full closed position.

OCONEE NUCLEAR STATION

Loss Of Component Cooling

1. Purpose

This procedure provides the actions necessary to maintain the the plant in a safe condition following a loss of component cooling.

2. Symptoms

- "CC CRD RETURN FLOW LOW" statalarm (1SA-9, B-1)
- "CC COMP COOLING RETURN FLOW LOW" statalarm (1SA-9, C-1)
- "CC COMP COOLING SURGE TANK LEVEL HIGH/LOW" statalarm (1SA-9, D-1)
- "HP LETDOWN TEMP HIGH" statalarm (1SA-2, C-1)
- Increasing CRD stator temperatures (Use TURN-ON CODE [CRD)
- Increasing LDST temperature
  O1A1240 HPI LDST TEMP

3. Automatic Systems Actions

- **IF** CC Total Flow decreases to 575 gpm,
  THEN standby CC pump starts.

- **IF** letdown temperature > 135°F,
  THEN 1HP-5 (LETDOWN ISOLATION) closes.

Loss Of Component Cooling

4. Immediate Manual Actions

NOTE 4.1:	If HPI Seal Injection and CC are both lost, RCP seals must be supplied from the SSF RCMU Pump within 10 minutes.

_____ 4.1 Verify automatic actions occur:

- **IF** Automatic Systems Actions have **NOT** occurred,

  THEN perform them manually.

_____ 4.2 Verify open:

- 1CC-7 (CC Return Inside Block)
- 1CC-8 (CC Return Outside Block).

Loss Of Component Cooling

CAUTION 4.3:	1. CRDM stator temperatures will reach 180°F in ≈ 4 minutes if stator cooling is lost. 2. 1CC-8 (CC Return Outside Block) cannot be operated from the Control Room if it is manually opened. 3. The operator must remain with 1CC-8 (CC Return Outside Block) and in constant communication with the Control Room until it is closed or returned to the "AUTO" position.

_____ 4.3 **IF** 1CC-8 (CC Return Outside Block) **CANNOT** be opened electrically,

THEN manually open 1CC-8 (CC Return Outside Block):

_____ 4.3.1 Unlock with key **OR** break shackle on lock associated with 1CC-8 (CC Return Outside Block) chain.

_____ 4.3.2 Position 1CC-8 (CC Return Outside Block) local selector lever to "MANUAL".

_____ 4.3.3 Manually open 1CC-8 (CC Return Outside Block) by rotating attached handwheel counter-clockwise.

_____ 4.3.4 Establish constant communication with the Control Room.

_____ 4.4 Check CC pump operation:

- **IF** CC pumps have tripped,
  THEN verify CC Surge Tank level $\geq$ 12 inches
  AND attempt to restart both pumps.

Loss Of Component Cooling

NOTE 4.5:	A computer alarm will be received each time a CRD stator temperature exceeds 150 °F.

_____ 4.5 Monitor CRD stator temperatures:

4.5.1 **IF** two or more stator temperatures exceed 180°F,

THEN manually trip the Reactor:

- **REFER TO** EP/1/A/1800/001 (Emergency Operating Procedure).

Loss Of Component Cooling

5. Subsequent Actions

_____ 5.1 Monitor RCP operating parameters:

_____ 5.1.1 **IF** there is a Loss of HPI

AND a loss of Component Cooling

THEN perform the following:

_____ manually trip the Reactor

_____ manually trip the RCPs.

- **REFER TO** EP/1/A/1800/001 (Emergency Operating Procedure)
- **REFER TO** OP/1/A/1103/006 (RCP Operation)
- **REFER TO** AP/0/A/1700/025 (Standby Shutdown Facility Emergency Operating Procedure)
- **REFER TO** AP/1/A/1700/016 (Abnormal Reactor Coolant Pump Operation).

_____ 5.1.2 **IF** any limit is exceeded,

THEN manually trip the RCP.

- **REFER TO** OP/1/A/1103/006 (RCP Operation)
- **REFER TO** AP/1/A/1700/016 (Abnormal Reactor Coolant Pump Operation).

_____ 5.2 Check all heat exchangers for temperature increase:

- **REFER TO** Enclosure 6.1 "Heat Exchanger Temperature Limits".

QUESTION # 97

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	______	**3**
Group #	______	**1**
K/A #	**G 2.1.33**	
Importance Rating	______	**4.0**

Technical Reference(s): **ITS 3.6.5**
PNS-RBC

Proposed references to be provided to applicants during examination: **ITS 3.6.5**

Learning Objective: **PNS-RBC #15 & #16**

Question Source:
Bank # ________
Modified Bank # **ADM-002**
New ________

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ______
55.43 __X__

Comments:

1 POINT

QUESTION # 97

SRO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Time = 0800
- RCS temperature = 350°F
- RCS pressure = 890 psig
- 1C RBCU is OOS due to a failed motor bearing

CURRENT CONDITIONS:

- Time = 1000
- 1LPSW-16 (Inlet to 1A RBCU coil) fails closed

Which ONE of the following is correct?

SEE ATTACHMENT

A. Before 0800 tomorrow restore either the 1C RBCU or LPSW-16 to OPERABLE or be in MODE 5 in 36 additional hours.

B. Before 1000 tomorrow restore either the 1C RBCU or LPSW-16 to OPERABLE or be in MODE 5 in 36 additional hours.

C. Immediately enter LCO 3.0.3, within 1 hour initiate actions and be in MODE 4 in 19 hours.

D. Immediately enter LCO 3.0.3, and be in MODE 4 in an additional 18 hours.

3.6 CONTAINMENT SYSTEMS

3.6.5 Reactor Building Spray and Cooling Systems

LCO 3.6.5 Two reactor building spray trains and three reactor building cooling trains shall be OPERABLE.

---NOTE---
Only one train of reactor building spray and two trains of reactor building cooling are required to be OPERABLE during MODES 3 and 4.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One reactor building spray train inoperable in MODE 1 or 2.	A.1 Restore reactor building spray train to OPERABLE status.	7 days <u>AND</u> 14 days from discovery of failure to meet the LCO
B. One reactor building cooling train inoperable in MODE 1 or 2.	B.1 Restore reactor building cooling train to OPERABLE status.	7 days <u>AND</u> 14 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One reactor building spray train and one reactor building cooling train inoperable in MODE 1 or 2.	C.1	Restore one train to OPERABLE status.	24 hours
D. Required Action and associated Completion Time of Condition A, B, or C are not met.	D.1	Be in MODE 3.	12 hours
E. One required reactor building cooling train inoperable in MODE 3 or 4.	E.1	Restore required reactor building cooling train to OPERABLE status.	24 hours
F. One required reactor building spray train inoperable in MODE 3 or 4.	F.1	Restore required reactor building spray train to OPERABLE status.	24 hours
G. Required Action and associated Completion Time of Condition E or F not met.	G.1	Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Two reactor building spray trains inoperable in MODE 1 or 2. <u>OR</u> Two reactor building cooling trains inoperable in MODE 1 or 2. <u>OR</u> Any combination of three or more trains inoperable in MODE 1 or 2. <u>OR</u> Any combination of two or more required trains inoperable in MODE 3 or 4.	H.1 Enter LCO 3.0.3.	Immediately

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 — LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 — Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 — When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 12 hours;

b. MODE 4 within 18 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 — When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)	Exceptions to this Specification are stated in the individual Specifications. LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.
LCO 3.0.5	Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.
LCO 3.0.6	When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.16, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

1 POINT

QUESTION # 97

G2.1.33 SRO ONLY (CFR 43.2 / 43.3 / 45.) RSI/GCW 04/18/00

Question setup:
The unit is operating in MODE 3. ITS 3.6.5 requires that 2 RBCU trains be OPERABLE for MODE 3 operating conditions. At 0800 one of the 3 RBCUs is reported INOPERABLE. This requires no action because both required units are operable. At 1000 the second of the 3 RBCUs becomes INOPERABLE. This places the unit in Condition E of TS 3.6.5 therefore requiring one of the RBCU to be restored within 24 hours.

A. Incorrect, 24 hour time limit starts at 1000 because until then the LCO is met.

B. Correct, Restore one or be in mode 5 in 36 hours per Condition G.

C. Incorrect, If in MODE 1 or 2 would enter 3.0.3. and would be required to be in MODE 4 in 18 hours. The 1 hour to initiate action is inclusive of the 18 hours.

D. Incorrect, If in MODE 1 or 2 would enter 3.0.3. 18 hour time would be correct if 3.0.3 applied.

12. Describe two conditions that will activate a RBCU "Cooler Rupture" alarm. (R13)

13. Given a set of conditions, determine the proper operation / alignment of the RBC System and the basis for that specific operation / alignment. (R5, R7, R9)

14. Given a set of plant conditions, analyze RBC System operation and determine system status and any required actions / corrective actions. (R14, R15)

15. Given a copy of ITS / SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS / SLC LCO's. (R11)

16. Apply all ITS / SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R18)

17. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITS / SLC's. (R19)

C. Reactor Building Temperature

Refer To OP-PNS-RBC-11

1. A multipoint recorder located on VB-1 trends reactor building temperatures
2. Points monitored include:
    a) Aux Fan cooling coil inlet and outlet.
    b) RB Cavity
    c) Steam Generator Compartments
    d) Reactor Vessel Cavity
    e) RB Dome

2.7 Improved Tech Specs (ITS) Requirements – Refer to ITS.

A. ITS 3.6.5 (RB Spray and Cooling Systems):

1. Requires that one train of RBS and two trains of RBC be operable during MODES 3 and 4.
2. Requires that two trains of RBS and three trains of RBC be operable during MODES 1 and 2.
3. Lists the surveillance requirements associated with the RBS and RBC systems.

B. SLC 16.6.3 (Containment Heat Removal Verification Frequency) verifies containment heat removal capability is sufficient. Frequency is determined by the RBCU (and LPI) fouling rate.

3. SUMMARY

3.1 This lesson has discussed the design and functions of the Reactor Building Cooling System. By applying this knowledge in conjunction with the proper operating procedures, the operator should be able to recognize proper Reactor Building Cooling System operation during normal and accident situations, and be able to diagnose and correct identified system problems.

Exam Question Report

27-Jan-99

Question ID:	LP-ADM002	Revision No:	0	Revision Date	10/29/1999
Question Description:	LP-ADM002				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	TS 3.3
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:		Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

520 ONLY

Question

KA: 2.1.33

Unit 3 is operating at 100% power with no major equipment out of service and no LCO's in effect. The Operations Test Group informs us at 0400 hours that 3A and 3C RBCU's were discovered to be inoperable at 0000 hours today. How does this affect the units operation as defined by T.S.3.3 (Emergency Core Cooling, RB Cooling, RB Spray, and LPSW)? (.25)

A) operation may continue as long as the Rx Building Spray System is operable.

B) must be in hot shutdown by 1600 hours today.

C) must be in hot shutdown by 1200 hours today.

D) operation may continue at rated power, only one RBCU is required to be operable.

Answer

C

Lessons

ID	Description

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

QUESTION # 98

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	_____	**3**
Group #	_____	**3**
K/A #	**G 2.3.4**	
Importance Rating	_____	**3.1**

Technical Reference(s): **RAD-RPP**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **RAD-RPP # 8**

Question Source:
- Bank # **RAD-51**
- Modified Bank # _______
- New _______

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 _____

Comments:

1 POINT

QUESTION # 98

SRO ONLY

An NLO on your shift is listed on the REC (Radiation Exposure Control) printout as an exposure CLASS 3.

- The NLO's current radiation exposure status is as follows:
  - ➢ TEDE = 3.0 Rem
  - ➢ SDE = 49.5 Rem
  - ➢ CEDE = 1.0 Rem
  - ➢ DDE = 2.0 Rem
  - ➢ Dose extension has been approved AS REQUIRED

Which ONE of the following exposure combinations would be allowable under the individual's CURRENT extension, AND WHO authorized this CURRENT EXTENSION?

A. 0.5 Rem DDE, 1.0 Rem CEDE; RP Manager, Station Manager and Site Vice President.

B. 0.5 Rem DDE, 1.0 Rem CEDE; ~~Operations Manager~~ sect (Superintendent of Ops) and RP Manager.

C. 1.0 Rem DDE, 1.5 Rem CEDE; RP Manager, Station Manager and Site Vice President.

D. 1.0 Rem DDE, 1.5 Rem CEDE; ~~Operations~~ sect Manager (Superintendent of Ops) and RP Manager.

can station manager act for?

section manager
vice ops manager

what is the minimum level of signature authority that is required

1 POINT

QUESTION # 98

G 2.3.4 (2.5/3.1) SRO ONLY T3-C3 #120 RSI/PRA 5-1-00)

Answer explanation: CLASS 3 is a normal radiation worker with no other limits. Only if the exposure class were a 2 (declared pregnant female) would the classification become a deciding factor. Otherwise, the whole body dose (DDE) also counts as a Shallow dose (SDE). An additional DDE of 1.0 Rem would cause the workers SDE limit of 50 rem to be exceeded.

DPC Approval for extensions:
The Section Manager (Superintendent of Operations) and the RP Manager will approve the extension up to TEDE of 4.5 rem/yr. To exceed the 4.5 rem/yr TEDE the RP Manager, Station Manager, and Site VP must approve the extension.

A. Incorrect, - See answer explanation above. The additional dose would place the NLO at 4.5 rem TEDE. This would not require and additional extension. The current extension approval is incorrect due to the Operations Manager not being involved with the extension approval.

B. Correct – TEDE = 4.5. The Section Manager (Ops Supt.) and the RP Manager will approve the extension up to TEDE of 4.5 rem/yr.

C. Incorrect – CEDE + DDE = TEDE = 5.5. 5.5 rem/yr exceeds the NRC allowable dose. Approval would be correct if < 5.0 rem/yr

D. Incorrect - CEDE + DDE = TEDE = 5.5. 5.5 rem/yr exceeds the NRC allowable dose. Approval would be correct is < 4.5 rem/yr.

TRAINING OBJECTIVES

TERMINAL OBJECTIVES

1. Review, discuss, and maintain familiarity with those radiation and radiation protection terms needed to comply with the proper radiation protection practices for personnel dose and contamination control at Duke Power Company Nuclear Stations, and particularly at Oconee Nuclear Station.

2. Understand and explain the personal dose limits for ionizing radiation set forth by federal regulations, as well as those established by Duke Power Company; and explain the required personnel monitoring equipment for individuals at Oconee Nuclear Station, and the individual's responsibilities for proper personnel monitoring and dose control.

3. Understand and explain the requirements set forth for controlling contamination limits at the site, including individual responsibilities for complying with contamination controls and in implementing the contamination control program at Oconee.

4. Explain the ALARA concept for maintaining radiation exposures low, and discuss the individual's responsibilities in meeting the ALARA principles at the station.

ENABLING OBJECTIVES

1. Define the following nuclear radiation terms, and discuss where and under what operating conditions at the station the specific type of ionizing radiation is most likely to be encountered: (R1)

   1.1 alpha (α)

   1.2 beta (β)

   1.3 gamma (γ)

   1.4 neutron (η)

2. Discuss the specific personnel hazard associated with each of the four types of ionizing radiation that can be encountered at Oconee, and describe the most effective means of shielding individuals from each type of radiation. (R2)

3. State the Annual Limit on exposure for Total Effective Dose, Eye Dose Equivalent, Shallow Dose Equivalent, minors, and declared pregnant women, as established by the NRC; state both the basic and maximum administrative limits for each, as established by Duke Power Company. (R3)

ENABLING OBJECTIVES (continued)

4. State the approval requirements for an individual at Duke Power Company to exceed the **basic** permissible exposure limit of 2.0 rem. (R4)

5. State the special dose limits established for the general public. (R5)

6. Describe the special dose control measures used to protect the fetus of a "declared" pregnant radiation worker. (R6)

7. Recognize that in "exceptional situations", it is possible to allow an adult radiation worker to receive additional exposure, apart from normal occupational exposure. (R7)

8. Define and describe the specific site area for each of the following terms relating to the control of station areas: (R8)

    8.1 Unrestricted Area

    8.2 Restricted Area

    8.3 Controlled Area

    8.4 Radiation Control Area (RCA)

    8.5 Radiation Control Zone (RCZ)

    8.6 Radiation Area (RA)

    8.7 High Radiation Area (HRA)

    8.8 Extra High Radiation Area (EHRA)

    8.9 Very High Radiation Area (VHRA)

    8.10 Airborne Radioactivity Area

    8.11 Hot Spot

    8.12 Significant Dose Contributor

    8.13 Low Exposure Waiting Area

    8.14 Contaminated Area

ENABLING OBJECTIVES (continued)

10. Explain the purpose of the Daily Exposure Time Record Card (DETRC or dosecard) and explain the proper use of it, including: (R10)

    10.1 when the DETRC would be used in lieu of the EDC System

    10.2 how to properly enter name on the card.

    10.3 when entries must be made on the card.

    10.4 what legal record the dosecard represents.

    10.5 in what increments exposure is to be recorded on the dosecard.

11. Recognize that even if a zero dose is accumulated during an entry, a dosecard entry must still be made, and the dosecard turned in (whenever the EDC is out of service). (R11)

12. Explain the purpose of the Radiation Exposure Control Daily Report (REC) and demonstrate the ability to use the REC by: (R12)

    12.1 calculating allowable remaining yearly dose from a REC printout.

    12.2 explaining what is meant by the exposure status codes A (alert) and E (excluded).

13. Define the following terms: (R13)

    13.1 contamination.

    13.2 contaminated area.

14. List the five types of personnel contamination monitoring devices used at Oconee. (R14)

15. State the minimum frisking requirements for the following situations: (R15)

    15.1 exit from an RCZ or the RCA.

    15.2 exit from a contaminated area.

    15.3 changing elevations in the RCA

    15.4 removal of hand-held items from a contaminated area.

16. Explain the proper procedure for exit from the RCA in an emergency, when exit must be made through a non-designated entrance/exit point. (R16)

17. Describe the correct procedure to use if a personnel contamination monitoring device indicates contamination on an individual or piece of equipment. (R17)

a) **Dose Equivalent** is the absorbed dose modified by the QF and any other necessary modifying factors.

b) **Deep-dose Equivalent (DDE)** applies to external, whole-body exposures. (Measured at a depth of 1 cm).

c) **Shallow-dose Equivalent (SDE)** applies to exposures to the skin or the extremities. (Measured at a depth of .007 cm).

d) **Eye-dose Equivalent** applies to external exposures to the lens of the eyes. Also referred to as Lens Dose Equivalent (LDE).

e) **Committed Dose Equivalent** relates to the internal uptake of isotopes into the body and the effects on internal organs or tissues during the 50 years following the uptake. Another modifying factor is the **Effective Dose Equivalent** which takes into account the relative weights of the specific isotopes taken in. By summing the products of the weighting factors associated with each irradiated organ (or tissue) and the Committed Dose Equivalent to these organs (or tissues) we come up with the **Committed Effective Dose Equivalent (CEDE).**

f) **Total Effective Dose Equivalent (TEDE)** is the sum of the deep-dose equivalent (external exposures) and the committed effective dose equivalent (internal exposures). [TEDE = DDE + CEDE]

4. Rem (r)

a) The unit of dose measurement used almost exclusively in radiation work is now the rem, which is the acronym for "Roentgen Equivalent Man."

b) While QF defines the different ionizing potential for a uniformly distributed dose of ionizing radiation, doses received by the human body are not uniformly distributed.

c) The rem is the special unit of any of the quantities expressed as dose equivalent. The dose equivalent in rems is equal to the absorbed dose in rads multiplied by the quality factor.

5. While 10CFR20 dose limits deal with doses from "external" sources, and 40CFR190 deals with radiation doses to the public resulting from operation of the plant, period, 10CFR50 deals with radiation doses from liquid and gaseous effluents from the station. 10CFR50 limits exposures to unrestricted areas, brought about from activities in liquid effluents released from the station (ONS), to 9 mrem/year whole body, and 30 mrem/year to any organ. Dose limits resulting from gaseous activity released from the station (ONS) are limited to dose equivalents of 30 mrads for gamma, and 60 mrads for beta; and the dose to any organ from iodine activity is limited to 45 mrem/year. (These limits are listed in SLC 16.11-1 and 16.11-2)

D. Control of External Exposure

1. All dose up to 2.0 rem/yr will be controlled by the Section Manager of the group; that is, Section Managers are able to set dose targets for the individuals in their areas, up to the 2.0 rem/yr plateau, by making entries directly into the radiation monitoring and control (RM & C) computer program.

   Authorization for a worker to exceed the basic permissible dose limit of 2.0 rem/yr, up to 4.5 Rem/yr, must be supplied by the Radiation Protection Manager and the Section Manager.

   Dose extensions in excess of 4.5 Rem/yr must be approved by the RP Manager, Station Manager, and Site VP.

2. Dose targets set by each section will be treated as dose limits, so that if a dose target for an individual is exceeded, it will be investigated by radiation protection.

3. Workers will normally be excluded from the RCA if they reach **4.5 rem for the year**. To exceed 4.5 rem in a year (up to the federal limit of 5.0 rem) requires the approval of the Radiation Protection Manager, the Station Manager, and the Site Vice President. No worker will be allowed to exceed the NRC normal occupational exposure limit of 5.0 rem/year.

4. In addition to the 10% of normal limits for persons less than 18 years old (500 mrem/yr), minors should not be allowed into any high radiation areas.

5. Once a female worker declares her pregnancy (i.e., informs management) her dose should be limited to, typically, no more than 50 mrem per month, not to exceed a total of 500 mrem for the entire pregnancy. Entries into airborne radiation areas, or into high radiation areas by declared pregnant workers should be limited.

E. Planned Special Exposure

1. While the "basic administrative" annual limit for Duke Power Company workers is 2.0 rem, extensions up to the maximum administrative limit of 4.5 rem can be allowed, with properly approved dose extensions; for "normal" situations, the 5.0 rem 10CFR20 annual limit cannot be exceeded.
2. In "exceptional" situations, it is possible to allow an adult radiation worker to receive additional radiation exposure, apart from and accounted for totally separate from normal occupational doses.
3. At present, there is no clear definition of what constitutes an "exceptional" situation; what is clear, though, is that planned special exposures cannot be used as a means to routinely extend the normal occupational dose limits for workers.
4. For planned special exposures, workers may be allowed to receive up to an additional 5 rem TEDE, 15 rem LDE, and 50 rem SDE within a given annual accounting period; however, the additional dose received for a planned special exposure cannot exceed a total lifetime dose for planned special exposures of 5 x the annual limits.

   For example:

   a) a worker has received a normal occupational dose of 1.5 rem thus far in the year; the worker may be allowed to receive up to an additional 0.5 rem of "normal occupational exposure" during the remainder of the year.
   b) a "special and exceptional" situation has arisen where it is necessary for this particular worker (because of his skill, knowledge, willingness, and unavailability of anyone else) to perform a job that will involve receipt of approximately 4.0 rem.
   c) If the worker agrees to perform the job and receive this additional exposure, the exposure received will be recorded and reported separate from the normal occupational exposure of 1.5 rem already received for the year.
   d) The actual amount of dose received will be recorded as a "planned special exposure" dose and go into a "bank" in which the worker starts out with 25.0 rem. If the worker receives 4.0 rem during this planned special exposure (assuming he has not performed any planned special job before, i.e. has 25 rem in his "bank") he will have left, for his entire working life, 21 rem of permissible dose for planned special exposure.

I. REC Printout

1. The **REC (Radiation Exposure Control)** printout is a hard-copy report of the current exposure for each radiation worker at the station. The REC data base contains the current External Radiation Exposure History NRC Form 5 information; it also contains internal exposure data in the form of DAC-hours. Copies of this printout are maintained at various locations throughout the plant.

2. Information appearing on the REC printout may be gathered from several sources, including TLD and MG readings, or by RP estimates. Internal exposure data (DAC-hours) is recorded from DAC-hour logs maintained for the various jobs involving internal exposure.

3. Routine updating of the REC is accomplished via the EDC System or by inputting data from the DETRC. The daily cumulative external doses based on MG readings are updated when the quarterly TLD readings are received.

4. The REC printout is usually updated daily when used with Daily Dose Cards. This would happen when the EDC System is inoperable.

5. Each section of the REC is arranged by work group, such as Ops, Maintenance, etc.; after turning to the section of the printout for their applicable work group, each individual can scan the printout for their RP Badge number and name to obtain the latest information on their exposure status.
6. Next on the printout is the exposure classification of the individual; the three most probable classifications are:
   a) Class 1-incomplete CURRENT YEAR records; MAE = 2000
   b) Class 2-declared pregnant female
   c) Class 3-active worker; complete NRC-4 on file; MAE up to 4500 (requires extension > 2000); PSE = yes
7. The next line on the REC printout lists the "home plant" for Duke Power employees:
   a) CA is Catawba Nuclear Station.
   b) GO is the General Office, in Charlotte.
   c) MG is McGuire Nuclear Station.
   d) OC is Oconee Nuclear Station.
8. The **Maximum Allowable Exposure (MAE) Limit** for the year for each individual is listed, followed by the current year-to-date whole body exposure that the individual has received. These are the numbers used for calculating the REMAINING DOSE ALLOWED when making DETRC entries.
9. An EXPOSURE STATUS column is used to call attention to individuals who are approaching their yearly (Y) limits for exposure.
   a) An "A" in the column means ALERT; the person is at 80% of his MAE.
   b) An "E", for EXCLUDE, means that the individual has reached 90% of his MAE and should not receive further exposure without an extension of the MAE.

J. Body Burden Analysis

1. The total amount of a given radionuclide which is contained in the body at any given time is called the body burden.
2. The most effective way of maintaining control over internal uptakes is through good radiological practices. Once the activity has entered the body, analysis becomes much more difficult. Several methods used to monitor uptake are:
   a) Monitoring of body wastes (bioassay) to determine how quickly the radionuclide is being eliminated, and from this, inferring how much remains in the body.

b) Using whole-body scintillation detectors in a low-background chamber (body burden analysis). Generally, whole body counters are set up to measure the gamma energy emitted by Co-58, Co-60, Cs-134, Cs-137, and I-131, since these are representative of the normal radionuclides resulting from reactor operations.

3. A baseline body burden analysis must be obtained for each person prior to his initial work assignment in potentially contaminated atmospheres. Subsequent body burden analyses are then used to detect and evaluate internal exposures.

4. Generally, all Duke employees who are issued permanent dosimetry receive, at a minimum, an annual body burden analysis update.

5. When an individual receives more than four (4) DAC hours of exposure in a consecutive seven-day period, a BBA will be conducted.

6. 10CFR20 states that each licensee must monitor the occupational intake of radioactive material to people, and assess the **committed effective dose equivalent** to:

   a) adults likely to receive in 1 year, an intake in excess of 10% of the applicable ALI(s) in Table 1, Columns 1 and 2, Appendix B; and

   b) Minors and declared pregnant women likely to receive in 1 year, a committed effective dose equivalent in excess of 50 mrem.

2.6 Respiratory Protection Program

A. Objective

1. The primary objective of the respiratory protection program is to limit the significant inhalation of airborne radioactive materials by personnel.

2. RP will evaluate and analyze any job suspected of having a radiological respiratory hazard. As already discussed, any area where airborne concentration levels exceed or are expected to exceed 25% of the Derived Air Concentration (DAC) listed in Appendix B of 10CFR20 is classified as an Airborne Radiation Area.

   Note: The manager of Safety and Health Services will evaluate the need for respiratory protective equipment for jobs involving respiratory hazards other than radiological, such as work involving asbestos or lead.

3. The preferred method for ensuring that these limits are met is to reduce the amount of airborne contaminants present, through engineering controls, such as ventilation systems or exhaust equipment. When engineering controls are not possible, or are inadequate, respiratory protective equipment will be used.

B. When Respiratory Protective Equipment Should Be Worn

44. EDC (electronic dose capture) System:

    computer-based system for recording and keeping track of each individual's exposure.

45. DETRC (daily exposure time record card):

    manual recording system for keeping track of an individual's exposure when then the EDC System is inoperative.

46. REC (radiation exposure control) Printout computer-based daily exposure printout for all individuals at the plant who are issued permanent dosimetry.

47. Exposure Class 1 - incomplete CURRENT YEAR records; MAE = 2000

48. Exposure Class 2 - declared pregnant female

49. Exposure Class 3 - active worker; complete NRC-4 on file; MAE up to 4500(requires extension > 2000); PSE = yes

50. Exposure Class 5 - Pulled dosimetry

51. Exposure Class 7 - MAE Restricted

    - Place minors and workers whose training has been waived in Exposure Class 7.

52. Exposure Class 8 - Duke worker temporarily off-system (DESI, INPO, etc.)

53. Exposure Class 9 - Complete CURRENT YEAR; incomplete previous years; MAE up to 4500; no PSE

54. Level I Radiation Worker - employee at the plant who is not trained extensively in radiation work because he is not typically expected to be exposed; administrative office workers are usually classified as Level I workers.

55. Level II Radiation Worker - all other employees at the plant who are not classified as Level I workers; Level II workers receive extensive training in protection practices.

56. IRW (independent radiation worker) - individual who receives additional training in radiation protection practices so as to be able to assist the health physics technicians at the plant in performing some of the more routine radiation protection work.

57. Maximum Allowable Exposure (MAE) Limit - the normal administrative whole body, yearly exposure limit established for radiation workers at the plant.

58. Exposure Status "A" - "Alert" for persons who have reached 80% of the MAE.

B. Limits

1. NRC (10CFR20) Limits

a)	Total Effective Dose Equivalent (Whole Body)*	**5.0 rem/yr**
b)	Eye Dose Equivalent (Lens of the Eye)	**15 rem/yr**
c)	Shallow Dose Equivalent	
	·Skin	**50 rem/yr**
	·Extremities**	**50 rem/yr**
d)	Minors (persons under 18 years of age)	**10% of Above Limits**
e)	Declared Pregnant Woman	**0.5 rem for Entire Pregnancy**

Note that there is no longer any distinction in 10CFR20 (NRC Limits) between "Basic" and "Maximum" external exposure limits; they are now one and the same, that being the "Annual Limit" on exposure.

*Although listed under "external exposure" the Total Effective Dose Equivalent limit listed is a combination of whole body external and whole body internal exposure; i.e., both whole body external and whole body internal exposure must be summed to arrive at the Total Effective Dose Equivalent that a worker has received.

**Extremities, per 10CFR20, are now defined as the hand, elbow, and arm below the elbow, and the foot, knee, and leg below the knee.

2. Duke Power Administrative Controls

   Same as NRC limits except an administrative limit for Total Effective Dose Equivalent of 2.0 rem/yr exists.

3. Dose Limit for Individual Members of the Public

   100 mrem/yr

4. Maximum Allowable Dose Rate for Any Unrestricted Area

   2 mrem/hr

5. Planned Special Exposure Limit

   Additional 5 rem TEDE, 15 rem LDE, and 50 rem SDE within a given annual accounting period not to exceed a lifetime Planned Special Exposure limit of 5 x annual limit.

6. Limit before SRD must be re-zeroed - 60% of available scale.

Exam Question Report

27-Jan-99

Question ID:	RAD051	Revision No:	0	Revision Date	10/29/1999
Question Description:	RAD051				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	RAD-RPP - Radiation Protection Practices
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 3/4; LRO = 3/4; SRO = 3/4	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Given the following concerning a 25 year old maintenance technician's current radiation exposure status:

- TEDE = 3.0 Rem
- SDE = 49.5 Rem
- CEDE = 1.0 Rem
- DDE = 2.0 Rem
- Extension(s) have been approved AS REQUIRED

Which ONE of the following exposure combinations would be allowable under his CURRENT extension, AND WHO authorized his CURRENT EXTENSION? (.25)

A) .5 Rem DDE, 1.0 Rem CEDE; Section Manager and RP Manager.

B) 1.0 Rem DDE, .5 Rem CEDE; Section Manager and RP Manager.

C) .5 Rem DDE, 1.0 Rem CEDE; RP Manager, Station Manager and Site Vice President.

D) 1.0 Rem DDE, .5 Rem CEDE; RP Manager, Station Manager and Site Vice President.

Answer

A

Whole body dose (DDE) also counts as a Shallow dose (SDE). An additional DDE of 1.0 Rem would cause the workers SDE limit of 50 rem to be exceeded.

Lessons

ID	Description
RAD-RPP	Radiation Protection Practices (RAD-RPP)

QUESTION # 99

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	_____	3
Group #	_____	C4
K/A #	G 2.4.30	
Importance Rating	_____	3.6

Technical Reference(s): **ADM-OMP 1-14**
NSD-202

Proposed references to be provided to applicants during examination: **NSD-202/OMP 1-14 Encl. 5.1**

Learning Objective: **ADM-OMP 4.3**

Question Source:

Bank # **ADM-845**
Modified Bank # _______
New _______

Question History:

Previous NRC Exam _______
Previous Quiz / Test _______

Question Cognitive Level:

Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:

55.41 _____
55.43 __X__

Comments:

1 POINT

QUESTION # 99

SRO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS:

- ES Channel 5 actuated five (5) minutes ago

CURRENT CONDITIONS:

- All Channel 5 ES equipment has been returned to normal
- Reactor power = 90% and steady
- RCS pressure = 2155 psig and steady
- RCS Tave = 579°F and steady

Which ONE of the following is correct concerning NRC notification per OMP 1-14 and NSD-202 and what level of approval (if any) was required prior to repositioning ES Channel 5 components?

SEE ATTACHMENT

A. 1 hour / no approval required, procedure directed

B. 4 hour / no approval required, procedure directed

C. 1 hour / CRSRO

D. 4 hour / CRSRO

1 POINT

QUESTION # 99

G2.4.30 (2.2/3.6) SRO ONLY CFR 43.5 PRA 5-1-00

A. Incorrect – It is a 4 hour notification. 1 hour would be correct for discharge to the RCS. ES Channel 5 will not cause discharge to the RCS. Repositioning safety systems requires SRO approval.

B. Incorrect – First part is correct due to ESF actuation. Repositioning safety systems requires SRO approval.

C. Incorrect – 1 hour would be correct for discharge to the RCS. ES Channel 5 will not cause discharge to the RCS. The second part is correct. Repositioning safety systems requires SRO approval.

D. Correct – NSD 202, step 202.8.2.b NRC notification is required if a safety system has actuated unnecessarily. Repositioning safety systems requires SRO approval.

OMP 1-14
Notifications
Attachment B
NRC Event Notification Worksheet

NRC Event Notification Worksheet				
Notification Time	**Facility or Organization**	**Unit**	**Caller's Name**	**Call Back #**
	Oconee Nuclear Station			ENS 256-9931 (864) 885-

NRC Operations Officer Contacted:	NRC Event Number:

Event Time/Zone	Event Date	Power/Mode Before	Power/Mode After

Event Classifications

- ☐ General Emergency
- ☐ Site Area Emergency
- ☐ Alert
- ☐ Unusual Event
- ☐ 50.72 Non-emergency (see other columns)
- ☐ 72.75 Spent Fuel (ISFSI)
- ☐ 73.71 Physical Security
- ☐ Transportation
- ☐ 20.2202 Material/Exposure
- ☐ 26.73 Fitness for Duty
- ☐ Other:

1-Hour Non-emergency 10 CFR 50.72 (b)(1)

- ☐ (i) (A) TS Required Shutdown
- ☐ (i) (B) TS Deviation
- ☐ (ii) Degraded Condition
- ☐ (ii) (A) Unanalyzed Condition
- ☐ (ii) (B) Outside Design Basis
- ☐ (ii) (C) Not Covered by OPs/EPs
- ☐ (iii) Earthquake
- ☐ (iii) Flood
- ☐ (iii) Hurricane
- ☐ (ii) Ice/Hail
- ☐ (iii) Lightning
- ☐ (iii) Tornado
- ☐ (iii) Other Natural Phenomenon

1-Hour Non-Emergency (continued)

- ☐ (iv) ECCS Discharge to RCS
- ☐ (v) Lost ENS
- ☐ (v) Lost Other Assessment/Communication
- ☐ (v) Emergency Siren INOP
- ☐ (vi) Fire
- ☐ (vi) Toxic Gas
- ☐ (vi) Rad Release
- ☐ (vi) Other Hampering Safe Operations

4-Hour Non-Emergency 10 CFR 50.72 (b)(2)

- ☐ (i) Degraded While Shutdown
- ☐ (ii) RPS Actuation (scram)
- ☐ (ii) ESF Actuation
- ☐ (iii) (A) Safe S/D Capability
- ☐ (iii) (B) RHR Capability
- ☐ (iii) (C) Control of Rad Release
- ☐ (iii) (D) Accident Mitigation
- ☐ (iv) (A) Air Release > 20x App B
- ☐ (iv) (B) Liquid Release > 20x App B
- ☐ (v) Offsite Medical
- ☐ (vi) Offsite Notification

OMP 1-14
Notifications
Attachment B
NRC Event Notification Worksheet

Event Description (Include systems affected, actuations and their initiating signals, causes, effect of event on plant, actions taken or planned, etc.)
Event:
Initial Safety Significance:
Corrective Action(s):

Anything unusual or not understood?	☐ Yes (Explain above)	☐ No
Did all systems function as required?	☐ Yes	☐ No (Explain above)
Mode of operations until corrected:	Estimated restart date:	

Does event result in a radiological release, RCS leak, or steam generator tube leak?	☐ Yes (complete page 3)	☐ No

Does the event result any of the units experiencing a transient?	☐ Yes (complete Oconee Plant Status sheet)	☐ No

Notifications			
NRC Resident:	Y/N/will be	Plant Manager:	Y/N/will be
Notified By:	Time:	Notified By:	Time:
State(s):	Y/N/will be	Operations Superintendent:	Y/N/will be
Notified By:	Time:	Notified By:	Time:
Local:	Y/N/will be	Other Government Agencies:	Y/N/will be
Notified By:	Time:	Notified By:	Time:
Media/Press Release:	Y/N/will be	Other:	Y/N/will be
Notified By:	Time	Notified By	Time

Operations Shift Manager/Emergency Coordinator Approval:	**Date/Time:**

NRC Notification Complete by Caller/NRC Communicator:	**Date/Time:**

OMP 1-14
Notifications
Attachment B
NRC Event Notification Worksheet

Additional Information for Radiological Releases			
Radiological Release (check as applicable with specific details in event description including release path)			
☐ Liquid Release	☐ Gaseous Release	☐ Unplanned Release	☐ Planned Release
☐ Monitored	☐ Unmonitored	☐ Off-Site Release	☐ TS Exceeded
☐ Personnel Exposed or Contaminated	☐ Rad Mon Alarms	☐ Off-Site Protected Actions Recommended	☐ Terminated
		☐ Areas Evacuated	☐ Ongoing

	Release Rate (Ci/sec)	% TS Limit	HOO Guide	Total Activity (Ci)	% TS Limit	HOO Guide
Noble gas:			0.1 Ci/sec			1000 Ci
Iodine:			10 µCi/sec			0.01 Ci
Particulate:			1 µCi/sec			1 mCi
Liquid (excluding tritium and dissolved noble gases):			10 µCi/min			0.1 Ci
Liquid (tritium):			0.2 Ci/min			5 Ci
Total Activity:						

	Plant Stack	Condenser/Air Ejector	Main Steam Line	SG Blowdown	Other
Rad Monitor Readings:					
Alarm Setpoints:					
% TS Limit (if applicable):					

Additional Information for Reactor Coolant Leaks and Steam Generator Tube Leaks			
Location of the leak (e.g. SG, valve, pipe, etc.)			
Leak Rate:	Units (gpm/gpd):	TS Limit:	Sudden or Long Term Development:
Leak Start Date:	Time:	Coolant Activity & Units: Primary - Secondary -	
List of Safety Related Equipment Not Operational:			

Duke Power
A Duke Energy Company

NUCLEAR POLICY MANUAL

Nuclear System Directive: 202. **Reportability**

Process/Program Owner: **Regulatory Compliance Managers BEST**

REVISION NUMBER	ISSUE DATE
0	11/01/92
1	02/28/94
2	06/09/94
3	08/22/94
4	12/12/94
5	06/14/95
6	06/13/96
7	03/12/97
8	06/16/97
9	06/16/98
10	12/20/98
11	04/30/99
12	02/23/00

CATAWBA	MCGUIRE	OCONEE
Approved By/Date	Approved By/Date	Approved By/Date
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Issued By: C. J. Thomas
Manager, Nuclear Regulatory & Industry Affairs

DOCUMENT REVISION DESCRIPTION

REVISION NO.	PAGES or SECTIONS REVISED AND DESCRIPTION
10	Section 202.3 is being revised to reflect the approved version of NUREG 1022, Rev. 1. Section 202.6.1 is being revised to reflect changes made by the NRC to Part 21. Section 202.7.1, reportable example 'a' and 'b' are being revised to reflect implementation of the ITS at MNS. Section 202.7.2 is being revised to include information on the reportability policy regarding missed ASME Section XI required visual inspections after maintenance, to reflect a new philosophy concerning plants that have a delay of up to 24 hours in declaring an LCO or Tech Spec not being met (this philosophy resulted from the approval by the NRC of NUREG 1022, Rev. 1), to reflect new ITS section numbers for MNS, deletion of non-reportable example 'j' and a new example concerning multiple test failures was added under the "Reportable Examples". Section 202.8.2 is being revised to add a "Non-Reportable" example regarding Oconee's Emergency Feedwater system. Section 202.10 is being revised to add Fire Protection Program reporting information and more guidance on the reporting of Tech Spec Safety Limits violation. Appendix A, item #12 is being revised to reflect the new ITS section number for MNS.
11	Revised Section 202.4, entitled it, "Roles and Responsibilities" and renumbered remaining sections of NSD. Revised Section 202.6.4 to reflect that Tech Spec 3.03 is generic to all three sites. This is due to the implementation of the ITS. Added guidance on Past Operability in Section 202.6.4.1. Section 202.7.2 was also revised to reflect that Tech Spec 3.0.3 applies to all three sites. This is due to the implementation of the ITS at all three sites. Example 'e' was also revised to reflect to pressurizer heatup and cooldown rates no longer being in Tech Specs and Tech Specs no longer requiring D/G Special Reports. Items 'k' and 'd' were reused. Section 202.7.3 was revised to clarify what is considered a deviation and Reportable example 'b' was deleted. Section 202.7.4, item, #2 was revised to reflect renumbering of NSD sections. Section 202.8.1 was revised to reflect the new CNS ITS numbers governing tube plugging and examples that referenced these sections were also revised. Section 202.10, Tech Spec Safety Limit, was revised to reflect the change from 14 to 30 days for the submission of a written report. This is due to the implementation of the ITS. Throughout the directive, "TS" was changed to "Tech Spec".
12	Revised Section 202.6.4.1, 6th paragraph, inserted the phrase "have to" in the discussion "How far back do I look;" inserted the phrase "is generally sufficient" in place of "ago" in the 1st sentence; added the phrase "however, is there is reason to believe that the SSC was inoperable ...;" deleted the sentence addressing an exception at the end of the 1st sentence, replacing it with the phrase "or until an inoperability in excess of the Tech Spec CT is discovered;" replaced the phrase "are not appropriate" with "are not necessary" in the last sentence of the paragraph.

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202. REPORTABILITY

202.1 INTRODUCTION

The NRC's regulations set forth in Title 10, Chapter I, Code of Federal Regulations require the reporting of various events or conditions in connection with licensee activities. This directive is presented to ensure nuclear station compliance and consistency with the reporting regulations of the 10CFR.

202.2 PURPOSE

The purpose of the directive is to provide guidance for use in determining reportability of station events or conditions under the provisions of the Immediate Notification Requirements of Significant Events (10CFR 50.72), the Licensee Event Reporting System (10CFR 50.73), 10CFR Part 21, Reporting of Defects and Noncompliance, 10CFR 50.9, Completeness and Accuracy of Information, to ensure proper and consistent reporting. Security event notifications are addressed in the Duke Nuclear Security Manual, 10CFR 20 radiological notification requirements and 10CFR 72, ISFSI, notification requirements are included in the NSD. For Part 50.72 reporting, this directive only addresses [50.72(b)] one/four hour notifications for non-emergency events, since adequate guidance currently exists in each stations' Emergency Plan's implementing response procedures for emergency events [50.72(a)] and their classifications.

202.3 REFERENCES

1. 10 Code of Federal Regulations 50.72, 50.73, Part 20, 21, 72 and 50.9
2. Federal Register; Vol. 48, No. 144 and 168; July 26, 1983/August 29, 1983; "Licensee Report System" and "Immediate Notification Requirements of Significant Events"; final rule
3. NUREG 1022, Sept. '83, and Supplement 1, Feb. '84
4. BWR Owners' Group LER/JCO Committee Consolidated Event Reporting Guidance document
5. NUREG 1397, Feb. '91, "An Assessment of Design Control Practices and Design Reconstitution Practices in the Nuclear Industry"
6. Generic Letter 91-18, "Operable/Operability: Ensuring the Functional Capability of a System or Component"
7. Standard Review Plan (NUREG 800)
8. Federal Register (56 FR 36081); July 31, 1991; "10 Code of Federal Regulations Parts 21 and 50.55(e)"
9. NUREG 1022, Revision 1.

202.4 ROLES AND RESPONSIBILITIES

202.4.1 OPERATIONS

1. Responsible for reportabilty timeliness
2. Makes reportability determination (i.e., knowledgeable of issue, performs reasonableness review, ensures license requirements are met)
3. Makes notification to NRC and other required parties (Note: Reg Compliance may perform this function for some issues: e.g., design basis issues)

4. Ensures license and NRC requirements are met

202.4.2 REGULATORY COMPLIANCE

1. Reviews PIPs on front end (screening meeting) for potential reportability issues; follows up on screened PIPS that are unclear with respect to reportability
2. Independently assures timeliness is in accordance with NRC requirements
3. Provides lead role in making reportability recommendations to Operations Shift Manager (OSM) (For engineering related reportability evaluations, works closely with engineering to ensure the right questions are being asked and performs QV & V to ensure licensing basis assumptions are valid and proper)
4. Notifies OSM immediately when sufficient evidence exists that indicates an item is reportable
5. Provides support to the OSM as needed for NRC notification (may make NRC notification for some issues: e.g., design basis issues)
6. Ensures NSD conformance with NRC requirements
7. Ensures process is implemented in accordance with NSD

202.4.3 ENGINEERING

1. Notifies OSM and Regulatory Compliance immediately when sufficient evidence exists that indicates an item is reportable. (Note: Engineering scope for reportability also includes SSC past operability evaluations (50.72 and 50.73), and Part 21 evaluations)
2. Keeps OSM and Regulatory Compliance informed of status of reportability evaluation and timeline to complete
3. Ensures engineering analysis and calculations assure SSE's can perform required functions
4. Determines if component failures meet 10 CFR 21 reporting criteria

202.5 GENERAL CONSIDERATIONS

Guidance is presented in this directive in order to ensure nuclear station compliance and consistency with the reporting regulations of 10CFR 50.72, 50.73, Part 21, and 50.9.

The purpose of the emergency notification requirements in 10CFR 50.72 is to inform the NRC of deficient conditions or events that have immediate safety significance or that may require NRC awareness or action in response to potential public interest. The purpose of the LER Rule in 10CFR 50.73 is to identify the types of deficient conditions or events that are significant to the NRC so that the NRC may perform engineering studies of operational anomalies, trends and pattern analyses of operational occurrences. In general, many of the conditions and events that require immediate notification will also require an LER, as is reflected by the many parallel requirements specified in 50.72 and 50.73. Therefore, the event reporting guidance is arranged in the sequence of the Emergency Notification Rule, along with the corresponding sections of the LER Rule.

In some cases, such as discovery of an existing but previously unrecognized condition, it may be necessary to undertake an evaluation to determine if an event or condition is reportable. An evaluation should generally proceed on a schedule commensurate with the safety significance of the question. Plant operation may continue provided there is reasonable expectation that the equipment in question is operable. Whenever this reasonable expectation no longer exists, or significant doubts begin to arise, the equipment should be considered inoperable and appropriate actions, including reporting, should be taken promptly (Refer to NSD 203, "Operability" for more guidance on operable/inoperable equipment).

In evaluating a potentially reportable item, this document should be reviewed to identify all possible sections of the event reporting rules which might be applicable. It should be noted that an item can be reportable under several

criteria and, in accordance with 50.72 and 50.73, a reportable item must be reported under all applicable criteria. In this directive, an example in a specific section is only evaluated for reportability under that specific criterion. In actual application, that same example might be reportable under other criteria. For ENS calls, the report should be made in accordance with the most stringent criterion that applies in order to fulfill all 50.72 requirements (e.g. an event that falls under a 1 hour and 4 hour notification should be reported within 1 hour, which also satisfies the 4 hour requirement). For LERs that are reportable under more than 1 criterion, all applicable blocks should be marked on the LER form.

202.6 SPECIFIC GUIDANCE

202.6.1 PART 21 REPORTING

10CFR Part 21, "Reporting of Defects and Noncompliance", should be considered for those components or materials in which a condition is discovered that would render the item incapable of performing its design function.

For items not yet installed, but received and accepted for installation; or installed in the system, but the system has not been declared operable and is not required to be operable, an operability evaluation shall be performed on the degraded component and its effect on the system if it was installed and required to be operable. If the results of the evaluation show that the system would have been inoperable, then the determination of whether a substantial safety hazard could be created, shall be made.

If this evaluation cannot be completed within 60 days from the discovery of the deviation, an interim report is to be written and submitted to the NRC. This report should describe the deviation that is being evaluated and should also state when the evaluation will be completed.

A notification to the NRC of a deviation is not necessary if the Licensee has actual knowledge that the NRC has been notified in writing of the deviation.

A substantial safety hazard is defined as a loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety. Moreover, if a system function is lost, and that system is taken credit for in the accident analysis (UFSAR Ch. 15), then a substantial safety hazard is created. [NOTE: Loss of a system's safety function means loss of both trains of a 2 train system. If the degraded component only affected 1 out of 2 trains or channels, then a loss of system safety function cannot occur.]

If it is determined that a substantial safety hazard is created, then the item(s) shall be considered reportable pursuant to Part 21. Initial notification shall be made within 2 days (calendar) of the reportability determination to the Director, Nuclear Reactor Regulation (NRR) or Nuclear Material Safety and Safeguards (NMSS), as appropriate. The station Senior Resident Inspector and the Region should also be included in the notification. The verbal notification shall be followed up by a written report within 30 days of the reportability determination.

For items installed and the affected system declared operable and the system is required to be operable, follow the same procedure as above, except the verbal notification shall be made per 50.72(b)(1) (ii) within 1 hour. If the item is discovered during shutdown and the system is not required to be operable, the notification shall be made per 50.72(b)(2)(i) within 4 hours. An LER will then be required to be written and submitted within 30 days per 50.73(a)(2)(ii) (in addition to this paragraph, most probably other sections within 50.73 will apply and should be indicated on the LER form). Both the ENS notification and the LER should indicate that the condition is also Part 21 reportable. This can be accomplished on the LER form by marking the "Other" block and typing "Part 21". The appropriate vendor shall be notified of the problem immediately, if not already.

202.6.2 10CFR 50.9 REPORTING

The stated intent for 10CFR 50.9(a) is that information provided to the NRC be complete and accurate in all material respects. Sections 50.72 and 50.73 contain provisions for updating and revising reports that should be used to

correct material incompleteness or inaccuracies that are discovered. For example, submitting a revised LER would be appropriate to correct any previously submitted inaccuracies of a material nature.

10CFR 50.9(b) states that any licensee information with significant public health and safety, or common defense and security implications be reported to the NRC, except where a specific reporting requirement exists. The Statements of Consideration for 50.9 refer to such information as "residual information" that could affect licensed activities. The provisions of 50.9 should not be used to report information that is required to be reported under other reporting rules such as 50.72, 50.73, and Part 21.

If a condition is determined to be reportable under Part 50.9, the station shall notify the Region within 2 working days of the discovery of the information. A special report shall be written and submitted to the Region within 30 days of discovery of the event. The report should contain all relevant information pertaining to the circumstances involved, as well as, any planned corrective actions to be taken to prevent recurrence.

202.6.3 EMERGENCY NOTIFICATION SYSTEM REPORTING

202.6.3.1 Reporting Timeliness

The timing for ENS reporting is described in 10CFR 50.72 as "immediate" and "as soon as practical and in all cases within one (or four) hour(s)" of the occurrence of an event (depending on its significance). The intent is to require reportability decisions to be made in a timely manner so that ENS notifications are made to the NRC as soon as practical, keeping in mind the safety of the plant comes first. The event reportability timeclock generally starts at the time of the event or the discovery of the condition. For example, the reportability time clock would start at the time of a Reactor trip or initiation of plant shutdown in accordance with Tech Specs. In come cases, such as discovery of an existing but previously unrecognized condition, it may be necessary to undertake an evaluation in order to determine if an event or condition is reportable. This evaluation should generally proceed on a schedule commensurate with the safety significance of the question. When evaluating more complex issues such as design basis questions, the clock should start once appropriate station management makes a decision with respect to the operability of the system or component. For example, the reportability time clock begins for a past operability evaluation once the evaluation concludes that the associated system was inoperable. When evaluating an event for reportability, consideration should be given to the requirements contained in section 202.8, Follow-up Notifications.

It is recognized that in the short time frame between the event and the ENS notification, there may not be enough time for an evaluation of the cause, effect, or compensatory measures taken. It is more important that the NRC be quickly made aware of the situation than it is for the station to answer every NRC question at the time of the initial notification. In other words, when evaluating a potentially reportable item, and there is doubt regarding whether to report or not, the NRC's policy is that licensees should make the report. Update ENS notifications should be made to provide additional information or analysis as it becomes available as appropriate.

202.6.3.2 Voluntary/Courtesy Notifications

The station may make voluntary or courtesy ENS notifications about events or conditions the NRC may be interested in. The NRC will evaluate and respond to any voluntary notification of an event or condition, as its safety significance warrants, regardless of the reporting classification of the reporting requirement. If it is determined later that the event is reportable, then another ENS notification should be made under the appropriate 50.72 criterion.

202.6.3.3 ENS Notification Retractions

If the station makes a 50.72 notification and later determines that the event or condition was not reportable, the appropriate station personnel should contact the NRC Operations Center to retract the previous notification and explain the reasons for the decision.

202.6.4 SPECIFIC REPORTING GUIDANCE TO 50.72 AND 50.73

The sections that follow will address guidelines for reporting one and four hour non-emergency events and LERs. Specific guidance for reporting Emergency classifications will not be provided in this directive since adequate guidance currently exists under the Emergency Plan implementing response procedures. If guidance is needed with respect to the reportability of an environmental event, Environmental Management should be contacted for assistance.

In addition to the specific guidelines given under each section, a descriptive list of examples, some of which have occurred, of nuclear station events and conditions that have been determined either reportable or non-reportable pursuant to 10CFR 50.72 and 50.73 are provided. A single event may fall under several reporting criteria. Although this list will not reflect all reporting sections applicable to a specific example, the example will be included under its most immediate reporting requirement, with the exception of those events that may also be reported under an Emergency Class declaration.

Each entry into Tech Spec 3.0.3 requires an ENS notification (red phone notification). This is a decision that has been conservatively established by Duke Management (reference Regulatory Compliance Assessment report number RGC-01-97). In addition, each event that requires entry into Tech Spec 3.0.3 shall be reported separately, even if a plant shutdown is in progress or Tech Spec 3.0.3 has already been entered for another event.

202.6.4.1 Past Operabilty Determinations

Past operability determinations are performed to support reportability. Although there is no attendant duty of protecting the public, past operability determinations should be completed in a timely manner. For example, there are one hour, four hour, and 30 day reporting requirements associated with past operability issues. The reporting sections that should be consulted and are most applicable for past operability determinations are operations prohibited by Tech Specs {50.73 (a)(2)(i)(B)}, common-mode failures of independent trains or channels {50.73(a)(2)(vii)}, events or conditions that could have prevented the fulfillment of a safety function {50.72(b)(2)(ii)} and the plant in a degraded or unanalyzed condition {50.72(b)(1)(ii)) or 50.72(b)(2)(i).

In most cases, it is expected that past operability determinations can be made in concert with the reportabilty determination (e.g., there is firm evidence that Tech Spec Completion Time has been exceeded, etc.). In other cases, additional information regarding past operability may be needed to complete the reportability determination. For these cases, it is expected that the required information can be obtained and the reportability determination completed within thirty days. Some few exceptional cases may take longer.

Also, in most cases, engineering judgement by a technically qualified individual is all that is needed to support the past operability determination. A documented engineering analysis is not a requirement as a basis for an engineering judgement for all events or conditions – it's only necessary for particularly complex situations requiring in-depth analysis. When exercising engineering judgement, however, the NRC recommends that licensees record in writing that a judgement was exercised by identifying the individual making the judgement and the date made, and briefly documenting the basis for the judgement.

Past Operability determinations are only required for conditions that may have existed prior to discovery and affect operability of SSCs subject to the Tech Specs through the definition of operability. In general, for the purpose of evaluating the reportability of situations found during surveillance tests, it should be assumed that the situation occurred at the time of discovery, unless there is firm evidence to believe otherwise. For example, if a standby component with a seven day Limiting Condition for Operation (LCO) is found to be inoperable because it was assembled improperly during maintenance conducted thirty days previously, then there is firm evidence that it had been inoperable for the entire thirty days, and an LER is required.

When performing past operability determination, the impact of the condition on the affected component (s) is first evaluated. If it is determined the condition renders the component (s) inoperable, the effect of the component's inoperability is evaluated with respect to Tech Spec LCOs. If an LCO does not exist for an affected component, then the component's inoperability is evaluated with respect to its impact on the operability of any associated train, channel, system or structure. Because the Tech Specs do not directly specify an LCO for many items that perform

supporting functions, a knowledge of the plant design basis is essential to determine which support systems can affect operability.

A common question when performing past operability determinations is "How far back do I have to look?" If the SSC could have been inoperable in excess of its Tech Spec CT in the past, a look back of two years or two refueling intervals is generally sufficient. However, if there is reason to believe that the SSC was inoperable in excess of its Tech Spec CT greater than two years or two refueling intervals ago, then the look back should cover the entire questionable period or until an inoperability in excess of the Tech Spec CT is discovered. The intent is to perform a reasonable search for a condition that could be reportable. In general, exhaustive searches or in-depth analyses are not necessary.

202.7 1-HOUR ENS NOTIFICATIONS AND LERS

This section addresses 50.72(b)(1) 1-hour notifications for non-emergency events and the associated LER. If not reported as a declaration of an emergency class under 50.72(a), the station is required to notify the NRC as soon as practical and in all cases within 1 hour of the discovery of any of the events specified.

In addition to similar reporting criteria under both 10CFR 50.72 and 50.73, several requirements for only 50.72 notifications or only LERs are included in this section because of the sequential numbering scheme used.

202.7.1 PLANT SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATIONS

§50.72(b)(1)(i)(A)	**50.73(a)(2)(i)(A)**
Licensees shall <u>report</u>: "The <u>initiation</u> of any nuclear plant shutdown required by the plant's Technical Specifications."	Licensees <u>shall submit a Licensee Event Report on</u>: "The <u>completion</u> of any nuclear plant shutdown required by the plant's Technical Specifications."

1. 50.72

   The 50.72 reporting requirement is intended to capture those events for which Tech Specs require the initiation of reactor shutdown to provide the NRC with early warning of safety significant conditions. "Initiation" is the performance of any action to start reducing reactor power to achieve an operational condition or mode that requires the reactor to be subcritical, as a result of a Tech Spec requirement. This includes any means of power reductions such as control rod insertion or boron concentration changes.

2. 50.73

   For 50.73 reporting purposes, the phrase "completion of any nuclear plant shutdown" is defined as the point in time during a Tech Spec required shutdown when the plant enters Mode 3 (MNS and CNS) or Hot Shutdown (ONS). Therefore, if a failure can be corrected before the unit is required to be in Mode 3 (MNS and CNS) or Hot Shutdown (ONS), an LER is not required. This includes a situation where the plant is shutdown, the problem is fixed and the unit is returned to power before the completion of shutdown was required by Tech Specs. The shutdown is reportable, however, if the failure cannot be corrected before the unit was required to be shutdown.

EXAMPLES

<u>Reportable</u>

a. Two out of three channels for a certain ESF function failed. Tech Specs require the unit to be placed in Mode 3 (Hot Standby)(MNS and CNS) within 6 hours with less than the minimum required channels operable. After 1 hour, the station began a load reduction from full power at 20% per hour. Within 15 minutes of the initial load reduction, an ENS notification was made. The station made an update ENS call

3 hours later after the equipment was repaired, the channels were declared operable, and the power reduction was stopped before completion of the shutdown.

An ENS notification per 50.72 was required because the power reduction was an initiation of plant shutdown. {Note, however, an LER was not required because the shutdown was never completed (i.e., Mode 3(MNS and CNS) was not entered).}

b. When leakage around the primary containment ventilation exhaust dampers exceeded the maximum allowable combined secondary bypass leakage rate, the plant Tech Specs required the plant to be in Hot Shutdown within 12 hours. The station commenced a reactor shutdown at 10% per hour and made an ENS call within 10 minutes of entering the LCO Action. Hot Shutdown was reached 10 hours later.

c. While the unit was at 100%, the unit's nuclear service water pump discharge valve failed its monthly periodic test. Because the station knew repairs could not be made during the remaining time allowed by Tech Specs (72 hour Action), the unit was placed in Cold Shutdown within 1 day. The ENS call was made 30 minutes after the initial load decrease (even though there were 50 hours left on the Tech Spec clock).

Non-Reportable

d. Two out of three channels for a certain ESF function failed. Tech Specs require the unit to be placed in Mode 3 (Hot Standby)(MNS and CNS) within 6 hours with less than the minimum required channels operable. Since IAE personnel felt the repairs could be made within 3 hours, the Shift Supervisor decided to hold power for 3 hours. The equipment was repaired and the station declared the failed channels operable 4 hours later. No ENS call was made since there was no shutdown initiated.

e. While the unit was at 100%, the unit's nuclear service water pump discharge valve failed its monthly periodic test. Because the station thought repairs could be made during the remaining time allowed by Tech Specs (72 hour Action), the unit held at full power. The ENS call was not required since the valve work took only 40 of the remaining 50 hours left on the Tech Spec clock, and no power reduction had begun.

202.7.2 TECHNICAL SPECIFICATION PROHIBITED OPERATION OR CONDITION

10 CFR 50.72	§50.73(a)(2)(i)(B)
[There is no corresponding Part 50.72 requirement. However, for certain operations or conditions prohibited by a plant's Tech Specs, other reporting requirements may apply, such as 50.72(b)(1)(ii) and (b)(2)(iii); 50.36(c)(1) and (2); 20.2202; and 20.2203.]	Licensees shall report: "Any operation or condition prohibited by the plant's Technical Specifications."

1. General

An LER is required under this criterion if an LCO and associated Action statement are not met. The time constraints included in the associated Action statements are based on the safety significance of the component or system being removed from service. The NRC is interested in the frequency of occurrence and the Tech Spec involved in events which a shutdown did not occur within the given time constraint. The condition is reportable even if the condition was not discovered until later and was corrected upon discovery. Therefore, if an inoperable component or system is discovered, an investigation is required in order to determine how long the component has been in the degraded condition. Reportability per this section can be determined based upon the results of the investigation.

The LER rule does not address violations of License Conditions in documents other than Tech Specs. Such notifications are reportable as specified in a plant's license or other applicable document.

2. Inoperable Upon Discovery

If, through the course of the investigation of the inoperable component or system, it cannot be determined how long it was in the as found condition, there are 2 different assumptions to be made in order to reach a reportability decision. If the inoperable condition was discovered during the course of a surveillance, maintenance, or inspection, the condition is assumed to have occurred at the time of discovery (same philosophy as operability), provided no firm evidence exists that would indicate when the failure occurred. If, however, the condition was discovered by chance (e.g., an operator discovers a mispositioned valve during a walkdown), and it is obvious that the degraded condition was caused by some personnel action, it is assumed that the condition has existed since the component (or system) was known to be operable. Once these determinations are made, the most stringent Action statement should be compared to the time the item was inoperable. If the most limiting time constraints of the LCO Action were exceeded, the condition is reportable per this criterion. Since these determinations may be subjective at times, the evaluator should consider what is reasonable based upon the circumstances surrounding the as found condition.

3. Tech Spec 3.0.3

Tech Spec 3.0.3 establishes requirements for actions when an LCO is not met and no Action statement is provided. If Tech Spec 3.0.3 is entered for any reason, this means the LCO cannot be met. Since there is no existing corresponding Action statement to comply with, then a condition prohibited by Tech Specs exist, and the condition is reportable. Keep in mind that merely entering Tech Spec 3.0, regardless of the amount of time in 3.0.3, is reportable per this section. Additionally, the plant condition that caused entry into Tech Spec 3.0.3 shall be evaluated for reportability under the most applicable 50.72 reporting criteria (see sections 202.7.4 and 202.8.3).

4. Missed Surveillance

To determine the reportability of a missed or previously performed, inadequate surveillance, consideration must be given to both the requirements and allowances of Tech Specs Surveillance Requirements (SR) 3.0.1 and 3.02. Tech Spec SR 3.0.2 establishes the maximum allowable time interval between surveillances. Tech Spec SR 3.0.1 requires equipment or systems to be declared inoperable once the maximum allowed surveillance interval (extension) is exceeded. Tech Spec SR 3.0.1 also states that surveillances do not have to be performed on inoperable equipment. In order for a condition prohibited by Tech Specs to exist, the LCO and associated Action statement have not been met (see Section 202.7.2 Item 1). Therefore, for a condition to be reportable, tte maximum allowed surveillance interval (including ext.) plus the time constraints of the associated LCO Action statement for the given system must be exceeded. This philosophy also applies to systems, subsystems, trains or components that are required by Tech Specs to be tested on a staggered test basis and the tests were not staggered. However, if a train or component is required by Tech Specs to be tested on a staggered test basis and the testing was <u>performed within</u> the specified interval but not staggered, then this should be treated as a condition prohibited by Tech Specs and reportable.

Tech Specs allow a delay of up to 24 hours in declaring an LCO or a Tech Spec requirement not met if it is found that a surveillance was not performed within its specified frequency or interval. However, this does not change the fact that the condition existed longer than allowed by Tech Specs. Failure to perform a surveillance within its frequency or interval is still reportable. The delay merely specifies appropriate remedial action.

5. IST Requirements per Tech Spec 5.5

Tech Specs 5.5.8 covers IST requirements for ASME Class 1, 2, and 3 components. Missed or deficient IST/ASME surveillances are reportable when, as a result of the missed or deficient surveillance, a Tech Spec controlled system must be declared inoperable and the LCO action statement has been exceeded. The reportability evaluation should proceed per the guidance in Section 202.7.2, 1 through 4 (above), as applicable.

Failure to perform a visual inspection required by ASME Section XI does not in itself affect the operability of the component. As such, failure to perform visual inspections will not be reported as a condition prohibited by Technical Specifications. However, these missed inspections shall be reported to the NRC Resident Inspector for inclusion in any inspections as determined appropriate.

6. Administrative Requirements

Tech Specs include administrative requirements that are required to be followed. Failure to meet such requirements is a violation of Tech Specs. Whether it is reportable as an LER depends on whether it results in a condition that affects the safe operation of the plant, or is reportable under other provisions of the LER Rule. Occasionally, purely administrative requirements are also found under the LCO Action and Surveillance sections of Tech Specs (e.g., Tech Spec Special Report required to be submitted in 10 days). If the report is submitted 2 days late, this is a violation of an administrative requirement of Tech Specs, but since the safe operation of the plant is unaffected by the delay, the incident is not reportable as an LER.

Radiological conditions and events that are prohibited by Tech Specs should be evaluated for reportability under the requirements of 10CFR 20.403 and 20.405. Redundant reporting is not required.

EXAMPLES

<u>Reportable</u>

a. Doghouse water level instrumentation functional test was not performed on 1 train of channels. Tech Specs require this surveillance to be performed once every refueling outage. The missed test was discovered 1 month later and the Action statement requires continuous level monitoring with 1 or more trains inoperable.

b. The IWV Program lists valve NV-150 as a valve that requires a VST quarterly. With NV-150 inoperable, Train A of the Chemical and Volume Control (NV) system is inoperable. This valve has been tested only during Cold Shutdown.

c. Unit 1 operated at greater than 100% licensed thermal power for a period greater than the Tech Specs allow.

d. During testing of valve NI-144, the valve closed and would not reopen, rendering 'B' train Safety Injection(NI) and ECCS inoperable. Since 'A' train was already inoperable due to KC heat exchanger work, Tech Spec 3.0.3 was entered. Tech Spec 3.0.3 was exited 30 minutes later after valve NI-144 was declared operable.

e. While preparing to perform a surveillance on an air operated valve, a technician discovered the instrument air line disconnected from the port. The inoperable valve renders its respective train inoperable. Upon investigation, it was determined that the line was most probably not connected properly after maintenance performed 2 weeks earlier. The Tech Spec Action for this train is 72 hours. The valve was not immediately retested following maintenance.

f. While performing surveillances on the main steam safety valves, of the 20 valves tested, 17 were out of tolerance (13 with set points above Tech Specs by as much as 4 percent). The existence of similar discrepancies in multiple valves is an indication that the discrepancies arose over a period of time and therefore reportable.

<u>Non-Reportable</u>

g. The IWV Program lists valve NV-151 as a valve that requires a VST quarterly. With NV-151 closed and incapable of opening, Train A of the NV system is inoperable. This valve has not been tested in 9 months. Upon this discovery Operations confirmed that the valve had been in the open position for the entire period, thus, in its safety position and train 'A' NV was capable of performing its intended safety function.

   Even though the IST program was violated, an LER is not required because the failure to test the valve's movement did not render its associated system or train inoperable.

h. Upon entering Mode 4(MNS and CNS), operators observed that the NC system heatup rate had been exceeded during the removal of the 'A' reactor coolant pump. The Tech Spec LCO Action requires the rate to be restored within 30 minutes (which it was) and an engineering evaluation performed on the integrity of the pressurizer. The evaluation was performed immediately and confirmed the structural integrity acceptable, thus complying with the Action statement.

i. A certain containment isolation valve failed to meet its stroke timing test of 5.0 seconds during an outage. The subsequent investigation failed to reveal any evidence as to why the valve was slower for this surveillance. After maintenance, the valve tested in under 5.0 seconds and was declared operable.

j. Failure to perform a visual inspection after maintenance as required by ASME Section XI does not in itself affect the operability of the component. As such, failure to perform visual inspections will not be reported as a condition prohibited by Tech Specs.

202.7.3 TECH SPEC DEVIATION PER 10CFR 50.54(X)

§50.72(b)(1)(i)(B)	§50.73(a)(2)(i)(C)
Licensees shall report: "Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x) of this part."	Licensees shall report: "Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x) of this part."

1. General

10CFR 50.54(x) generally allows the station to take reasonable action in an emergency even though the action is in violation of the License Condition or Tech Specs provided: (1) the action is immediately needed to protect the health and safety of the public (including station personnel), and (2) no action consistent with the License Conditions and Tech Specs is obvious that can immediately provide adequate protection. In accordance with 50.54(y), such action requires, as a minimum, prior approval by a licensed Senior Reactor Operator.

Deviation from an Emergency Procedure which alters the intent of the procedure without prior approval is also a violation of Tech Specs and would require reporting under this section.

EXAMPLES

<u>Reportable</u>

a. With the unit at 100% power, the upper containment airlock inner door was opened to allow a technician to exit from the containment while the upper door was inoperable, resulting in a loss of containment integrity. The Technician was inside containment when the lower airlock failed, requiring exit through the upper door.

The decision to open the upper containment airlock inner door exercised an allowable option under 10CFR 50.54(x). Immediate action was considered necessary for the technician to exit the containment for his personal safety. An ENS call was made within 1 hour of the breech of containment.

202.7.4 OPERATING PLANT IN A DEGRADED OR UNANALYZED CONDITION, OR OUTSIDE DESIGN BASIS

§50.72(b)(1)(ii)	§50.73(a)(2)(ii)
Licensees shall report: "Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being:	Licensees shall report: "Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or that resulted in the nuclear power plant being:
a. In an unanalyzed condition that significantly compromises plant safety;	a. In an unanalyzed condition that significantly compromised plant safety;
b. In a condition that is outside the design basis of the plant; or	b. In a condition that was outside the design basis of the plant; or
c. In a condition not covered by the plant's operating and emergency procedures."	c. In a condition not covered by the plant's operating and emergency procedures."

Note: This reporting criteria shall be reviewed for applicability to any plant condition that caused entry into Tech Spec 3.0.3.

1. Definitions

   a. Principal Safety Barriers: The principal safety barriers involve the functionally controlling or bounding accident and transient analysis barriers:

      - Fuel Cladding
      - Reactor Coolant System (RCS) Pressure Boundary
      - Primary and Secondary (MNS/CNS Annulus) Containment

   b. Serious Degradation: Serious degradation is plant degradation beyond that analyzed for in the UFSAR.

   c. Unanalyzed Condition: The current UFSAR transient and accident analyses define the limiting conditions for operation and confirms the ability of the station's systems, structures, and components to prevent or mitigate the consequences of postulated transients and accidents. An event or condition that places the plant outside the bounds of any of these analyses represents an unanalyzed condition.

   d. Significantly Compromises Plant Safety: An unanalyzed condition significantly compromises plant safety if it results in serious degradation, or has the potential to result in serious degradation of one of the principal safety barriers.

   e. Design Basis of the Plant: Rather than referring to the design of individual systems or components, meeting the design basis of the plant means staying within the design basis of the principal safety barriers. The specific safety function of these principal safety barriers is the protection of public health and safety through limiting the release of radioactive material. The controlling parameters for each of the principal safety barriers is contained in the UFSAR. Typical parameters include:

      - Offsite Dose
      - Fuel Clad Temperature

- Hydrogen Generation
- Core Geometry
- Primary Containment Integrity
- Reactor Coolant Pressure Boundary Integrity

The specific value or ranges of values chosen for each controlling parameter along with final verification of principal safety barrier performance is contained in the station's UFSAR.

f. Condition Not Covered by Procedures: This is an event or condition for which there is no existing procedure to prevent a significant compromise to plant safety.

g. During operation: During operation is defined as when the reactor is critical (i.e. the neutron chain reaction is self-sustaining and K[eff] = 1).

2. General

If the event or condition affects more than a single safety system or structure, or one of the principal safety barriers, reportability under this section should be reviewed. The stations are designed and licensed to adequately handle its Design Basis Accident along with its most limiting single failure. If an event occurs or condition exists that results in more equipment or systems being inoperable than covered by the plant's safety analysis, it may be in an unanalyzed condition and outside the design basis of the plant. The definitions provided in 202.7.4.1 for these concepts need to be applied to determine reportability.

It is not intended that this section apply to minor variations in individual parameters, or to problems concerning single pieces of equipment. Any failure, or minor error in performing surveillance tests could produce a situation in which 2 or more often unrelated, safety-related components are out of service. Technically, this is an unanalyzed condition. However, these events should be reported only if they involve functionally related components or if they significantly compromise plant safety. For instance, if an event occurred where there could have been a failure of a safety system to properly complete a safety function, Section "Event or Condition That Alone Could Have Prevented the Fulfillment of Safety Function of Systems or Structures" should be reviewed for reportability. If an event occurred where a single cause actually made a component or group of components inoperable in redundant or independent trains or channels, of one or more systems having a safety function, Section "Common-Mode Failures of Independent Trains or Channels" should be reviewed for reportability.

EXAMPLES

Reportable

a. Three studs were discovered missing on the horizontal missile shields located over the reactor vessel. Engineering analysis determined that the remaining studs were not sufficient to hold the shields during a postulated accident, thus, putting a principal safety barrier in a potentially degraded position.

b. Two weeks after painting the Diesel Generators, an operability test was performed and neither D/G was capable of starting since paint had been applied to key D/G components, preventing any movement. Upon discovery, the D/G components had to be scraped clean before the D/G's were able to start and load. The Unit had been at full power during the entire period.

c. During unit operation, a local leak rate test determined that a containment purge exhaust line penetration was leaking at 0.9La. This made the total Type B and C leakage 1.1La, which is outside the design basis of the analyzed value of leakage for the containment structure (La).

d. Engineering determined that instrument loop inaccuracies could result in safety injection initiation on low pressurizer pressure at a lower RCS pressure than assumed in the accident analysis.

Non-Reportable

e. A main steam isolation valve closed while the plant was at 100% power as a result of a solenoid failure. Operations personnel reduced reactor power because of asymmetric power tilt and feedwater oscillations.

No procedure existed for operating the unit in these conditions while the solenoid was being replaced. The event is not reportable because this condition would not have significantly compromised plant safety.

f. Upon review of historical test data on the 2A and 2B Component Cooling(KC) heat exchangers, both HX's were unknowingly inoperable during the same time period due to excessive fouling. {Note, although not reportable under these criteria, the condition is reportable as a loss of safety function for the KC system under Section 202.7.3 "Event or Condition That Alone Could Have Prevented the Fulfillment of Safety Function of Systems or Structures."}

g. While in Cold Shutdown and mid-loop operation, the 2A Containment Spray (NS) pump suction valve was opened for VST with ND-1 and 2 open. This subjected the 2A NS train to Reactor Coolant pressure conditions. Overpressurization of the NS heat exchanger and a 10,000 gallon spill resulted.

202.7.5 NATURAL PHENOMENON OR CONDITION THREATENING PLANT SAFETY (EXTERNAL THREAT)

§50.72(b)(1)(iii)	§50.73(a)(2)(iii)
Licensees shall report: "Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant."	Licensees shall report: "Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant."

1. General

This section applies only to acts of nature (e.g., tornadoes, earthquakes, fires, hurricanes, floods) and external hazards (e.g., industrial and transportation accidents). This section requires events to be reported if the threat or actual damage challenges the ability of plant personnel to continue to operate in a safe manner, including the orderly shutdown and maintenance of safe shutdown conditions. It is expected, that in the area of external threats, there may be a significantly greater amount of 50.72 notifications than 50.73 LERs.

2. Actual Threat

Judgment should be used to determine if a condition actually threatens the plant. For example, a small brush fire in a remote area of the site that was quickly controlled and did not present a threat to the plant need not be reported. However, a major forest fire or hurricane moving in the direction of the plant and thus threatened plant equipment are reportable. There are no prescribed limits, but in general, situations involving only monitoring by the plant's staff are not reportable. But when preventative actions are taken or if there are serious concerns, then the situation should be carefully reviewed for reportability.

3. Significantly Hampers Personnel

To be reportable, an event need not prevent station personnel from performing their duties. It is only necessary that they be significantly hampered, hindered, or interfered with in the performance of safety-related activities. If the condition makes performing routine safety-related functions significantly more difficult, it is reportable. For example, in a snowstorm, judgment may be based on the amount of snow, the extent to which additional assistance could have been available in an emergency, and the length of time the condition existed. If station management decides to allow all non-essential personnel to go home early as a conservative, precautionary step, considering the safety of the employees during their travel, the condition is not reportable.

EXAMPLES

Reportable

a. The station had been provided detailed hydrological information indicating a flood would occur that would overflow portions of the plant and put the plant into an emergency class. An ENS call is required because a prediction of a flood that is expected to affect the safety of the plant is sufficient cause to initiate emergency preparations.

b. The station made an ENS call when Hurricane Hugo was within 150 miles of the plant and appeared to be heading toward the direction of the plant. Since the force of the hurricane had diminished significantly by the time it neared the station and an insignificant amount of damage was done, an LER is not required.

Non-Reportable

c. One day in February it began snowing considerably. When the accumulation reached 4 inches, station management made a decision to allow non-essential plant personnel to leave early that afternoon since the snow was expected to continue into the night and decreasing temperatures would make traveling home especially hazardous in the evening.

202.7.6 ECCS DISCHARGE INTO THE REACTOR COOLANT SYSTEM

§50.72(b)(1)(iv)	**10 CFR 50.73d**
Licensees shall report: "Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal."	[ECCS discharge is a subset of §50.73(a)(2)(iv), actuation of an engineered safety feature (ESF), as discussed in Section 3.3.2. Therefore, an LER is required.]

1. General

Those events that result in either automatic or manual actuation of the ECCS, or should have resulted in ECCS discharge into the reactor coolant system if some component had not failed or an operator action had not been taken, are reportable. Reporting exceptions include preplanned actuations and the ECCS is properly removed from service and not required to be operable.

2. Valid Signal

Valid signal refers to those signals that are automatically initiated by the measurement of an actual physical system parameter that was within the established set point band of the sensor that provides the signal to the protection system's logic, or manually initiated in response to plant conditions. Valid signals should also include those passive system actuations that occur as a function of system conditions like differential pressure (i.e., cold leg accumulators) whereby no SSPS or other electrical signal is involved. The validity of an ECCS signal may not be determined within 1 hour, ECCS signals that result or should have resulted in injections should be considered valid until firm evidence proves otherwise. Invalid ECCS injections are still considered ESF actuations and therefore require a 4 hour NRC notification (unless a 1 hour notification was made per this section) and LER.

EXAMPLES

Reportable

a. While in Mode 3, valve 2NC-29 stuck open resulting in a rapid decrease in reactor coolant (NC) system pressure. This caused the Cold Leg Accumulators to actuate and inject approximately 1100 gallons of borated water into the reactor coolant system. An ENS call is required to be made within 1 hour and an LER is required because of the ESF actuation.

b. During a reactor vessel pressure test while in Cold Shutdown, a low pressure coolant injection pump (LPCI) automatically started when a reactor recirculation pump start caused a perturbation in reactor vessel level instrumentation readings. Because the reactor vessel pressure was above the LPCI pump shutoff head, no water was injected into the vessel. An ENS call is required because this was a valid ECCS signal that should have resulted in an ECCS discharge into the reactor vessel.

Non-Reportable

c. While surveillance testing containment isolation valves, a test pushbutton was inadvertently released, which initiated a 'B' train containment isolation and safety injection. High pressure ECCS pumps injected 300 gallons of borated water from the RWST into the reactor before pumps were secured, while the reactor remained at 94% power. The event is not reportable as a 1 hour ENS call under this section, even though it was an ECCS injection. The signal that caused the injection was an inadvertent, manual signal (i.e., plant conditions did not require a manual safety injection), thus, not a "valid" signal. The event is reportable, however, as an ESF actuation, and a 4 hour ENS call and a LER is required (ref. Section "Actuation of an Engineered Safety Feature or the Reactor Protection System").

202.7.7 LOSS OF EMERGENCY ASSESSMENT, RESPONSE, OR COMMUNICATIONS

§50.72(b)(1)(v)	10 CFR 50.73
Licensees shall report: "Any event that results in a major loss of emergency assessment capability, or communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system)."	[No corresponding Part 50.73 requirement.]

1. Loss of Emergency Assessment Capability

   Emergency assessment capability is defined in the station Emergency Plan and implementing procedures. A major loss of emergency assessment capability would include those events or conditions that significantly impair the operators ability to determine the status of the key station parameters and take the proper course of actions in the event of an emergency. Engineering judgment may be needed to determine the significance of the loss in terms of the equipment and the length of time involved. For example, the unavailability of 1 redundant component or train such as a radiation monitor or OAC, for a period of time as permitted by Tech Specs or administrative procedures, generally is not reportable.

2. Loss of Offsite Response Capability

   A major loss of offsite response capability includes those events that would significantly impair the fulfillment of the station's Emergency Plan. Loss of offsite response capability may typically include the loss of plant access, emergency offsite response facilities, or public prompt notification system (a loss of more than 25% of the station's total sirens (for more than 1 hour) would be considered a major loss and, therefore, reportable per this section).

3. Loss of Communications Capability

   A major loss of communications capability (for more than 1 hour) would include the loss of the ENS, or Selective Signal phone and commercial telephone lines. If the NRC Headquarters Operations Officer notifies the station of an inoperable ENS line, that discussion constitutes the required ENS notification and no further notification is necessary.

EXAMPLES

Reportable

a. More than 25% of the stations' total alert sirens were disabled for more than 1 hour because of loss of power as a result of severe weather.

b. ENS phone line was discovered to have been cut while crews were digging.

c. The local sheriff notified the station that all roads to and from the plant were closed because of a heavy snow storm. The station had 2 full shift crews on site to support plant operations and no emergency declaration was made. An ENS call is required because the road closing may prevent the plant staff from adequately staffing the TSC, or from fully responding to some emergencies.

Non-Reportable

d. Because of some major work being performed on the Emergency Notification System, the NRC Headquarters Operations Officer (HOO) notified the station of an inoperable ENS line. No separate 50.72 notification by the station is necessary since the discussion with the HOO constitutes the required ENS notification.

e. It was observed during siren testing that 5 of 52 alert sirens around the EPZ failed to function. This was not considered to be a major loss of the offsite response capability.

f. York County was performing a scheduled quarterly full cycle siren test and as they were performing the procedure there was a step requiring the turning of a key in order to make the sirens sound. The sirens did not sound; however, within minutes the individual realized an improper arming configuration, rearmed the siren, turned the key, and the sirens function properly. This event would not be reportable because the sirens were not disabled for more than 1 hour.

202.7.8 INTERNAL THREAT TO PLANT SAFETY

§50.72(b)(1)(vi)	§50.73(a)(2)(x)
Licensees shall report: "Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases."	Licensees shall report: "Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases."

1. General

This section pertains to threats internal to the station. Fires, toxic gas releases, and radioactive releases are not the only threats that may require reporting under these provisions. The criterion to be applied in each case is whether the event poses an actual threat to the safety of the plant or significantly hampers personnel in the performance of duties necessary for the safe operation of the plant. The significant hampering criterion is pertinent to "the performance of duties necessary for safe operation of the nuclear power plant." One way to evaluate this is to ask if one could seal the room in question (or disable the function in question) for a substantial period of time and still operate the plant safely. Actions such as room evacuations that are purely precautionary would not constitute significant hampering if the performance of duties necessary for the safe operation of the plant can still be performed in a timely manner. Refer to Section 202.7.5, "Natural Phenomenon or Condition Threatening Plant Safety (External Threat)" of this directive for additional discussion on "actual threats" and "significantly hampering personnel".

EXAMPLES

Reportable

a. The station reported a fire in the main generator excitor housing. The reactor was manually tripped and taken to Cold Shutdown. The fire brigade quickly and successfully extinguished the fire, and no offsite fire-fighting assistance was required. A 1 hour ENS call is required because the fire threatened the safety of the nuclear power plant.

b. A turbine building evacuation was ordered when a large area of the floor was contaminated. Condensate demineralizer resin was being transferred through a cleaner to a mix-and-hold tank. As the tank was being pressurized, a mispositioned inlet valve allowed 50 to 100 gallons of water/resin to blow out into the turbine building. The ventilation system spread loose surface contamination through various turbine building locations. Eight operators and construction workers were contaminated.

   An ENS call is required because plant operators were significantly hampered in the performance of their duties because they were evacuated from areas containing safety-related equipment and would have been delayed in their duties during an emergency.

Non-Reportable

c. A small hydrazine leak occurred in the yard as a result of transporting a drum that was inadvertently punctured. An ENS notification is not required under this reporting section since the toxic gas leak posed no threat to the safety of the plant, nor did it significantly hamper personnel in the performance of their duties necessary for plant safety. However, depending on the circumstances, this event may be reportable under other 50.72 criteria such as Section 202.8.7, "News Release or Other Government Notifications" (notification to outside agencies) if not an emergency class declaration.

202.8 4-HOUR ENS NOTIFICATIONS AND LERS

This section addresses 50.72(b)(2) 4-hour notifications for non-emergency events and the associated LER. If not reported as a declaration of an emergency class under 50.72(a) or as a non-emergency 1 hour report under 50.72(b)(1), the station is required to notify the NRC as soon as practical and in all cases within 4 hours of the discovery of any of the events specified.

In addition to similar reporting criteria under both 10CFR 50.72 and 50.73, several requirements for only 50.72 notifications or only LERs are included in this section because of the sequential numbering scheme used.

202.8.1 SHUTDOWN PLANT FOUND IN DEGRADED OR UNANALYZED CONDITION

§50.72(b)(2)(i)	10 CFR 50.73
Licensees shall report: "Any event found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety."	[Events found while the reactor is shutdown that involve degradation of the principal safety barriers or unanalyzed conditions that significantly compromise plant safety are addressed by §50.73(a)(2)(ii). Therefore, an LER is required. See Section 3.2.4.]

1. General

   As previously indicated in Section 202.7.4, "Operating Plant in a Degraded or Unanalyzed Condition, or Outside Design Basis," similarities exist between 50.72(b)(1)(ii) reporting and this section for degraded or unanalyzed plant conditions. The difference in the reporting time frame (1 hour vs. 4 hours) is warranted since

this section pertains to events found while the reactor is shutdown and Section "Operating Plant in a Degraded or Unanalyzed Condition, or Outside Design Basis" applies to events or conditions occurring while the plant is in operation.

Guidelines for reporting under 50.72(b)(2)(i) above are the same as provided in Section 202.7.4 "Operating Plant in a Degraded or Unanalyzed Condition, or Outside Design Basis" of this directive. Any condition that is discovered while the unit is shutdown, that existed previously while the unit was in operation should be evaluated for reportability under this criterion. For example, Steam Generator Tube plugging in accordance with Tech Spec 5.5.9, Table 5.5-2 (CNS only) would be reportable per this section. It is sometimes difficult to determine how long a degraded condition of a system or component has existed once the station is in an outage and several systems are out of service. For these cases, the same philosophy for determining if a condition prohibited by Tech Specs exist, should be applied (see Section 202.7.2, "Technical Specification Prohibited Operation or Condition").

EXAMPLES

Reportable

a. With the unit in Mode 6, ultrasonic testing revealed a number of failed fuel rods (233 were identified in 88 of 109 fuel assemblies scheduled for reinsertion) that far exceed the anticipated number of failures. An ENS call is required because a principal safety barrier (fuel cladding) was found seriously degraded.

b. Steam Generator Tube plugging in accordance with Tech Spec 5.5.9, Table 5.5-2 (CNS only) would require an ENS notification per this section.

{Examples under this section are similar to those in Section 202.7.4, "Operating Plant in a Degraded or Unanalyzed Condition, or Outside Design Basis," except these conditions are discovered while the unit is shutdown}.

202.8.2 ACTUATION OF AN ENGINEERED SAFETY FEATURE OR THE REACTOR PROTECTION SYSTEM

§50.72(b)(2)(ii)	§50.73(a)(2)(iv)
Licensees shall report "any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that results from and is part of the preplanned sequence during testing or reactor operation need not be reported."	Licensees shall report "any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported."

1. Definitions

a. Engineered Safety Feature (ESF): Engineered Safety Features are the provisions in the plant which serve to: (1) control reactor fission products which may leak from the fuel by assuring their retention in the Reactor Coolant System (RCS), (2) control and limit the consequences of energy and radioactivity within the containment, and (3) provide adequate cooling of the core under all circumstances. Those ESF systems specific to each station are listed in Appendix A.

b. ESF/RPS Actuation: (1) Receipt of a Solid State Protection System (SSPS) signal(s) necessary to activate the ESF/RPS system, or (2) manual or automatic actions that activate the ESF/RPS system without the presence of an SSPS signal(s).

c. Preplanned Actuation: A preplanned ESF actuation is the initiation of a particular ESF as called for by an approved operating or testing procedure.

d. Properly Removed From Service: The component or system is intentionally mechanically or electrically disabled such that is not capable of performing its intended safety function, and station procedures for removing equipment from service have been implemented (e.g., required clearance documentation, equipment and control board tagging, etc.).

2. Reportability

All ESF actuations, including actuations of the RPS, are reportable regardless of the plant operating mode or the significance of the structure, system, or component that initiated the event or whether initiated manually or automatically. The fact that the safety analysis assumes that an ESF system will actuate automatically under certain plant conditions does not preclude the need to report such actuations.

ESFs are provided to mitigate the consequences of a significant event and, therefore:

a. they should work properly when called upon, and

b. they should not be challenged frequently or unnecessarily.

The NRC is interested both in events where an ESF was needed to mitigate the consequences (whether or not the equipment performed properly) and events where an ESF operated unnecessarily. Generally, the NRC would not consider this to include single component actuations because single components of complex systems, by themselves, usually do not mitigate the consequences of significant events. However, in some cases a component would be sufficient to mitigate the event (i.e., perform the ESF function) and its actuation would then be reportable.

Since single trains do mitigate the consequences of significant events, train level actuations are reportable. In this regard, actuation of a diesel-generator is considered to be an actuation of a train and not an actuation of a single component because a diesel generator is needed to mitigate the event (performs the ESF function).

The ECCS contains systems which have no other operating function as well as systems which are shared with other systems. Actuations of ECCS systems which are shared with other systems is reportable only when they are performing their ESF function.

3. Reporting Exceptions

Actuations that need not be reported are those initiated for reasons other than to mitigate the consequences of an event (e.g., preplanned actuations, actuations that occur after the safety function has already been completed and ESFs that have been properly removed from service (i.e., plant procedures for removing equipment from service have been implemented) and not required to be operable). However, if the ESF actuates during the planned operation or test in a way that is not part of the planned procedure, such as at the wrong step, that event is reportable.

Invalid Actuations of certain specified systems are not reportable. These systems are limited to:

- control room emergency ventilation system
- reactor building ventilation system
- fuel building ventilation system
- auxiliary building ventilation system

EXAMPLES

Reportable

Note: {For the reportable examples provided, assume the actuation is not part of a pre-planned sequence in a procedure and the system has not been removed from service.} This note applies to examples a-k.

a. Any manual or automatic actuation of the reactor trip switchgear is reportable.

b. Initiation of a containment isolation signal constitutes an ESF actuation whether or not the containment isolation valve actually repositions.

c. The opening of a Hydrogen Skimmer fan header isolation valve and the subsequent starting of a Hydrogen Skimmer fan is an ESF actuation.

d. The starting or speed change of a Reactor Building Cooling Unit fan, as a result of a valid or spurious ES Channel 5 or 6 signal, is reportable. (ONS)

e. The starting of any of the ECCS pumps to mitigate the consequences of a significant event is an ESF activation.

f. The automatic start of a train of Control Room Ventilation from a valid signal constitutes an ESF actuation. (MNS and CNS)

g. Any manual or automatic actuation of the Auxiliary Feedwater(CA) system is reportable. (MNS and CNS)

h. Unplanned Diesel Generator starts, and Keowee starts resulting from ES Channel 1 or 2 signals, are reportable.

i. Emergency power switching logic actuations of 4160V breakers which result from ES 1 or 2 signals [ONS] reportable.

j. The operation of Auxiliary Building ventilation in the filtered mode is an ESF function.

k. During a significant operational transient, an "ice condenser door open" alarm was received in the Control Room. This is a reportable event because if the Ice Condenser doors are off their seals, the equipment is considered actuated.

Non-Reportable

a. Swaps of Nuclear Service Water pump's suction from the lake to the Standby Nuclear Service Water pond is not reportable.

b. Equipment actuation because of a signal generated by EMF's (radiation monitors) is not considered to be an ESF actuation and therefore, is not reportable.

c. RPS actuates after all control rods and banks have already been inserted in the core.

d. During surveillance testing of the main steam isolation valves (MSIVs), an operator incorrectly closed MSIV "D" when the procedure specified closing MSIV "C". This event is not reportable because the event is an inadvertent actuation of a component of an ESF system.

e. Movement of a single ESF valve swapped the suction of the Nuclear Service Water System to the Auxiliary Feedwater pump suction. Since only a single component was actuated and the valve could not mitigate the consequences of an event by itself, the ESF valve movement is not reportable as an ESF actuation.

f. Actuations of the Emergency Feedwater system at the Oconee Nuclear Station are NOT reportable.

202.8.3 EVENT OR CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF SAFETY FUNCTION OF SYSTEMS OR STRUCTURES

§50.72(b)(2)(iii)	§50.73(a)(2)(v)
Licensees shall report: "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: a. Shut down the reactor and maintain it in a safe shutdown condition; b. Remove residual heat; c. Control the release of radioactive material; or d. Mitigate the consequences of an accident." 10 CFR 50.72 [The Statements of Consideration for 10 CFR 50.72 contain wording similar to those of 50.73(a)(2)(vi). §50.73(a)(2)(vi).]	Licensees shall report: "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: a. Shut down the reactor and maintain it in a safe shutdown condition; b. Remove residual heat; c. Control the release of radioactive material; or d. Mitigate the consequences of an accident." §50.73(a)(2)(vi) "Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if redundant equipment in the same system was operable and available to perform the required safety function".

Note: **This reporting criteria shall be reviewed for applicability to any plant condition that caused entry into Tech Spec 3.0.3.**

1. General

   The intent of this section is to capture those events where there could have been a failure of a safety system to properly complete a safety function, regardless of when the failures were discovered or whether the system was needed at the time. The event must be reported regardless of the situation or condition that caused the system to be unavailable, and regardless of whether or not an alternate safety system could have been used to perform the safety function.

   The applicability of this section includes those safety systems designed to mitigate the consequences of an accident (e.g., containment isolation). Hence, minor operational events involving a specific component such as valve packing leaks, which could be considered a lack of control of radioactive material, should not be reported under this section.

2. Alone Could Have Prevented

   The phrase "alone could have prevented" means the event or condition was, or would be, sufficient by itself to prevent the performance of the safety function of a system or structure (i.e., no additional single failure is assumed or needed to prevent the function).

3. Single Train/Common-Mode Failure

   These reporting criteria are not meant to require reporting of a single, independent component failure that makes only one functionally redundant train inoperable. The following conditions, however, are reportable:

   - an actual single event or condition that disabled multiple trains of a safety-related system

- an actual event or condition that disabled one train of a safety-related system and could have affected a redundant train
- a condition or potential single event that could have disabled multiple trains of a safety-related system

Engineering judgement should be used when these criteria are applied to those few systems with more than 2 redundant trains (e.g., MNS/CNS CA system).

4. Non-Reportable Events or Conditions
   - failures that affect inputs or services to systems that have no safety function
   - defective component(s) that has not been installed
   - unrelated component failures in several different safety systems
   - a single stuck control rod that alone would not have prevented the fulfillment of a reactor shutdown

OTHER EXAMPLES

Reportable

a. During a refueling outage, the equipment hatch was discovered open 1/4" after containment integrity had been established.

b. It was determined that when the Control Room smoke purge exhaust fan (SPE) is running, the Control Room Ventilation (VC) system cannot perform its safety function. The SPE fan has been operated numerous times in the past without adequate comp measures to ensure this fan does not operate when the OAPFT'start. Therefore, there have been several occasions in the past where both trains of the VC system are considered to have been inoperable.

c. Lower Annulus Ventilation (VE) doors on Unit 1 were opened for painting without proper COMP measures. Since these doors are common to both VE trains, Tech Spec 3.0.3 was entered.

d. While train 'A' VC/YC was inoperable due to maintenance, the 'B' train YC chiller tripped and could not be restarted. Tech Spec 3.0.3 was entered for 90 minutes because both trains of the VC system were inoperable. There was no load reduction since operators felt that 1 of the trains would be back in service within 2 hours. This event is reportable as a loss of safety function and since Tech Spec 3.0.3 was entered, it is also reportable as a condition prohibited by Tech Specs.

Non-Reportable

e. While performing a main steam line Pressure Instrument Functional Test and Calibration, a switch was found to actuate at 853 psig. The Tech Spec limit is 825 + 15 psig head correction. The redundant switches were operable. The cause of the occurrence was setpoint drift. The switch was recalibrated, tested successfully per procedure and returned to service. The event is not reportable due to the drift of a single pressure switch unless it alone could have caused a system to fail to fulfill its safety function.

202.8.4 COMMON-MODE FAILURES OF INDEPENDENT TRAINS OR CHANNELS

10 CFR 50.72	§50.73(a)(2)(vii)
[No corresponding Part 50.72 requirement.]	Licensees shall report: "Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to: a. Shut down the reactor and maintain it in a safe shutdown condition; b. Remove residual heat; c. Control the release of radioactive material; or d. d. Mitigate the consequences of an accident."

1. General

This section requires those events to be reported where a single cause made a component or group of components to become inoperable in redundant or independent trains or channels, of one or more systems having a safety function (common-mode failures). Failures reported under this part of the rule should be <u>actual failures</u>, not potential ones.

Such failures can be simultaneous which occur from a single initiating cause, or sequential (i.e., cascade failures), such as the case where a single component failure results in the failure of one or more additional components.

To be reportable, however, the event or failure must result in or involve the failure of independent portions of more than one train or channel in the same or different systems. For example, if a single cause or condition resulted in inoperable components in Train "A" of the KC System and Train "B" of the Nuclear Service Water (RN) system (i.e., train that is assumed in the safety analysis to be independent) the event is reportable. Additionally, one function of the "B" train of the RN system is to provide cooling for the "B" train of the KC System, and since "B" train of the RN system cannot perform its cooling function, then "B" train of the KC system is also inoperable. Thus, both trains of the KC system are inoperable and unable to perform their safety function.

EXAMPLES

<u>Reportable</u>

a. Events reportable under Section 202.8.3, "Event or Condition That Alone Could Have Prevented the Fulfillment of Safety Function of Systems or Structures" of this directive are also reportable per this section provided: (1) the system involved has 2 or more trains or channels, and (2) the inoperable condition is as a result of "actual" failures.

b. The station found 11 inoperable snubbers during periodic testing. All the snubbers failed to lock up when required. These failures rendered trains in 3 systems inoperable. This condition is reportable because the condition indicated a generic common-mode problem that caused numerous multiple independent trains in one or more safety systems to become inoperable.

Non-Reportable

c. Design investigation indicated that electrical power feed to the VE filter train heaters can be postulated to drop to a sustained voltage that would place power dissipation outside the required range. Both trains of VE were considered inoperable. This condition is not reportable under this section because the condition was not an actual failure of both trains, but a postulated event that "could have" prevented the fulfillment of the safety function of the VE system, and is reportable under Section 202.8.3, "Event or Condition That Alone Could Have Prevented the Fulfillment of Safety Function of Systems or Structures," ref. Example 2.

202.8.5 AIRBORNE OR LIQUID EFFLUENT RELEASE EXCEEDING 20 TIMES APPENDIX B

§50.72(b)(2)(iv)	50.73(a)(2)(viii)
Licensees shall report: a. "Any airborne radioactive release that exceeds 20 times the applicable concentrations of the limits specified in Appendix B, Table 2 of Part 20 of this chapter in unrestricted areas, when averaged over a time period of one hour." b. "Any liquid effluent release that exceeds 20 times the applicable concentrations of the limits specified in Appendix B Table 2 of Part 20 of this chapter at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour." ("Immediate notifications made under this paragraph [§50.72(b)(2)(iv)] also satisfy the requirements of paragraphs (a)(2) and (b)(2) of §20.2202 of Part 20 of this chapter".)	Licensees shall report: a. "Any airborne radioactivity release that exceeded 20 times the applicable concentrations of the limits specified in Appendix B, Table 2 of Part 20 of this chapter in unrestricted areas, when averaged over a time period of one hour." b. "Any liquid effluent release that exceeded 20 times applicable concentrations of the limits specified in Appendix B Table 2 of Part 20 of this chapter at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour." §50.73(a)(2)(ix) "Reports submitted to the Commission in accordance with paragraph (a)(2)(viii) of this section also meet the effluent release reporting requirements of paragraph 20.2203(a)(3) of Part 20 of this chapter".

1. General

This section is similar to Parts 20.2202 and 20.2203, but places a lower threshold for reporting events at commercial power reactors. The lower threshold is based on the significance of the breakdown of the station's program necessary to have a release of this size, rather than on the significance of the impact of the actual release.

For a release that takes less than 1 hour, normalize the release to 1 hour (e.g., release of 15 minutes, multiply by 4). For releases that last more than 1 hour, use the highest release for any continuous 60 minute period. It often takes a period of time to assess the magnitude of a radioactive release. If preliminary estimates determine that the release has exceeded the reporting criterion, an ENS notification is required, followed up by a more precise estimate in the LER. If it is determined later than the release was less than this criterion, the ENS notification should be retracted.

EXAMPLE

Reportable

a. During routine maintenance on a pressure actuated valve in the waste gas system, an unplanned radioactive release to the environment was detected by a radiation alarm. The release occurred when an isolation valve, required to be closed, was inadvertently left open. This allowed radioactive gas from the waste gas decay tank to escape through a pressure gage connection that had been opened to vent the system. The concentration at the site boundary, averaged over 1 hour, was estimated by the station to exceed the limits specified in Appendix B of Part 20.

202.8.6 CONTAMINATED PERSON REQUIRING TRANSPORT TO OFFSITE MEDICAL FACILITY

§50.72(b)(2)(v)	10 CFR 50.73
Licensees shall report: "Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment."	[No corresponding Part 50.73 requirement.]

1. General

Contaminated, in this case, refers to either contaminated clothing, the person, or both. If the initial onsite survey is incomplete and there is a potential for contamination, the station should assume the individual is contaminated and make the ENS notification. Often the full extent of radioactive contamination on an injured individual may not be known until after arrival at the hospital. If no potential for contamination is present, reporting of the transport to offsite medical facilities is not required.

EXAMPLES

Reportable

a. A contract worker experienced a back injury lifting a tool while working in the reactor building and was considered to be potentially contaminated because his back could not be surveyed. An ENS call was made immediately. The individual was later found not to be contaminated and an update ENS notification was made.

Non-Reportable

b. The station transported a high school student from its PAP to a medical office because the student had stomach pains. This event is not reportable because no potential for contamination was present.

c. A CMD employee cut his head in the containment pipe chase. RP reported that the individual was not contaminated but was being transported to the hospital. The event is not reportable because no potential for contamination was present.

202.8.7 NEWS RELEASE OR OTHER GOVERNMENT NOTIFICATIONS

§50.72(b)(2)(vi)	10 CFR 50.73
Licensees shall report: "Any event or situation, related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an on-site fatality or inadvertent release of radioactively contaminated materials."	[No corresponding Part 50.73 requirement.]

1. General

The purpose of this section is to ensure the NRC is made aware of issues that will cause heightened public or government concern related to radiological or environment events. The NRC Operations Center does not need to be made aware of every press release or offsite notification made. Only those issues that are perceived by the public to be related to the radiological health and safety of the public, onsite personnel, or protection of the environment, need be reported. When in doubt, the ENS notification should be made by the station.

Generally, the following types of events require a report under 50.72:

- onsite plant or animal disease outbreaks
- inadvertent release of radioactively contaminated materials to public areas
- fish kills
- inadvertent releases of radioactivity
- unanticipated non-radioactive releases\spills that would generate interest from local government agencies or the EPA
- mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973
- inadvertent public notification system operation for which a news release is planned
- excessive bird impaction events
- Release of a Reportable Quantity (RQ) of a Superfund Amendment Reauthorization Act (SARA) extremely hazardous substance
- increase in nuisance organisms or conditions casually related to station operation

The NRC does not generally need to be informed under this section of:

- administrative matters such as company management changes, SALP ratings, civil penalties, or media inquiries
- minor deviations from permitted effluent limits
- routine reports of effluent releases to other agencies
- minor onsite chemical spills that would not generate interest from local agencies or the EPA
- peaceful strikes or civil demonstrations
- exceedance of groundwater monitoring well parameter

For the following events, a report under 50.72 is left to the discretion of the Environmental Compliance Manager; however, the resident inspector should be informed:

- underground storage tank(ust) or ust piping leak
- brown scum on the waters of the state
- release of a dye to waters of the state
- groundwater contamination
- any drinking water maximum contaminate level violation for which posting is required

The 4-hour ENS notification clock starts at the time the decision is made to report to other agencies or to make a press release. In some cases, a decision to issue a press release may not be made until long after the specific event. In all cases, however, the notification to the NRC should be made <u>before</u> the press release is issued to the media.

Notifications to other Federal agencies does not relieve the station of the requirement to report to the NRC Operations Center via ENS. Likewise, if current procedures require reporting of events to other areas within the NRC, such as Region II, this too does not fulfill the reporting requirement of 50.72. [Note: If an event is significant enough to warrant reporting to the Region, as required by current procedures, this in itself meets the reportability threshold under 50.72. However, informal notifications to the Resident Inspector for awareness are not reportable per this criterion.]

OTHER EXAMPLES

<u>Reportable</u>

a. A man fell into the discharge canal while fishing and failed to resurface. The station notified the sheriff, state police, and state emergency agencies. The local media was granted onsite access to cover the event. An ENS call is required because of the fatality onsite, the other notifications made, and the media involvement.

b. The station informed the county government and other organizations of a spurious actuation of several alert sirens in a county. The station also planned a press release. An ENS notification is required within 4 hours of the initial contact with any county agency regarding the inadvertent actuation of part of the public notification system.

c. The station transported 2 secondary side filters to the county dump as non-radioactive waste, but later determined that they were contaminated. The station notified appropriate state agency and NRC resident inspector. An ENS call is required.

d. The station notified its state environmental protection agency and the NRC resident of a fish kill involving several species in the circulating water discharge canal, possibly resulting from thermal water conditions. An ENS call is required because of the state notification of a significant fish kill, which the media or public could perceive as an environmental or public hazard.

e. Oil spills to waters of the state require an ENS call.

f. A spill of ≥ 1 pound (≥ 50 ppm) of PCB's to the environment or any impervious surface requires an ENS call.

<u>Non-Reportable</u>

g. The station notified the state, EPA, and Dept. of Transportation that 5 gallons of diesel fuel oil had spilled onto gravel covered ground inside the protected area. The spill was cleaned up by removing the gravel and dirt. Such notifications to other agencies such as this do not require an ENS notification. These kind of events do not pertain to the radiological health and safety of the public, or protection of the environment.

h. As a result of a local newspaper article regarding the findings of an NRC regional inspection, a station representative was interviewed on local television and radio stations. The station also notified State officials and the NRC resident. An ENS call is not required in this case because the station was responding to findings raising by the NRC.

i. Notification of when a hazardous waste manifest is returned from an out-of-state facility does not require an ENS call.

j. A hazardous waste manifest not returned within the 45 day limit does not require an ENS call.

k. A licensee notified the U.S. EPA that the circulation water temperature rise exceeded the release permit allowable. This event was caused by the unexpected loss of a circulation water pump while operating at 92 percent power. The licensee reduced power to 73 percent so that the circulating water temperature would decrease to within the allowable limits until the pump could be repaired. An ENS notification is not needed because this event is routine and has little safety significance.

202.8.8 SPENT FUEL STORAGE CASK NOTIFICATIONS

§50.72(b)(2)(vii)	10 CFR 50.73
Licensees shall report: "Any instance of: a. A defect in any spent fuel storage cask structure, system, or component which is important to safety; or b. A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use of the storage cask under a general license issued under §72.210 of this chapter. A followup written report is required by §72.216(b) of this chapter including a description of the means employed to repair any defects or damage and prevent recurrence, using instructions in §72.4, within 30 days of the report submitted in paragraph (a). A copy of the written report must be sent to the administrator of the appropriate Nuclear Regulatory Commission regional office shown in Appendix D to Part 20 of this chapter."	[No corresponding Part 50.73 requirement.]

1. General

This information is necessary to inform the NRC of potential hazards to the public health and safety relating to spent fuel storage casks. The term "defect" as defined in Part 21, may also be applied to this section. If the defect is evaluated and reported per this section, then as indicated in Section 202.6.1, "Part 21 Reporting" of this directive, the evaluation and notification obligations of Part 21 have been met.

{No Reporting Examples available for this section}.

202.9 FOLLOWUP NOTIFICATION

10CFR 50.72(c), "Followup Notification", is in addition to making the required initial ENS notification under 50.72(a) or (b). Reporting under this section is intended to provide the NRC with timely notification when an event becomes more serious or additional information or new analysis clarify the event.

It is important that the station record the NRC 50.72 Report number in the appropriate procedure for the initial ENS phone call, so when notifications are made per this section, the station can provide the NRC the proper report number. Any new information to be given will be recorded as such on the NRC's original 50.72 report as an update.

The followup notification is required for data or analysis results that clarify the plant conditions. Anytime a determination is made that a followup notification is required under 10CFR 50.72c, a formal notification shall be made using the ENS phone. Notification to the NRC Resident, other NRC representatives on site, or informally communicating on the open ENS line during an event is not a substitute for a 50.72 notification.

Since this criterion primarily deals with changes in plant status or analyses associated with emergency events, no discussion on the specific parts of the rule will be included in this directive, since current Emergency Plan implementing procedures provide adequate guidance (as stated in the Purpose of this directive).

202.10 OTHER EVENTS REQUIRING "IMMEDIATE NOTIFICATION"

This section addresses immediate notification requirements for sections other than 50.72. The station is required to notify the NRC as soon as practical and in all cases within 1 hour of the occurrence of any of the events specified. There are no examples available for these reporting sections.

10CFR 20.2202a

Each Licensee shall immediately report any incident involving byproduct, source or special nuclear material which may have caused or threatens to cause the following:

1. Individual Exposure
   Greater than or equal to 25 Rem total effective dose equivalent (TEDE)

   or

   Greater than or equal to 75 Rem eye dose equivalent (EDE)

   or

   Greater than or equal to 250 Rads shallow dose equivalent to the skin or extremities (SDE)

2. Release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot cells or process enclosures).

10CFR20.1906(d)(1) and (d)(2)

Notification to the NRC Regional Office, Region II, Atlanta, GA. following receipt of a package of radioactive materials where:

Removable radioactive surface contamination exceeds the limits of 10 CFR 71.87 (i)

or

External radiation levels exceed the limits of 10 CFR 71.47

Loss of one working week or more of the operation of any unit

Damage to property in excess of $200,000

Tech Spec Safety Limit Violation

The station is required to notify the NRC as soon as practical and in all cases within 1 hour of the occurrence of a Safety Limit violation. A follow-up written report is to be submitted within 30 days of the event.

Fire Protection Program

The Facility Operating Licenses require that Duke notify the NRC as soon as practical and in all cases within 24 hours of the occurrence of any violations of the Fire Protection program (refer to license condition for additional reporting requirements). In general, only programmatic breakdowns of the Fire Protection Program need to be reported. Programmatic breakdowns do not include failures to execute the program. Individual problems with Fire Protection Program Remedial Actions (e.g., fire watches) or fire protection/detection equipment should be evaluated with respect to actual cause. If the cause is related to a failure to put in place an element or elements of the Fire Protection Program, then this would be reportable. If however, the problem is related to isolated failures to execute due to human performance problems or isolated equipment failures, these conditions would not be reportable as programmatic breakdowns.

202.11 FOR OCONEE NUCLEAR STATION ONLY

Reporting Requirements

Independent Spent Fuel Storage Installation (ISFSI)

10CFR72.75 (a) Emergency Notifications

Adequate guidance currently exist in Oconee's Emergency Plan's implementing response procedures for emergency events and their classifications. For an ISFSI that is located on the site of a nuclear power reactor licensed for operation by the Commission, the emergency plan required by 10CFR50.47 shall be deemed to satisfy the requirements of this section.

The sections that follow address guidelines for reporting four and twenty-four hour notifications for non-emergency events and the associated event report. There are no examples available for these sections.

10CFR72.75 (b) Four hour reports

The station is required to notify the NRC as soon as practical and in all cases within 4 hours of the occurrence of any of the following events or conditions involving spent fuel or high-level radioactive waste:

1. An event that prevents immediate actions necessary to avoid exposures to radiation or radioactive materials that could exceed regulatory limits, or releases of radioactive materials that could exceed regulatory limits (e.g., events such as fires, explosions, and toxic gas releases).
2. A defect in any spent fuel storage structure, system, or component which is important to safety.
3. A significant reduction in the effectiveness of any spent fuel storage confinement system during use.
4. An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under this part when the action is immediately needed to protect the public health and safety and no action consistent with license or certificate of compliance conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent.
5. An event that requires unplanned medical treatment at an offsite medical facility of an individual with radioactive contamination on the individual's clothing or body which could cause further radioactive contamination.
6. An unplanned fire or explosion damaging any spent fuel or HLW, or any device, container, or equipment containing spent fuel or HLW when the damage affects the integrity of the material or its container.

10CFR72.75 (c) Twenty-four hour reports

The station is required to notify the NRC as soon as practical and in all cases within 24 hours of the occurrence of any of the following events or conditions involving spent fuel or high-level radioactive waste:

1. Any unplanned contamination event that requires access to the contaminated area by workers or the public to be restricted for more than 24 hours by imposing additional radiological controls or by prohibiting entry into the area.

2. An event in which safety equipment is disabled or fails to function as designed when:

   a. The equipment is required by regulation, license condition, or certificate of compliance to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposures to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident; and

   b. No redundant equipment was available and operable to perform the required safety function.

10CFR72.75(d)(2) Written report

The station is required to submit a written followup report within 30 days of an initial report required by paragraph (a) or (b) of this section. Written reports prepared pursuant to other regulations may be submitted to fulfill this requirement if the reports contain all of the necessary information and the appropriate distribution is made.

202.12 APPENDIX

Appendix A, "202. Engineered Safety Features"

VERIFY HARD COPY AGAINST WEB SITE IMMEDIATELY PRIOR TO EACH USE

APPENDIX A. 202. ENGINEERED SAFETY FEATURES

Engineered Safety Feature	CNS	MNS	ONS
1. Containment Isolation Systems			
a. Phase A	X	X	X
b. Phase B	X	X	X
c. NW	X		
2. Containment Heat Removal			
a. Ice Condenser	X	X	
b. Air Return Fans	X	X	
c. Containment/Reactor Building Spray	X	X	X
d. Reactor Building Cooling Units			X
3. Secondary Containment			
a. Annulus Ventilation	X	X	
4. Combustible Gas Control in Containment	X	X	
a. Hydrogen Recombiners	X	X	
b. Air Return and Skimmer Fans	X	X	
c. Hydrogen Purge	X	X	
d. Hydrogen Igniters	X	X	
5. Emergency Core Cooling System			
a. NV/HPI	X	X	X
b. NI	X	X	
c. ND/LPI	X	X	X
d. CLA/CFT	X	X	X
e. FWST/BWST	X	X	X
1) Containment Sump Swapover	X	X	
6. Habitability Systems			
a. Control Room Ventilation (S[s] or Blackout Signal)	X	X	
7. 7. ESF Filter Systems			
a. Auxiliary Building Filtered Exhaust (S[s] or Blackout Signal)	X	X	
b. Penetration Room Ventilation			X
8. Auxiliary Feedwater System	X	X	
9. Diesel Generator starts	X	X	
10. Keowee starts (see Section 202.7, example "h")			X
11. Reactor Protection System	X	X	X

Engineered Safety Feature	CNS	MNS	ONS
12. Turbine Trip per Tech Sec Table 3.3.2-1	X	X	
13. Steam Line Isolation	X	X	
14. Feed Water Isolation	X	X	
15. 4KV Undervoltage	X	X	

BASIS/BACKGROUND FOR ENGINEERED SAFETY FEATURES AND ASSOCIATED SYSTEMS

The American Nuclear Society published ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, which defined ESFs as features that "serve to control and limit the consequences of releases of energy and radioactivity" inside the Containment. The ESF systems were considered to be: (1) the containment; (2) containment cooling; and (4) containment air cleanup systems.

At about the same time that the ANSI N18.2 document was published, Westinghouse developed a Reference Safety Analysis Report (RESAR) as part of an effort towards design and licensing standardization of its Nuclear Steam Supply System. The RESAR was referenced by the McGuire (MNS) and Catawba (CNS) PSARs for a description of their respective Engineered Safety Features. Westinghouse originally defined ESFs as the provisions in the plant which retain the leakage of fission products from the fuel in the reactor coolant, and which ensure retention of fission products by the Containment for operational and accidental releases beyond the reactor coolant boundary. This Westinghouse definition of ESFs is consistent with the American National Standard and was adopted by Duke Power as indicated by the station Safety Analysis Report submittals.

Several vendor and NRC documents appear to have expanded the scope of the original ANSI N18.2 and RESAR definitions of what constitutes an ESF. A later revision of the RESAR (RESAR 3S-July 1976) identified auxiliary feedwater as an ESF system. NUREG 0800, "Standard Review Plan", now also lists auxiliary feedwater as an example of an ESF.

Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" itemizes the commonly used ESFs in light water cooled power reactors. This document mentions "ESF Filter Systems" and, although it contradicts other NRC documents (e.g., NUREG 0800), forms the basis for including ESF filter systems in Duke Power's listing of ESF systems.

Page 1 of this Attachment is a listing of the ESFs incorporated into the design of the nuclear stations and the systems which perform the associated protective functions. This listing is consistent with the intent of what constitutes an ESF, as gleaned from the documents discussed above. This listing is formatted in the same manner as Reg. Guide 1.70.

It is important to note that page 1 is a complete listing of the 3 station's ESF systems. In the past, various safety systems have been incorrectly classified as ESFs for reporting purposes presumably due to their actuation by the Engineered Safety Feature Actuation System (ESFAS). The ESFAS consists of process instrumentation and logic circuits and controls to sense accident situations and initiate the operation of necessary Engineered Safety Features. These non-ESF safety systems have included RN, KC, the Emergency Diesel Generators, and certain HVAC systems. As clearly stated in the MNS UFSAR and Westinghouse RESAR, "the following systems are required for support of the engineered safety features," and therefore, are not in themselves ESF systems:

Nuclear Service Water System (RN)
Component Cooling Water System (KC)
Electrical Power Distribution Systems

This differentiation between ESF systems and ESF support systems is also apparent in NUREG 0800.

Section 8.2 of this document titled "Engineered Safety Features Systems," discusses the difference between Engineered Safety Features (ESF) systems and what the NRC designates as essential auxiliary supporting (EAS) systems.

NUREG 0800 includes the following lists as examples of what constitutes an ESF and an EAS:

Typical ESF systems are:
Containment and Reactor Vessel Isolation Systems
Emergency Core Cooling Systems (ECCS)
Containment Heat Removal and Depressurization Systems
Pressurized Water Reactor (PWR) Auxiliary Feedwater Systems

Containment Combustible Gas Control Systems

Typical EAS systems are:

Electric Power Systems
Diesel Generator Fuel Storage and Transfer Systems
Instrument Air Systems
Heating, Ventilating, and Air Conditioning (HVAC) Systems for ESF Areas
Essential Service Water and Component Cooling Water Systems

Section 202.8.2, "Actuation of an Engineered Safety Feature or the Reactor Protection System" of this directive establishes what constitutes a reportable ESF actuation for the 3 nuclear stations. This section not only identifies examples of reportable ESF actuations, but those items that are not considered to be ESF actuations.

REFERENCES

1. ANSI N18.2-1973, *Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.*
2. Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*, Revision 3.
3. 10CFR50, *Domestic Licensing of Production and Utilization Facilities.*
4. NUREG 1022, *Licensee Event Report System*, and Supplements 1 and 2.
5. NUREG 0800, *Standard Review Plan* [Section 7.3]
6. Catawba Nuclear Station UFSAR
7. McGuire Nuclear Station UFSAR
8. Westinghouse Reference Safety Analysis Report (RESAR)

1-14 Jcw

4.3 OMP ~~1-10,~~ Usage and Testing of the NRC Emergency Notification System (ENS) (R26)

A. Use and testing of the ENS.

B. Determine reportability of an event.

C. Making a report, including follow-up as needed.

4.4 OMP 1-12, NRC License Maintenance (R27)

A. Requirements for maintaining an active NRC license.

B. Methods for restoring an inactive license to active status.

4.5 OMP 1-24, Operations Communications Standard (R32)

A. Responsibilities

B. Principles of Effective Communications

C. Crew Update

D. Crew Briefing

4.6 OMP 2-1, Duties and Responsibilities of On-Shift Operations Personnel (R5)

A. Roles and responsibilities of the OSM

B. Responsibilities of the Control Room SRO

C. Responsibilities of the Plant SRO

D. Responsibilities of the Reactor Operators

E. Responsibilities of non-licensed operators

F. Responsibilities of Refueling SRO and RB SRO

G. Normal lines of communication and shift organization during plant operation

H. Required boundaries within the control room to ensure the controls are adequately monitored by the operator

I. Shift Staffing Requirements

4.7 OMP 2-2, Unit Log (R24)

A. Maintain the Unit Log in a manner that documents the shift activities, accomplishments and problems.

4.8 OMP 2-7, SSF LCO Required Actions (R25)

A. Define actions required when SSF equipment is declared inoperable.

B. Identify criteria for determining SSF operability.

Exam Question Report

27-Jan-99

Question ID:	ADM845	Revision No:	0	Revision Date	10/29/1999
Question Description:	ADM845				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	NSD
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	Reference: NSD202	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Unit 1 has had an inadvertent ES channel 1 actuation at 100% power. All ES channel 1 components performed as required. Operators have taken action to re-position components as needed. Based on NSD 202 (Reportability), which ONE of the following is correct? (.25)

See Attachments

This event should be classified as ...

A) not reportable.

B) reportable within 1 hour.

C) reportable within 4 hours.

D) reportable within 24 hours.

Answer

C
A. Incorrect.
B. Incorrect. Even though, ECCS discharged into the RCS, it was not the result of a valid signal (step 202.6.6).
C. Correct. NSD 202, step 202.7.2.b, NRC is also interested in events where ESF actuated unnecesarily.
D. Incorrect.

Lessons

ID	Description

Enabling Objectives

QUESTION # 100

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		**3**
Group #		**4**
K/A #	**G 2.4.21**	
Importance Rating		**4.3**

Technical Reference(s): **PNS-RBP**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-RBP #9 & #10**

Question Source:
- Bank # **WE-90**
- Modified Bank # ______
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test __X__

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 __X__

Comments:

1 POINT

QUESTION # 100

SRO ONLY

As the CRSRO, evaluate the following sets of atmospheric conditions and determine which ONE is the **most** favorable for a purge of Unit 1's Reactor Building?

A. Cloudy daytime, $\Delta T = +10$, and wind speed 2 mph at 15°.

B. Cloudy daytime, $\Delta T = -5$, and a wind speed 10 mph at 183°.

C. Clear calm nighttime, $\Delta T = -10$ and a wind speed 2 mph at 15°.

D. Clear, calm nighttime, $\Delta T = +5$, and a wind speed 10 mph at 183°.

1 POINT

QUESTION # 100

G2.4.21 (3.7/4.3) SRO ONLY T3-C4 /PRA 5-1-00 Bank WE-90

Question setup:

A note in the RB Purge Release enclosure states the GWRs should be coordinated with favorable meteorological conditions. Unfavorable conditions are:

- Temperature inversion indicated by (+) on charts.
- Low wind speeds

A. Incorrect – An inversion is indicated by the (+)10° which is not favorable for a release. The wind speed is low at 2 mph which is also not favorable for a release

B. Correct – An inversion is not indicated because the ΔT is (-)5°. The wind speed is high which also favorable.

C. Incorrect - An inversion is not indicated because the number is (-)10°. However the wind speed is low at 2 mph.

D. Incorrect – An inversion is indicated by the (+)5°. However the wind speed is favorable at 10 mph.

4. Explain the purpose for each RBP system interlock and when given plant conditions: (R4)

   - Predict system/component/indication response to RBP system interlock actuation.
   - Describe necessary actions and/or plant status required to return system/component/indication to normal operating status.

5. Describe the response of the RBP system to a "High" alarm on RIA-45 or RIA-46. (R5)

6. Given the enclosure for removal/restoration of the Equipment Hatch: (R6)

   - Explain the personnel safety considerations that result in stopping the RB Purge fan prior to Equipment hatch removal/replacement
   - Describe the purpose of the desired minimum flow limit on RB Purge flow with the Equipment hatch removed
   - Describe the major difference in the enclosures for Equipment hatch removal/replacement with and without the RB Purge in operation

7. Describe the response of the RBP system to an actuation of Engineered Safeguards channels 1&2. (R7)

8. Given a specific RBP Limit and Precaution describe the purpose for the Limit or Precaution. (R8)

9. Given a specific set of conditions, determine if "favorable" or "unfavorable" conditions exist for a release. (R9)

10. Discuss the "required level of approval" for the various releases which may be in progress at the Station. (R10)

11. Given a copy of ITS/SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's.

Releases at the Station in Progress (including this one)	Required Level of Approval
1/3 Station Limit-1 unit releasing	SRO
1/3 Station Limit-2 units releasing.	Operations Shift Manager
1/3 Station Limit-3 units releasing	Operations Shift Manager
2/3 Station Limit-1 unit releasing	Operations Shift Manager
1/3 Station Limit on 1 unit AND 2/3 Station Limit on another unit	Operations Superintendent

e) Adjust PR-3 (RB Purge Control) to the desired purge rate, as indicated on the Purge Flow recorder in the control room.

f) Mark Meteorological charts (Temperature, Wind Direction, and Wind Speed) at beginning of release with

"Begin GWR ________________"

g) Mark Unit Log at beginning of release with

"Begin GWR ________________"

h) Monitor the unit vent radiation monitoring during the release, do not allow the alarm limits to be reached.

i) For continuous purges (i.e., those lasting longer than approximately 24 hours), contact Performance to measure Reactor Building relative humidity weekly during the continuous purge.

j) R.B. Gaseous Waste Release Form should only be used for approximately 24 hours.

- For continuous purges (i.e., those lasting longer than approximately 24 hours), initiate a new sample request daily on day shift, complete section 3.0 of the Gaseous Release Form, and proceed to the Continuous Release Form.

- This RTD is located downstream of the inlet heat exchanger and controls the steam flow to the heating coil when steam is valved into the coils.

2.6 Limits and Precautions

A. R.B. Purge System

1. Filters should be changed when the radiation level exceeds 20 mR/hr @ 1 ft. or the filter DP reaches the alarm point.
2. If low flow is experienced through the filters, the filters should be performance tested.
3. If equipment hatch removal is required while RB Purge is NOT available, the RB should be vented (via PR- 1, 2, &3) to the stack prior/during removal.
4. If Unit Vent RIA's inoperable during RB Purge operations, alternate sampling should be established per SLC 16.11.3-2.
5. Purge valve operation is allowed if RCS is in MODE 5, 6, or NO MODE. (Instructor Note: To clarify when Purge operation is allowed).
6. When the equipment hatch is open with the RB Purge system available, the RB Purge flow should be > 10,000 cfm to prevent release to the environment. The mini-purge should not be operated.
   - **Instructor Note: Tests have shown that a low flow rate does not ensure positive air flow into the RB through the hatch.**
7. There is also an enclosure in OP/1/A/1102/014 that allows opening the Equipment Hatch with the Purge Fan not available. This enclosure vents all positive RB pressure out of the stack prior to removal of the Equipment Hatch.
8. During a refueling outage when the Equipment Hatch is installed, the RB Purge System provides a vent flowpath for compressed air used inside the Reactor Building.
   - If the Purge Fan is stopped, Reactor Building pressure will gradually increase, resulting in a gradual decrease in Fuel Transfer Canal level and a subsequent increase in SFP level.
9. Reactor Building purges should be coordinated, as far as practicable, with favorable meteorological conditions.

- Unfavorable meteorological conditions are:
  1. Very low windspeed.
  2. Temperature inversion exists
     - The use of vertical temperature gradients is the most practical and universally accepted method of determining atmospheric stability. An inversion is defined as: "air at ground level colder than air aloft." This is indicated on Unit 1 Control Room chart recorder (Temp Difference). Any positive reading indicates an inversion (temperature near top of tower minus temperature at bottom of tower).
     - Unusually stable atmospheric conditions exist when an inversion exists, meaning that vertical air movement is stifled. Clear, calm, nightime conditions are usually very stable because the earth's surface cools rapidly, thus cooling the ground surface air. This is usually the time of day that an inversion will exist. The absence of winds prevents this cool air from mixing with the warmer air above, It is under these unfavorable conditions that the release of radioactive gases would not be desired.

2.7 Power Supplies

A. R.B. Purge Fans
1. Main Purge Fan(1)(2)(3)XR
2. Mini Purge Fan (1)(2)(3)XR

B. R.B. Purge Valves
1. PR-1(1)(2)(3)XS1
2. PR-6(1)(2)(3)XS1

2.8 Associated Technical Specifications and SLC's.

A. Tech Specs
1. Section 3.6 (CONTAINMENT SYSTEMS)
2. ITS 3.9.3 (Containment Penetrations)

B. SLC's
1. 16.11.2 (Radioactive Gaseous Effluents)

2. 16.11.3 (Radioactive Effluent Monitoring Instrumentation)
3. 16.6.9 (Containment Purge Valve Testing)

3. SUMMARY

3.1 This lesson has discussed the design and functions that are provided by the Reactor Building Purge System. With this knowledge, and by using the proper written procedures to operate this system, the Reactor Building Purge System can be safely operated both during Normal and Abnormal conditions.

Exam Question Report

27-Jan-99

Question ID:	WE090	Revision No:	000	Revision Date	04/18/2000
Question Description:	WE090				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	WE-GWD - Gaseous Waste Systems
Last Used Date:	04/20/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:		Max. Point Value:	0.25
		Passing Point Value:	0.25

27-Jan-99

Question

SRO ONLY

Which ONE of the following sets of atmospheric conditions is favorable for conducting a gaseous waste release? (.25)

A. Clear, calm nighttime, ΔT = +5, and a wind speed 10 mph at 183°.

B. Cloudy daytime, ΔT = -5, and a wind speed 10 mph at 183°.

C. Clear calm nighttime, ΔT = -10 and a wind speed 2 mph at 15°.

D. Cloudy daytime, ΔT = +10, and a wind speed 2 mph at 15°.

Answer

B

Lessons

ID	Description

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

INITIAL SUBMITTAL

OCONEE EXAM 2000-301
50-269, 270, AND 287/2000-301

JULY 10 - 14, 18, 19, AND 20, 2000

INITIAL SUBMITTAL

RO ONLY
WRITTEN EXAM

RO ONLY

Oconee
2000

NRC Copy

QUESTION # 76

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	____
Group #	**1**	____
K/A #	**000024 K 3.02**	
Importance Rating	**4.2**	____

Technical Reference(s): **EOP Rule #1**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E26 #5**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

QUESTION ✓

000024 K3.02 (GCW) 2-1-00

Question overview:
Condensate flow rapidly decreasing to zero will cause both Main FDW Pumps to trip on low suction pressure. (Actual) flow decreasing should provide understanding that it does not assume that indicated flow has failed. This should cause the reactor to trip on "Loss of FDWP" anticipatory trip. "A" HPI Header Flow rate = flashing 8s indicates that the "A" flow indication has failed high. The "A" and "B" HPI Pump would be operating due to low RCP seal injection flow of ≤ 30 gpm. Emergency Boration is required because the reactor has not tripped and power is greater than 5%. Emergency Boration requires the following:

- Open HP-24/25/26/27
- Check running "A" or "B" HPI Pump
- Start "C" HPI Pump
- Verify flow in both headers
  - If "A" flow is less than Enclosure 7.2 open HP-410 (HP-26 Bypass)
  - If "B" flow is less than Enclosure 7.2 open HP-409 (HP-27 Bypass)

B. ~~A.~~ Incorrect - Not required to throttle HPI flow ≤ 475 gpm if two HPI pumps are operating in the header.

A. ~~B.~~ Incorrect - 3HP-410 (HP-26 bypass) will put flow in the "A" header

C. Incorrect - The 3B HPI Pump should remain on if the operator opens 3HP-409 (3HP-27 Bypass) to establish flow in the "B" header.

D. Correct – Two HPI pumps and two headers are required to be operating. "A" or "B" and "C" HPI Pumps should be operating.

7

QUESTION ______

Unit 3 plant conditions:

INITIAL CONDITIONS:

- Power level = 46%
- Condensate flow (actual) rapidly decreases to 0 gpm

CURRENT CONDITIONS:

- Power level = 38%
- RCS pressure = 2310 psig
- 3HP-26 (3A HP INJECTION) fully open
- "A" HPI Header Flow rate = flashing 8's
- "B" HPI Header Flow rate = 0 gpm
- Seal Inlet HDR Flow = 20 gpm

Which ONE of the following operator actions is correct?

SEE ATTACHMENT ______

A. Throttle 3HP-410 (3HP-26 BYPASS) to achieve ≤ 475 gpm in the 3B HPI Crossover Header

B. Throttle 3HP-26 (3A HP INJECTION) to achieve ≤ 475 gpm in the 3A HPI Header

C. Stop 3B HPI Pump and open 3HP-409 (3HP-27 BYPASS)

D. Stop 3B HPI Pump and Start 3C HPI Pump

~~still no valves have been opened to establish flow in B HDR.~~

c 3

1 POINT

QUESTION #76

RO ONLY

Unit 3 plant conditions:

INITIAL CONDITIONS:

- Power level = 46%
- Condensate flow (actual) rapidly decreases to 0 gpm

CURRENT CONDITIONS:

- Power level = 38%
- RCS pressure = 2410 psig increasing
- 3HP-26 (3A HP INJECTION) fully open
- "A" HPI Header Flow indication = FAILED high
- "B" HPI Header Flow rate = 0 gpm
- Seal Inlet HDR Flow = 20 gpm and decreasing

Which ONE of the following operator actions is correct?

A. Throttle 3HP-410 (3HP-26 BYPASS) to achieve ≤ 475 gpm in the 3B HPI Crossover Header.

B. Throttle 3HP-26 (3A HP INJECTION) to achieve ≤ 475 gpm in the 3A HPI Header.

C. Stop 3B HPI Pump and open 3HP-409 (3HP-27 BYPASS).

D. Stop 3B HPI Pump and Start 3C HPI Pump.

1 POINT

QUESTION # 76

000024 K3.02 (GCW) RO ONLY 2-1-00 (GTH)

Question overview:
Condensate flow rapidly decreasing to zero will cause both Main FDW Pumps to trip on low suction pressure. (Actual) flow decreasing should provide understanding that it does not assume that indicated flow has failed. This should cause the reactor to trip on "Loss of FDWP" anticipatory trip. "A" HPI Header Flow rate = flashing 8s indicates that the "A" flow indication has failed high. The "A" and "B" HPI Pump would be operating due to low RCP seal injection flow of ≤ 30 gpm. Emergency Boration is required because the reactor has not tripped and power is greater than 5%. Emergency Boration requires the following:

- Open HP-24/25/26/27
- Check running "A" or "B" HPI Pump
- Start "C" HPI Pump
- Verify flow in both headers
  - If "A" flow is less than Enclosure 7.2 open HP-410 (HP-26 Bypass)
  - If "B" flow is less than Enclosure 7.2 open HP-409 (HP-27 Bypass)

A. Incorrect - Not required to throttle HPI flow ≤ 475 gpm if two HPI pumps are operating in the header. 3HP-410 (3HP-26 BYPASS) aligns flow to the "A" header not the "B" header.

B. Incorrect - 3HP-26 HPI is not required to throttle HPI flow ≤ 475 gpm if two HPI pumps are operating in the header.

C. Incorrect - The 3B HPI Pump should remain on if the operator opens 3HP-409 (3HP-27 Bypass) to establish flow in the "B" header.

D. Correct – Two HPI pumps and two headers are required to be operating. "A" or "B" and "C" HPI Pumps should be operating. Emergency Boration implies 2 HPIPs in 2 headers.

OBJECTIVES

TERMINAL OBJECTIVE:

1. Describe the use of Section 506, Unanticipated Nuclear Power Production, of the Emergency Operating Procedure in order to perform the required actions of a Nuclear Control Operator during an UNPP event.

ENABLING OBJECTIVES:

1. State when Section 506 of the EOP should be implemented. (R1)

2. Explain the basis for the entry conditions of this section. (R2)

3. Recognize that actual industry events have occurred where CRD breakers have failed to trip on demand and that prompt operator action was necessary to insure reactor shutdown. (R4)

4. Briefly explain why the reactor coolant pumps should remain in operation, even if RCS subcooling margin is lost. (R11)

5. State the three major operator actions that should be performed during an UNPP event based on Rule #1 (ATWS Actions). (R3)

6. Briefly explain why it is important that the turbine be tripped if NO Main FDW is available during an UNPP event.(R5)

7. State the two primary reasons why RCS expansion will occur during an UNPP event, requiring RCS letdown to be re-established.(R6)

8. Explain why the operator should control feedwater to match Rx power production during an UNPP event until RCS temperature stabilizes, i.e. no heatup or overcooling. (R7)

2. PRESENTATION

The major EOP step numbers are in parenthesis at the end of the lesson plan steps.

If SCMs are lost during the UNPP event, RCPs should not be tripped; they should remain in operation until power is ≤ 1% to provide flow through the core for heat removal if possible

SRO may deem it necessary to secure an RCP in a loop if RCP damage is immanent.

Maintaining forced RCS flow is the preferred method to remove core heat (due to the increased heat transfer available).

2.1 If any PR NI > 5% FP, then shutdown the Rx:

A. Ensure Rule #1 (ATWS Actions) is in progress or complete.

Rule #1:

NOTE:	Steps in this Rule may be performed concurrently.

1. Drive Control Rods to the "In Limit".

Manual insertion of all control rods which have not completely inserted should be initiated immediately to place the reactor in a subcritical state. This step is initiated first since it will be the quickest and most effective means available to shut the reactor down from the control room.

The operator must make sure to place the Diamond in MANUAL to accomplish this; several times during simulator training, this has been neglected, so that while the Joy stick has been latched to the insert position, rods were not actually moving.

This action will not result in the safety rods inserting if there has been a DSS trip since the Aux Power Supply is not available. (NOTE: If a DSS trip has occurred, the Rx. will probably be ≤ 5% power).

It is important that the SRO(s) know whether or NOT the Control Rods can be MANUALLY tripped FROM THE CONTROL ROOM AND whether or NOT DSS has inserted Groups 5,6,& 7. He will use this information in determining the Emergency Classification for the event.

- *IF the Control Rods CAN be manually tripped OR IF DSS HAS inserted Groups 5-7, THEN the event will be classified as an ALERT (IF based solely on the ATWS).*

- ***IF the Control Rods <u>CANNOT</u> be manually tripped AND DSS has NOT inserted Groups 5-7, THEN the event will be classified as an <u>SITE AREA EMERGENCY</u> (IF based solely on the ATWS).***

2. Initiate emergency boration:
   a) Open HP-24, 25, 26 & 27
   b) Ensure 'A' or 'B' HPIP operating
   c) Start 'C' HPIP

   Emergency boration from the BWST should be initiated to increase the boron concentration in the RCS which will lead to a reduction in reactor power.

   Boration is a sure means of adding negative reactivity to the core. However, this process is slow since time is necessary to increase the boron concentration to the value necessary to completely shut down the reactor.

   This Emergency Boration can only be secured when ALL Wide Range NIs < 1% power.

3. Dispatch operator(s) to de-energize the CRD System.

   Barring mechanical binding of the control rods, interrupting power to the CRDMs should cause all rods to drop; operators should be immediately dispatched to the cable room to manually open the AC and DC breakers. Another operator should be sent to open the 600VAC primary and secondary CRD supplies in case the AC or DC breakers fail to operate.

4. Ensure adequate HPI header flow in the acceptable region per the EOP curve provided.

5. If at any time (IAAT) only one HPIP is running in a header, limit HPI header flow to ≤ 475 gpm/pump (including seal injection).

B. If Main FDW is NOT available, then trip the Turbine Generator.

   Although the turbine offers a means of relieving the primary system energy, fission power generation will still exist with a condition that warrants a reactor trip. Thus by tripping the turbine promptly, a mismatch will occur between the core power generation and the secondary heat removal rate (the RCS will heat up).

   Both the moderator temperature and Doppler reactivity feedback effects will begin to reduce reactor power to a level, which will more closely match the secondary heat removal capabilities.

   In the absence of control rod insertion, these effects are the only means of promptly reducing reactor power until the slower means of reactivity control via boration becomes effective.

Emergency Operating Procedure

Rule #1

ATWS Actions

NOTE:	Steps in this Rule may be performed concurrently.

1. Drive the Control Rods to the "In Limit" by performing the following:

_____ 1.1 Place Diamond in "MANUAL."

_____ 1.2 Use manual command switch to insert control rods.

2. Initiate emergency boration by performing the following:

2.1 Open the following valves:

_____ 1HP-24 (1A HPI BWST SUCTION)
_____ 1HP-25 (1B HPI BWST SUCTION)
_____ 1HP-26 (1A HP INJECTION)
_____ 1HP-27 (1B HP INJECTION).

_____ 2.2 Ensure 1A or 1B HPI Pump is operating.

_____ 2.3 Start 1C HPI Pump.

3. Dispatch operator(s) to de-energize CRD system:

_____ 3.1 Open all Unit 1 AC and DC CRD breakers (Cable Room).

3.2 Open the 600v CRD breakers:

_____ "UNIT 1 CRD NORMAL SUPPLY BKR" (1X9) (Equipment Room)

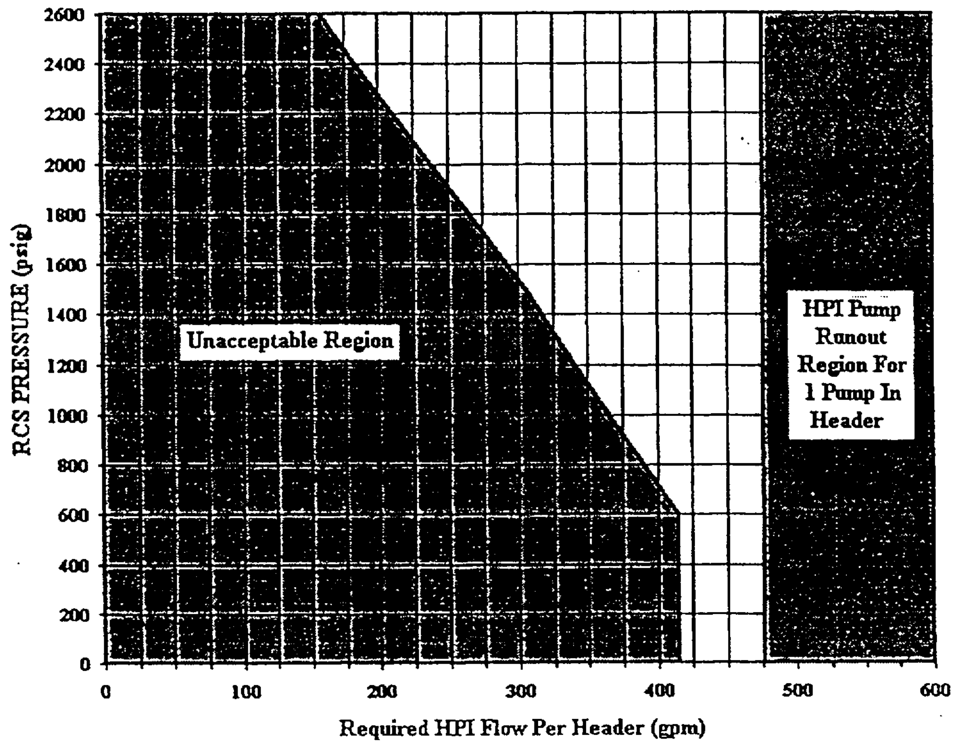
_____ "CONTROL ROD DRIVE ALTERNATE SUPPLY BKR TO UNIT 1" (2X1). (TB-3/E-28)

CONTINUED ON REVERSE SIDE

ATWS Actions

____ 4. **IF** at least two HPI Pumps are operating,

THEN ensure adequate HPI header flow by performing the following:



____ **IF** 1A HPI header flow is in the Unacceptable Region,

THEN open 1HP-410 (1HP-26 BYPASS).

____ **IF** 1B HPI header flow is in the Unacceptable Region,

THEN open 1HP-409 (1HP-27 BYPASS).

____ 5. **IF** AT ANY TIME only one HPI Pump is running in a header,

THEN limit HPI header flow to $\leq$ 475 gpm/pump (include seal injection).

____ 6. Notify the Procedure Director to **GO TO** Section 506, Unanticipated Nuclear Power Production.

◇◇◇ END ◇◇◇

QUESTION # 77

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	_____
Group #	**1**	_____
K/A #	**000026 A1.03**	_____
Importance Rating	**3.6**	_____

Technical Reference(s): **CF-EF**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-EF #11 & #12**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

000026 A1.03 RO ONLY (PRA) 1-30-00

A. Incorrect – 1LPSW-137 has failed closed. The limit switch from MS-93 TD Steam Admission has actuated as the TD is operating as indicated by it's discharge pressure indication. does this limit switch open 1LPSW-137?

Does not explain why distractor is wrong

B. Incorrect – On a loss of power to 1LPSW-137 or loss of air to LPSW-137 and/or 1HPSW-184 will deenergize their solnoid and the valves will fail open.

same as B

C. Incorrect – On a loss of power to 1LPSW-137 or loss of air to LPSW-137 and/or 1HPSW-184 will deenergize their solnoid and the valves will fail open.

D. Correct – Since 1LPSW-137 has failed in the closed position and power and air is available to LPSW-138 and HPSW-184 manual positioning of HPSW-184 is required.

explain why distractor is wrong not how system normally operates.

RO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 100%
- Both Main FDWPT trip

CURRENT CONDITIONS:

- 1A and 1B MDEFDWP fails to start
- TD EFDWP (1MS-93) position indicates "RED" on UB1
- TD EFDWP discharge pressure = 1380 psig on UB1
- 1LPSW-137 (LPSW to TD EFDWP Cooling Jacket) position indicates "GREEN" on 1VB1
- Instrument Air header pressure = 100 psig

Which ONE of the following describes how ~~the~~ cooling water ~~source~~ will be aligned to the TD EFDWP?

A. Placing the TD EFDWP switch to "RUN" will open 1LPSW-137 (LPSW to TD EFDWP Cooing Jacket).

B. 1LPSW-138 (TDEFDWP Cooling Bypass) will automatically open due to LPSW-137 open limit switch not energized with the TD running.

C. 1HPSW-184 (TDEFDWP Cooling Bypass) will automatically open due to LPSW-137 open limit switch not energized with the TD running.

D. 1HPSW-184 must be manually opened from its local switch on the SG Level panel.

explain

3c

1 POINT

QUESTION # 77

RO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 100%
- Both Main FDWPTs trip

CURRENT CONDITIONS:

- 1A and 1B MD EFDWPs fail to start
- TD EFDWP is operating
- 1LPSW-137 (LPSW to TD EFDWP Cooling Jacket) failed closed
- Instrument Air header pressure = 100 psig

Which ONE of the following describes how the cooling water will be aligned to the TD EFDWP?

A. Placing the TD EFDWP switch to "RUN" will open 1LPSW-137 (LPSW to TD EFDWP Cooing Jacket).

B. 1LPSW-138 (TD EFDWP Cooling Bypass) will automatically open due to LPSW-137 open limit switch not energized with the TD EFDWP running.

C. 1HPSW-184 (TD EFDWP Cooling Bypass) will automatically open due to LPSW-137 open limit switch not energized with the TD EFDWP running.

D. 1HPSW-184 must be manually opened from its local switch on the SG Level panel.

1 POINT

QUESTION # 77

000026 A1.03 RO ONLY (PRA) 1-30-00 (GTH)

A. Incorrect – 1LPSW-137 has failed closed. The limit switch from MS-93, TD EFDP Steam Admission has actuated as the TD EFDWP is operating as indicated by it's discharge pressure indication.

B. Incorrect – A loss of power to 1LPSW-137 or loss of air or loss of solenoid power to LPSW-138 will cause the valve to fail open. 1LPSW-138 will not automatically open.

C. Incorrect - 1HPSW-184 will not automatically open if LPSW-137 does not energize its open limit switch. 1HPSW-184 will automatically open when the local switch is placed into the bypass position or a loss of air or power occurs to its solenoid.

D. Correct – Since 1LPSW-137 has failed in the closed position and power and air is available to LPSW-138, HPSW-184 manual positioning of HPSW-184 is required.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

1. After this lecture, the student will have an understanding of the components, indications, controls and operation of the EFDW System. He/she will be able to relate the operation of the EFDW System to the safe operation of the plant and its impact on accident mitigation. Along with this, the student will be aware of the conditions that could possibly lead to rendering the EFDW System inoperable and what actions to take to mitigate the consequences of these situations. Identify and be able to discuss an understanding of the other systems that integrate with the EFDW System, such as, ICS, FDW, Main Steam, Auxiliary Steam, Condensate, IA, and Electrical Power. (T1)

ENABLING OBJECTIVES

1. State the purpose of the EFDW System. (R1)
2. List the power supplies for the MDEFDWPs. (R5)
3. List the cooling medium for the MDEFDWP motors. (R6)
4. Describe the operation of the cooling water system for the MDEFDWP motors, including the failure mode on loss of power. (R26)
5. Describe the normal AND alternate suction and discharge flow paths for the EFDW System, include major pumps and valves.(R2)
6. Draw a one-line diagram of the EFDW System that indicates the normal suction and discharge flow paths (R57)
7. List the normal suction supplies to the TDEFDWP and the MDEFDWPs. (R3)
8. List the alternate suction supplies to the TDEFDWP and the MDEFDWPs and describe what conditions must exist to be able to use this source. (R4)
9. Describe the minimum recirculation flow paths for the MDEFDWPs, including the function of the ARC valve. (R7)
10. List the cooling medium for the TDEFDWP bearing cooling jacket. (R8)
11. List the backup source of cooling water to the TDEFDWP bearing-cooling jacket. (R9)

12. Describe the operation of the TDEFDWP bearing cooling jacket water system. (R10)
13. Describe the minimum recirculation flow path for the TDEFDWP. (R11)
14. List the sources of steam for the TDEFDWP. (R12)
15. Describe the purpose and operation of MS-93. (R24)
16. Describe how to manually open MS-93. (R43)
17. Explain the operation of the Primary Relay associated with the TDEFDWP including a description of the operation of the Speed Governor and Pilot Valve. (R13)
18. Explain how to use the Hand-Start lever of the Primary Relay to start the TDEFDWP in the event that the Auxiliary Oil Pump does not start when MS-93 opens. (R44)
19. List the two functions of the Trip Throttle valve and explain the operation of the Trip Throttle and Operating Valve associated with the TDEFDWP. (R14)
20. Describe how to reset the Trip Throttle Valve and what to look for to verify that it is reset. (R15)
21. Explain the operation of the Overspeed Governor and Emergency Relay associated with the TDEFDWP. (R16)
22. Describe how control oil and lube oil are supplied to the TDEFDWP during startup and operation. (R17)
23. List the normal and backup cooling medium for the TDEFDWP oil cooler. (R18)
24. Explain how the EFDW Systems can be cross-connected between units. (R19)
25. Describe the MANUAL and AUTOMATIC (including Auto 1 & AUTO 2) control available for the MDEFDWPs and their purposes. (R20)
26. Explain how to stop the MDEFDWPs following an the AUTO START or a MANUAL START.(R21)
27. Describe or make a sketch of the logic/conditions that will AUTO START the MDEFDWPs when the respective control switches are in AUTO, including a description of AMSAC and DRY OUT PROTECTION (R22)
28. Describe the purpose for AMSAC/DSS, including actuating setpoints and functions they provide following actuation. (R61)

4.3 Manual Operation of FDW-315 and FDW-316 (Fig OC-CF-EF-11)

A. Valves located in East (FDW-315) and West (FDW-316) Penetration Rooms

1. Handwheel on top of each valve.
2. Normally "backed-off" so the controller can operate the valve.
3. If the handwheel is not fully backed off the valve travel will be limited. This has caused EFDW train inoperability at ONS.
4. Can <u>only be used</u> to close valve or throttle valve if it fails open, i.e. loss of IA.
5. There is a locking nut on the handwheel stem that has to be "backed-off" of the bottom of the stem in order to allow free operation of the handwheel.(This nut is normally backed off)

5. A.C. Independence (Figure OC-CF-EF-19)

5.1 Loss of all power - steam driven TDEFDWP available

A. Cooling Water

1. LPSW-138 (TDEFDWP Cooling Bypass)
   a) LPSW-138 is an air-operated valve, normally shut.
   b) On a loss of power to LPSW-137, the solenoid valve keeping air on LPSW- 138 de-energizes, allowing LPSW-138 to fail open, bypassing LPSW-137.
2. HPSW-184 (TDEFDWP Cooling Bypass)
   a) HPSW-184 is an air operated valve, normally shut.
   b) On a loss of power to (1/2/3 - XS3) the EFDWPT oil cooling water pump becomes inoperable, the solenoid valve keeping air on HPSW-184 must also de-energize, to allow HPSW-184 to open on loss of power. This provides HPSW to the TDEFDWP oil cooler.
      1) Cooling water supply to the oil cooler is limited to CCW with HPSW backup. New piping removes LPSW from the flowpath and provides independent HPSW backup supplies. (Refer to OC-CF-EF-19a)
3. HPSW-184/LPSW-138 Control Switch

a) Two position switch, Normal/Bypass located on Steam Generator Level Control (SGLC) panel used for testing, in a loss of LPSW scenario, or a loss of power only to 1XS3.

b) Indicator lights show actual valve position, regardless of switch position.

c) **Normal** - HPSW-184/LPSW-138 shut.

1) Solenoids are energized, air is supplied to HPSW-184/LPSW-138 operators.

d) **Bypass** - HPSW-184/LPSW-138 open.

1) Solenoids are de-energized; air is bled off of valves.

4. Power Supplies for cooling water:

Note: Power supplies are not the same for each unit. PIP: 99-0971 was written to correct the DBD, which incorrectly describe the power supplies.

VALVE ↓ UNIT →	UNIT 1	UNIT 2	UNIT 3
LPSW – 137	1XC	2XAA	3XAA
LPSW – 138 / HPSW – 184	1XC	2XC	3XC

B. Nitrogen Backup

1. Following valves have N² backup. This insures adequate steam regulation and level control on a loss of IA for at least **2 hours**.

- FDW-315 (SG "A" EFDW Control Valve)
- FDW-316 (SG "B" EFDW Control Valve)
- MS-87 (MS to TDEFDWP Control)
- MS-126 (MS to AS Control)
- MS-129 (MS to AS Control)

C. FDW-315 and 316 may be throttled locally by a manual handwheel, if required.

D. MS-93 does not require AC power to open.

E. Starting oil pressure supplied by DC oil pump.

QUESTION # 78

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	____
	Group #	**1**	____
	K/A #	**000040 A 1.24**	
	Importance Rating	**3.8**	____

Technical Reference(s): **SAE-L22**

Proposed references to be provided to applicants during examination: ________

Learning Objective: **SAE-L22 OBJ. #1**

Question Source:
- Bank # ____
- Modified Bank # ____
- New __X__

Question History:
- Previous NRC Exam ____
- Previous Quiz / Test ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ____

Comments:

000040 A1.24 (RO ONLY) (PRA) 2-2-00

A. Correct - "A" will increase to post trip pressure as the leak is isolated when the MSSV close on the reactor/turbine trip. "B" will decrease as this leak is large enough to decrease the MS pressure.

B. Incorrect – If the "B" MS line leak was smaller and or positioned ~~upstream~~ of the MSSV (cold or hot reheat - MSRH's) this would be correct. Since the leak is ~~down~~-stream of the MSSV the indication would decrease.

C. Incorrect – If both leaks were up stream of the MSSV's and large enough to overcome post trip FDW valve leakage then this would be correct.

D. Incorrect – If the leakage on the MS line were swapped and "A" was the larger leak upstream of the MSSVs then this would be correct.

RO ONLY

Unit 3 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 100%

CURRENT CONDITIONS:

- A small steam leak develops on the 3A MS line in the Turb Bldg. (≈ 2" break) at 3SD-273 (CV Above Seat Drain to Condenser)
- A large steam leak develops on the 3B MS line in the Pent Rm. (≈12" break)
- The RO manually trips the reactor

SEE ATTACHMENT ___?___

Which ONE of the following describes the response of the MS pressure indication when the OATC trips the reactor?

"A" MS pressure indication will ____ / "B" MS pressure indication will ____.

A. increase / decrease

B. increase / increase

C. decrease / decrease

D. decrease / increase

Barely C 2

Rx Trip → Turbine trip → MSSV shuts

1 POINT

QUESTION # 78

<u>RO ONLY</u>

Unit 3 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 100%

CURRENT CONDITIONS:

- A small steam leak develops on the 3A MS line in the Turbine Building (≈2" break) at 3SD-273 (CV Above Seat Drain to Condenser)
- A large steam leak develops on the 3B MS line in the Pent Rm. (≈12" break)
- The RO manually trips the reactor

Which ONE of the following describes the response of the MS pressure indication when the OATC trips the reactor?

"A" MS pressure indication will ____ / "B" MS pressure indication will ____.

A. increase / decrease

B. increase / increase

C. decrease / decrease

D. decrease / increase

1 POINT

QUESTION # 78

000040 A1.24 (RO ONLY) (PRA) 2-2-00 (GTH)

A. Correct - “A” will increase to post trip pressure as the steam leak is isolated when the MSSV close on the reactor/turbine trip. “B” will decrease, as this leak is large enough to decrease the MS pressure.

B. Incorrect – If the “B” MS line leak was smaller and or positioned upstream of the MSSV (cold or hot reheat - MSRH’s) this would be correct. Since the leak is down stream of the MSSV the indication would decrease.

C. Incorrect – If both leaks were up stream of the MSSV’s and large enough to overcome post trip FDW valve leakage then this would be correct.

D. Incorrect – If the leakage’s on the MS line were swapped and “A” was the larger leak upstream of the MSSV’s then this would be correct.

LESSON SPECIFIC OBJECTIVES

Terminal Objective

1. The Crew should perform the actions required to diagnose a MS Line Rupture outside containment, isolate the SG with the MS Line Rupture and stabilize the Unit once the SG is isolated. During the evolution, plant status and operator actions will be monitored by the Oversight SRO. (T1)

Enabling Objectives

1. Utilize control board indication and plant response to diagnose a MS Line Rupture in the "1A" MS Line, outside containment. (R1)
2. Demonstrate the ability to perform the correct actions to mitigate the MS Line Rupture. (R2/R3/R3a/R7/R10)
   - 2.1 Manually trip the Reactor if an automatic trip has not already taken place. (R2)
   - 2.2 Perform IMAs (R3)
     - A. The OATC should first perform from memory and then verify with the SRO.
   - 2.3 Perform a symptoms check: (R3)
     - A. If the RCS is saturated, run Rule #2. (R7)
     - B. If the RCS is subcooled, run Rule #7. (R10)
   - 2.4 Stabilize RCS CETCs when the overcooling is isolated. (R3a)
3. Compare the Unit response to the expected response for a LBLOCA and verify the diagnosis of a MS Line Rupture and not a LOCA. (R5)
4. When rule #6 has been completed, run ES-505 if ES has actuated. (R6)
5. Use the Turbine Bypass Valves on the intact steam generator to maintain CETCs constant. (R9)
6. Diagnose the symptoms of a subcooled pressurizer and initiate proper corrective actions: (R14)
   - 6.1 Limit primary heatup and insurge into the pressurizer.
   - 6.2 Energize all pressurizer heater banks until saturated conditions exist in the pressurizer.
7. Determine if operation in the TSOR is required. (R15)

INSTRUCTOR NOTE: Make sure that the Crew understands that the MSLB Isolation Circuit does nothing to the MD EFDWPs or FDW-315&316. It is still the responsibility of the Operator to secure EFDW flow to a faulted generator within three minutes.

2. The SG Operating Range level will immediately increase to 100% and then begin to return to normal because of the instantaneous change in pressure within the SG. The level detector sees the ΔP change as a high level. The MFDWP trip circuit will see this increase in level and trips the MFDWPs when SG level indicates >98% on the Operating Range. Even if the MSLB circuit does not work, the MFDWPs will trip automatically.

C. The affected steam generator is quickly identified after turbine trip due to separation of the MS headers. MS pressure will continue to decrease on the "A" side, and will begin to recover on the "B" side.

3.2 Instruct the students to monitor the plant for the symptoms just discussed and activate Timer #1 to cause a Main Steam Line break outside containment on the "1A" SG.

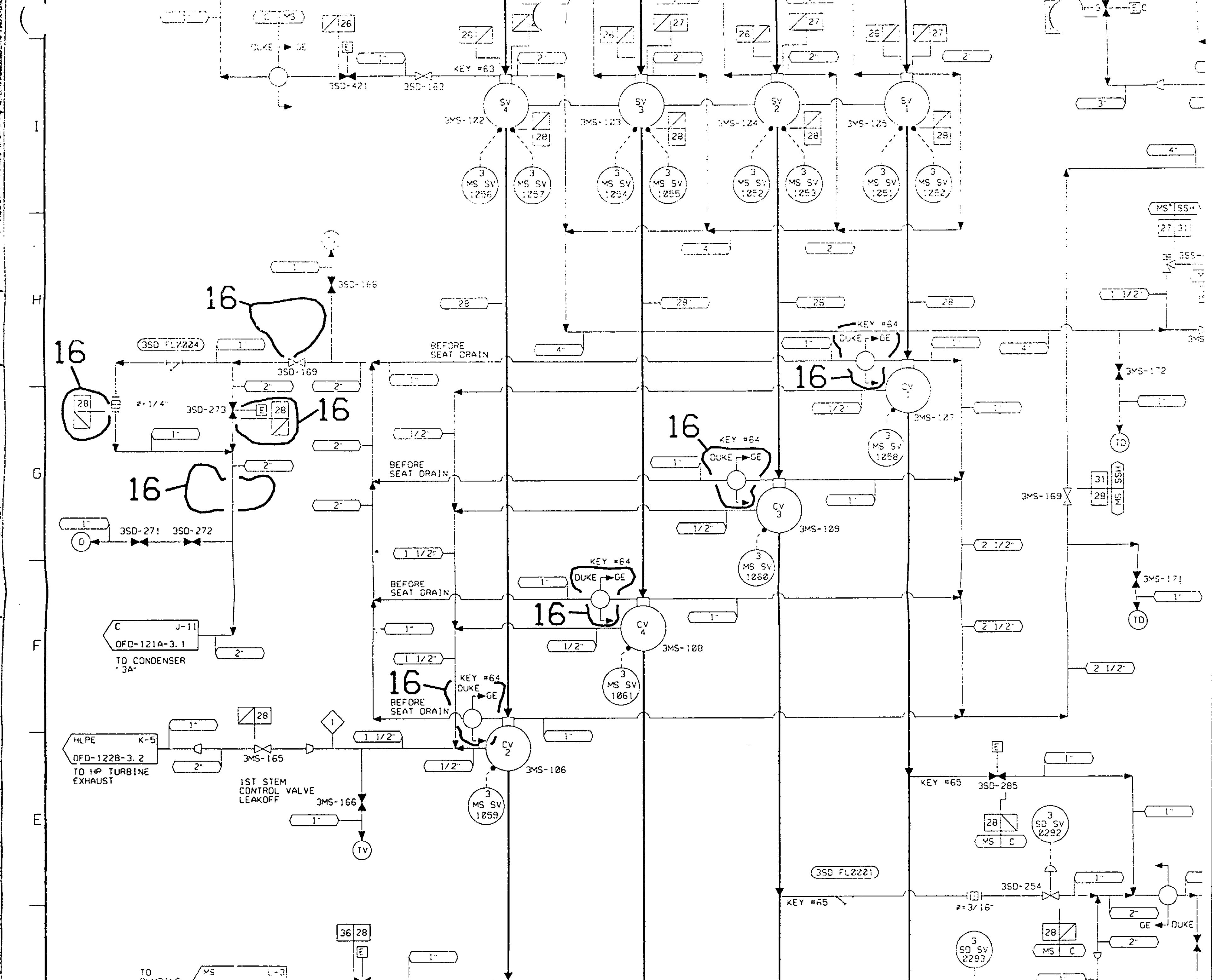
INSTRUCTOR NOTE: During the blowdown of the affected SG, water that is in the MFDW and EFDW lines up to the first isolation valve will be sucked into the SG (becomes low pressure area as steam is blown out). On the simulator this phenomenon is accomplished by the addition of mass to the SG inventory. This addition occurs in three distinct dumps of water into the SG. Because of this the blowdown, depending upon break size, appears to stop and linger at approximately 600, 400 and 200 psig. If the students ask about the blowdown pauses, this is the reason.

3.3 *Freeze* the simulator once the A SG has depressurized and SCM = 0, then walk through the following:

A. Discuss with the students that the event is different from a LOCA in the following ways:

1. RCS pressure and temperature decrease not just pressure like a LOCA.
2. RB pressure and temperature do not increase.
3. There is no increase in RB RIA's.

B. Relate the Pressure/Temperature (P/T) display to a Steam Line Rupture.



SV 4
SV 3
SV 2
SV 1
3MS-122
3MS-123
3MS-124
3MS-125
KEY #63
3SD-421
3SD-162
3SD-168
3SD-169
3SD-273
3SD-271
3SD-272
3SD FL2224
BEFORE SEAT DRAIN
KEY #64
DUKE
GE
CV 1
CV 3
CV 4
CV 2
3MS-127
3MS-109
3MS-108
3MS-106
3MS-165
3MS-166
3MS-169
3MS-171
3MS-172
OFD-121A-3.1
TO CONDENSER "3A"
HLPE
OFD-122B-3.2
TO HP TURBINE EXHAUST
1ST STEM CONTROL VALVE LEAKOFF
KEY #65
3SD-285
3SD-254
3SD FL2221
16

QUESTION # 79

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	____
	Group #	**2**	____
	K/A #	**000001 G2.4.10**	
	Importance Rating	**3.0**	____

Technical Reference(s): **IC-CRI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-CRI OBJ. #6 & #7**

Question Source: Bank # **IC-318**
Modified Bank # ____
New ____

Question History: Previous NRC Exam ____
Previous Quiz / Test ____

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 __X__
55.43 ____

Comments:

1 POINT

QUESTION # 79

RO ONLY

Which ONE of the following is the MINIMUM individual rod position deviation that will generate the CRD position error statalarm and give the associated fault lamp status listed below?

A. SEVEN (7) inches from its Relative Position Indication (RPI) group average and the ASYMMETRIC FAULT lamp on the Diamond Control panel is OFF.

B. NINE (9) inches from its Absolute Position Indication (API) group average and the individual FAULT lamp on the Position Indicating panel is OFF.

C. SEVEN (7) inches from its Absolute Position Indication (API) group average and the individual FAULT lamp on the Position Indicating panel is ON.

D. NINE (9) inches from its Relative Position Indication (RPI) group average and the ASYMMETRIC FAULT lamp on the Diamond Control panel is ON.

1 POINT

QUESTION # 79

000001 G2.4.10 RO ONLY rsi/gcw 04/20/00

A. Incorrect - The CRD position statalarm is generated by the Absolute Position Indication (API) and NOT the Relative Position Indication (RPI).

B. Incorrect - This is correct in that the CRD Position Error alarm will be in alarm but, it is NOT the MINIMUM individual rod position deviation.

C. Correct - The alarm is generated when API is 7 inches or greater than the average group API position. The individual ASYMM FAULT lamp on the Diamond is lit.

D. Incorrect - The CRD position statalarm is generated by the API and NOT the RPI and it is NOT the MINIMUM individual rod position deviation.

OBJECTIVES

TERMINAL OBJECTIVE

Describe the operation of the Control Rod Drive Instrumentation System including Control Room Diamond Panel, PI Panel, and associated power supplies. Evaluate proper system operation in accordance with procedures and alarm response manual to ensure safe operation of the Control Rod Drive System.

ENABLING OBJECTIVES

1. Describe the purpose of the following rod groups associated with the Control Rod Drive System: (R1)

   1.1 Safety Groups 1 through 4.

   1.2 Regulating Groups 5 through 7.

   1.3 Axial Power Shaping Rods Group 8.

2. Given a schematic of the CRD system power supply, identify the normal and alternate power supply paths for each Oconee unit including the purpose for the following: (R2)

   2.1 Source Interruption Device

   2.2 Auxiliary Power Supply

   2.3 Regulating Power Supply

   2.4 DC Hold Power Supply

3. Describe basically how the sequential energizing of the six phases of a CRD stator results in movement of the control rod into or out of the reactor core. (R3)

4. Discuss the functions of the Silicon Controlled Rectifiers used in the Auxiliary and Regulating power supplies of the CRD system. (R4)

5. Explain the reason for a CRD to automatically step back from three stator coils energized to two stator coils energized if a CRD stops with three stators energized. (R5)

6. Explain how each of the CRD Position Indication signals (Absolute and Relative) is derived. (R6)

7. Given an I&C output location or CRI circuit, identify whether the variable is fed by Absolute or Relative Position Indication. (R7)

B. Position Indicator Panel

REFER to OP-OC-CRI-7 & 7a

1. The position indicator panel is mounted on the vertical board above the Diamond Control Panel and consists of the following indications for ALL 69 control rods:
   - position meter
   - "out limit" lamp (100%)
   - "in limit" lamp (0%)
   - "7 alarm" lamp (Asymmetric Alarm)
   - "control on" lamp
2. There is a "Position Select" switch which is used by the operator to select either the API signal or the RPI signal to be monitored by the PI meters.
   - NOTE: The signal selected here will also be the one selected as the input to the Operator Aid computer).
3. There is a "Position Reset" switch used by the operator to reset a Relative Position Indicator after a dropped rod to match the API signal.
   a) Used before start-up to calibrate the RPI signals before withdrawing rods.
   b) Position Reset switch is used in conjunction with the Group and Single Select switches on the Diamond to select the rod(s) needing RPI resetting.

C. Relative Position Indication (RPI)

REFER to OP-OC-CRI-6

1. The relative position indication is derived from a three turn potentiometer whose output is proportional to rod travel.
2. The potentiometer is synchronized through the gear train driven by a three phase motor.
   a) This three phase motor has its windings tied to phases A, C, and BB of the stator of the CRD motor, synchronizing these two motors.

b) The three turn potentiometer is connected to a stable 5 VDC reference source which allows a 0 to 5 VDC linear signal to be fed from the potentiometer to an operational amplifier.

3. When the rod is at the bottom of the core, the relative position indicator can be reset to zero by a reset pulser located on the Position Indicator Panel (PI) to establish the zero point as a reference.

4. Every two steps of the CRD motor results in one step of the relative position stepping motor and a corresponding change in the potentiometer output.

5. The accuracy is ±2.5 inches and is dependent upon the accuracy of the three turn potentiometer.

6. This RPI 0 to 5 VDC output signal is fed to the same switch as API which will allow the operator to select which indication system will be used as an input to the **Computer** and the **PI panel**.

7. The RPI average signals for Groups 5 thru 7 are used as reference signals to supply indication to the **Sequence Monitor**.

    1) Should the rod groups fail to operate in sequence, resulting in overlap to be <20% or >30% between sequential groups, the sequence monitor will cause:

        - Sequence Fault statalarm (1SA1/E1)
        - Sequence Inhibit lamp on the Diamond panel illuminate
        - Diamond panel will trip to "hand" if in automatic.

D. Asymmetric Alarm and Fault

1. Both alarms and fault circuits are identical except for adjustment of the feedback resistors. A signal will be generated when a rod becomes misaligned with its group average by a preset amount .

    a) The **ALARM**; 1SA2/B10, CRD Position Error, is adjusted for 7 inches deviation from the group average.

    b) The **FAULT** is adjusted for 9 inches deviation from the group.

        1) The Fault logic provides input into the ICS Asymmetric Rod Runback circuitry.

2. The misaligned rod is calculated in the group average as fed by API.

Exam Question Report

27-Jan-99

Question ID:	IC318	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC318				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-CRI - CRD Instrumentation
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	LRO = 9; SRO = 9 Reference: ANO 94 NRC EXAM	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

WHICH ONE (1) of the following is the MINIMUM individual rod position deviation that will generate the CRD position error statalarm and give the associated fault lamp status listed below?

A) SEVEN (7) inches from its RPI group average and the ASYMM FAULT lamp on the Diamond Control panel is OFF.

B) NINE (9) inches from its API group average and the individual FAULT lamp on the Position Indicating panel is OFF.

C) SEVEN (7) inches from its API group average and the individual FAULT lamp on the Position Indicating panel is ON.

D) NINE (9) inches from its RPI group average and the ASYMM FAULT lamp on the Diamond Control panel is ON.

Answer

C Ref: ANO 94 NRC Exam

Lessons

ID	Description
IC-CRI	Control Rod Indication (IC-CRI)

Enabling Objectives

ID	Description

Referenced Documents

QUESTION # 80

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	T1	______
Group #	G2	______
K/A #	000007 A1.03	
Importance Rating	4.2	______

Technical Reference(s): **CP-605, SAE-L39**
EAP-E35

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **SAE-L39 #5 / EAP-E35 T #1**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 80

RO ONLY

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 100%
- 475 EFPD

CURRENT CONDITIONS:

- 2TA and 2TB lockout
- Core Subcooling Margin = 58°F

If an RCS cooldown is initiated at 0212, which ONE of the following is correct?

One (1) hour later (0312) RC temperature should be monitored using _____ indication and _____ is the lowest RCS temperature allowed that does not exceed the maximum allowable cooldown rate.

A. CETC / 495°F

B. CETC / 535°F

C. T_C / 460°F

D. T_C / 500°F

1 POINT

QUESTION # 80

000007 A1.03 RO ONLY GCW 2-7-00 (GTH/JRS)

A. Incorrect - uses CETCs (normal indication with no RCPs) and assumes 45°F/half-hour (normal C/D rate) and starting at 585°F CETCs which is 550T_C "post trip Tc" + 35 degree ΔT (normal end of life natural circulation ΔT).

B. Incorrect - uses CETCs (normal indication with no RCPs) and assumes 25°F/half-hour and starting at 585°F which is 550 "post trip Tc" + 35 degree ΔT (normal end of life natural circulation ΔT).

C. Incorrect - uses T_C a C/D rate of 45°F/half-hour (normal forced circulation C/D rate), starting at 550°F which is Tc post trip.

D. Correct - use T_C on a natural circulation C/D per EOP section 605 and directs a C/D rate < 25°F/half-hour and starting at 550°F, which is Tc post trip.

LESSON PLAN OBJECTIVES

Terminal Objectives

1. Describe the use of CP-605 (Subcooled Cooldown) of the Emergency Operating Procedure in order to perform the required actions of a Nuclear Control Operator during an event involving the use of this procedure.
2. Describe the use of CP-605 (Subcooled Cooldown) of the Emergency Operating Procedure in order to direct the required actions of a Nuclear Control Operator during an event involving the use of this procedure.

Enabling Objectives

1. Recognize that CP-605 can only be entered from the Solid Plant Cooldown section of the EOP. (R1)
2. Explain why a plant cooldown using natural circulation cooling should be avoided if possible. (R2)
3. Explain the bases for the following actions associated with a natural circulation cooldown: (R3)
   - 3.1 maintaining RCS pressure at 2155 psig when $T_h > 450°F$
   - 3.2 maintaining subcooled margin > 150° when $T_h < 450°$ F
4. Explain how opening the RV Head Vents can help prevent void formation in the RV Head during a natural circulation cooldown. (R4)
5. List three methods that may be used to eliminate a void in the RCS outside of the PZR, other than ambient cooling, and discuss the advantages and disadvantages of each. (R5)
6. State the main disadvantage involved with using ambient cooling to eliminate a void in the RCS. (R6)
7. Evaluate plant conditions for indication of possible RCS voids. (R7)

8. Maximize RB cooling if RB pressure is < 3psig (Step 7.8)
    a) Since the RV Head Vents discharge to the RBCU duct work in the RB, maximum cooling is provided to help remove this energy.
    b) It must be recognized that opening the RV Head Vents may eventually create a hostile RB environment that can challenge the availability of instrumentation, since some instrumentation is not considered environmentally qualified. Elevated RB temperatures may also require manual temperature compensation calculations for the SG level instrumentation being used, per the EOP enclosure.
9. Establish RCS P/T to prevent RV head voiding (Steps 7.9 and 7.10)
    a) If Th and CETCs are > 450°F, maintain RCS pressure at 2155 psig; IF Th and CETCs < 450°F, maintain loop SCMs > 150°F.
        1) The object is to maintain RCS pressure high enough to prevent void formations in the system during C/D by providing a large saturation margin for the RCS.
        2) Maintaining pressure high also helps to promote flow through the RV Head Vents and therefore helps increase flow around the stagnant portions in the RV Head area to promote cooling in this area.
        3) After the C/D has been underway for some time (Th and CETCs are < 450°F) RCS pressure must be allowed to decrease to meet NDT curve requirements. By maintaining SCMs above 150°F for the remainder of the C/D, adequate margin to saturation is accomplished while NDT concerns are considered. (Step 7.14)
10. Close breakers for and Open the RV Head Vents. (Step 7.11and 7.12)
    a) Duke Power Company has made the commitment that a natural circulation cooldown will not be performed unless the RV Head Vents are maintained open for reasons already discussed.
    b) Will be kept open until LPI is in service.
    c) Promotes flow through the RV head area, which would otherwise have very little flow during a NCC. This promotes cooling in the RV head area, and minimizes the likelihood of a RV head void.
11. The maximum allowable C/D rate while in natural circ is < 25°F/half-hour. Tcold is used for the cooldown. (Step 7.13)
    a) Permits RV Head cooling to keep up with the rest of the RCS.
12. PZR level is maintained between 100-300" to ensure positive RCS pressure control throughout the C/D. (Step 7.14)

EXERCISE GUIDE OBJECTIVES

Terminal Objectives

1. When 1LPSW-6 fails closed, manually trip the reactor and all operating reactor coolant pumps prior to exceeding any limits, stabilize the plant at hot shutdown in natural circulation, and initiate an RCS natural circulation cooldown of less than 25°F per 1/2 hour in accordance with EP/1/A/1800/01, Emergency Operating Procedure.

Enabling Objectives

1. Based on control room indication, correctly diagnose that 1LPSW-6 has failed closed: (R1)
   - 1.1 Computer alarm
   - 1.2 Change of state on control switch
2. Refer to the limits and precautions of OP/1/A/1103/06, Reactor Coolant Pump Operation, to determine the criteria for securing reactor coolant pumps (RCPs) due to high motor bearing temperatures. (R2)
3. When it has been determined that manual tripping of the RCPs is required, manually trip the reactor first to prevent challenging the Reactor Protective System. (R3)
4. After the RCPs have been secured, monitor plant systems and indications to verify that natural circulation is being established, taking manual control of feedwater and/or turbine bypass valves as necessary to prevent excessive heat transfer. (R4)
   - 4.1 Steam generator level increasing to 50% on the operating range
   - 4.2 Startup feedwater control valves limited to 40% demand
   - 4.3 Cold leg RCS temperatures tracking saturation temperature for existing steam generator pressure (~ 555° F for 1010 psig)
   - 4.4 30° to 40° ΔT between the RCS hot leg and cold leg temperatures.
   - 4.5 Turbine bypass valves maintaining turbine header pressure at ~ 1010 psig.
5. Initiate a natural circulation cooldown, open the reactor vessel vent valves and throttle open the turbine bypass valves to establish and maintain a < 25° F per 1/2 hour cooldown rate while maintaining RCS pressure at 2155 psig. (R5)

7.11 Open RxV Head vents:

_____ 1RC-159 (RXV HEAD VENT)

_____ 1RC-160 (RXV HEAD VENT).

_____ 7.12 Using T_c, establish and maintain < 25°F/ ½ hr cooldown rate.

_____ 7.13 Maintain PZR level between 100-300".

_____ 7.14 <u>WHEN</u> T_h and CETCs ≤ 450°F,

<u>THEN</u> reduce RCS pressure while maintaining the following:

- Loop subcooling margins > 150°F.
- RCS pressure (based on T_C) < NDT limit:
  - **REFER TO** the appropriate Encl 7.1, "P/T Curves."

_____ 7.15 Monitor Vessel Head Level for void formation:

- Maintain Vessel Head Level > 163" by reducing cooldown and depressurization rates.

NOTE 7.16:	RxV Head venting must be maintained to prevent void formation until the LPI system is in operation.

_____ 7.16 Continue cooldown to MODE 5:

- **REFER TO** OP/1/A/1102/010 (Controlling Procedure For Unit Shutdown).

QUESTION # 81

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	_____
	Group #	**2**	_____
	K/A #	**000011 A2.02**	
	Importance Rating	**3.3**	_____

Technical Reference(s): **EOP Rule #7**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E47 OBJ. #15**

Question Source:
- Bank # ________
- Modified Bank # ________
- New ___X___

Question History:
- Previous NRC Exam ________
- Previous Quiz / Test ________

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis _____

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 __X__

Comments:

1 POINT

QUESTION # 81

RO ONLY

Unit 3 plant conditions:

INITIAL CONDITIONS:

- A small break LOCA occurred at 0100
- RCS pressure decreased to 900 psig and steady
- Reactor building pressure peaked at 16 psig and is slowly decreasing
- BWST level = 33 feet decreasing

CURRENT CONDITIONS:
All operator actions required by EOPs performed ~~[illegible]~~ in a timely manner -

- Time = 01:50
- RCS pressure rapidly decreases to 30 psig
- Reactor building pressure rapidly increases to 30 psig
- BWST level = 30 feet decreasing

Which ONE of the following is the correct operator action in response to these conditions?

A. Start all LPI pumps.

B. Start the "A" and "B" LPI pumps.

C. Reset ES analog channels and allow LPI to actuate automatically.

D. Throttle Reactor Building Spray to less than 1000 gpm flow per header.

1 POINT

QUESTION # 81

000011 A2.02

The Tech Spec setpoint for LPI actuation is $\geq$500 psig. The shutoff head of the LPI pumps is < 200 psig. ES 3&4 start A&B LPIP and align to RV due to RB pressure > 3 psig. LPIPs are manually secured following ES per Rule # 7 when RCS pressure is above LPIP shutoff head.

A. Incorrect, Only the “A” and “B” LPIP should be started. The “C” LPIP is only used if both “A” and “B” LPIPs fail.

B. Correct, LPI pumps started when ES occurred but should have been secured after 30 minutes operation at > shutoff head (200 psig). ES is in manual and requires operator action to re-initiate LPI.

C. Incorrect, When RCS pressure is < 200 psig the operator is responsible for manually starting the LPI pumps. The LPIPs will not actuate automatically due to high RB pressure as the ES digital channels will not reset automatically.

D. Incorrect, BWST is not at the proper level for swapping suction to the RBES and throttling RBS to < 1000 gpm/header.

ENABLING OBJECTIVES

13. Explain how Rule 7 provides LPI pump suction line overpressure protection in the event that one of the LPI pumps fails during ES actuation. (R-16)
14. Discuss the actions in Rule 7 to establish LPI flow capability in both trains of LPI with 1 or 2 LPI pumps available.(R-17)

    14.1 Explain why these actions are TIME CRITICAL

15. Recognize that SRO approval is not required prior to securing LPI pumps following ES actuation, and that the crew should be notified of this action. (R-12)
16. Recognize that high RB temperatures will cause inaccurate level indications in the control room. (R-6)
17. Demonstrate the ability to use EOP Enclosure 7.5 to correct level indications for degraded containment. (R-7)
18. Differentiate between plant conditions that allow the operator to exit Rule 7 and those conditions that will require Rule 7 to remain in the "OPEN" status.(R-18)

L. **Step 9.12** determines if the LPI pumps are running against shutoff head and provides guidance to stop the LPI pumps.

   1. This step is TIME CRITICAL (30 minutes, from the time the LPI pumps were started) to prevent pump damage (over heating) from running at shutoff head conditions.

M. **Step 9.12.1** requires the operator to inform the rest of the crew that the LPI pumps have been stopped and must be restarted when RCS pressure decreases below LPI pumps shutoff head.

2.10 **Step 10** has the operator start both Outside Air Booster Fans:

A. This step is TIME CRITICAL (30 minutes). The issue here is Operator dose in the Control Room area.

2.11 **Step 11** If RCS Pressure is > 550 psig procedure step directs the operator to re-establish RCP support systems.

A. If RCS Pressure is < 550 psig, the RCS leak size is large enough to depressurize the RCS and the need to restart a RCP should not occur.

B. This Step is TIME CRITICAL (10 Minutes from the time of the break) to establishes RCP supports in the anticipation of a RCP restart is needed.

   1. A RCP restart may be needed to ;

      a) Mitigate PTS for a MSLB inside the Reactor Building.

      b) SBLOCAs in which RCPs are needed for RCS cooldown.

2.12 **Step 12** checks for proper CFT operation.

A. **Step 12.1** provides guidance that if either CFT fails to dump, the operator is to open or verify open CF-1 & CF-2.

   1. Step 14.1 is to insure that CF-1 and CF-2 are open when CFTs would be required.

   2. This step is required because CF-1 and CF-2 may be closed during the normal startup/shutdown process.

      a) The EOP still applies for transients when shutdown.

_____ 9.10 **IF** AT ANY TIME both of the following valves are open:

- 1LP-9 (1C LPIP DISCH TO 1A LPI HDR)
- 1LP-10 (1C LPIP DISCH TO 1B LPI HDR),

AND both of the following valves operate from the Control Room:

- 1LP-12 (1A LPI COOLER OUTLET)
- 1LP-14 (1B LPI COOLER OUTLET),

THEN maximize LPI flow in each header ≤ 1100 gpm.

_____ 9.11 **IF** AT ANY TIME LPI flow is required,

AND NEITHER of the following LPI Pumps are available:

- 1A LPI Pump
- 1B LPI Pump,

THEN **REFER TO** AP/1/A/1700/007, (Loss of LPI System).

CAUTION 9.12: LPI Pump damage may occur if operated in excess of 30 minutes against a shutoff head.

_____ 9.12 **IF** LPI Pumps are operating against a shutoff head,

THEN shutdown LPI Pumps:

_____ 1A LPI Pump

_____ 1B LPI Pump.

_____ 9.12.1 Notify Control Room crew that LPI Pumps must be restarted when RCS pressure decreases below LPI Pump shutoff head.

QUESTION # 82

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	____
	Group #	**2**	____
	K/A #	**000038 A1.29**	
	Importance Rating	**3.5**	____

Technical Reference(s): **EAP-E24**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E24 OBJ. #9**

Question Source:
Bank # ____
Modified Bank # **EAP-039**
New ____

Question History:
Previous NRC Exam ____
Previous Quiz / Test ____

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ____

10 CFR Part 55 Content:
55.41 __X_
55.43 ____

Comments:

1 POINT

QUESTION # 82

RO ONLY

Unit 1 plant conditions:

- 350 gpm Steam Generator Tube Rupture is in progress 1A1 & 1B2 RUNNING
- ~~RCPS secured~~ 1RC-4 Failed closed.

Which ONE of the following is correct if the RCS cooldown is delayed?

BWST volume is ______ / and the leak rate ______.

A. recoverable / can be reduced

B. recoverable / cannot be reduced

C. not recoverable / can be reduced

D. not recoverable / cannot be reduced

NO PZR SPRAY AVAILABLE (But can use PORV)

1 POINT

QUESTION # 82

000038 A1.29 (3.5/3.3) RO ONLY RSI/PRA 4-25-00

A. Incorrect - During a SGTR BWST inventory is lost when it leaks through the ruptured tube. The SGTR event is different than a SBLOCA or LBLOCA where inventory is recoverable via the Emergency Sump and long term recirculation. Decreasing SCM (RCS Pressure) can reduce the leak rate.

B. Incorrect - During a SGTR BWST inventory is lost when it leaks through the ruptured tube and is not recoverable. The second part is true, decreasing SCM (RCS Pressure) can reduce the leak rate.

C. Correct – During a SGTR BWST inventory is lost when it leaks through the ruptured tube. Decreasing SCM (RCS Pressure) can reduce the leak rate.

D. Incorrect – The first part is true BWST inventory is lost. However decreasing SCM (RCS Pressure) can reduce the leak rate.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

Describe the use of Section 504 (SG Tube Leak) of the Emergency Operating Procedure in order to perform the required actions during an event involving a primary to secondary leak greater than Tech. Spec. limits. Be able to discuss Section 504 procedure steps and their bases in an oral or written format. Discuss in an overview format how Section 504 mitigates a SGTR event and places the plant into MODE 5 with the affected SG(s) isolated and heat removal via LPI.

ENABLING OBJECTIVES

1. Using an overview format describe the intent of this procedure. (R1)
2. Given a set of conditions, be able to identify a OTSG with a tube leak. (R2)
3. Explain why it is important to place the TBVs in hand just prior to manually tripping the reactor. (R3)
4. During a SGTL shutdown explain the importance of maintaining PZR levels $\geq$ 220, $\geq$ 200, and > 80 inches at different times during the shutdown and cooldown to 532°F. (R4)
5. Explain how the affected SG is isolated and why it is not done until an RCS temperature of 532°F is reached. (R5)
6. Explain the reason for maintaining the subcooled margin as close as possible to 0°F during the cooldown. (R6)
7. Given a set of conditions determine the proper Cooldown Plateaus and state the bases behind the specified plateaus. (R13)
8. Given a situation understand that throttling FDW flow may be the only method for controlling the cooldown rate when feeding a SG with a steam leak. (R7)
9. Describe the reason for the increased concern over available BWST inventory during a SGTR event if the cooldown rate is not properly maintained. (R8)
10. Given a set of plant conditions calculate and explain the basis for the limit(s) for SG tube-to-shell ΔT. (R9)
11. Explain how an uncontrolled release to the environment can possibly result if the MS lines are allowed to fill with primary water from the affected SG. (R10)

NOTE Step 28.2: SG pressure control for a SG with a steam leak may be difficult or impossible. Throttling FDW flow may be the only method for controlling cooldown rate.

28.2 IF cooldown using SG with the steam leak is selected **THEN** perform the following:

A. **REFER TO** OP/1102/01 (Controlling procedure for Unit Shutdown)

B. Maintain cooldown rate < 50°F/½ Hr

1. It is important to maintain a cooldown rate as close to 50°F/½ Hr as possible. The goal is to reach LPI DHR operations as quickly as possible so the SG(s) can be isolated. The makeup water being supplied to the RCS is NON-RECOVERABLE.

28.3 IF a steam line rupture exist outside the RB THEN clear personnel from the affected area before feeding any affected SG.

1. The area of the steam leak will be totally covered with steam as soon as the SG steaming is commenced.

28.4 If "A" SG has a Steam leak, Start the "A" MDEFDWP and Throttle EFDW flow with FDW 315 ("A" EFDW Control).

28.5 If "B" SG has a Steam leak, Start the "B" MDEFDWP and Throttle EFDW flow with FDW 316 ("B" EFDW Control).

28.6 Control the cooldown by the following methods:

- Throttle EFDW flow

  For large steam leaks the cooldown may be controlled by throttling EFDW supply to the affected SG.

- Use the TBVs to cool down the RCS.

  TBVs will be used if the steam leak is small enough that the amount of steam leaking cannot maintain an adequate cooldown rate.

A. IF TBVs are not available THEN dispatch two operators to perform ENCLOSURE 7.7 "Operation of the ADVs".

NOTE 28.7: Continued steaming to the Reactor Building may cause RB P/T concerns

28.7 IF any SG is being steamed to the RB THEN notify the TSC

This is a concern in that RB pressure will continue to increase as an effected SG continues to steam inside the RB. The TSC will formulate plans to prevent over pressurization of the RB.

NOTE 29: Use Tc with RCPs or CETCs without RCPs

Exam Question Report

27-Jan-99

Question ID:	EAP039	Revision No:	0	Revision Date	10/29/1999
Question Description:	EAP039				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-E24 - SG Tube leak
Last Used Date:	01/26/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	LRO = 1; SRO = 1	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exean Question Report

27-Jan-99

Question

During an SGTR, if the cooldown of the RCS is stopped for a long period of time, you as the operator should be concerned with the BWST level due to which ONE of the following? (.25)

A) The BWST's inadequate inventory during SGTR's.

B) The BWST's inventory is not recoverable during SGTR's.

C) The BWST's concentration can't maintain 1% delta k/k SDM during SGTR's.

D) The BWST's concentration will maintain 1% delta k/k SDM but only for a brief time period.

Answer

B - correct, BWST inventory is not recoverable. Therefore a sustained shutdown with a SGTR could empty the BWST and the BWST is the fastest source of highly borated water.
A - incorrect since the BWST will, unless a sustained shutdown without leak isolation depletes the BWST, have plenty of inventory for a SGTR.
C & D - incorrect, BWST concentration will provide enough boron for > 1% SDM down to Mode 1.

Lessons

ID	Description
EAP-E24	SG Tube Leak EAP-E24

Enabling Objectives

ID	Description

QUESTION # 83

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	____
Group #	**1**	____
K/A #	**001000 K5.10**	
Importance Rating	**3.3**	____

Technical Reference(s): **PT/600/01, COLR**
Unit #1 Cycle #19

Proposed references to be provided to applicants during examination: **p.24 –27 of COLR**

Learning Objective: **NONE**

Question Source:
- Bank # ____
- Modified Bank # ____
- New __X__

Question History:
- Previous NRC Exam ____
- Previous Quiz / Test ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ____

Comments:

1 POINT

QUESTION # 83

RO ONLY

Unit 1 plant conditions:

- Reactor power = 60%
- 1A2 RCP secured
- PT/1/A/600/01, Periodic Instrumentation Surveillance, Enclosure 13.1, Periodic Checks Schedule Sheet (RCS greater than 200 degrees F) in progress
- Unit 1 OATC is verifying adequate SDM

Which ONE of the following is the MINIMUM allowable position for Control Rods?

SEE ATTACHMENT

Group _____ / _______ withdrawn.

A. 6 / 63%

B. 7 / 3%

C. 7 / 22%

D. 7 / 55%

1 POINT

QUESTION # 83

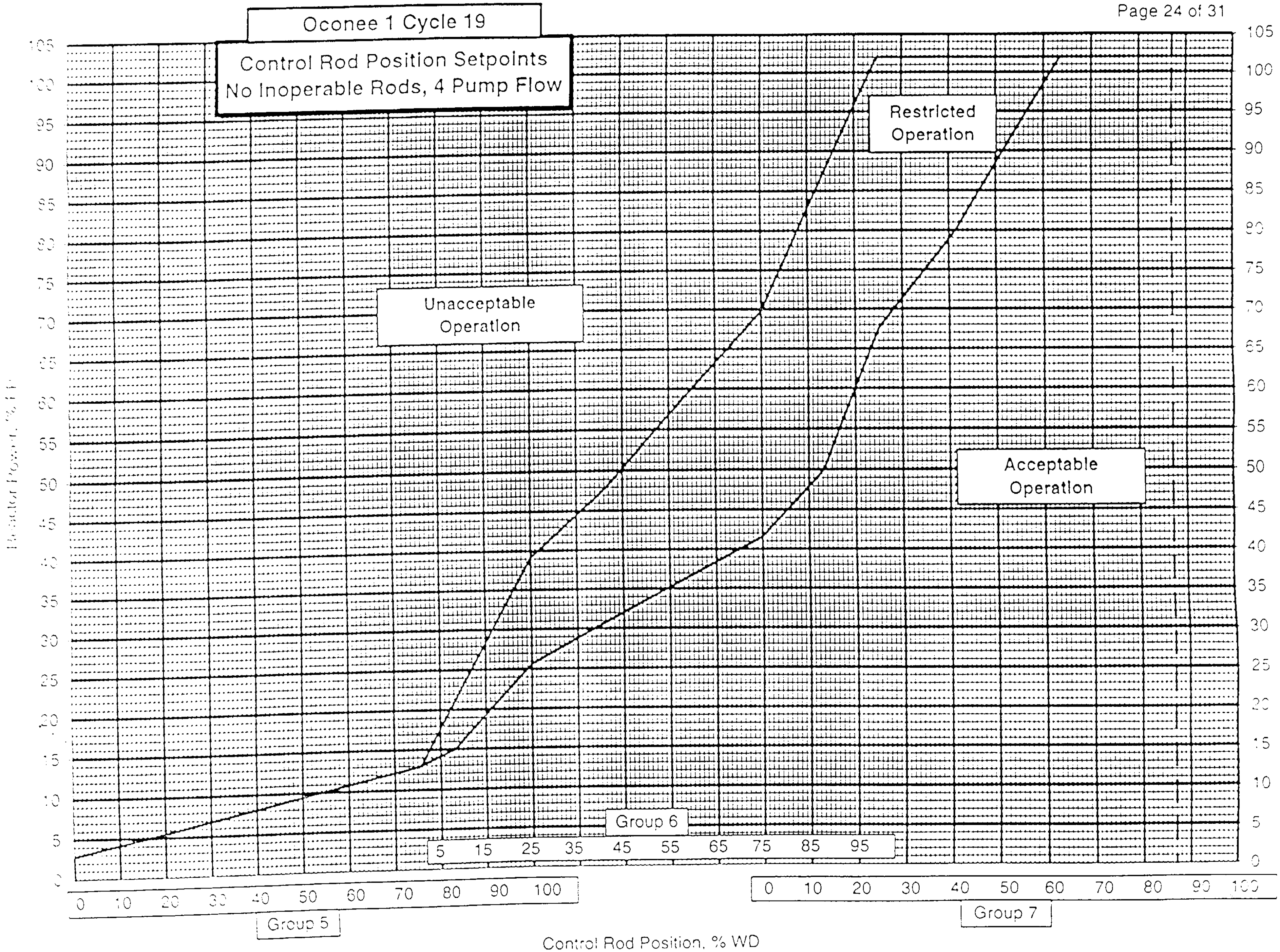
001000 K5.10 (3.3/3.7) (RO ONLY) RSI/PRA 4-25-00
Attachment Page 24-27 of COLOR

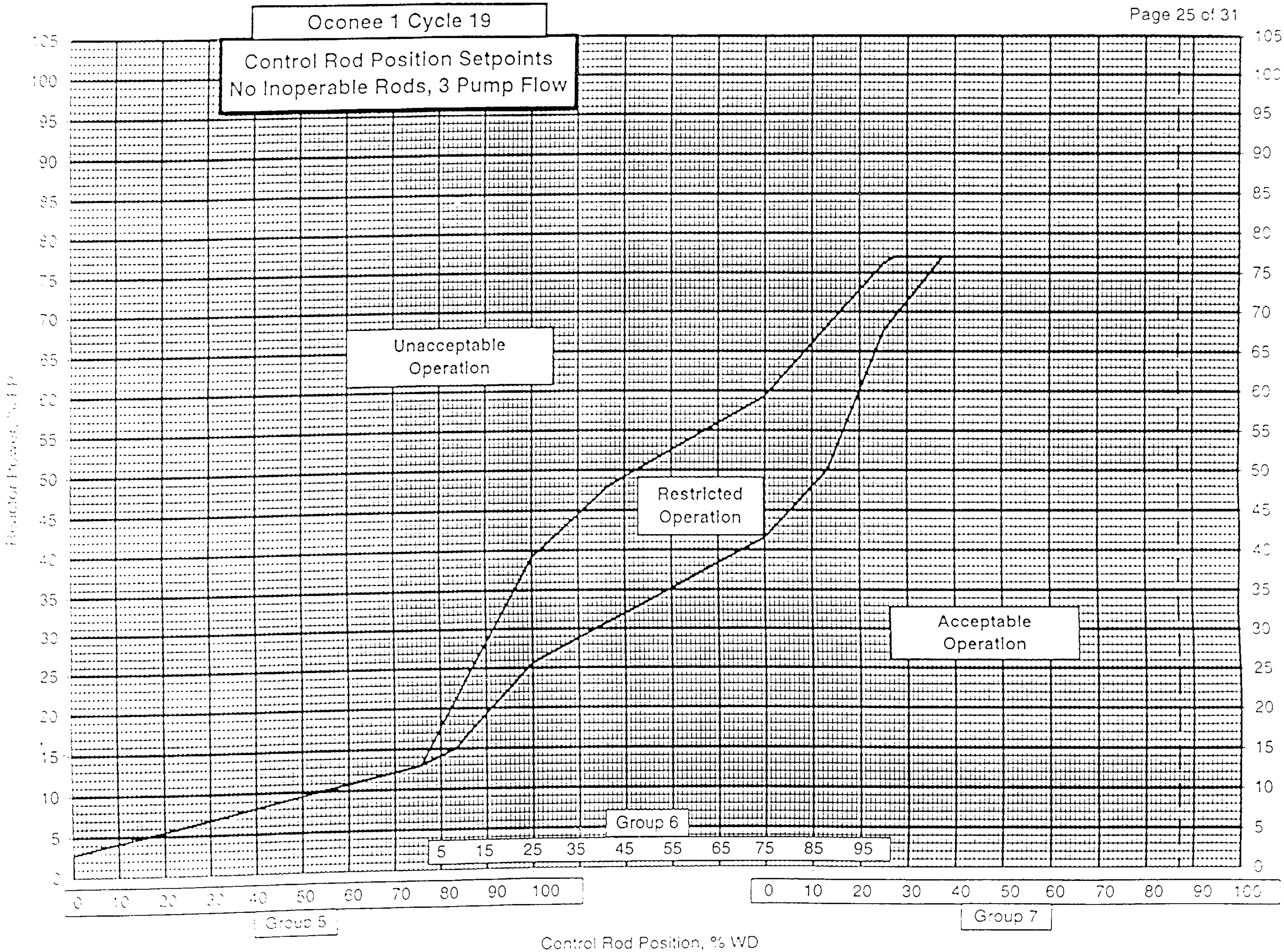
A. Incorrect – Would be correct for 4 RCP flow (page 24 of 31).

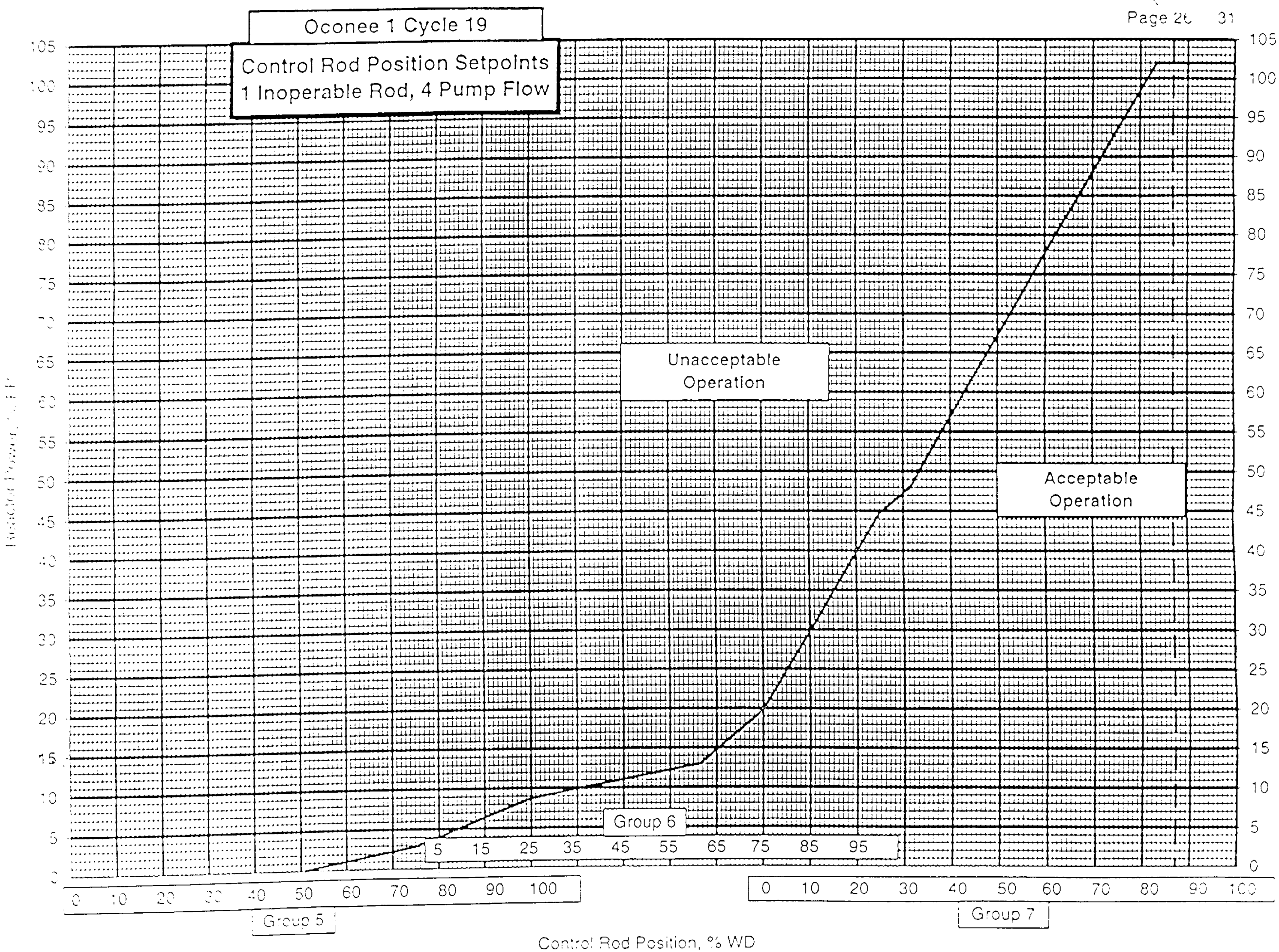
B. Correct - From the SDM curve (page 25 of 31) the minimum CR position is 3% on Group 7.

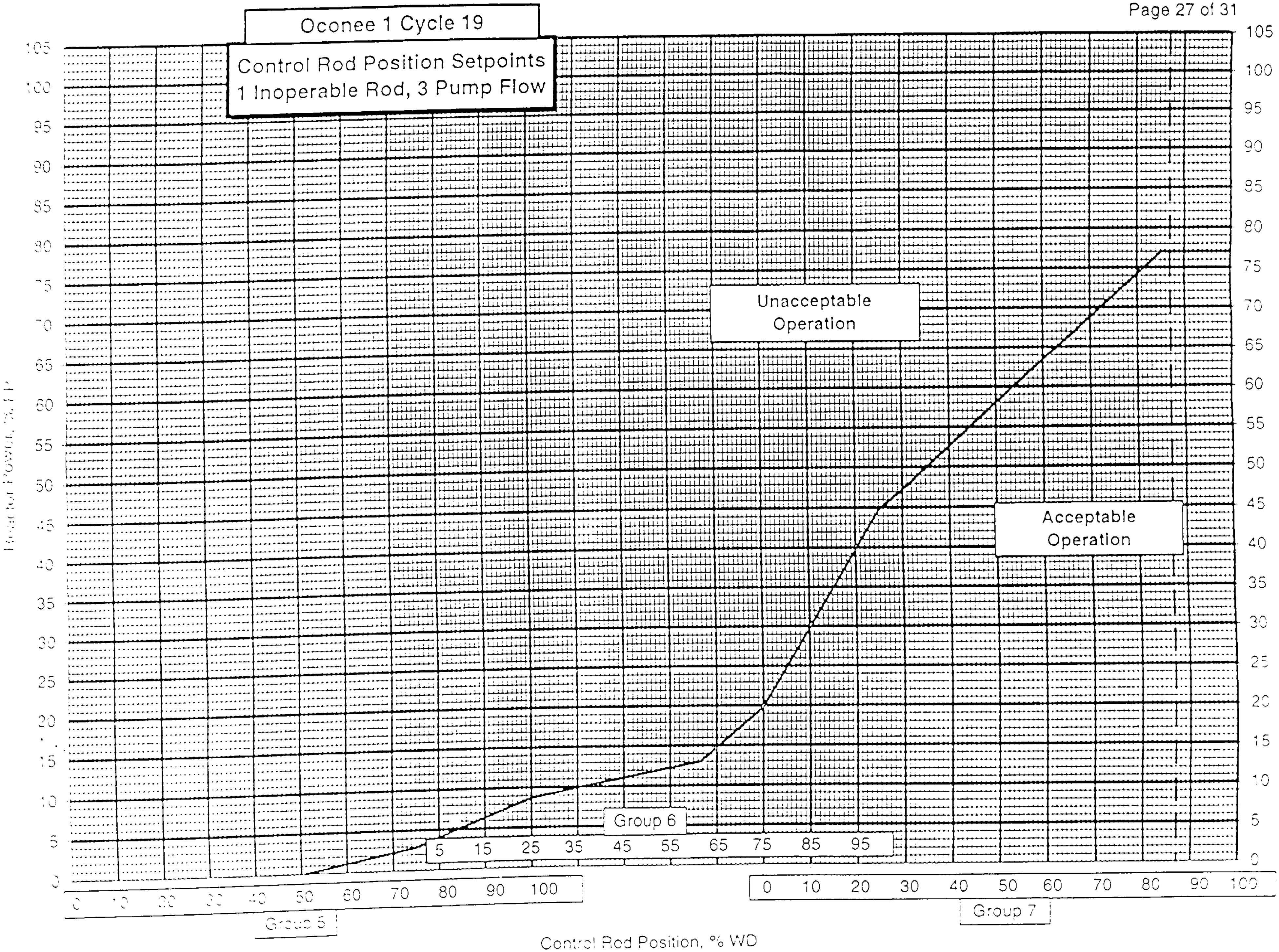
C. Incorrect - From the SDM curve (page 25 of 31) the minimum allowable insertion limit for Group 7 is 3% for 60% power. 22% is correct based on the 2 hour curve.

D. Incorrect - Would be correct for 3 RCP flow with one inoperable CR (page 27 of 31).


Oconee 1 Cycle 19
Control Rod Position Setpoints
No Inoperable Rods, 4 Pump Flow
Restricted Operation
Unacceptable Operation
Acceptable Operation
Reactor Power, % FP
Group 5
0 10 20 30 40 50 60 70 80 90 100
Group 6
5 15 25 35 45 55 65 75 85 95
Group 7
0 10 20 30 40 50 60 70 80 90 100
Control Rod Position, % WD


Oconee 1 Cycle 19
Control Rod Position Setpoints
No Inoperable Rods, 3 Pump Flow
Unacceptable Operation
Restricted Operation
Acceptable Operation
Reactor Power, % FP
Group 5
Group 6
Group 7
Control Rod Position, % WD


Oconee 1 Cycle 19
Control Rod Position Setpoints
1 Inoperable Rod, 4 Pump Flow
Unacceptable Operation
Acceptable Operation
Group 5
Group 6
Group 7
Control Rod Position, % WD


Oconee 1 Cycle 19
Control Rod Position Setpoints
1 Inoperable Rod, 3 Pump Flow
Unacceptable Operation
Acceptable Operation
Reactor Power, % FP
105
100
95
90
85
80
75
70
65
60
55
50
45
40
35
30
25
20
15
10
5
0
Group 6
5 15 25 35 45 55 65 75 85 95
0 10 20 30 40 50 60 70 80 90 100
Group 5
0 10 20 30 40 50 60 70 80 90 100
Group 7
Control Rod Position, % WD

QUESTION # 84

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	____
	Group #	**1**	____
	K/A #	**003000 K5.03**	____
	Importance Rating	**3.1**	____

Technical Reference(s): **IC-RCI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-RCI OBJ. #7**

Question Source:
- Bank # ____
- Modified Bank # ____
- New __X__

Question History:
- Previous NRC Exam ____
- Previous Quiz / Test ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ____

Comments:

1 POINT

QUESTION # 84

RO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Power = 65%
- Loop A Tave = 581°F
- Loop B Tave = 577°F

CURRENT CONDITIONS:

- 1A2 RCP tripped
- 1SA-2/A-3 ("A" Loop RC Flow LOW) actuates

Which ONE of the following is correct?

ASSUME Reactor Power remains constant at <u>65%</u> and loop A and B Tave values remain the same.

Prior to 1A2 RCP tripping, Tave input to ICS was _______°F. Following 1A2 RCP tripping, Tave input to ICS is ________°F.

A. 579 / 579

B. 579 / 577

C. 581 / 579

D. 581 / 581

Low load limits

1 POINT

QUESTION # 84

003000 K5.03 RO ONLY RSI/GCW NEW 04/27/00

A. Incorrect – Prior to the RCP trip the ICS selects Unit Tave, which is the average of the 2 Loops. This would be correct if the Tave input to ICS did not swap to the Loop with the highest RC flow when a RCP is tripped.

B. Correct - After RCP 1A2 trips Tave input to ICS will shift to the "B" Loop Tave. B Loop has 2 RCPs operating vs. "A" Loop 1 RCP operating.

C. Incorrect – This would be correct if the highest Tave was input to the ICS during 4 RCP operation. If ICS used an average of the two Loops the second part of the distracter would be correct.

D. Incorrect - This would be correct if the highest Tave was input to the ICS during 4 RCP operation and a swap did not occur when RC Loop flows are different. During SG Low Level limit operation this would be correct as the Tave input selects the highest Loop Tave.

4. During all modes of operation, analyze proper operation of "Dixon" meters and differentiate between a loss of power, overranged, and underranged instrument (R21)

5. Given a set of conditions describe the required operator actions when selecting an alternate controlling signal. (R20)

6. Applying the knowledge of simplified instrumentation drawings be able to determine how various indications and control functions are processed for RCS temperature, pressure, level and flow including: (R2, 3, 62)

   6.1 Range of the indicator

   6.2 Source of the signal

7. Given a set of conditions analyze proper operation of RCS TEMPERATURE indications that the operator uses to monitor and control unit operation including the following: (R3, 4, 5, 6, 10)

   7.1 RCS T-hot

   7.2 RCS T-cold

   7.3 Core exit temperature (CETCs)

   7.4 Pressurizer temperature

8. Given a set of conditions analyze proper operation of RCS PRESSURE indications that the operator uses to monitor and control unit operation including the following: (R6, 7, 9, 63, 10)

   8.1 RCS Loops

   8.2 ICCM WR Pressure

   8.3 Low Range Cooldown

9. Given a set of conditions analyze proper operation of RCS LEVEL indications that the operator uses to monitor and control unit operation including the following: (R13, 15, 16, 17, 18)

   9.1 Pressurizer level and pressure

   9.2 Reactor Vessel (LT-5)

   9.3 Ultrasonic Level Indication (ULI)

   9.4 Tygon tubing

QUESTION # 85

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	_____
	Group #	**1**	_____
	K/A #	**004000 G2.2.22**	
	Importance Rating	**3.4**	_____

Technical Reference(s): **CP-016**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CP-016 OBJ. #4 & #11**

Question Source:
- Bank # **CP-194**
- Modified Bank # ________
- New ________

Question History:
- Previous NRC Exam ________
- Previous Quiz / Test ___X___

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 _____

Comments:

1 POINT

QUESTION # 85

RO ONLY

Unit 1 plant conditions are as follows:

- 100% power
- RCS Boron concentration = 600 ppm
- "1A" BHUT concentration = 1000 ppm
- "1B" BHUT concentration = 0 ppm

It is desired to add 350 gallons to the LDST tank with **no** resulting rod movement.

Which ONE of the following describes the correct "1A" BHUT and the "1B" BHUT additions to the LDST, and the sequence in which they should be added?

A. 140 gallons from "A" BHUT then 210 gallons from "B" BHUT

B. 210 gallons from "A" BHUT then 140 gallons from "B" BHUT

C. 140 gallons from "B" BHUT then 210 gallons from "A" BHUT

D. 210 gallons from "B" BHUT then 140 gallons from "A" BHUT

1 POINT

QUESTION # 85

004000 G2.2.22 RO ONLY Bank CP194 exam #1 GCW 04/27/00

$C_1V_1 + C_2V_2 = C_fV_f$

(1000 ppm)(x) + 0 = (600 ppm)(350 gal)
1000x = 210000
x = **210 gals ("A" BHUT)** 350 gal – 210 gals = **140 gals ("B" BHUT)**

A. Incorrect, calculation backwards.

B. Correct, calculation is correct and high boron water is added first, with demin water added last to flush the line.

C. Incorrect, correct numbers but wrong order. Demin water is added last to flush the line.

D. Incorrect, calculation backwards and in wrong order.

OBJECTIVES

TERMINAL OBJECTIVE

The student will be able to relate the use of the Soluble Poison Concentration Control Procedure to overall plant response and coordination with other procedures.

ENABLING OBJECTIVES

1. Perform C_1V_1 calculations for the following situations. (R1)
   1.1 Makeup to LDST
   1.2 Makeup to change RCS concentration – no Tave change
   1.3 Makeup to change RCS concentration – Tave changing
2. Given "Steam Tables" and RCS volume in ft^3 perform calculations to determine amount of makeup water required to be added when cooling down or amount to be letdown when heating up between two RCS temperatures. (R3)
3. Given "Steam Tables", LDST gallons/inch conversion and Pressurizer gallons/inch conversion, calculate makeup volume required to raise Pressurizer level while maintaining LDST level constant and/or change in LDST level for a given change in Pzr level. (R4)
4. Discuss why the higher concentrated boric acid should be completely added prior to adding the lower concentration makeup water during an RCS cooldown. (R2)
5. Given the Feed and Bleed hand calculation formula, perform Feed and Bleed calculations for DW additions and/or using an unused deborating demineralizer. (R5)
6. Analyze for use of the "CBAST (PPM Cl^-) vs. Pumping Rate Per 30 Minute Period" enclosure of OP/O/A/1108/01 (Curves and General Information. (R7)
7. Given the "CBAST (PPM Cl^-) vs. Pumping Rate Per 30 Minute Period" enclosure, perform calculations of CBAST pumping rate using CBAST pump or BHUT pump. (R8)
8. Examine the process, steps, and considerations when using CBAST to borate the local RCS or another unit's RCS. (R9)
9. State the considerations particular to using 'A' BHUT pump to pump CBAST. (R10)
10. State the considerations particular to using 'B' BHUT pump to pump CBAST. (R11)
11. Describe the batch process of makeup from 'A' or 'B' BHUT. (R12)
12. Describe the feed and bleed process of makeup from 'B' BHUT. (R13)

B. Examples of Calculations

1. Makeup to LDST - no concentration change desired

   a) $C_1V_1 + C_2V_2 = C_fV_f$

   b) C_1 = DW or low boric acid concentration source

   c) V_1 = Volume addition of D.W.

   d) C_2 = Boric acid source concentration

   e) V_2 = Volume addition from boric acid source

   f) C_f = LDST boron concentration (before and after)

   g) V_f = LDST volume change $= V_1 + V_2$

   h) Since V_f is known (amount of water needed to bring LDST level to desired level)

   $$V_1 = V_f - V_2$$

   i) Substituting $V_f - V_2$ in place of V_1 will allow the equation to be worked with only one unknown.

   j)

   $$C_1(V_f - V_2) + C_2V_2 = C_fV_f$$
   $$C_1V_f - C_1V_2 + C_2V_2 = C_fV_f$$
   $$C_2V_2 - C_1V_2 = C_fV_f - C_1V_f$$
   $$V_2(C_2 - C_1) = V_f(C_f - C_1)$$
   $$V_2 = \frac{V_f(C_f - C_1)}{C_2 - C_1}$$

   k) If C_1 is 0 (Zero) ppmb (i.e., 'B' BHUT), the formula can be simplified to:

   $$V_2 = \frac{V_fC_f}{C_2}$$

   and

   $$V_1 = V_f - V_2$$

l) Assume operator wants to add 600 gallons to LDST from 'A' BHUT @ 900 ppmb and from 'B' BHUT @ 0 ppmb and wants LDST and RCS boron concentration to remain constant at 500 ppmB. How much should be added from each source?

$$V_2 = \frac{V_f C_f}{C_2}$$

$$V_2 = \frac{600\ gal\ (500\ ppm)}{900\ ppm}$$

$$V_2 = 333\ gallons\ of\ A\ BHUT\ at\ 900\ ppm$$

$$V_1 = V_f - V_2 = 600\ gal - 333\ gal$$

$$V_1 = 267\ gallons\ of\ B\ BHUT\ at\ 0\ ppm$$

2. Makeup to Raise or Lower RCS Concentration - no Tave Change

a) This calculation will assume that RCS volume is allowed to increase by the amount of makeup water. It is known that at power RCS volume is held constant. The small volume of makeup added in relation to RCS volume is so small that the calculation will not be affected greatly.

b)

$$C_1 V_1 + C_2 V_2 = C_f V_f$$

V_f = new RCS volume - unknown

C_f = new RCS concentration - desired

V_1 = initial RCS volume

V_2 = makeup volume

C_1 = initial RCS concentration

C_2 = makeup concentration

Exam Question Report

27-Jan-99

Question ID:	CP194	Revision No:	0	Revision Date	10/29/1999
Question Description:	CP194				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	016
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 1; SRO = 1	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Assume the operator wants to add 350 gallons to the LDST and maintain RCS boron concentration constant at 600 ppmB. Which ONE of the following describes the correct amounts to be added from the 'A' BHUT and the 'B' BHUT, and the sequence in which they should be added? (.25)

- 'A' BHUT concentration = 1000 ppmB
- 'B' BHUT concentration = 0 ppmB

A) 140 gal. from 'A' BHUT then 210 gal. from 'B' BHUT

B) 210 gal. from 'A' BHUT then 140 gal. from 'B' BHUT

C) 140 gal. from 'B' BHUT then 210 gal. from 'A' BHUT

D) 210 gal. from 'B' BHUT then 140 gal. from 'A' BHUT

Answer

B

Lessons

ID	Description

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

QUESTION # 86

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	_____
	Group #	**1**	_____
	K/A #	**004000 K1.29**	
	Importance Rating	**3.4**	_____

Technical Reference(s): **PNS-PZR**
IC-RCI

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PZR #26 & #27**

Question Source:
Bank # ________
Modified Bank # ________
New __X__

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 86

RO ONLY

Unit 2 Plant Conditions:

- RCS heatup is in progress
- RCS pressure is 475 psig and increasing
- PORV setpoint selector switch in LOW
- Quench tank level is 84 inches and increasing
- PZR Relief Valve tailpipe temperatures:
  - RC-66 (PORV) = 211°F and slowly increasing
  - RC-67 (PZR Safety RV) = 235°F and increasing
  - RC-68 (PZR Safety RV) = 213°F and slowly increasing

Which ONE of the following would result in these plant conditions?

A. The PORV has lifted as required and has not reseated.

B. The PORV is at the lift setpoint and is "chattering".

C. One of the PZR code safety valves is not properly seated.

D. Both of the PZR code safety valves are "chattering".

1 POINT

QUESTION # 86

004000 K1.29 (3.4/4.0) RO ONLY RSI/PRA 5-1-00

A. Incorrect, - The PORV pressure relief setpoint when selected to LOW is 530 psig. Current RCS pressure is 55 psig below the PORV lift setpoint. Elevated tail pipe temperature is not indicative of a lifting PORV.

B. Incorrect, - Current RCS pressure is 55 psig below the PORV lift setpoint. Tailpipe temperature is normal.

C. Correct, - PORV/Safety tailpipes are monitored by thermocouples. If one valve is leaking, the other tailpipes will tend to heatup due to thermal conductance and close proximity of tailpipes for PORV and safeties. At 475 psig isoenthalphic expansion of code safety leakage yields a maximum temperature of 250 degrees F and provides a lower temperature when ambient losses equalize that heat transfer from RCS via leak. The highest tailpipe temperature indicates the source of the leak. All tailpipe temperatures would increase via conduction heating of the piping due to the common connecting header downstream of the RVs prior to entering the Quench tank.

D. Incorrect, - Safety valve setpoints are not affected by selecting LOW. Safety valve setpoints remain at 2500 psig.

ENABLING OBJECTIVES (continued)

13. Discuss the operation of the pressurizer heaters including: (R7)
    13.1 Three purposes of pressurizer heaters.
    13.2 Purpose and level of interlock associated with pressurizer heaters.
    13.3 On/off setpoints for pressurizer heater banks 2, 3and 4.
14. Describe the physical operation of the PORV including what causes the Pilot Valve to operate and how this causes the PORV to open or close. (R8)
15. Explain the purpose of the two opening setpoints associated with the PORV. (R9)
16. Explain how to manually operate the PORV. (R37)
17. Given a set of conditions, determine operability of the PORV following a loss of power. (R30)
18. Discuss the reason for the pressurizer safeties and their setpoint. (R12)
19. Given a set of plant conditions, determine the response of Pressurizer level. (R14)(R15)
20. Explain the operation of SASS as it relates to pressurizer level control. (R31)
21. Given a set of conditions, determine how pressurizer level control/indication is affected by a loss of SASS and/or ICCM. (R35)
22. Discuss the use of Pressurizer Saturation Pressure Indication by the operator. (R16)
23. Discuss the forming of a pressurizer steam bubble including any precautions to be taken during the evolution. (R17)
24. Given a completed copy of PT/0/A/201/04 PORV Operability Test apply compare data taken to acceptance criteria to determine PORV operability. (R10)(R11)
25. Differentiate between a pressurizer steam space leak and a water space leak. (R32)
26. Given a set of plant conditions, determine the position of the PORV. (R13)
27. Given a set of conditions, calculate the expected PORV discharge temperature. (R34)
28. Given a copy of a Limit and Precaution from OP/A/1103/05, Pressurizer Operation, be able to state the reason for that limit and precaution. (R18)
29. Apply ITS/SLC's rules to determine applicable Conditions and Required actions for a given set of Pressurizer conditions. (R24)

3. Operation of RC-66

   a) With the RC-66 setpoint selector positioned to either HIGH or LOW, an RCS pressure increase to the corresponding pressure setpoint will cause the PORV to automatically open. Once RCS pressure decreases below the setpoint, the valve will automatically close.

   b) RC-66 can also be manually operated.

      1) The PORV can be opened manually by the operator selecting OPEN on the setpoint selector and then depressing the OPEN PERMIT pushbutton.

      2) When the OPEN PERMIT pushbutton is depressed there is a two second time delay before the PORV opens. The pushbutton does not need to be held for the two seconds.

      3) If the PORV has been opened manually and needs to be closed, the operator must select either "LOW" or "HIGH" with the setpoint selector switch. The PORV will then close provided no automatic opening signal is present.

4. PORV operability under various loss of power situations

   a) RC-66 pilot valve DC solenoid is powered from DIB Panelboard breaker #24. Therefore the PORV can be manually operated following a loss of KI (i.e., ICS) power.

   b) The primary control power for automatic operation of the PORV at the high-pressure setpoint is supplied from **KI Auto** power. Backup control power is supplied from **KI Hand** power. The low range RC pressure signal is not an input to ICS. Therefore, operation of the PORV is unaffected by a loss of KI power when LOW is selected on the RC-66 setpoint selector.

      1) If KI AUTO power is lost, control automatically swaps to KI Hand power. When KI Auto power is restored, control automatically swaps back to Auto power.

      2) Loss of KI MANUAL power:

         No swap takes place, as KI Auto power is the primary control power.

      3) Loss of KI Auto and Manual Power

         NR RCS pressure signal will fail low and the PORV will be automatically inoperable. However, the PORV will still be manually operable.

      4) Loss of DIB breaker #24:

         PORV will **NOT** open under any conditions. If the PORV is open and DIB lost, the PORV will close regardless of selector switch position.

D. Pressurizer Code Safety Relief Valves (PNS-PZR-1)

1. Two pressurizer code safety valves (RC-67 & 68) are mounted on individual nozzles on top of the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston.

2. The pressurizer code relief valves are set at 2500 psig ± 1% to ensure that the 2750 psig high pressure safety limit is not exceeded. The codes relieve through a common line with the PORV to the quench tank.

E. Relief Valve Effluent

1. Effluent from the PORV and code safety valves discharges into the quench tank which condenses and collects the relief valve effluent. The quench tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the pressurizer electromatic relief valve.

   a) The tank contents can be cooled by the component drain pump and quench tank cooler of the Coolant Storage System. The tank fluid is circulated from the tank through the cooler and returned to the tank by spraying into the tank vapor space.

   b) The quench tank can be remotely vented to the Gaseous Waste Disposal System. Excess water can be transferred to storage. Quench Tank pressure, level, and/or temperature can be used as a diverse indication of an open relief valve, as well as leakage past the seat.

2. A Pressurizer Relief Valve Flow Monitor is installed on each unit. This is a reliable, single channel acoustical monitoring system that is powered from a battery backed vital bus (KVIA).

   a) A piezo-electric transducer mounted on the tailpipe of each of the three relief valves converts flow induced vibration into a flow signal.

   b) The monitor panel is equipped with a series of red indicating lights for each relief valve. The number of lights that are illuminated is proportional to the flow through that valve.

   c) When indicated PZR relief valve flow exceeds a flowrate, which represents a 25% open relief valve (**activation of five or more lights**), statalarm SA-18/A-1, Pressurizer Relief Valve Flow, actuates.

3. Temperature sensors located downstream of the PORV and safety valves provide backup valve position indication.

F. Pressurizer Surge Line

1. A 10 inch diameter surge line connects the bottom of the pressurizer to Loop A hot leg on Unit 1 and Loop B hot leg on Units 2 and 3.

1) RC-66 tailpipe
2) RC-67 tailpipe
3) RC-68 tailpipe
4) Pressurizer Surge Line
5) Pressurizer Spray Line

7. Incore Thermocouples TYPE-K, Chromel-Alumel (0-2500°F)
   1) Core Exit Thermocouple (CETCs)
   2) 52 total Incore thermocouples, 47 for control room usage and 5 for the SSF Control Room.
      - 12 CETCs for Train A ICCM
      - 12 CETCs for Train B ICCM
      - 23 CETCs for Control Room indications

      47 total provided to the Control Room
      - 5 independent CETCs for the SSF

      52 total

QUESTION # 87

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	____
	Group #	**1**	____
	K/A #	**013000 K4.08**	
	Importance Rating	**3.1**	____

Technical Reference(s): **IC-ES**

Proposed references to be provided to applicants during examination: **IC-ES-3A**

Learning Objective: **IC-ES-12.1**

Question Source:
- Bank # **IC-346**
- Modified Bank # ______
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 87

RO ONLY

???

Unit 2 plant conditions:

- LOCA in progress
- RB Pressure peaked at ~~14~~ 34 psig
- RB Pressure is currently stable at 11 psig
- RCS Pressure is ≈11 psig
- ALL ES power supplies are operating properly
- Lights on the ES analog channels are per Attachment

Which ONE of the following is correct concerning the state of the ES <u>digital channels</u>?

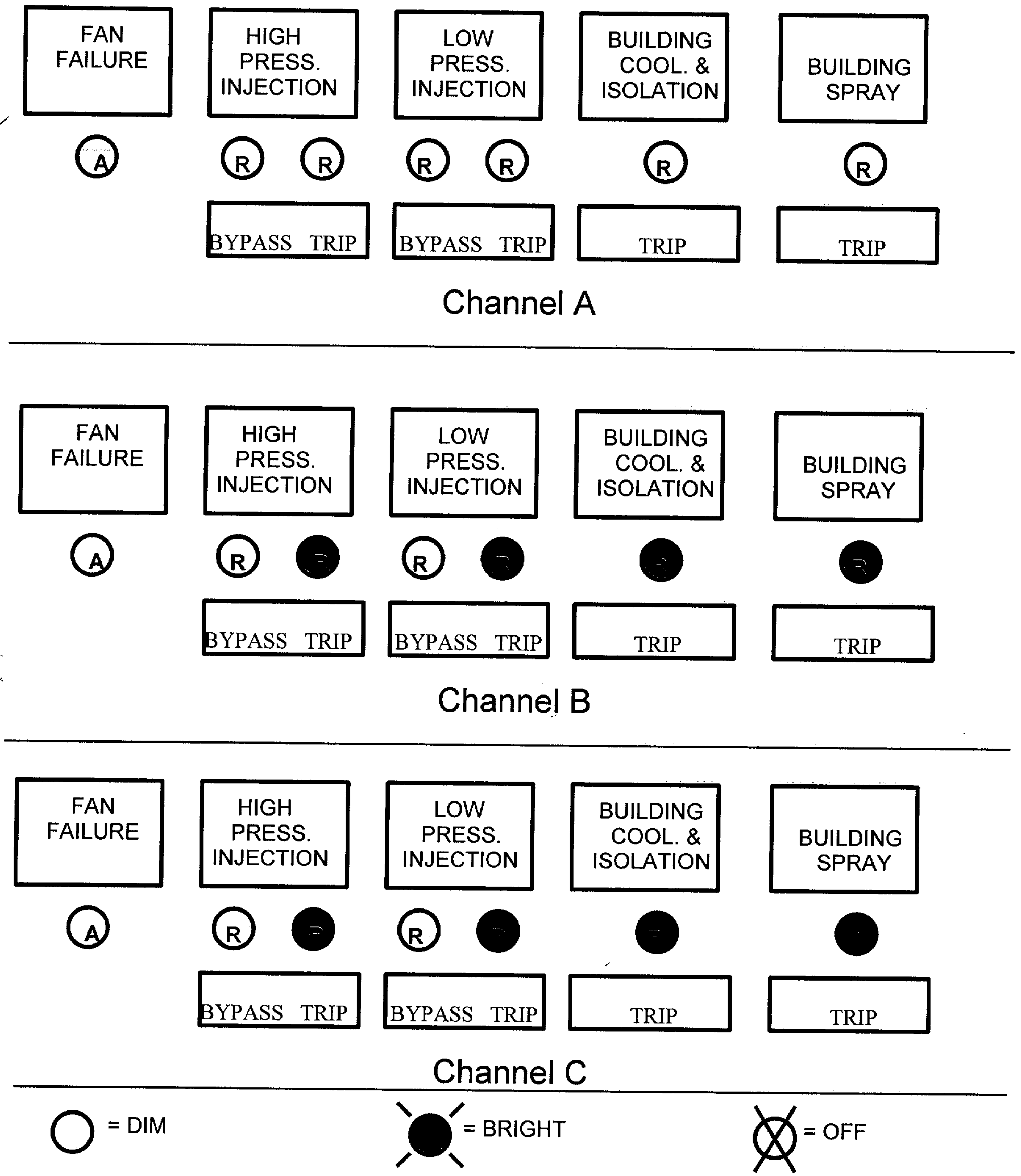
SEE ATTACHMENT

A. All digital channels have tripped initiating a full ES actuation.

B. The odd digital channels did not trip because the "A" analog channel failed to initiate a trip signal.

C. Only digital channels 1 through 6 have tripped since the TS set point for Building Spray was not reached.

D. All digital channels have tripped EXCEPT 7. Channel 7 did not trip because the "A" analog channel failed.



TITLE	NOTES:	ID#: OC-IC-ES-3A	DATE: 11/10/97
ENGINEERED SAFEGUARDS	ANALOG CHANNELS A, B, & C	REF: Design Basis Document	
		DRN by WAH:	APR BY:
		TRAINING USE ONLY	

1 POINT

QUESTION # 87

013 K4.08 (3.1/3.4) RO ONLY T2-G1 #135 RSI/PRA 5-1-00

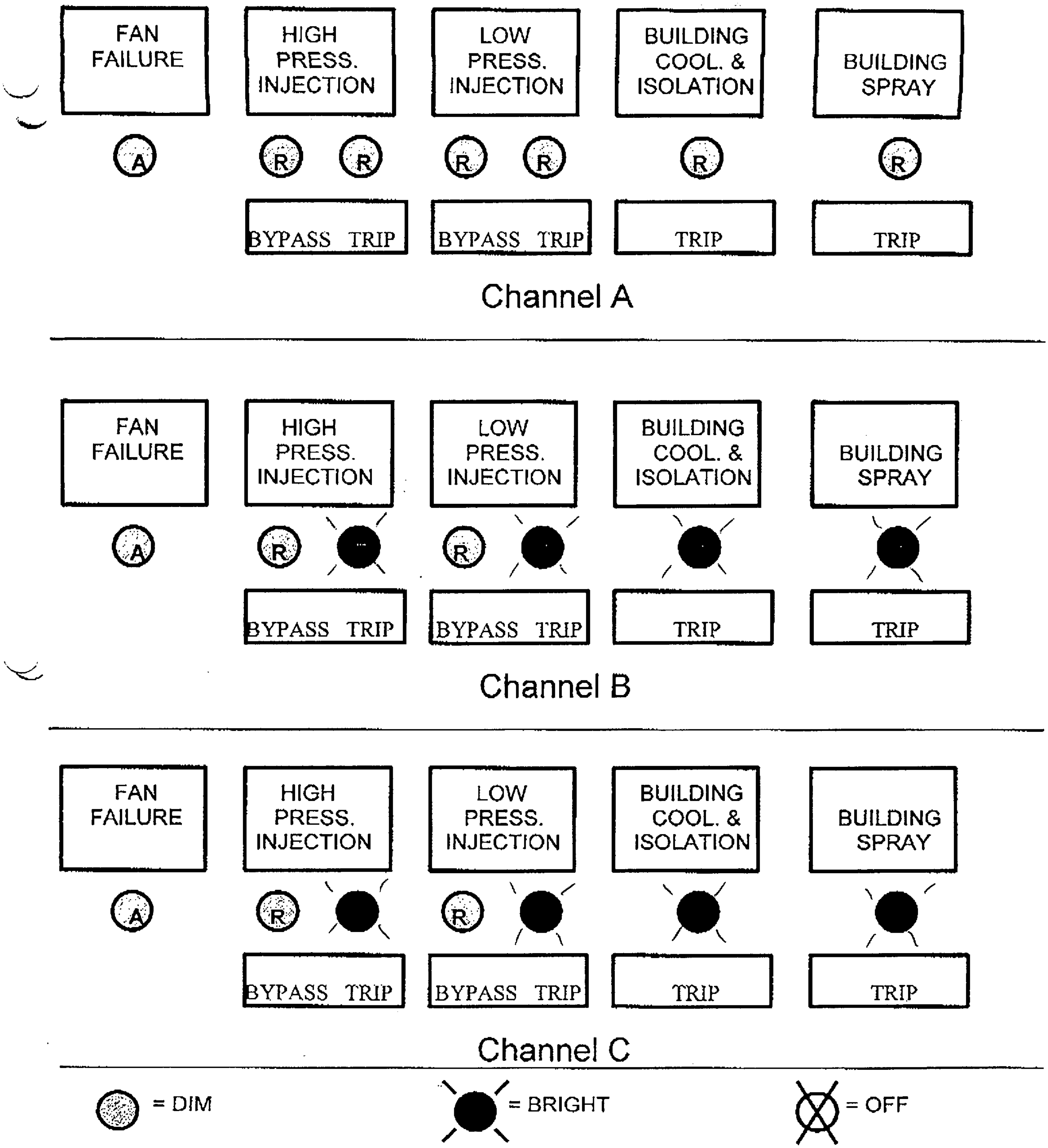
Requires OC-IC-ES-3A as an attachment. This drawing must be marked up to indicate that the HPI TRIP, LPI TRIP, RB Cooling & Isol. TRIP, and Building Spray TRIP lights for analog channels B and C are BRIGHT. All other lights should be DIM.

A. Correct, - Outputs from the three analog channels are combined in a two-out-of-three logic scheme. This logic generates a digital ES actuation signal if any two analog channels actuate.

B. Incorrect, - Analog channels 2 out of 3 logic will trip the digital channels. Analog Channel B and C tripping are sufficient to trip all the digital channels. A loss of power to the "A" analog channel will cause the failure of the odd channels

C. Incorrect, - The TS setpoint is less conservative that the actual setpoint. Channels 7 & 8 should have actuated at 10 psig in the RB.

D. Incorrect, - Channels 7 and 8 operate independent of the Analog channels A, B, & C which control the Digital channels 1-6. "A" Analog Channel for Channel 7 is independent of Channels 1-6. If Channel 7 was not independent then this would be a correct answer.



TITLE	NOTES:	ID#: OC-IC-ES-3A	DATE: 11/10/97
ENGINEERED SAFEGUARDS	ANALOG CHANNELS A, B, & C	REF: Design Basis Document	
		DRN by WAH:	APR BY:
		TRAINING USE ONLY	

12. Predict the response of ES analog and digital channels following a loss of power to: (R12)

    12.1 Analog channels

    12.2 Digital channels

    12.3 Analog and Digital channels simultaneously

13. Explain the actions necessary to manually trip and/or reset an analog or digital ESG channel. (R13)

14. Predict the emergency operation of the ESG analog and digital channels in response to a LOCA that results in RCS pressure gradually decreasing to ≈100 psig accompanied by a gradual increase in Reactor Building pressure to ≈15 psig. (R14)

15. Discuss the proper operation of all RZ Module controls and indications located on a unit's vertical control board in the Control Room. (R15)

16. Discuss and properly apply the guidance associated with repositioning ES equipment following an ES actuation. (R16)

17. Describe the actions necessary to properly return HPI pumps, Reactor Building Cooling Units and Keowee Hydro Units to normal operation following ES actuation. (R17)

This setpoint will allow 5 psi of margin to the Tech Spec setpoint of 4 psig to account for instrument uncertainty.

4. The purpose of the Reactor Building Spray initiation is to protect the Reactor Building by removing heat from the RB via the LPI system decay heat coolers.

   In addition, the RBS system can help provide protection while not actually removing heat from the RB.

   The RBS system can be configured in many different ways while being used. The RBS can be aligned to spray cool water from the BWST into the RB atmosphere. As the spray condenses steam in the RB, the energy formerly in the steam is transferred to the liquid entering the RB sump. The water in the sump is not cooled until the LPI system is realigned to pass water through the LPI decay heat coolers. Current station procedures allow the water from the sump to be sprayed back into the RB without being cooled. As with the injection mode of operation, this mode will condense steam but will not remove heat from the RB. However, the condensation of steam by the spray will work to decrease the internal pressure of the RB by reducing the partial pressure of the steam. In this manner, the RB spray will help protect the RB integrity without actually providing RB cooling.

   The UFSAR assumes that RBS is actuated at 30 psig. The actual ES setpoint is 10 psig. The 20-psi difference between the safety analysis assumption and the plant setpoint is ample to account for instrument uncertainty.

   The UFSAR states that a 2.4-second ESFAS delay is assured for Reactor Building Spray actuation.

   T. S. basis for the 4 psi and 15 psi setpoints is to establish a setting which would be reached immediately in the event of a DBA, cover the entire spectrum of break sizes, and yet be far enough above normal operation pressures to prevent spurious initiation.

C. System Logic -

Refer to OC-IC-ES-2

The system is comprised of three analog channels and eight digital channels. The analog channels each receive input from RCS and Reactor Building pressure. Should these inputs get outside acceptable operating ranges the associated analog channel will trip. Outputs from the three analog channels are combined in a two-out-of-three logic scheme. This logic generates a digital ES actuation signal if any two analog channels indicate an actuation setpoint has been reached. In addition, the ability to manually initiate ES is provided on the main control board. This design

reduces the likelihood of spurious actuations while providing sufficient redundancy to prevent a single failure from blocking necessary safeguards action. Once an automatic ES actuation has occurred, all components actuated will remain in their actuated mode after the ES channel is reset. Each actuated component must then be removed from ES control by deliberate operator action to return it to manual control.

2.2 Analog Channels

A. Three identical analog channels - A, B, C powered from KVIA, KVIB and KVIC respectively

B. Inputs

1. Pressure Transmitters

a) There are three identical but independent wide range RC pressure transmitters - one for each analog channel. They are located inside the RB on the second level.

b) Likewise there are three identical but independent narrow range reactor building pressure transmitters - one for each analog channel. They are located inside the East and West Penetration rooms.

2. Pressure Switches

There are six identical independent RB pressure switches, two for each analog channel. They are located inside the East and West Penetration Rooms.

C. Analog Cabinets - one for each channel

1. There are 3 analog cabinets powered from vital busses A, B, and C.

2. Modules contained in each channel

NOTE: Bistable Modules convert an analog input to a digital output (in the form of relay contacts) when setpoint is reached.

a) HPI Trip Bistable

1) Receives input from WR RC pressure transmitter.

2) Will trip if RC pressure decreases below 1600 psig unless bypassed.

3) Output is fed through an OR gate to digital channels 1 & 2.

4) Once tripped, must be manually reset.

b) HPI Inhibit Bistable

1) Allows manually bypassing the HPI trip bistable when RC pressure is < 1750 psig.

2) Bypass is automatically removed when RC pressure increases above 1750 psig.

c) LPI Trip Bistable

1) Receives input from WR RC pressure transmitter.

2) Will trip if RC pressure decreases below 550 psig unless bypassed.

3) Output is fed through an OR gate to digital channels 3 & 4.

4) Once tripped, must be manually reset.

d) LPI Inhibit Bistable

1) Allows manually bypassing the LPI trip bistable when RC pressure is < 900 psig.

2) Bypass is automatically removed when RC pressure increases above 900 psig.

e) RC Pressure Test Module

1) Used by I&E to check trip setpoints.

2) When placed in the TEST position all associated outputs go to the tripped state.

f) RB Pressure Trip Bistable

1) Receives input from RB pressure transmitter.

2) Will trip if RB pressure increases above 3 psig.

3) Output is fed to digital channels 5 & 6 and also to digitals 1, 2, 3 & 4 through OR gates.

4) Once tripped, must be manually reset.

g) RB Pressure Test Module

1) Used by I&E to check trip setpoints.

2) When placed in the TEST position all associated outputs go to the tripped state.

h) High RB Pressure Contact Buffers

NOTE: The contact Buffer Modules provide an isolating interface between input from the RB pressure switches and ESFAS.

1) Receive input from the RB pressure switches.

2) Will trip if RB pressure increases above 10 psig.

3) One provides output to digital channel 7 and the other provides output to digital channel 8.

4) Automatically reset when RB pressure decreases below 10 psig.

3. Lamp indication is provided above each cabinet to indicate the status of the analog systems. All lights are normally on and dim. When the function being monitored is initiated the light will brighten.

Refer to OC-IC-ES-3

a) Fan Failure - monitors the cabinet cooling fan which is normally in operation.

b) HPI Bypass/Trip - associated lights will brighten if HPI portion of ES (for that particular channel) is bypassed or tripped.

c) LPI Bypass/Trip - associated lights will brighten if LPI portion of ES (for that particular channel) is bypassed or tripped.

d) Rx Bldg. Cooling & Isolation Trip - indicates trip signal being sent to digital channels 5&6.

e) Bldg. Spray Trip - indicates trip signal being sent to digital channels 7&8.

2.3 Digital Channels

A. Eight separate digital channels numbered 1-8. There are four sets of redundant digital channels. Either redundant channel is capable of actuating the necessary safeguards action.

1. ES 1&2 High Pressure Injection, Keowee Emergency Start and Non-Essential RB Isolation
2. ES 3&4 Low Pressure Injection and Low Pressure Service Water
3. ES 5&6 Reactor Building Cooling, Penetration Room Ventilation, Essential Reactor Building Isolation
4. ES 7&8 Reactor Building Spray

B. The odd digital channels are powered from KVIA and the even digital channels are powered from KVIB.

C. Inputs - Each of the eight digital channels receives an input from each of the three analog channels plus one manual trip/reset switch.

D. Digital Cabinets -

1. There are a total of 4 digital cabinets (logic cabinets 4-7), with two digital channel logic systems per cabinet.

   a) Cabinets housing odd numbered digitals are powered from KVIA and even numbered digital systems powered from KVIB.

Exam Question Report

27-Jan-99

Question ID:	IC346	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC346				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = R4; SRO = R4 Reference: OP-OC-IC-ES	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

KA: 013000K4.08

Refer to Attachment ______.

Following a LOCA on Unit 2, the following indications are noted:

- RB Pressure peaked at 14 psig
- RB Pressure is currently stable at 11 psig.
- RCS Pressure is approximately 11 psig.
- Lights on the ES analog channels are per Attachment ____.

Which ONE of the following is correct concerning the state of the ES digital channels? (0.25)

a. The odd digital channels should not have tripped because the A analog channel failed to initiate a trip signal.

b. All digital channels should have tripped initiating a full ES actuation.

c. Only digital channels 1 through 6 should have tripped since the TS set point for Building Spray was not reached.

d. All digital channels should have tripped EXCEPT seven. Channel seven could not have tripped because the A analog channel failed.

Answer

B

Requires OC-IC-ES-3A as an attachment. This drawing must be marked up to indicate that the HPI TRIP, LPI TRIP, RB cooling & Isol. TRIP, and Building Spray TRIP lights for analog channels B and C are BRIGHT. All other lights should be DIM.

A. Analog channels B and C tripping are sufficient to trip all the digital channels.
C. The TS setpoint is less conservative that the actual setpoint. Channels 7 & 8 should have actuated at 10 psig in the RB.
D. Channels B & C should be sufficient to trip both channels 7 & 8.

Lessons

ID	Description

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

QUESTION # 88

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	______
	Group #	**1**	______
	K/A #	**015000 A1.01**	
	Importance Rating	**3.5**	______

Technical Reference(s): **CP-012**
PIP 99-766

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **OP-012 OBJ. #3.2**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 _____
55.43 _____

Comments:

1 POINT

QUESTION # 88

RO ONLY

Unit 2 conditions:

INITIAL CONDITIONS:

- Startup from a refueling outage to 100% power is in progress
- Group 7 = 50% withdrawn
- Reactor Power (NIs) = 50.0%
- Thermal Power Best = 50.0%

CURRENT CONDITIONS:

- Power = 80%
- Group 7 = 80% withdrawn

Which ONE of the following is correct?

During the **above power escalation**, Power range NI indication compared to Thermal Power Best will become _______ due to _______ neutron leakage.

A. Conservative / increased

B. Conservative / decreased

C. non-conservative / increased

D. non-conservative / decreased

1 POINT

QUESTION # 88

015000 A1.01 RO ONLY RSI/GCW NEW (IPE) 04/27/00

A training package, ON-OPS 99-44 NI Cal During Power Increase, was prepared and sent to all licensed individuals with the following information:

Lessons we should learn:
Power increases with large group 7 rod movement will result in the effects of un-shadowing overcoming the effects of T-cold decrease. Highly conservative NI readings will develop.
When above ~85% power, the effects of group 7 movement may not be enough to overcome T-cold and NIs will move toward non-conservative.

A. Correct, Power increases with large group 7 rod movement will result in the effects of un-shadowing overcoming the effects of T-cold decrease. Highly conservative NI readings will develop.

B. Incorrect, The first part is true. Indication will be conservative due to increased neutron leakage.

C. Incorrect, This would be true with previous plant maneuvering practices on startups due to the affect of T-cold decreasing. The indication will be conservative due to increased neutron leakage.

D. Incorrect, The first would be true with previous plant maneuvering practices on startups due to the affect of T-cold decreasing.

OBJECTIVES

TERMINAL OBJECTIVES:

1. State the purposes of the Operation at Power procedure and its enclosures, describe the underlying reasons for specific tasks or considerations, and recognize the allowable maneuvering rate limits.

ENABLING OBJECTIVES

1. State the purposes of the Operation at Power procedure. (R1)

2. Discuss the basis for applicable limits and precautions: (R2)

3. Concerning the power escalation section of the procedure: (R3)

   3.1 State the general purpose of and the entry path to the enclosure.

   3.2 Explain when NI's must be calibrated and how Thermal Power Best is used during calibration.

   3.3 State the action, which must be performed after the second MFDWP is placed in service.

C. Maintain CRD Groups 5-8 within the required position limits during power operations per PT/600/01.

- If any CRD Groups are in the restricted region, boration must begin within 15 minutes per TS 3.2.1. Operation within the restricted region is limited to 2 hours.

  The restricted region curve is used to meet ECCS power peaking and ejected rod worth criteria, and coincides with the shutdown margin curve at low rod indices.

D. Maintain core power imbalance and quadrant power tilt per PT/600/01.

E. At ≈ 225 MWE, transfer unit electrical loads from the startup to the auxiliary transformer.

F. Prior to exceeding 30% Rx power, verify or reduce UST temperature to ≤ 125°F.

G. Check NI calibration at 30% FP (compared to Thermal Power Best).

- If any two NI's are non-conservative, have I&E calibrate all NI's 3 to 5% conservative and resume power increase after calibration.

  {PIP O99-0766}: During the power increase, the difference between NIs and thermal power increased (had been calibrated conservatively). As T_C decreases during a power increase, shielding of the NIs occurs (greater density of water). This would normally cause NIs to decrease during a power increase. For this reason, NIs have historically been calibrated 3 to 5% conservative during the startup.

  Recent changes in ONS maneuvering philosophy have changed the overall effect on NIs during a power increase. In the past, a typical power increase was performed with Group 7 further out and with relatively less rod motion during the power increase. Current philosophy is to keep Group 7 about equal to power in order to minimize imbalance swings. During power increases using large Group 7 pulls, NIs tend to increase relative to power due to the influence of 'unshadowing' that occurs as Group 7 uncovers more fuel. This results in a power distribution that strongly increases leakage to the excores.

  Ops will have to determine a set maneuvering philosophy and then determine whether calibrating NIs conservatively during power increases is the correct approach.

- If any two of the four NI's become ≥ 2% non-conservative during the power level increase, stop the power increase and have all NI's recalibrated 3 to 5% conservative.

- Reset "Output Memory" on MT Trip Bypass Bistables on all four RPS channels.

Problem Investigation Process
Oconee Nuclear Station

PIP Serial No:	Action Category:	LER No:	Other Report:
O-99-00766	3		

Problem Identification

Discovered Time/Date: 17:00 02/27/1999 **Occurred Time/Date:**

Unit(s) Affected:

Unit	Mode	%Power	Unit Status Remarks
2	1		Reactor Power Increase in Progress.

System(s) Affected:

NI Nuclear Instrumentation System

Affected Equipment

(No Equipment Affected)

Location of Problem:

Bldg: Column Line: Elev:

Location Remarks:

Method Used to Discover Problem:
Observation

Brief Problem Description:
NI Power Ranges did not respond as expected during the reactor startup.

Detail Problem Description:
During the power escalation on 2/26/99, NIs were calibrated ~ 4%FP conservative such that the NIs would decrease toward the reactor thermal power. This behavior is the typical behavior, since the cold leg water becomes more dense as reactor power increases. However, during the unit 2 power escalation, the difference between NIs and thermal power increased as reactor power increased. Reactor power was held at 85 %FP (99 % NI Power) for 2 hours to calibrate NIs. This time the NIs were set 1%FP behind reactor thermal power at 84 % NI power. On the subsequent escalation, the NIs responded by decreasing as reactor power increased even with rod pulls, which normally make the NIs read higher than originally calibrated. It is not understood if something has changed such that the rod withdrawal affect is stronger than the cold leg effect on NIs. This should investigated and procedures adjusted accordingly.

This same behavior was observed on the unit 1 startup .

Originated By: TPG4205: GILLESPIE JR, T P Team: NEC3262 Group: OPS Date: 02/27/99

Other Units/Components/Systems/Areas Affected(Y,N,U): Y

Industry Plants Affected(Y,N,U): U

Immediate Corrective Actions:
Calibrated NIs.

Problem Investigation Process
Oconee Nuclear Station

Originated By: TPG4205: GILLESPIE JR, T P Team: NEC3262 Group: OPS Date: 02/27/99

Immediate Corrective Action Documents / Work Orders:

	Indiv	Team	Group	Date
Problem Identified By:	TPG4205	NEC3262	OPS	02/27/1999
Problem Entered By:	TPG4205	NEC3262	OPS	02/27/1999

Present Operability

Responsible Group: Status:

Sys/Comp Operable? (Y,N,C,E,T):

Required Mode:

Comments:

No Current Signatures For This Section

Past Operability:

Responsible Group: Status:

Sys/Comp Operable?(Y,N,C,E,T):

Required Mode:

Comments:

No Current Signatures For This Section

Reportability

Responsible Group: Status:

Problem Reportable(Y,N,E):

Reportable Per:

Comments:

No Current Signatures For This Section

Investigation Report:

esponsible Group: Act Date:

Investigator: Group:

Due Date:

Problem Investigation Process
Oconee Nuclear Station

Date Due to VP or Sta. Mgr:
Date Regulatory or Agency Rpt Due:
Date Investigation Report Approved:

NRC Cause Codes:

Problem Evaluation

Event	Cause Code	Cause Description	Primary	Causing Groups
F5	P2b	Effect of changing operating parameters not proper	Yes	CEN

Problem Evaluation From: Resp. Group: CEN Status: Closed OEDB Checked: Yes

Nuclear Engineering has reviewed NI vs TPB and Gp 7 data for several Unit 1 and 2 startups including the 2/26/99 Unit 2 startup. (Startups reviewed for Unit 1 were: 8/17/86, 2/14/89, 2/1/93, 6/6/90 and 8/25/98. Shutdowns reviewed were 8/6/86, 4/26/90 and 8/8/98).

The current Operations training has taught the Operators to expect that NI response should decrease as cold leg temperature gets colder between 15-100% power (due to increased density of the water therefore causing more of a shielding effect). After reviewing the data for the above startups and shutdowns, it appears that group 7 has a more significant affect on NI response than Tcold. During the periods of ɔwer increase using large Group 7 pulls, NIs tend to increase relative to power. This behavior was seen by plotting Gp 7, TPB and the ᴊelta between TPB and NI Power Ranges. There is a strong correlation between NI increase and Gp 7 withdrawal. More recent core designs have placed higher powered assemblies in Group 7 locations which can strongly affect the power distribution response of the core during Group 7 withdrawals.

This data and conclusions should be reviewed with Operations and GO Nuclear Design to determine if further evaluation must be performed. The GO may be able to use standard power manuevering models to perform a local peaking comparison between a power increase using boron vs a power increase using gp 7 withdrawal. Once this review is performed, Operations Training must be updated and Operations should make the appropriate changes to the startup procedures to give the operators guidance on expected NI response during power increases.

Originated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 03/29/99

Last Updated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 05/11/99

Last Updated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 05/11/99

OEDB Comments:

Remarks Comments:

Signature Type	Indiv	Team	Group	Date
Accepted By:	FJV9133	FJV9133	CEN	03/05/1999
Assigned To:	LLW9962	FJV9133	CEN	03/05/1999
Due Date:	05/11/1999			

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	LLW9962	FJV9133	CEN	05/11/1999
Approval Assigned To:		BRL7315	CEN	05/11/1999
Approved By:	BRL7315	CAL7344	CEN	05/11/1999

Corrective Actions

CA Seq. No: 1

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
CEN	Closed	CEN	F5	B3	P2b

Proposed Corrective Action:
This data should be reviewed with Operations and GO Nuclear Design to determine if further actions must be performed. The GO may be able to use standard power manuevering models to perform a local peaking comparison between a power increase using boron vs a power increase using gp 7 withdrawal. In addition, the GO should be able to determine what has changed with the core designs that would have increased the affect of a Group 7 withdrawal.

After this review is performed, an additional corrective action should be added to update Ops Training on the information learned from this investigation. Another corrective action should be added for Operations to change the appropriate procedures to add guidance on what NI response to expect during large Group 7 withdrawals during power increases.

Originated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 05/11/99

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	LLW9962	FJV9133	CEN	05/11/1999
Approval Assigned To:	BRL7315	BRL7315	CEN	05/11/1999
Approved By:	BRL7315	CAL7344	CEN	05/11/1999

General: Outage: N/A Mode: N/A

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: B3aStatus: Closed Due Date: 10/14/1999

Last Updated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 10/14/1999

This data was reviewed with GO Oconee Nuclear Design Group and Operations.

'I Response Evaluation (PIP-99-0766)

Background:

During a Unit 2 power escalation on 2/26/99 and 2/27/99, NI's were calibrated ~ 4% FP conservative since it was expected that as

thermal power increased toward 100% FP NI's would decrease toward Reactor Thermal Power Best. This calibration has historically been an appropriate method based on the power maneuvering method performed by the Operators. However, during the Unit 2 power escalation, the difference between NI's and thermal power increased as reactor power increased. Reactor power was held at 85% FP (99% NI Power) for 2 hours to calibrate NIs. This time, the NI's were set 1% FP behind thermal power at 84% NI power. On the subsequent escalation, the NI's responded by decreasing as reactor power increased even with rod pulls. This same behavior has been observed on other unit startups since February of 1998.

Evaluation:

PI data was collected and reviewed for unit startups from 1986 to August of 1999. The data from these startups was discussed and reviewed with the Oconee Nuclear Design Group. A determination has been made as to the change that has occurred to cause a difference in NI response. A detailed review of the data was performed and the following possible causes have been discarded:

1. No significant changes have occurred with core designs that would have increased the affect of a Group 7 withdrawal.
2. No change in the control rod pattern has occurred.
3. No Tave change or downcomer temperature change has occurred that would effect leakage to excores.

The result of this evaluation concluded that the cause of the NI response is a direct effect of a gradual change in power maneuvering philosophy used to control the plant. During February of 1998, the Imbalance Failure Investigation Process Team recommended specific guidance and training for Operations on the control philosophy to be used during power maneuvers. After this guidance was implemented, the NI response change during power escalation became more obvious. The power maneuver control philosophy (i.e., rule of thumb) essentially states that "to minimize imbalance swings, between 50% and 90% FP, Group 7 position should be about equal to reactor thermal power. This ensures that sufficient control rods are available for a power increase, and minimizes imbalance swings from control rod insertion."

ı the past, a typical power increase was performed using boron adjustments while allowing Group 7 to be almost fully out during the entire power increase. Now, Group 7 and power are attempted to be closely matched during the power increase to help control imbalance. The influence of Group 7 shadowing the excores produces an increase in excore power as the control rods are being withdrawn. Therefore, as Group 7 is being withdrawn, there is an influence of unshadowing as Group 7 uncovers more fuel. This results in a power redistribution that strongly increases the leakage to the excores.

At low powers, the NI's are typically calibrated ~4% greater than Thermal Power Best. As a result of this calibration, while Group 7 is withdrawn out of the core, the excores see an increase in power. As power increases, the NI's get extremely conservative. At 85% FP during the Unit 2 power escalation, there was not a lot of Group 7 effect left. Therefore, a normal decrease in NI powers occurred between 85% and 100% FP.

Solution:

The solution is to determine what method of power maneuvering Operations will use for power increases. Then, determine how to appropriately calibrate the NI's such that they never become too conservative or too non-conservative. For example, if Group 7 position will closely match reactor thermal power during a given power increase, then NI's should not be calibrated any higher than ~+2% above Thermal Power Best. If Group 7 will be allowed to move out to >50%wd at low powers and the power increase will be performed primarily with boron adjustments, then a more conservative NI calibration may be performed at the time (>+2%). Nuclear Engineering has made a procedure change to the Power Escalation Testing procedure that performs NI calibrations for low power testing. The requirement used to be to calibrate +5/-2% at low powers and that has been changed to +2/-2%.

Recommendations to Operations:

1. Determine what maneuvering philosophy will be used for power increases.
2. Provide training to Operators on NI response behavior based on the power maneuvering method chosen.
3. Revise appropriate procedures to ensure that NI's are not calibrated too conservatively for a given power increase.

A corrective action was added to update Ops Training on the information learned from this investigation.

Another corrective action was added for Operations to change the appropriate procedures to add guidance on NI calibration during large

Group 7 withdrawals (for power increases). NI calibration at low powers needs to either be avoided or the calibration should be no greater than ~2% above Thermal Power Best if Group 7 is <50% power.

Originated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 10/12/1999

Signature Type	Indiv	Team	Group	Date
Accepted By:	FJV9133	FJV9133	CEN	05/13/1999
Assigned To:	LLW9962	FJV9133	CEN	05/13/1999
Due Date:	10/14/1999			
Approval Assigned To:	FJV9133	FJV9133	CEN	10/14/1999
Ready For Approval:	FJV9133	FJV9133	CEN	10/14/1999
Approved By:	FJV9133	FJV9133	CEN	10/14/1999

CA Seq. No: 2

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OSP	Delete	CEN	F5	B3a	P2b

Proposed Corrective Action:

Reason for Delete:-

Dupicate with number 3, wrong group specified.

Last Updated By: FJV9133: VERBOS JR, FRANKLIN J Team: FJV9133 Group: CEN Date: 10/14/1999

This CA was created in error and will be deleted.

Originated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 10/14/1999

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	LLW9962	FJV9133	CEN	10/14/1999
Approval Assigned To:	FJV9133	FJV9133	CEN	10/14/1999

General: Outage: Mode:

Other Tracking Processes

Type Number Text

Problem Investigation Process
Oconee Nuclear Station

Actual Corrective Action:
Actual CAC: Status: Delete Due Date: 10/24/1999

Signature Type	Indiv	Team	Group	Date
Assigned To:			OSP	10/14/1999
Due Date:	10/24/1999			

CA Seq. No: 3

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	CEN	F5	C	P2b

Proposed Corrective Action:

ast Updated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 11/30/1999

Operations should provide appropriate training to Operators on NI response behavior based on power maneuvering method chosen. The results of this evaluation (as described in ACA#1) should be communicated to Operators so that a knowledge of NI response can be gained.

Originated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 10/14/1999

Signature Type	Indiv	Team	Group	Date
Assigned To:	LLW9962	FJV9133	CEN	11/01/1999
Ready For Approval:	LLW9962	FJV9133	CEN	11/30/1999
Approval Assigned To:	FJV9133	FJV9133	CEN	11/30/1999
Approved By:	FJV9133	FJV9133	CEN	12/01/1999

General: Outage: Mode:

Other Tracking Processes
Type Number Text

Actual Corrective Action:
ʻctual CAC: C9Status: Closed Due Date: 01/19/2000

Last Updated By: ASH0555: HOLLINGSWORTH, ANTHONY S Team: ASH0555 Group: OPS Date: 01/19/2000

Problem Investigation Process
Oconee Nuclear Station

A training package, ON-OPS 99-44 NI Cal During Pwr Increase, was prepared and sent to all licensed individuals with the following information:

NI calibration during power increase

We have typically been taught that as reactor power increases, NI detector response will result in NIs moving in the non-conservative direction. This was due to Tcold becoming colder and therefore fewer neutrons reaching the detectors. In anticipation of this, our procedures have us calibrate NIs 3-5% conservative.
In February of 1999, a startup did not produce this effect. As power increased, NIs became more and more conservative; at 85% TPB, the NIs were reading 99%. PIP-99-766 was written to determine why this occurred. The following is an excerpt from the evaluation describing why this is occurring:

Background:
During a Unit 2 power escalation on 2/26/99 and 2/27/99, NI's were calibrated ~ 4% FP conservative since it was expected that as thermal power increased toward 100% FP NI's would decrease toward Reactor Thermal Power Best. This calibration has historically been an appropriate method based on the power maneuvering method performed by the Operators. However, during the Unit 2 power escalation, the difference between NI's and thermal power increased as reactor power increased. Reactor power was held at 85% FP (99% NI Power) for 2 hours to calibrate NIs. This time, the NI's were set 1% FP behind thermal power at 84% NI power. On the subsequent escalation, the NI's responded by decreasing as reactor power increased even with rod pulls. This same behavior has been observed on other unit startups since February of 1998.

Evaluation:
PI data was collected and reviewed for unit startups from 1986 to August of 1999. The data from these startups was discussed and reviewed with the Oconee Nuclear Design Group. A determination has been made as to the change that has occurred to cause a difference in NI response. A detailed review of the data was performed and the following possible causes have been discarded:

1. No significant changes have occurred with core designs that would have increased the affect of a Group 7 withdrawal.
2. No change in the control rod pattern has occurred.
3. No Tave change or downcomer temperature change has occurred that would effect leakage to excores.

The result of this evaluation concluded that the cause of the NI response is a direct effect of a gradual change in power maneuvering philosophy used to control the plant. During February of 1998, the Imbalance Failure Investigation Process Team recommended specific guidance and training for Operations on the control philosophy to be used during power maneuvers. After this guidance was implemented, the NI response change during power escalation became more obvious. The power maneuver control philosophy (i.e., rule of thumb) essentially states that "to minimize imbalance swings, between 50% and 90% FP, Group 7 position should be about equal to reactor thermal power. This ensures that sufficient control rods are available for a power increase, and minimizes imbalance swings from control rod insertion."

In the past, a typical power increase was performed using boron adjustments while allowing Group 7 to be almost fully out during the entire power increase. Now, Group 7 and power are attempted to be closely matched during the power increase to help control imbalance. The influence of Group 7 shadowing the excores produces an increase in excore power as the control rods are being withdrawn. Therefore, as Group 7 is being withdrawn, there is an influence of unshadowing as Group 7 uncovers more fuel. This results in a power redistribution that strongly increases the leakage to the excores.

At low powers, the NI's are typically calibrated ~4% greater than Thermal Power Best. As a result of this calibration, while Group 7 is withdrawn out of the core, the excores see an increase in power. As power increases, the NI's get extremely conservative. At 85% FP during the Unit 2 power escalation, there was not a lot of Group 7 effect left. Therefore, a normal decrease in NI powers occurred between 85% and 100% FP.

Solution:

The solution is to determine what method of power maneuvering Operations will use for power increases. Then, determine how to appropriately calibrate the NI's such that they never become too conservative or too non-conservative. For example, if Group 7 position will closely match reactor thermal power during a given power increase, then NI's should not be calibrated any higher than ~+2% above Thermal Power Best. If Group 7 will be allowed to move out to >50%wd at low powers and the power increase will be performed primarily with boron adjustments, then a more conservative NI calibration may be performed at the time (>+2%). Nuclear Engineering

has made a procedure change to the Power Escalation Testing procedure that performs NI calibrations for low power testing. The requirement used to be to calibrate +5/-2% at low powers and that has been changed to +2/-2%.

Recommendations to Operations:

1. Determine what maneuvering philosophy will be used for power increases.
2. Provide training to Operators on NI response behavior based on the power maneuvering method chosen.
3. Revise appropriate procedures to ensure that NI's are not calibrated too conservatively for a given power increase.

A corrective action was added to update Ops Training on the information learned from this investigation.

Another corrective action was added for Operations to change the appropriate procedures to add guidance on NI calibration during large Group 7 withdrawals (for power increases). NI calibration at low powers needs to either be avoided or the calibration should be no greater than ~2% above Thermal Power Best if Group 7 is <50% power.

Lessons we should learn:
· Power increases with large group 7 rod movement will result in the effects of 'un-shadowing' overcoming the effects of Tcold decrease. Highly conservative NI readings will develop.
· When above ~85% power, the effects of group 7 movement may not be enough to overcome Tcold and NIs will move toward non-conservative.

This communication is the fastest way to get all licensed individuals this information. Since Reactor Engineering personnel are monitoring power escalations they may have input in your selection during calibration. Use their knowledge.

/e will:
· Change the Ops at Power procedure to clarify the guidance on NI calibration. The procedures group has a corrective action for this.
· Enhance the appropriate lesson plans to train on this behavior.

Thanks, Scott Hollingsworth

Originated By: ASH0555: HOLLINGSWORTH, ANTHONY S Team: ASH0555 Group: OPS Date: 01/05/2000

Signature Type	Indiv	Team	Group	Date
Accepted By:	MAP7314	HRL7353	OPS	12/07/1999
Assigned To:	ASH0555	RDL3572	OPS	12/07/1999
Due Date:	01/19/2000			
Ready For Approval:	ASH0555	ASH0555	OPS	01/19/2000
Approval Assigned To:	RDL3572	RDL3572	OPS	01/19/2000
Approved By:	MAP7314	HRL7353	OPS	01/19/2000

CA Seq. No: 4

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
TRN	Closed	CEN	F5	C7	P2b

Proposed Corrective Action:
Operations should provide training on core behavior versus NI power as discussed in this PIP. The NI response for differing power

maneuvering methods should be more understood by licensed operators.

Originated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 10/21/1999

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	LLW9962	FJV9133	CEN	10/21/1999
Approval Assigned To:	FJV9133	FJV9133	CEN	10/21/1999
Approved By:	FJV9133	FJV9133	CEN	10/21/1999

General: Outage: Mode:

Other Tracking Processes

Type Number Text

Actual Corrective Action:

Actual CAC: C7Status: Open Due Date: 09/29/2000

The changes described below have also been incorporated into lesson plan CP-14, Operation at Power. In addition, the HLP team has been assigned a tracking number to ensure that all current license class candidates receive training on this event before they assume licensed duties.

Last Updated By: CHE7135: EFLIN, CAMDEN H Team: PMS9238 Group: TRN Date: 04/27/2000

During the March 29, 2000 Continuing Training mini-TPRC meeting, PIP O-99-00766 was reviewed to determine the appropriate changes that need to be made the the Operator Training programs as a result of this PIP. A review of the PIP indicated that a summary of the event described in PIP O-99-00766, and the associated lessons learned, needed to be incorporated in lesson plans IC-NI, Nuclear Instrumentaion, and CP-018, Power Manuever Imbalance Control, and simulator exercise guide SNO-L005, Power Escalation to 100% Power. These changes will be completed by May 1, 2000. It was further concluded during this meeting that training package ON-OPS-99-44 addressed in CA #3 of this PIP provided sufficient short term training requirements for the operators.

Last Updated By: CHE7135: EFLIN, CAMDEN H Team: PMS9238 Group: TRN Date: 04/24/2000

Training may be scheduled after EOP revision training and exams are completed. Approx. expected completion Sept. 2000.

Originated By: WHC0645: CAUDILL, WILLIAM H Team: DJS7490 Group: TRN Date: 04/24/2000

Signature Type	Indiv	Team	Group	Date
Accepted By:	WHC0645	PMS9238	TRN	10/25/1999
Assigned To:	CHE7135	PMS9238	TRN	10/25/1999
Due Date:	09/29/2000			

CA Seq. No: 5

Problem Investigation Process
Oconee Nuclear Station

Resp Group	Status	Orig Group	Event Code	Prop CAC	Cause Code
OPS	Closed	CEN	F5	A3	P2b

Proposed Corrective Action:
Operations should revise 1102/04 - Ops at Power procedure. Currently it states to calibrate NIs 3-5% conservative which is not correct if the power increase will be performed with Group 7 matching Thermal Power Best. Contact Reactor Engineering for appropriate guidance to add to the procedure.

Originated By: LLW9962: WATROBSKI, LORI L Team: FJV9133 Group: CEN Date: 10/21/1999

Signature Type	Indiv	Team	Group	Date
Ready For Approval:	LLW9962	FJV9133	CEN	10/21/1999
Approval Assigned To:	FJV9133	FJV9133	CEN	10/21/1999
Approved By:	FJV9133	FJV9133	CEN	10/21/1999

General: Outage: Mode:

Other Tracking Processes
Type Number Text

Actual Corrective Action:
Actual CAC: A3Status: Closed Due Date: 03/14/2000

Revised procedures OP/1,2,3/A/1102/004 (Operation At Power) per redmarked procedure received from Reactor Engineering. These changes removed the wording that stated to calibrate NIs 3-5% conservatively and replaced it with wording that says to maintain NIs calibrated within +/- 2% of Core Thermal Power. These changes were made in the following revisions:

OP/1/A/1102/004 Rev. 85
OP/2/A/1102/004 Rev. 59
OP/3/A/1102/004 Rev. 55

Last Updated By: RTR7312: RIDINGS, ROYCE T Team: DBC7309 Group: OPS Date: 02/28/2000

2/15/00- received red marked copy of OP/1/A/1102/004 (Operation At Power) from L. Watrobski. Beginning work on procedure revision based on this red marked copy. Discussed proposed changes with CR personnel and changed some nomenclature to be more consistent with other procedures.

Last Updated By: RTR7312: RIDINGS, ROYCE T Team: DBC7309 Group: OPS Date: 02/16/2000

1/10/00- contacted L. Watrobski to obtain information to use in preparation for procedure change.

Originated By: RTR7312: RIDINGS, ROYCE T Team: KCM2861 Group: OPS Date: 01/10/2000

Signature Type	Indiv	Team	Group	Date
Accepted By:	MAP7314	HRL7353	OPS	10/21/1999
Due Date:	03/14/2000			
Assigned To:	RTR7312	DBC7309	OPS	01/11/2000
Ready For Approval:	RTR7312	DBC7309	OPS	02/29/2000
Approval Assigned To:	DBC7309	DBC7309	OPS	02/29/2000
Approved By:	DBC7309	DBC7309	OPS	03/01/2000

Problem Investigation Process
Oconee Nuclear Station

End of the Document for PIP No:	O-99-766
The status of this PIP is:	Screened
The duration of this PIP was:	6 days

QUESTION # 89

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	____
Group #	**1**	____
K/A #	**022000 K4.02**	
Importance Rating	**3.1**	____

Technical Reference(s): **OP/1104/015**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-RBC OBJ. #7**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ____
55.43 ____

Comments:

1 POINT

QUESTION # 89

RO ONLY

Unit 3 Plant conditions are as follows:

INITIAL CONDITIONS:

- Time = 1200
- 100% power
- 3A and 3C RBCU's operating normally
- Reactor Building pressure is "0" psig

CURRENT CONDITIONS:

- Time = 1300
- "3A" RBCU is secured by the BOP

Which ONE of the following describes how to place the "3A" RBCU back in service? Per OP 1104-15

A. Wait until 13:15 then place the selector switch to "HIGH".

B. Wait until 13:30 then place the selector switch to "LOW".

C. Wait until 13:30 then place the selector switch to "HIGH".

D. Immediately place the selector switch to "LOW".

1 POINT

QUESTION # 89

022000 K4.02 RO ONLY RSI/GCW NEW 04/027/00

Question setup:
OP/1104/15, Reactor Building Cooling System, is an Information Use Procedure (contains no signoff steps) and does not required to be in the possession of the user. A Limit and Precaution in this procedure gives starting guidance.

A. Incorrect, The fan motor shall be OFF for at least thirty (30) minutes, or allowed to operate for thirty (30) minutes prior to starting or changing speed. The second part of the distracter is correct.

B. Incorrect, The first part of the distracter is correct. Mixed speed combinations of RBCU fans should not be used. A fan running in HIGH creates an excessive backpressure against a fan operating in LOW causing fan and/or motor failure.

C. Correct, The fan motor shall be OFF for at least thirty (30) minutes, or allowed to operate for thirty (30) minutes prior to starting or changing speed. Mixed speed combinations of RBCU fans should not be used. A fan running in HIGH creates an excessive backpressure against a fan operating in LOW causing fan and/or motor failure.

D. Incorrect, The fan motor shall be OFF for at least thirty (30) minutes, or allowed to operate for thirty (30) minutes prior to starting or changing speed.

OBJECTIVES

TERMINAL OBJECTIVE

1. After this lecture, the student will have an understanding of the components, indications, controls, and operation of the Reactor Building Cooling (RBC) system. (T1)

2. The student will be able to assess the RBC system during normal and emergency conditions and determine corrective actions for improper system operation. (T2)

ENABLING OBJECTIVES

1. Describe the purpose of the RBC System during normal and emergency operation. (R1)

2. State the cooling medium for the RBCUs and the Reactor Building Aux Fans. (R2)

3. Describe the air circulation flow path throughout containment as provided by the RBCUs and the RB Aux Fans. (R3)

4. Discuss the purpose and location of the Fusible Dropout Plates. (R4)

5. Describe the flowpath of the cooling medium through the RBCUs and the RB Aux Fans during: (R6)

   5.1 normal operation

   5.2 ES operation

6. Given the limit and precaution, explain the basis for the limit and precaution. (R20)

7. Describe the manual starting sequence of an idle RBCU and the starting limits addressed in Limits and Precautions. (R8)

8. For PT/0160/002, RBCU Air Flow Test, describe: (R16)

   8.1 The purpose

   8.2 How the test is performed

9. Given a copy of PT/0160/002, RBCU Air Flow Test, and a set of data, evaluate if acceptance criteria is met. (R17)

2. To start a third RBCU in HIGH speed, contact Ops staff or System Engineering for guidance on starting sequence.

   There have been several incidents of RBCU's tripping the high-speed thermal overloads during starts to HIGH speed especially if the third fan is being started. One problem identified with the 'B' RBCU has to do with the power supply (PIP O98-5822). 'B' RBCU is fed from X10 (300 KVA load center) and the 'A' and 'C' RBCU's are fed from X8 and X9 (1000 KVA load centers). The duration of the inrush current is greater for the 'B' RBCU because the effect of the start produces a greater voltage drop. The combination of these factors along with damper conditions lengthens the time to achieve full speed, which also increases the likelihood the thermals will trip.

   Each RBCU breaker has a set of high-speed thermal overloads and a set of low-speed thermal overloads. During ES actuation, the overloads are shorted and CANNOT trip.

3. Although normal operation should be four RB Aux Fans in service and two RBCU's in HIGH with LPSW aligned as noted, the most effective combination of RBCUs and RB Aux Fans should be used.

D. Limits and Precautions

1. Fans should not be started, stopped, or speed changed to equalize run times.

NOTE: Either a single step of OFF to LOW speed (0-600 rpm) or a two step automatic sequence of OFF to HIGH speed (0-600-1200) is considered to be one start.

2. The motor shall be OFF for at least thirty (30) minutes, or allowed to operate for thirty (30) minutes, prior to starting or changing speed.

3. The 30 minute speed change time interval may be waived in emergencies.

   a) PIP-3-096-0528 was issued due to a RBCU tripping after restart.

      - The RBCU tripped off due to a loss of power (21 sec) on the unit. When power was restored the switch position was HIGH therefore the RBCU attempted to return to high speed. The RBCU tripped after starting in LOW and could not be restarted.

      - Investigation revealed that the thermal overloads on the breaker had tripped. The unit was tested and declared operable after the thermals were reset.

Reactor Building Cooling System

1. Purpose

To describe the proper method for operating Reactor Building Cooling System.

2. Limits And Precautions

2.1 RBCUs should NOT be started, stopped, or speed changed to equalize fan run times.

NOTE:
- A step of "OFF" to "LOW" (0-600 rpm) is considered one start.
- A step of "OFF" to "HIGH" (0-600-1200 rpm) is considered one start.

2.2 RBCU motor shall be off for 30 minutes, or allowed to operate for 30 minutes, prior to starting or changing speed.

2.3 30 minute speed change time interval may be waived in emergencies.

2.4 Manual speed changes should be minimized where possible.

2.5 During non-emergency operation, maximum RBCU motor bearing temperature: 220°F. (computer point: RBV CLR FAN IB/OB BRG TEMP).

2.6 1B RBCU may be operated while LPSW is diverted to Aux Fan Coolers.

2.7 Do NOT operate RBCUs in mixed speed combinations. Excess back pressure is placed on low speed fans.

2.8 Proper damper operation is NOT required for RBCU operability per Improved Technical Specifications (ITS).

- If dampers are NOT operating properly, high vibration and temperature problems may be encountered. {1}

2.9 When Reactor Building Cooling System Operability is required (TS 3.6.5, MODES 1, 2, 3, and 4), LPSW flow to all RBCUs must be ≥ 550 gpm. {2}

- If LPSW to an RBCU is < 550 gpm, LCO 3.0.3 applies. {2}

QUESTION # 90

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	_____
	Group #	**1**	_____
	K/A #	**056000 G2.1.28**	
	Importance Rating	**3.2**	_____

Technical Reference(s): **OP/1106/02**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-C OBJ. #39**

Question Source:
- Bank # **CF-145**
- Modified Bank # _______
- New _______

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 _____

Comments:

1 POINT

QUESTION # 90

RO ONLY

Unit 1 Plant Conditions:

- 100% power
- The pressure switch monitoring "1B" UST level fails low
- CST level = 3.5 feet and increasing
- 1DW-4 (#1 UST Makeup Control) closed

Which ONE of the following correctly describes the plant response?

Actual UST level will _______ and actual Hotwell level will ________.

A. increase / increase

B. remain the same / decrease

C. increase / decrease

D. remain the same / increase

Explain | CST pumps not running.

1 POINT

QUESTION # 90

056000 G2.1.28 RO ONLY Bank CF-145 GCW 4/27/00

Question setup: There are two pressure switches that monitor UST level. These pressure switches are set to actuate at an UST level of between 7'-0" and 7'-3". Only one pressure switch needs to sense a low level to actuate and de-energize the solenoid valves.

A. Incorrect – UST level would increase if the CST Pumps were in operation. The CST pumps will automatically start at 5 feet increasing and stop at 4 feet decreasing providing makeup to the UST. Hotwell level would increase if the M/U valves failed in the open position.

B. Correct – When the UST level instrument fails the Hotwell Makeup valves solenoids are de-energized (operating air is removed) and the valves close. Hotwell level will decrease due to normal Condensate system operation. Hotwell inventory cannot be increased or maintained at setpoint as UST cannot be established.

C. Incorrect –This would be correct if the CST was making up to the UST but the CST pumps are not operating.

D. Incorrect - The first portion is correct, as the UST will remain the same when the Hotwell Makeup valves fail closed.

TRAINING OBJECTIVES continued

39. Describe the interlock associated with the hotwell makeup control valves that prevents UST level from decreasing below 7 feet, including: (R44)

    39.1 Concerns that were identified

    39.2 Instrumentation that was added

    39.3 How interlock is actuated and subsequent automatic actions

    39.4 How control room operator is alerted to interlock actuation.

40. Draw a basic one-line diagram of the Condensate System, including the following components and valves: (R45)

    40.1 HWP's

    40.2 C-10

    40.3 Powdex and C-14/15

    40.4 Powdex Resin Trap

    40.5 Condensate Coolers

    40.6 Generator Hydrogen Coolers and Generator Stator Coolers

    40.7 Condenser Steam Air Ejectors

    40.8 Steam Packing Exhauster

    40.9 CBP's

    40.10 C,D,E,& F Heaters

    40.11 'C' Drain Coolers

    40.12 C-124/128

    40.13"E" and "D" Heater Drain Pump/Discharge

41. Given a copy of ITS/SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's.(R46)

42. Apply All ITS/SLC rules to determine applicable Conditions and Required actions for a given set of plant conditions.(R47)

43. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITS/SLC's.(R48)

44. Concerning PU/*/A/0152/003, Condensate System Valve Stroke Test, describe: (R49)

b) The loop seal is kept full by input through an orifice from the condensate supply to the main turbine exhaust hood spray.

10. Upper Surge Tank Indications and Alarms

a) Each upper surge tank is provided with a level transmitter. These transmitters send signals to UST level gauges, chart recorder (UST B only), and a UST low level statalarm SA6/A-11 (at 2 feet decreasing) in the control room.

b) A minimum inventory of ≥ 30,000 gallons, (5.4 feet) is required to be maintained in each unit's UST since it is the initial supply for the EFDWPs on that unit. Therefore, QA level indication is supplied for each UST.

11. Interlocks

a) A problem was identified where, during a casualty situation in which the hotwell makeup valves fail open, the UST level could decrease below the 6' level rendering the EFDWPs inoperable until their suction could be aligned to the hotwell. Even worse, there was the possibility that the UST's could drain to the hotwell followed by an auto start of the EFDWPs.

b) The UST level control system has been modified to prevent this situation from occurring.

c) Two new QA condition 1 pressure switches have been added to the UST's that will monitor UST level. These pressure switches are set to actuate at an UST level of between 7'-0" and 7'-3".

d) Three new QA condition 1 solenoid valves have been installed between the valve positioner (which generates the loader signal to the valve) and the diaphragms for C-192, C-187, and C-176.

1) When these solenoid valves are energized, they allow the hotwell level control system to operate normally.

2) When these solenoid valves are de-energized, the air is bled off of the valve diaphragms and the valve fail closed.

e) If UST level decreases to setpoint:

1) The pressure switches will de-energize the solenoid valves allowing the air to bleed off of the diaphragms causing the hotwell makeup valves to fail closed.

2) This is a one out of two logic so only one pressure switch needs to sense a low level to actuate and de-energize the solenoid valves.

3) Statalarm "UST TO HW MAKEUP VLVS FAIL CLOSED" (SA-6/D-10) will annunciate.

Note: OP/A/1106/02, Condensate and Feedwater, provides procedural guidance on recovering from this situation

3. Discharge header pressure on both MFDWP's < 770 psig (indicating that a FDWP is not running).

J. A Condensate Booster Pump will trip if either the suction or discharge valve is moved from full open position to 50% (+/- 25%) closed.

K. The 'E' and 'D' heater drain pumps will trip if there is:

1. An emergency low level in their respective feedwater heater or flashtank.
2. A low discharge header pressure on both MFDWP's (< 770 psig).

L. In the AUTO position, C-124 opens when C-128 position clears the closed limit switch and closes when C-128 actuates its closed limit switch.

M. If C-152 and C-153 are in AUTO, they will open if either C-128 or the feedwater recirc valve, FDW-82, comes off its closed seat and they will close when both C-128 and FDW-82 are closed. (C-152 and 153 are normally open by procedure).

N. If the UST level decreases to 7', the hotwell makeup valves fail closed.

O. In AUTO, the CST pumps will maintain the CST level between 4' and 5'.

2.6 Limits and Precautions

Note: Only the Condensate and Feedwater Procedure limits and precautions applicable to the Condensate System have been included here.

A. Hotwell and booster pumps will all trip off if there is a simultaneous low bearing oil pressure of ≤ 4 psig on a FWPT along with its respective FDWP suction valve open and a low FDWP discharge pressure of ≤ 770 psig on both MFDWP's.(Windmill Protection)

B. With only one 'D' HDP in service, power may be increased to a power level such that the FDWP suction pressure remains greater than 380 psig.

C. Condensate Booster Pumps will trip if either the suction or the discharge valve on the pump is moved from full open position to 50% (+/- 25%) closed.

D. If UST decreases to 7 ft., the hotwell makeup control valves will fail closed.

E. The following UST temperature limits shall be followed:

1. UST temperature ≤ 125° F (EFDW operability)
   a) Above 30% full power
   b) 2 hours following a Reactor Trip
   c) 2 hours after reducing power below 30%
2. UST temperature must be ≤ 145° F (system component design):
   a) For power operations up to 30% full power
   b) During shutdown conditions

Condensate and Feedwater System

1. Purpose

This procedure provides procedural guidance for operation of Condensate and Feedwater System and components.

2. Limits and Precautions

2.1 This procedure has the potential to affect Reactivity Management. {12}

2.2 When feeding SGs with Aux FDW Nozzles from Main FDW, Final FDW should be > 90°F. Minimum FDW temperature for Aux FDW Nozzle is 40°F. {2}

2.3 Maximum ΔT between FDW line temperature and SG lower downcomer is 450°F when using main FDW nozzles.

2.4 Maximum FDW flow to a SG per design basis is 5.7×10^6 lbm/hr.

- Design basis relates FDW flow affect on cross flow velocity in top of SGs.
- Design basis **NOT** dependent on number of running RCPs.

2.5 The following UST temperature limits apply: {1}

2.5.1 UST temperature ≤ 125°F (EFDW operability):

A. Above 30% full power.

B. 2 hours following a Reactor trip.

C. 2 hours after reducing power below 30%.

2.5.2 UST temperature must be ≤ 145°F (system component design): {1}

A. For power operations up to 30% full power.

B. During shutdown conditions.

2.6 Hotwell M/U Control Valves fail closed when UST level ≈ 7 ft.

2.7 If HWP discharge ≥ 168 psig, CBP discharge flow should be increased or number of HWPs reduced to lower pressure to < 168 psig. (Powdex overpressure concern) {6}

1) When RUN is selected, the associated pump will start and run with no automatic cutoff.

2) When OFF is selected, the associated pump will remain off unless CST level reaches the high high level of 11 feet, at which time the pump will start.

When level decreases enough to reset the 11-foot level pressure switch, the pump will stop. The pump will cycle on and off the high high level.

3) When AUTO is selected (normal alignment), the pumps will alternate pumping the CST down from the high level at 5 feet to the normal level of 4 feet. The second pump will start if the CST reaches the high high level. Both pumps will then stop when the CST returns to the normal level.

c) The common CST pump discharge piping contains a pneumatically operated control valve, C-148. This valve automatically opens whenever a CST pump starts.

d) An amber "low discharge pressure" light is provided at the control switch in the control room.

5. The contents of the CST can be heated during abnormally cold weather via plant heating steam. The controls are located at the CST (normally isolated).

6. CST Indications and Alarms

a) A CST level chart recorder is located in the control room.

b) "CST LEVEL LOW" SA6/B-11 (1 ft) statalarm is provided in the control room.

c) Each CST pump is provided with a local discharge pressure gauge.

W. Other Components/Systems Supplied by the Condensate System:

1. FDWP seal injection system.
2. Powdex backwash system.
3. Main turbine exhaust hood spray.
4. Condenser boot seal.
5. HWP, CBP, and 'D' HDP seal water to packing glands.
6. Seal water to vacuum valves.
7. Seal water to extraction valves.
8. FDWPT and Turbine Bypass Valve pumping traps.

2.4 System Operations

A. Normal Operations

Exam Question Report

27-Jan-99

Question ID:	CF145	Revision No:	0	Revision Date	10/29/1999
Question Description:	CF145				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	CF-C - Condensate System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 44; LRO = 44; SRO = 44 Reference: OP-OC-CF-C PAGE 45 OF 53 ITEM E.2	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Unit 1 is operating at 100% power. The pressure switch monitoring "1B" UST level fails, generating a low "1B" UST level signal. Which ONE of the following correctly describes plant response? (.25)

A) Hotwell level increases.

B) Hotwell level decreases.

C) "1B" UST level gauge fails low.

D) UST level low statalarm actuates.

Answer

B

A. incorrect, Hotwell level will decrease due to the failure of the M/U valves in the closed position. The air is removed from the valves when the soleniods are de-energized by the low level in the UST.

B. correct, See explanation above in "A".

C. incorrect, "1B" UST gage is not fed from the failed pressure switch and will indicate true UST level.

D. incorrect, Low level statalarm does not actuate from pressure switch. However an alarm will be received indicating the M/U valves have failed closed.

Lessons

ID	Description
CF-C	Condensate System (CF-C) Lesson Plan

Enabling Objectives

ID	Description

QUESTION # 91

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	_____
	Group #	**1**	_____
	K/A #	**059000 K4.02**	
	Importance Rating	**3.3**	_____

Technical Reference(s): **STG-ICS**

Proposed references to be provided to applicants during examination: **STG-ICS-4**

Learning Objective: **STG-ICS OBJ. #3**

Question Source:
- Bank # _______
- Modified Bank # _______
- New ___X___

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 _____

Comments:

1 POINT

QUESTION # 91

RO ONLY

Unit 1 Plant Conditions:

INITIAL CONDITIONS:

- 100% power
- "A" FW Loop Master in "Hand"
- All other ICS stations in "Auto"

CURRENT CONDITIONS:

- "1A" FDW PUMP trips

Which ONE of the following lists **ALL** of the lights that will **immediately** illuminate on the CTPD as a result of the trip of the "1A" FDWP?

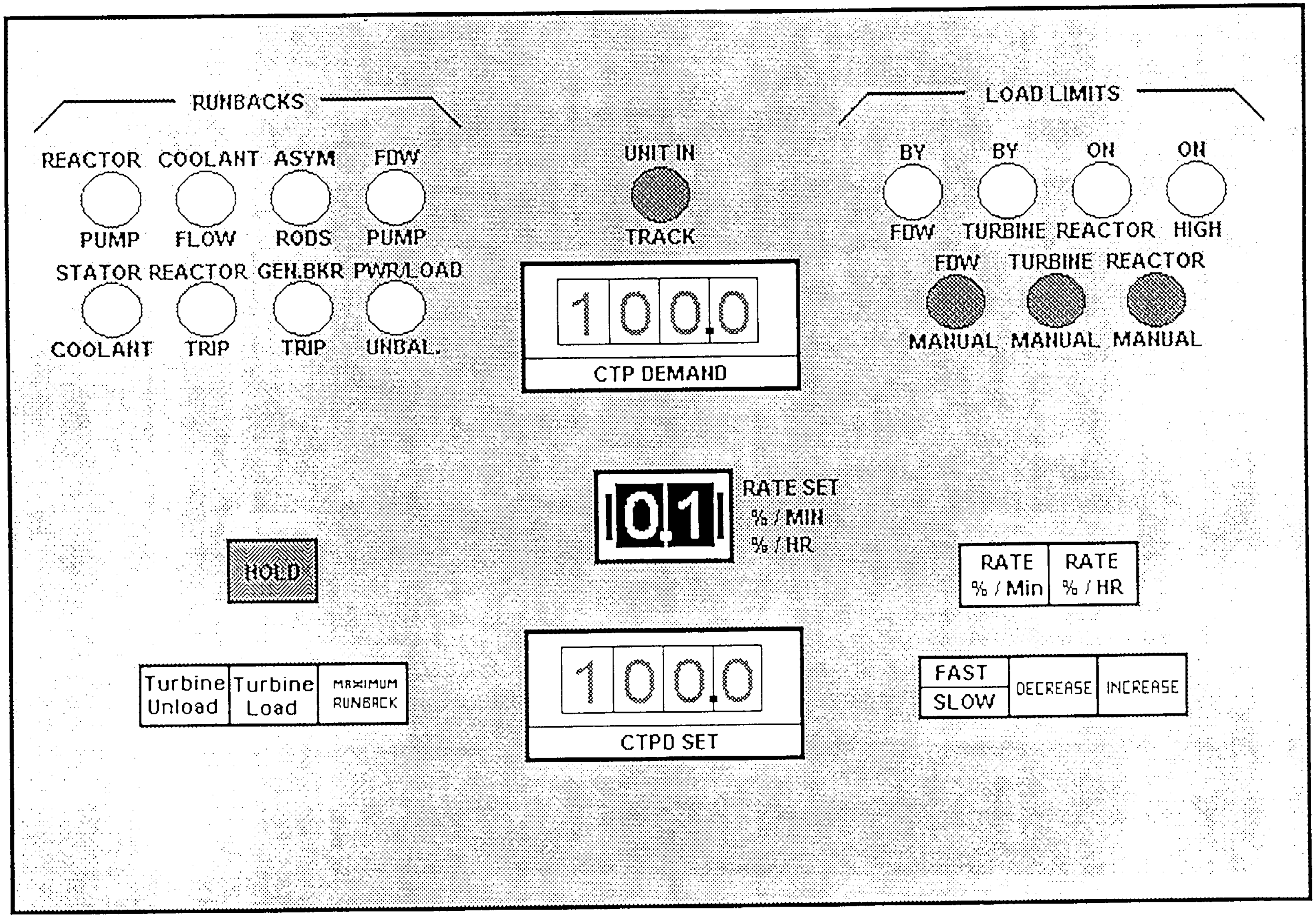
SEE ATTACHMENT:

A. WHITE - Load Limits By FDW
WHITE - Load Limits On High
RED - FDW Manual
RED - Unit In Track
WHITE - Runback FDW Pump

B. WHITE - Load Limits by FDW
RED - FDW Manual
RED - Unit in Track
WHITE - Runback FDW Pump

C. RED - Unit in Track
WHITE – Load Limit by FDW
WHITE - Load Limits On High
WHITE - Runback FDW Pump

D. WHITE - Runback FDW Pump



TITLE	NOTES		
INTEGRATED CONTROL SYSTEM	ICS Load Control Panel	ID.NO OC-STG-ICS - 4	DATE 7-24-96
		Ref NSM 2989	
		Drn By JRS	Apr By [signature]
		TRAINING USE ONLY	

1 POINT

QUESTION # 91

059000 K4.02 RO ONLY NEW RSI/GCW 04/27/00

A. Incorrect, The Red FDW Manual and Red Unit In Track are lit if <u>both</u> FDW Loop masters are in "Hand". Placing one in "Hand does not place the ICS in Track. The white Runbacks FDW Pump light would be lit.

B. Incorrect, The white Load Limits On High and white Runbacks FDW Pump lights would be lit.

C. Correct, Immediately after the "A" Main FDWP trips the white Load Limits On High light will be lit because power is above the runback target of 65%. The white Runback FDW Pump light would be lit because the FDWP is tripped. “Load Limits By FDW” will light when FDW tracking conditions are met.

D. Incorrect, - See c. This is a correct indication but it is not the only light lit.

OBJECTIVES

TERMINAL OBJECTIVE

1. Summarize the operational aspects of the Integrated Control System (ICS) with respect to the coordination of plant systems and controls. (T1)
2. Predict automatic actions performed by the ICS and identify corrective actions upon failure of the automatic actions. (T2)
3. Summarize the purpose and operation of the ICS indications and controls available to the operator. (T3)

ENABLING OBJECTIVES

1. Define the functions of the Core Thermal power Demand (CTPD) subsystem. (R1)
2. Given a set of conditions, determine the method to achieve a load change using the Load Control Panel (LCP) (R2)
3. Identify the operations of automatic and manual load limits including: (R3)
   - 3.1 LCP indications
   - 3.2 Load Limit values
   - 3.3 Runback Rates
   - 3.4 Over-riding conditions
4. Given a load limit condition, assess plant runback response and determine the source of any failure. (R4)
5. Define the purpose and operation of the HOLD push-button. (R5)
6. Identify the operation of the TRACKING mode including: (R6)
   - 6.1 Initiating conditions
   - 6.2 Tracking Parameters
   - 6.3 Operator interface
7. Describe the ICS response to a load change in the Integrated mode. (R7)
8. Describe the conditions and responses of the Integrated Master in maintaining turbine header pressure control. (R8)

1.1 Core Thermal Power Demand (CTPD)

A. The CTPD is designed to:

1. Provide operator a means to communicate demands to the ICS.
2. Recognize various limiting conditions of the plant and ensure operation within these limits by communicating them to the ICS.
3. Set "rate of change" to load transients.
4. Assume control of Feedwater, Reactor, and Turbine under certain conditions in order to stabilize the unit. **Referred to as TRACKING**

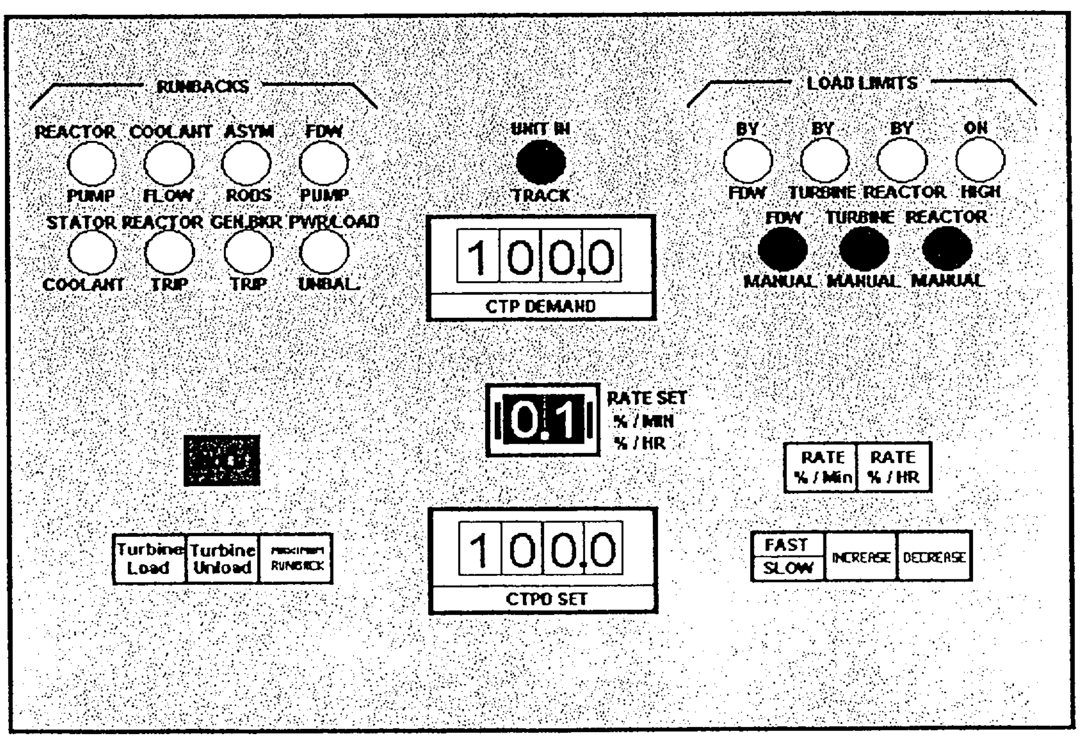
B. Modes of operation

- There are three modes of operation for the CTPD
  - ◊ OPERATOR SET (manual)
  - ◊ AUTOMATIC LOAD LIMITING
  - ◊ TRACKING

1. Operator Set

a) In this mode the operator manually selects the desired unit Core Thermal Power (CTP) demand using the Load Control Panel.

b) Load Control Panel



1) LCP is the primary operator interface to the ICS for Integrated Mode operation (Automatic).

2) Push-button control switches, digital meters, a digital thumb switch, and status lamps are provided for manipulation of the Core Thermal Power Demand (CTPD) Set, demand Rate Set, Turbine Load and Unload, Maximum Runback function, and status indication of Load limit and Tracking conditions.

3) Desired unit output is established by manipulating the INCREASE and DECREASE buttons on the LCP to establish a CTPD Set point. This setpoint is applied through the rate and load limits circuits to develop the actual CTPD for the unit.

4) The lower window on the LCP is the target demand (Called "CTPD SET").

   - This value is set manually by the operator depressing the increase or decrease push buttons on the LCP.
   - Can be set automatically by any of several automatically imposed load limits on the ICS or by a Tracking parameter value.

5) The "fast" and "slow" pushbuttons on the LCP select the rate at which the operator sets the **target load** into the window; they have no effect on the rate at which load actually changes.

6) The upper window on the LCP is "front end" CTP Demand actually being processed in the ICS.

7) Once the target value has been set into the lower window, the upper window will begin to move towards this value until the two match.

8) The rate at which the upper window changes is a function of the rate which the operator sets in with the "RATE SET" thumb-wheels on the LCP.

9) There are two possible scales for the RATE SET controls:

   (a) % / Minute

   (b) % / Hour

10) When either is selected, the operator can choose a value from 0.0% to 9.9%.

11) Rate of change control will be discussed in more detail later.

12) Indicating Lights:

(a) Runback lights → indicate to the operator the source of an ICS runback (A runback is initiated when unit load is above a maximum limit for the given conditions. Those conditions that invoke the load limits are listed below and are called "runbacks")

- REACTOR PUMP = loss of RCP
- COOLANT FLOW = degraded RC flow
- ASYMMETRIC RODS = Asymmetric rod
- FDW PUMP = loss of a feedwater pump
- STATOR COOLANT = loss of Stator Coolant
- REACTOR TRIP = Trip Confirm signal or DSS
- GEN.BKR TRIP = Both Generator Bkrs open
- PWR/LOAD UNBAL. = Power Load unbalance

(b) Load Limit lights → Indicates to the operator what the load limiting factor is in the ICS. (Indicates to the operator which section of the ICS control is the limiting factor.)

- BY FDW = FDW Total flow is the tracking parameter (i.e. Feedwater cross limit)
- BY TURBINE = Electrical megawatts is the tracking parameter (i.e. Turbine in manual)
- BY REACTOR = NI Flux is the tracking parameter or Reactor Demand is at the high limiter
- ON HIGH = A load limiting condition is present and the CTP Demand is above the limit value (i.e. Loss of a FWPT and CTP Demand is above 65%)

(c) Manual lights

(1) FDW MANUAL

On when the automatic control signal thru the FDW section is being blocked to BOTH FDW loops by bailey station(s) in "hand". Examples below:

(i) S/G Master in "hand"

(ii) BOTH FDW Masters in "hand"

(iii) BOTH controlling FDW valves in "hand"

(2) TURBINE MANUAL

(i) Turbine Master in "hand"

(3) REACTOR MANUAL

(i) Bailey in "hand"

(ii) Diamond panel in "hand"

(d) UNIT IN TRACK

(1) red light illuminates when unit is tracking.

13) HOLD Pushbutton

(a) When an operator initiated load change is about to be initiated, the HOLD pushbutton may be depressed which will simulate "0.0" %/min rate of change and override the setting on the thumbwheel thereby holding the unit where it is while the correct settings are entered.

(b) The HOLD button can also be depressed to stop a power increase or decrease that has already been set and is in progress.

(c) When HOLD is in effect, the HOLD pushbutton will be illuminated and depressing the button again will release the unit from HOLD.

(d) The HOLD button will NOT override automatic load limit changes in the CTPD except for the Asymmetric Rod load limit.

(1) i.e. During an Asymmetric Rod runback, the operator can stop the runback by selecting HOLD.

14) TURBINE LOAD/UNLOAD

(a) These two buttons will allow automatic loading or unloading of the turbine when CTPD is between 10% and 20% power.

(b) Turbine and TBVs must be in automatic for this to occur.

(c) The controlling signal to the turbine will be biased to change control valve position resulting in a THP change which will be compensated for by the TBVs. This will have the result of a transfer of control from/to TBVs for the steam header.

- More on this in the Integrated Master section.

15) MAXIMUM RUNBACK

(a) Inputs a load limit of 15% to the CTPD Set.

(b) Runback rate will be automatically input at 20% / min.

2) **Loss of one FDWPT = 65% power**

   (a) i.e. If only one main feedwater pump were operating, the unit would be limited to producing 65% power......therefore 65% is the load limit.

   (b) Sensed by a low hydraulic oil on the FWPT.

3) **Asymmetric rod = 55% power**

   (a) Sensed by a Diamond logic for asymmetric control rod from the Absolute Position Indication system.

4) **Loss of RC flow = Variable with flow degradation**

   (a) Sensed by the total of the two median selected Loop RC flow signals.

5) **Maximum Runback (when selected) = 15% power**

6) **Both Generator breakers OPEN = 20% power**

7) **Reactor Trip = 0% power**

d) If a load limit is reached, the appropriate light on the LCP panel will be illuminated indicating the source of the limit. This light will remain on until the CTP Demand is at or below the limit value.

   - The "On High" light will also be on as long as the CTP Demand is above the limit value.

e) If more than one load limit exists, the **MOST LIMITING** (lower limit) will be selected by ICS. If that particular limit were satisfied or no longer true, the next most limiting load limit would control.

f) **Load limits can only be applied in the Integrated Mode (automatic) of operation.**

   1) Manual operation will cause Tracking which inputs a demand signal downstream of the load limit signal input and will therefore block any load limit.

g) When a load limit is imposed to the ICS, the operator cannot adjust the ICS via the LCP.

QUESTION # 92

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	_____
Group #	**1**	_____
K/A #	**061000 A2.07**	_____
Importance Rating	**3.4**	_____

Technical Reference(s): **CF-EFW**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-EFW OBJ. #45**

Question Source:
- Bank # _______
- Modified Bank # _______
- New ___X___

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 _____

Comments:

1 POINT

QUESTION # 92

RO ONLY

Unit 1 Plant Conditions:

INITIAL CONDITIONS:

- Reactor power = 100%
- 1A and 1B Main FDW pumps trip

CURRENT CONDITIONS:

- 1FDW-315/316 (1A/B EFDW Control) controlling in automatic
- TD EFDWP operating
- EFDW flow to each OTSG is 220 gpm
- All RCPs operating

state

Which ONE of the following will occur ~~over the next **hour**~~ if IA and AIA pressures both stabilize at 35 psig?

ASSUME NO OPERATOR ACTIONS

A. OTSG's levels will automatically be controlled at setpoint.

begin to fill to MSL

B. OTSG's levels will ~~begin to~~ increase ~~to maximum.~~

C. TD EFDWP Main Steam supply will automatically isolate.

will

D. TD EFDWP speed control ~~fails~~ to the low speed stop.

1 POINT

QUESTION # 92

061000 A2.07 RO ONLY NEW RSI/GCW 04/27/00

A. Correct - FDW-315 and FDW-316 will automatically control OTSG level at 30 inches upon a loss of main feedwater. Both valves utilize either IA or AIA for automatic operation. Both valves have a backup nitrogen supply that ensures automatic OTSG level control on a loss of all air for at least 2 hours. Main steam pressure to the TD EFDWP is controlled by MS-87. MS-87 utilizes either IA or AIA for automatic operation. It also has a backup nitrogen supply that ensures steam regulation on a loss of all air for at least 2 hours.

B. Incorrect - Both FDW-315 and 316 have a two hour supply of nitrogen that will allow automatic OTSG level control if all air is lost. They do fail open if all air and nitrogen are lost.

C. Incorrect - MS-87 has a two-hour supply of nitrogen that will allow automatic steam pressure control to the TD EFDWP for two hours.

D. Incorrect - MS-95 controls turbine speed. MS-95 is controlled by a primary relay that is a speed sensitive mechanical/hydraulic power amplifier. Loss of air has no effect on speed control.

43. Describe the methods for throttling EFDW flow, available to the operator. (R49)
44. Describe how the TDEFDWP meets "AC Independence" criteria include how each component helps provide this independence. (R38)
45. Explain the purposes of the Nitrogen bottles associated with the EFDW System. (R39)
46. List the local indications available for monitoring MDEFDWP operation. (R40)
47. List the local indications available for monitoring TDEFDWP operation. (R41)
48. List the instrumentation available in the Control Room for monitoring the operation of the MDEFDWPs and the TDEFDWP. (R42)
49. Explain why the Jockey Pump is stopped when performing the TDEFWDW Pump Backup Cooling Water Supply Test (PT/1,2,3/A/0150/022L). (R62)
50. Describe the emergency feedwater flowpath during performance of the Turbine Driven Emergency Feedwater Pump Test (PT/1,2,3/A/0600/012). (R63)
51. Explain why the TDEFDW Pump is NOT tested on Main Steam if reactor power < 5%. (R64)
52. Given a set of data and a copy of PT/0600/012, Turbine Driven Emergency Feedwater Pump Test, determine if acceptance criteria are being met. (R65)
53. Given a copy of PT/0600/013, Motor Driven Emergency Feedwater Pump Test, a set of conditions, and applicable portions of ITS, determine what ITS actions are required to perform the test. (R66)
54. Explain why it is necessary to install a dP transmitter to verify combined flow from the MDEFDW Pumps when performing the Emergency Feedwater Pump Suction From Hotwell Test (PT/1,2,3/A/0600/014). (R67)
55. Describe the purpose of performing PT/0/A/251/14, FDW Check Valve functional test, following refueling outages. (R60)
56. Describe the operator action necessary to prevent an EFDW System AUTO actuation during a normal unit shutdown. (R48)
57. Describe the alternate ICS flow path to the SGs, using the MDEFDWPs, and the TDEFDWP. (R45)
58. Describe how to manually control FDW-315 & 316 locally and recognize that local control may be necessary to prevent EFDWP and/or SG damage if AUTO control fails. (R50)

a) Two position switch, Normal/Bypass located on Steam Generator Level Control (SGLC) panel used for testing, in a loss of LPSW scenario, or a loss of power only to 1XS3.

b) Indicator lights show actual valve position, regardless of switch position.

c) **Normal** - HPSW-184/LPSW-138 shut.

1) Solenoids are energized, air is supplied to HPSW-184/LPSW-138 operators.

d) **Bypass** - HPSW-184/LPSW-138 open.

1) Solenoids are de-energized; air is bled off of valves.

4. Power Supplies for cooling water:

Note: Power supplies are not the same for each unit. PIP: 99-0971 was written to correct the DBD, which incorrectly describe the power supplies.

VALVE ↓ UNIT →	UNIT 1	UNIT 2	UNIT 3
LPSW – 137	1XC	2XAA	3XAA
LPSW – 138 / HPSW – 184	1XC	2XC	3XC

B. Nitrogen Backup

1. Following valves have N² backup. This insures adequate steam regulation and level control on a loss of IA for at least **2 hours**.

- FDW-315 (SG "A" EFDW Control Valve)
- FDW-316 (SG "B" EFDW Control Valve)
- MS-87 (MS to TDEFDWP Control)
- MS-126 (MS to AS Control)
- MS-129 (MS to AS Control)

C. FDW-315 and 316 may be throttled locally by a manual handwheel, if required.

D. MS-93 does not require AC power to open.

E. Starting oil pressure supplied by DC oil pump.

2. FDW-94 & FDW-96 should not be opened until downstream feedwater valves have been repositioned. This is to preclude their having to operate against EFDW pump shutoff head.

2.5 Recirculation Flowpath

A. > 150 gpm continuous recirculation flow to the UST; limited by recirculation orifices; local flow indication only.

2.6 The TDEFDWP is also provided with a test line to the UST. This line is used for performance testing and for running the TDEFDWP in recirculation for training.

2.7 Steam Supplies (Figure OC-CF-EF-2)

A. Steam is supplied from Main Steam or Auxiliary Steam.
   - Main steam via MS-82 & 84 controlled by MS-87.
   - MS-82 & 84 come from "A" & "B" main steam lines and are controlled from the unit Control Room.

B. MS-87 is operated by a MOORE controller to maintain a steam pressure setpoint of 310 psig.
   - The controller has a battery backup that will prevent MS-87 from failing open on loss of power to the controller. This is designed to last for approx. 2 hours.
   - An alarm on SA-12 will alert the operator of such a loss of power and instruct the operator to isolate the MS supply to preclude MS-87 failure overpressurizing the line.
   - MS-87 controller is located near the valve and is programmed with no manual function.

C. MS-89 (Turbine Driven Steam Supply Block) This manual valve requires 104 turns to completely open and this valve also contains a pilot valve and a main valve. Refer to PIP 98-0444 for information on mis-operation of this valve during TDEFWP testing.

D. Auxiliary steam is supplied through block valve AS-38 which comes from the AS header.
   - The AS header is controlled at 300 psig by regulating valves MS-126 & 129.
   - The TDEFDWP exhausts to atmosphere.

E. Minimum steam pressure to assure operability is **250 psig**.

F. MS-93 (EFDWP Steam Supply) is normally positioned to closed. Air is supplied to the valve operator by both Instrument Air (IA) and Auxiliary Air (AIA).

- When the TDEFDWP is started air pressure is removed from the valve operator and the valve opens to allow steam to pass on to the turbine.
- If MS-93 fails to open when the TDEFDWP is started, air from AIA and IA can be isolated locally then bleed air off MS-93 to open the valve.

1. Manual Start of TDEFDWP
   a) MS-93 Failure.
   b) Take control switch to "Run" this should open MS-93 by de-energizing the solenoid.
2. MS-93 manual opening
   a) Isolate IA supply to MS-93.
   b) Isolate AIA supply to MS-93.
   c) Bleed air off regulator using moisture petcock.
   d) MS-93 should open & the TDEFDWP will start
3. If MS-93 fails to open following method in step above, isolate both air supplies and utilize the quick connector on the air supply line on top of MS-93.
   a) Insert hollow plug into quick connector to bleed air from MS-93.

2.8 Turbine

A. The TDEFDWP is driven by a single stage G.E. turbine rated at 930 HP.
   - The casing drain is left throttled open to prevent the accumulation of water.

B. TDEFDWP Speed Control Mechanism (Figures OC-CF-EF-5&9)
   1. Turbine speed is controlled by the Primary Relay which detects any changes in set speed and automatically controls the operating valve, MS-95, to maintain set speed. Turbine speed is controlled as follows:
      a) The primary relay is a speed sensitive mechanical/hydraulic power amplifier, which auto controls the operating valve (MS-95).
      b) The speed governor, which is mounted to the pump shaft, transmits speed changes to movement of the pilot valve.
      c) Constant oil pressure is supplied by the shaft driven oil pump or the auxiliary oil pump to the primary relay operating piston.

QUESTION # 93

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	_____
	Group #	**2**	_____
	K/A #	**014000 A1.02**	
	Importance Rating	**3.2**	_____

Technical Reference(s): **IC-CRI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-CRI #17.2**

Question Source:
- Bank # **IC-303**
- Modified Bank # _______
- New _______

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 _____

Comments:

1 POINT

QUESTION # 93

RO ONLY

Unit 1 plant conditions are as follows:

- Group 7 average position = 86%
- Rod 7-3 Position Indication (PI):
  - ➢ Relative rod position indication (RPI) = 85%
  - ➢ Absolute Position Indication (API) = 90%

Which ONE of the following is the proper operator action to **MATCH** Rod 7-3 RPI with Rod 7-3 API? (.25)

A. Position Rod 7-3 to match Group 7 average.

B. Position all Group 7 rods to match the current Rod 7-3 API.

C. Select Rod 7-3 on the Group/Single select switch then use the Reset Pulser to align RPI with API.

D. Select Group 7 rods on the Group/Single Select switch, and use the Reset Pulser to align Group 7 rods RPI with each rod's RPI OAC indication.

1 POINT

QUESTION # 93
014000 A1.02

A. Incorrect - Performing this action will not align the API and RPI. The API and RPI remain misaligned when the rod is moved. Rods are not moved to match API to RPI. The Reset Pulser is used to match RPI to API.

B. Incorrect - Performing this action will not align the API and RPI. This distracter would increase power by withdrawing Group 7 rods. The Reset Pulser is used to match RPI to API.

C. Correct - OP/0/A/1105/09 (CRD System), requires the operator to rotate the single select switch to the desired CRD and use the Reset Pulser switch.

D. Incorrect - Performing this action is not in compliance with the approved procedure OP/0/A/1105/09. RPI is not matched with API and matching RPI with the OAC indication will not correct the problem.

14. Explain the purpose for the Clamping Contactors associated with the CRD power supplies. (R14)

15. Explain the CRD Patch Panel including the associated S.L.C. requirement. (R15)

16. Given a Limit and Precaution from OP/O/A/1105/09, Control Rod Drive System, explain the basis of the limit or precaution. (R16)

17. Given the procedure, describe the bases of the steps involved in the following CRD system evolutions: (R17)

    17.1 Transferring between D.C. Hold, Auxiliary and Regulating power supplies for the CRDs.

    17.2 Latch and PI alignment of a safety group or any individual rod.

18. Describe the process for verifying the "A" and "CC" phases of Groups 1-4 stators operable. (R18)

19. Describe how the operator resets the Control Rod Drive Trip breakers from the Diamond Control panel. (R19)

20. Discuss the following concerning the Diverse Scram System (DSS): (R20)

    20.1 Operation and bases of DSS

    20.2 Signal inputs

    20.3 Actuation setpoints

    20.4 System reset

    20.5 Operability verification by the operator

21. Apply ITS/SLC's rules to determine applicable Conditions and Required actions for a given set of conditions. (R21)

22. Given a copy of Improved Technical Specifications, and associated Bases, analyze a given set of conditions for applicable ITS/SLC LCO's. (R22)

23. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITC/SLC's. (R23)

2. Transfer of a group/rod from auxiliary power supply to its normal power supply.
   a) Select group desired on the Group Select Switch.
   b) Select ALL or desired rod on the Single Select Switch.
   c) Press selector for SEQ OVERRIDE.
   d) Press selector for AUXILIARY.
   e) Select JOG on the Speed Selector.
      - NOTE: Insure Manual Transfer Sync. Light is lit before pressing clamp.
   f) Press selector for CLAMP.
   g) Press selector for MANUAL TRANSFER switch until the TRANSFER CONFIRM lamp and all CONTROL ON lamps on the PI panel go off.
   h) Press selector for CLAMP RELEASE.
   i) Press selector GROUP.
   j) Select SEQUENCE.
   k) Press selector for TRANSFER RESET.
- Transfer is complete.
   l) Select RUN

C. Latch and PI Alignment
- (Refer to proper Encl. of OP/0/A/1105/09)
- NOTE: It is not necessary to perform latch on the APSR's (Group 8).
1. Latch and PI alignment of safety rod group or <u>any</u> <u>individual</u> rod.
   a) Transfer safety group or any individual safety or regulating rod to its auxiliary power supply per appropriate encl.
   b) Select JOG on the Speed Selector.
   c) Press selector for the LATCH switch and insert rod for 15 seconds.
   d) Release LATCH switch.
   e) Compare absolute and relative readings on the PI panel.
   f) If it is required to reset the relative PI, rotate the single select switch to the desired CRD and use the PI reset raise/lower switch.
   g) Transfer group/rod back to its normal power supply per enclosure.
      - NOTE: This step is not required if the rod/group is to be withdrawn.
2. Latch and PI Alignment of regulating <u>group</u> as follows:

a) Press selector for GROUP.

b) Select CRD group on the Group Select Switch.

c) Press selector for SEQ OVERRIDE.

d) Select JOG on the Speed Selector.

e) Press selector for the LATCH switch and insert for approximately 15 seconds. Release the LATCH switch.

f) Compare absolute and relative readings on the PI panel.

g) If it is required to reset the relative PI, rotate the single select switch to the desired CRD and use the PI reset raise/lower switch.

D. CRD Movement Test

- Refer to PT/0/A/600/15

1. Purpose - to periodically test CRD operation under actual operating conditions.

2. Test Method

   a) CRD groups 1-6 will be inserted approx. 10% and returned to original position. CRD groups 7 & 8 will be inserted $\approx$ 2.5%. (On-Line)

   b) An alternate test method is the withdrawing of control rods a minimum of 2.5% during startup.

3. Acceptance criteria requires all control rods move as commanded.

4. **Control Rod Movement During Unit Startup**

   a) <u>A note in PT requires DIRECT supervision by SRO (Signature required).</u>

   b) If Group 8 not moved > 2.5% during startup then select group 8 and insert ~2.5% (~6 seconds) followed by a repositioning to original group 8 position.

   c) All other groups should be moved >2.5% due to power escalation and will be signed off as > 2.5% movement is verified.

Exam Question Report

27-Jan-99

Question ID:	IC303	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC303				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-CRI - CRD Instrumentation
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = R17; SRO = R17 Reference: NRC CR-11/93	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

KA: 014000A1.02

Given the following conditions for Unit 1:

- Group 7, rod 3 had been previously stuck.
- Rod 7-3 has been freed using a temporary procedure.
- However, the relative rod position indication (RPI) on the Position Indication (PI) panel for Rod 7-3 DOES NOT agree with its absolute position indication (API).

Which ONE of the following is required to realign Rod 7-3 RPI to match Rod 7-3 API? (.25)

A) Move Group 7 rods to Rod 7-3 API and realign all Group 7 rods with Rod 7-3 API.

B) Drive all Group 7 rods to the in-limit and realign the rods to the Zero position.

C) Select Rod 7-3 on the Group and Single Select switch, and use the Reset Pulser to align RPI with API.

D) Withdraw all Group 7 rods to the nearest zone indicating lamp and realign all Group 7 rods to the zone reference indication.

Answer

C

Lessons

ID	Description
IC-CRI	Control Rod Indication (IC-CRI)

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag
NRC CR-11/93	Reference created by conversion		

KA'S

ID	Description

QUESTION # 94

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	_____
Group #	**2**	_____
K/A #	**016000 A3.02**	
Importance Rating	**2.9**	_____

Technical Reference(s): **IC-RCI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-RCI #9**

Question Source:
Bank # **IC-381**
Modified Bank # _______
New _______

Question History:
Previous NRC Exam _______
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 94

RO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 100%
- SASS is DE-ENERGIZED
- PZR LEVEL #2 selected
- PZR TEMPERATURE "A" = 649° F
- PZR TEMPERATURE "B" = 649° F
- PZR TEMPERATURE "C" = 649° F
- 1HP-120 (RC Volume Control) is in AUTOMATIC

CURRENT CONDITIONS:

- PZR TEMPERATURE "A" = 649° F
- PZR TEMPERATURE "B" = 145° F
- PZR TEMPERATURE "C" = 649° F

Which ONE of the following actions will take place IMMEDIATELY?

RCS makeup flow will ______ and actual PZR level will _____.

A. remain the same / remain the same

B. decrease / increase

C. increase / decrease

D. increase / increase

1 POINT

QUESTION # 94

016000 A3.02 RSI/PRA 5-2-00

Question setup: The PZR has three RTD's "A", "B", and "C". RTD "A" feeds temp comp for level #1 and 2 (ICCM A). RTD "B" feeds temp comp for level #3 (ICCM B). RTD "C" feeds OAC indication only.

If RTD "B" fails low then corrected level #3 will be affected and fail low. If SASS is de-energized and Level #3 was selected, then an upset in the automatic level control circuit would occur. In this question no upset will occur, as the temperature compensation is not affected for the level channel selected.

A. Correct - PZR #2 fed by temp Compensation RTD A (NOT B). The level control circuit is not affected by this failure. No change in level will occur.

B. Incorrect - A false low temperature compensation will DECREASE indicated PZR level. This error between indicated controlling level vs. setpoint on HP-120 controller would cause HP-120 to OPEN and allow actual level to increase. As HP-120 OPENS MAKEUP FLOW increases causing actual PZR LEVEL and RCS inventory to increase.

C. Incorrect - PZR Level #2 is fed by temp compensation RTD "A" (NOT "B"), If PZR temperature compensation fails LOW (as indicated in PZR TEMP B that feeds Level #3) this would decrease indicated PZR level. If temperature failed low then PZR level indication would fail low and RC makeup flow would increase. The first portion would be correct if the controlling channel was affected.

D. Incorrect - This would be correct if level 3# was selected

5. Given a set of conditions describe the required operator actions when selecting an alternate controlling signal. (R20)

6. Applying the knowledge of simplified instrumentation drawings be able to determine how various indications and control functions are processed for RCS temperature, pressure, level and flow including: (R2, 3, 62)

   6.1 Range of the indicator

   6.2 Source of the signal

7. Given a set of conditions analyze proper operation of RCS TEMPERATURE indications that the operator uses to monitor and control unit operation including the following: (R3, 4, 5, 6, 10)

   7.1 RCS T-hot

   7.2 RCS T-cold

   7.3 Core exit temperature (CETCs)

   7.4 Pressurizer temperature

8. Given a set of conditions analyze proper operation of RCS PRESSURE indications that the operator uses to monitor and control unit operation including the following: (R6, 7, 9, 63, 10)

   8.1 RCS Loops

   8.2 ICCM WR Pressure

   8.3 Low Range Cooldown

9. Given a set of conditions analyze proper operation of RCS LEVEL indications that the operator uses to monitor and control unit operation including the following: (R13, 15, 16, 17, 18)

   9.1 Pressurizer level and pressure

   9.2 Reactor Vessel (LT-5)

   9.3 Ultrasonic Level Indication (ULI)

   9.4 Tygon tubing

10. Given a set of conditions analyze proper operation of RCS FLOW indications that the operator uses to monitor and control unit operation including the following: (R17, 18)

    10.1 Loop RC Flow

    10.2 Total RC Flow

    10.3 Pressurizer Relief Valve Flow

3. **Low Range (0 - 600 psi)**
   a) RC Low Range Cooldown Pressure meter
      1) Transmitter connected to one of the ICCM RVLIS impulse lines.
         (a) Transmitter always valved-in. The meter is turned on during a unit shutdown at approx. 600 psi. This provides a more accurate RCS pressure signal for the operator at low RCS pressures.
         (b) The Low Range Cooldown pressure signal is used in the RC-66 (PORV) **"LOW"** automatic opening circuit.
            (1) When the PORV select switch is **ON** and positioned to **LOW** the PORV setpoint becomes 475 psi and the controlling signal becomes the Low Range Cooldown pressure signal instead of Narrow Range RC pressure.
            (2) An Off/On switch located under the meter turns the indicator **ON/OFF.** When the PORV setpoint is in the **LOW** position the PORV will not open if switch is **OFF,** therefore the PORV is inoperable for RCS low pressure over pressure protection if the switch is **OFF**.
         (c) The digital Dixon meter will blink and bar graph will be "pegged" high when RCS pressure is >600 psi and the switch is **ON**.

C. RCS Level

1. Pressurizer Level Transmitter Operation (ROSEMOUNT)
   a) Three ΔP transmitters are used for indication and control of Pressurizer level. All three individually feed the Hi/Lo statalarm.
   b) Level #1 and #2 are fed through ICCM Train A (safety related). #2 tap also feeds SSF uncompensated PZR level and PZR pressure.
      1) Level tap #2 has a separate rosemount pressure/level transmitter to feed PZR uncompensated level and PZR pressure indication to the SSF control room.
   c) Level #3 is fed through ICCM Train B (safety related)
   d) Pressurizer (PZR) level control is protected by SASS.
   e) The reference legs of the three transmitters are connected to the upper (steam) portion of the Pressurizer and have individual taps.
   f) The variable leg of the transmitters are connected to the lower (water) portion of the Pressurizer and have individual taps.

1) Temperature/Pressure compensation:
2) Pressurizer temperature is supplied from one single well with three separate RTD elements located in the well
    (a) (Temp A through ICCM A, B through ICCM B, C fed to non-safety related indication to the OAC and provides input for the calculation of PZR saturation pressure on UB1).
3) Pressurizer level and temperature are used within the ICCM to develop temperature compensated level. The temperature compensated level signal, which is selected by the operator from UB1 or SASS is supplied to UB1 (PZR chart recorder), ICS level controller HP-120 (RC Volume Control), ASDP, Low level heater cutoff, Emergency hi/low level alarm.
4) RC Pressure fed to each ICCM Train is used to develop pressure compensated level for OAC indication <u>only</u>.
5) Temperature, pressure, and uncompensated pressurizer level is supplied to the OAC for indication. The operator has the ability to monitor Level 1, 2, and 3 temperature compensated PZR levels and channel A and B PZR temperature fed from ICCM to the "Dixon" meters on UB1.
6) The temperature of the water in the Pressurizer varies from near ambient (at cold S/D) to 650°F (at normal operating pressure). This large change in temperature results in a significant change in the density of the water in the Pressurizer. (variable leg)
7) The temperature of the water in the reference leg does not change significantly during any operation. The density does not change significantly.
8) During a startup, when PZR water temperature increases, causes a large density change in the variable leg with small density change in the reference leg, results in uncompensated PZR level indicating <u>lower</u> than actual.
9) Uncompensated PZR level will always indicate <u>lower</u> than actual.
10) To compensate for variations in the water density, PZR temperature and RC pressure signals are used to provide temperature and pressure compensated PZR level.
11) Temperature and pressure can be used as an accurate level compensation in a saturated system such as the Pressurizer because changes in liquid density can be determined by the use of steam tables.

2. Pressurizer Level recorder (0-400"):

   a) The operator or SASS selects one of three level signals. Whichever signal is selected is temperature compensated prior to being sent to the recorder.

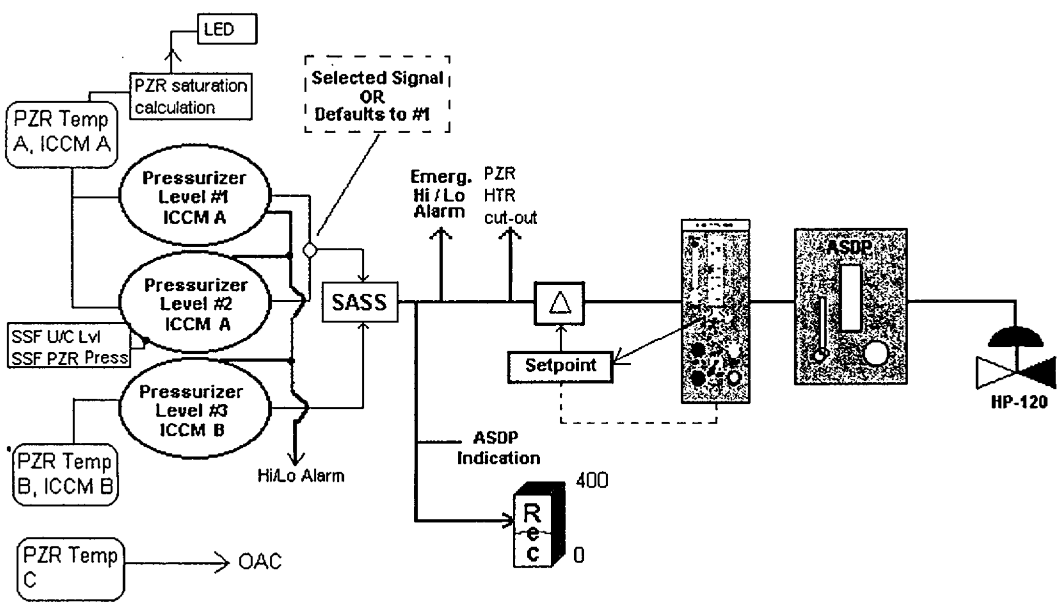
   b) whichever signal is selected is also used in:

      1) HP-120 (RC Volume Control) **AUTO** control circuit.

      2) Pressurizer Heaters low level cutoff circuit. Heaters will auto cutoff at 80" decreasing. At 80" increasing, the heaters will return to the mode they were in prior to level going below 80", *EXAMPLE: if the heaters were in AUTO, they will return to AUTO. If they were in MANUAL/ON, they will return to MANUAL/ON. If they were in MANUAL/OFF, they will return to MANUAL/OFF.*

      3) ASDP

      4) Emergency Hi/Lo level statalarm.



Exam Question Report

27-Jan-99

Question ID:	IC381	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC381				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-RCI - RCS Instrumentation
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 10; SRO = 10 Reference: (IC-RCI P21-23) OBJECTIVE 10,11,AND 13	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

KA: 016000A3.02

After taking the shift turnover on Unit 1 with all plant conditions normal, the following conditions are observed five (5) minutes after turnover:

- The plant is operating at 100% power.
- PZR LEVEL #2 selected.
- PZR TEMPERATURE "A" indicates 700□F.
- PZR TEMPERATURE "B" indicates 145□F.
- PZR TEMPERATURE "C" indicates 649□F.
- SASS is DEENERGIZED.
- 1HP-120 (RC Volume Control) is in AUTOMATIC.

Which ONE of the following describes the immediate effects on the RCS makeup system and actual pressurizer level? (.25)

<u>MAKEUP FLOW</u> <u>ACTUAL PZR LEVEL</u>

A) Increases Increases

B) Decreases Increase

C) Increases Decrease

D) Decreases Decreases

Answer

D

A. Incorrect - PZR Level #2 fed by Temp compensation RTD "A" (NOT "B"), If PZR temperature compensation fails LOW (as indicated in PZR TEMP B that feeds Level #3) this decreases <u>indicated</u> PZR level.
B. Incorrect - first portion correct; second portion incorrect.
C. Incorrect - neither portion is correct
D. CORRECT - PZR #2 fed by Temp Compensation RTD A (NOT B). As indicated PZR level increases from the false high temperature compensation (High scale of RTD = 700□F) INCREASES indicated PZR level. An error between indicated controlling level vs. setpoint on HP-120 controller is developed causing HP-120 to CLOSE and allowing actual level to decrease. As HP-120 CLOSES MAKEUP FLOW will decrease causing <u>actual</u> PZR LEVEL and RCS inventory to decrease.

Lessons

ID	Description
IC-RCI	Reactor Coolant System Instrumentation

Enabling Objectives

QUESTION # 95

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	____
	Group #	**2**	____
	K/A #	**026000 K1.01**	
	Importance Rating	**3.4**	____

Technical Reference(s): **PNS-BS, EAP-E31**
EAP-E25

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-BS 4.2 / EAP-E31-5**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 95

RO ONLY

Unit 1 Plant conditions:

- LBLOCA has just occurred
- BWST = 38 feet and decreasing
- RB pressure is 12 psig
- "A" RBS pump is out of service
- "B" RBS header flow is 1650 gpm.
- Statalarm "A" BS Header Flow High / Low is in alarm.
- All other ES components actuated as required and are performing as required.

Which ONE of the following is correct in response to the above conditions?

A. No action is required.

B. Throttle "B" RBS header flow to 900 gpm.

C. Throttle "B" RBS header flow to 1500 gpm.

D. Throttle "B" RBS header flow to 1700 gpm.

1 POINT

QUESTION # 95

026000 K1.01 RO ONLY

A. Incorrect, - Statalarm "A" BS Header Flow High / Low" is expected because "A" RBS pump is out of service and flow will be below the 1300 gpm setpoint, however flow in the "B" BS header is above the 1500 gpm limit.

B. Incorrect, - If RBES were aligned for suction, required RBS flow is required to be throttled to 900 -1000 gpm per pump.

C. Correct, - Flow should be throttled to limit spray to less than or equal to 1500 gpm to prevent pump runout.

D. Incorrect, - 1700 gpm is the BS Header High Flow alarm setpoint. Procedure direction requires ≤ 1500 gpm.

TRAINING USE ONLY

LPRO - TRAINING OBJECTIVES

The student will be able to: TIME: 1 Hours

A. TERMINAL OBJECTIVE

Describe the use of CP-601 (Cooldown Following Large LOCA) of the Emergency Operating Procedure, in order to perform the required actions of a Nuclear Control Operator during an event involving excessive heat transfer.

B. ENABLING OBJECTIVES

1. Recognize that even with adequate LPI flow, superheated conditions may exist for up to 10 minutes following a large break LOCA. (R1)

2. State the maximum allowable time for operation of the LPI pumps against a shutoff head.

3. Describe the three (3) actions taken to cool and depressurize the RCS if LPI and HPI flows are not adequate, or if HPI flow is adequate, but cannot be established to each header. (R3)

4. Explain the basis for checking for an immediate core flood tank blowdown and indicated LPI header flow with RCS pressure > 200 psig. (R4)

5. State the basis for the 1500 gpm/header limit on RBS flow. (R5)

6. State two benefits gained from caustic addition to the LPI System. (R6)

7. State the basis for the limit on BWST and RB levels prior

4. If HPI is not functioning correctly, then the RCS must be rapidly cooled and depressurized per step 4.

2.4 Initiate a Rapid RCS Cooldown and Depressurization

A. Raise SG levels to Loss fo SCM setpoint using EFDW (See EOP curve).

B. Lower SG pressure such that S/G Tsat is at least 90EF less than RCS Tsat, using TBV's (if available) or atmospheric dump valves (per EOP enclosure).

   1. Depressurize both S/G's to cool and depressurize the RCS as rapidly as possible. Tech. Spec. limits may be exceeded.

C. Depressurize the RCS via the PORV and the RV Head and Loop Vents.

   1. The lost inventory will be replaced by the CFT's.

   2. The PORV will remain open until LOW or HIGH is re-selected on its switch.

D. Rationale for step 4.

   1. Since LPI is not injecting, and HPI is not functioning correctly, a rapid RCS cooldown and depressurization is the only option left.

   2. Limit S/G fill rate to required XSUR level to . 200 gpm/hdr if both SGs are available. If flow cannot be established to both SGs, then increase flow to the available SG to . 400 gpm.

   3. Compare average of five (5) highest qualified CETC's (ICCM) to saturation value for S/G (from steam tables) to maintain at least 90EF delta T.

2.5 Monitor LPI flow and CFT level (5.0)

A. Check for a CFT/LPI line break which could cause only one CFT to empty, and make it appear as though a LOCA has occurred. If only one CFT has emptied and/or asymmetric LPI flow is observed, then a CFT/LPI line break may be indicated.

B. If a CFT/LPI line break has occurred, it would:

   1. Need to be isolated per step 5.

OBJECTIVES

TERMINAL OBJECTIVES

1. Describe the purpose, location and modes of operation in regard to the RBS System. The student should also recognize important power supplies associated with the system. (T1)

2. Assess the status of the RBS system during various system conditions to verify proper operation and determine any required corrective actions.(T2)

ENABLING OBJECTIVES

1. State the two (2) purposes of the RBS System. (R1)

2. Given a set of conditions, determine if containment design pressure and temperature limits will be met. (R16)

3. List the power supplies for the RBS Pumps (R3)

4. List the following flow values for the RBS pumps. (R2, R7)

   4.1 Minimum flow requirement

   4.2 Normal ES flow when taking suction from BWST

   4.3 Normal flow when taking suction from RB Emergency Sump (RBES)

5. Draw the RBS System labeling all major components and valves. Include the following: (R5, R10)

   5.1 BWST

   5.2 RBES

   5.3 Recirc flowpath to BWST (for testing)

6. State the setpoint, statalarms armed, and equipment actuated by ES Channels 7 and 8. (R6)

7. For PT/0204/007, RBS Pump Test, describe: (R12)

   7.1 The purpose

   7.2 How the test is performed

8. Given a copy of PT/0204/007, RBS Pump Test, and a set of data, evaluate if the acceptance criteria is met. (R13)

2) Non-safety related outputs:
   (a) BS Flow Train A and B flow gauge on VB2
   (b) RBS Flow High/Low statalarm
   (c) OAC Computer point (BS Line A Flow)
   (d) OAC Computer point (BS Line B Flow)

2.2 Modes of Operation

A. ES Mode (Channels 7 and 8)

1. Setpoint
   a) The RBS System automatically actuates if two of the three ESG RB pressure analog channels reach 10 psig.
      1) The ITS required setpoint is $\leq$ 15 psig RB pressure.
2. The following actions occur if the RBS System actuates:
   a) Both RBS pumps start.
   b) BS-1 and BS-2 open.
   c) LP-21 and LP-22 receive an open signal to supply RBS pumps from BWST.
      1) These valves are normally open, but receive an ES signal in case they are closed.
   d) Refer to OP-OC-BS-3 to explain the ES flowpath.
3. When ES-7 and 8 actuate, the "BS Header Flow High/Low" statalarm for each header is armed.
   a) Low flow on a given header alarms at 1300 gpm.
      1) The operator is directed to check RBS pump operation, BWST level, and suction and discharge valve position by the Alarm Response Guide.
   b) High flow on a given header alarms at 1700 gpm.
      1) The operator is referred to the EOP for guidance to throttle RBS flow to within operating limits.
4. Nominal RBS pump flow is 1500 gpm when suction to the pumps is from the BWST.
   a) BS-1 and BS-2 should be throttled as required to maintain this flow rate as RB pressure changes occur following a LOCA.
   b) Flow Indication and Recorders are located on VB2.

The corresponding RZ modules should be checked to insure all Blue "Auto" lights and White "Position" lights are on. These lights indicate that the appropriate ES components have actuated.

A. If either channel fails to fully actuate manually initiate the affected channel(s) by one or both of the following methods:

- Depress TRIP pushbutton for the affected ES channel(s).
- Position any affected components on the affected ES channels to the required ES position.

NOTE: This step was added in response to SOER 94-01.

B. Verify proper Reactor Building Spray Flow

1. Throttle RB Spray pump flow, using BS-1&2, to ≈ 1500 gpm per pump to prevent damage to a RBS pump due to pump run out.

C. If ES Channels 7 and 8 have actuated ALL ES Channels should have actuated.

2.6 If LPI pumps are operating against a shutoff head, THEN inform SRO.

A. If the RCS pressure is greater than 185 psig and not decreasing, then the continued operation of the LPI pumps against a shutoff head should not exceed 30 minutes to prevent overheating of the pumps. In this case, all LPI pumps should be stopped as long as both HPI trains are injecting.

B. SRO approval is required for the decision to secure these pumps.

C. The entire team should be made aware that the LPI pumps are being secured, and that if RCS pressure decreases to a point where they could inject into the reactor vessel (≈200 psig), the pumps should be restarted.

1 POINT

QUESTION # 96

RO ONLY

Unit 1 plant conditions:

- Reactor Power = 100%
- 1TD-2: Load Shed #1 Light is illuminated
- 1TD-16: Load Shed #2 Light is **not** illuminated (bulb checks good)

Which ONE of the following is correct?

The lights indicate ______ Load Shed channel(s) is/are operable on 1TD and ______ Load Shed channel(s) is/are required for Load Shed to actuate.

A. one / only one

B. two / only one

C. one / both

D. two / both

1 POINT

QUESTION # 96

063000 K3.02 (3.5/3.7) RO ONLY PRA 04/11/00 (new question)

Question setup:
The lights indicate if power is available for the Load Shed circuits associated with 1TD. There are two channels of Load Shed. If a light is out it indicates that the associated Load Shed circuit has no power and will not function. However only one Load Shed Channel is require for Load Shed to occur.

A Correct – The light indicates one Load Shed Channel associated with 1TD is inoperable because it has lost power. Only one Load Shed Channel is required for Load Shed to operate correctly

B Incorrect – First part is not correct. Only one Load Shed Channel is operable. Second part is correct.

C. Incorrect - First part is correct. Second part is not correct. Both channels are not required for Load Shed to actuate.

D. Incorrect – First part is not correct. Second part is not correct. Both channels are not required for Load Shed to actuate.

1 POINT

QUESTION # 96

RO ONLY

Unit 1 Plant conditions:

INITIAL CONDITIONS:

- Reactor Power = 100%
- 1TD-2: Load Shed #1 Light is illuminated
- 1TD-16: Load Shed #2 Light is **not** illuminated (bulb checks good)

CURRENT CONDITIONS:

- Turbine Trip
- CT-1 Lockout

Which ONE of the following is correct?

The 4160 loads on 1TD _____ Load Shed because...

A. will / the lights indicate Load Shed has actuated.

B. will / only one channel is required for Load Shed to function.

C. will **NOT** / the lights indicate that Load Shed will not actuate.

D. will **NOT** / both Load Shed channels are required for Load Shed to be operable.

1 POINT

QUESTION # 96

063000 K3.02 (3.5/3.7) RO ONLY PRA 04/11/00 (new question)

A. Incorrect – First part is correct. Only one Load Shed Channel is required for Load Shed to operate correctly. PSL lights in the Cable Room indicate operable/inoperable status. These lights will not indicate status and turn BRIGHT when the channel has actuated as some other light indications.

B. Correct – The light indicates one Load Shed Channel associated with 1TD is inoperable because it has lost power. Only one Load Shed Channel is required for Load Shed to operate correctly

C. Incorrect - First part is not correct, as Load Shed will occur with only one channel.

D. Incorrect – First part is not correct, as Load Shed will occur. Both channels are required for Load Shed to be operable.

QUESTION # 96

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	_____
	Group #	**1**	_____
	K/A #	**063000 K3.02**	
	Importance Rating	**3.5**	_____

Technical Reference(s): **EL-PSL**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EL-PSL OBJ. #3.3**

Question Source:
- Bank # _______
- Modified Bank # _______
- New ___X___

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge _____
- Comprehension or Analysis __x__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 _____

Comments:

1 POINT

QUESTION # 96

RO ONLY

Unit 1 Plant conditions:

INITIAL CONDITIONS:

- Reactor Power = 100%
- 1TD-2: Load Shed #1 Light is illuminated
- 1TD-16: Load Shed #2 Light is **not** illuminated (bulb checks good)

CURRENT CONDITIONS:

- Turbine Trip
- CT-1 Lockout

Which ONE of the following is correct?

The 4160 loads on 1TD _____ Load Shed because _________function.

A. will / the lights ~~indicates~~ indicate Load Shed will   because there is only 1 line.

B. will / only one channel is required for Load Shed to

C. will **NOT** / the lights ~~indicates~~ indicate that Load Shed will not

D. will **NOT** / both Load Shed channels are required for Load Shed to

A is correct –

D is reason for C ∴ Both cannot be correct

1 POINT

QUESTION # 96

063000 K3.02 (3.5/3.7) RO ONLY PRA 04/11/00 (new question)

A. Incorrect – First part is correct. Only one Load Shed Channel is required for Load Shed to operate correctly. PSL lights in the Cable Room indicates operable/inoperable status. If the light is OFF on the 4160 breaker cubical the associated Load Shed channel is inoperable.

B. Correct – The light indicates one Load Shed Channel associated with 1TD is inoperable because it has lost power. Only one Load Shed Channel is required for Load Shed to operate correctly

C. Incorrect - First part is not correct, as Load Shed will occur.

D. Incorrect – First part is not correct, as Load Shed will occur. Both channels are not required for Load Shed to function.

OBJECTIVES

Terminal Objective

1. Discuss the EPSL, including the various power supplies and how each power source can be aligned to supply power to ONS Units MFB during Design Bases Events. (T1)

2. For a given set of plant conditions evaluate the status of the MFB power sources including automatic system actions, time frames for re-energizing the MFB, and what contingency actions are required if automatic actions do not occur. (T2)

Enabling Objectives

1. Concerning the Design Bases for the 4KV Essential Auxiliary Power System, describe the following: (R1)

   1.1 The System Functional Design Bases

   1.2 The Design Bases Events

2. Concerning a Keowee Emergency Start, describe the following: (R2)

   2.1 Purpose

   2.2 Panel location

   2.3 Emergency Start signals

3. Describe the following for the Load Shed Logic:

   3.1 Purpose (R3)

   3.2 Panel location (R4)

   3.3 The conditions, which will initiate a load-shed signal and the logic, involved. (R5)

   3.4 Loads which will be load shed (R6)

   3.5 How to reset a load shed signal (in the Cable Room) (R7)

   3.6 How to reset a load shed signal (R8)

   3.7 The location of the fuses for load shed in the 4160V switchgear (R9)

   3.8 How to verify power is available to the load shed trip relays (R10)

4. Initiate closing of the startup breakers if the remainder of the automatic closing logic is satisfied.

F. A modification to the 4160V switchgear provided a dedicated fuse block to supply power to the load shed trip initiate relays to ensure that the load shed circuits will be operable even if the 30 amp pull out fuse blocks are removed.

1. This modification was necessary because of incidents where the 30 amp control power fuse block located in certain 4160V breaker compartments were pulled to isolate that breaker.

    a) These fuse blocks are provided in two breakers on each 4160V bus - TC, TD, TE

    b) Each fuse block provides power to the load shed relay for channel 1 or 2 for all load shed components within that switchgear.

    c) Removal of one of these fuse blocks renders one channel of load shed inoperable to that 4160V switchgear string (TC, TD, TE).

2. The dedicated fuse block provides power to the load shed trip initiate coil to ensure operability.

3. Computer alarms are provided to alert the control room if any dedicated circuit loses power.

4. A power available lamp is provided on the front of each breaker cubicle containing a dedicated fuse block to allow the operator to determine the specific circuit which has lost power. The lamp should be on to verify power available during normal operation.

G. Load shed panel

1. The load shed panels are included as a part of the EPSL panels located in the cable room of each unit.

    a) Contains a channel A and a channel B panel

2. Indications and Controls

    a) Main Feeder Bus 1 (2) Unit 1 (2)(3) PT

        1) amber light

        2) monitors power on associated MFB (uses potential transformer)

        3) normally on

DESCRIPTION OF EQUIPMENT	RANGE	NIGHT	DAY	REMARK
4160 V SWITCHGEAR 1TD				
Indicating Lights	Operable			
Targets	None			
1TD-0: MDEFWP Test Delay "NORMAL" light	Lit			
1TD-2: 1X5 Trip & Time Delay Reclose Cir. 'A' Light	Lit			
1TD-2: 1TD Load Shed #1 and 1X5 Trip & Time Delay Reclose Cir. 'B' Light	Lit			
1TD-16: 1TD Load Shed #2 Light	Lit			
4160 V SWITCHGEAR 1TC				
Indicating Lights	Operable			
Targets	None			
1TC-2: 1TC Load Shed #1 Light	Lit			
1TC-15: 1TC Load Shed #2 Light	Lit			
600 V LOAD CENTERS 1X1-1X7				
Indicating Lights	Operable			
Gas Press.	13-7.5-4.5 psig			
Gas Temp.	200-80-N/A °C			
Normal Bkrs.	Closed			
Emergency Bkrs.	Open			
Bkr. Control Selector Sw.	Auto			
Pzr. Htr. Power Supply Ground Monitor Lights	Not Lit			
Charging Spring Switches	On			

QUESTION # 97

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	_____
	Group #	**3**	_____
	K/A #	**027000 K1.01**	_____
	Importance Rating	**3.4**	_____

Technical Reference(s): **PNS-BS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **AM-7 #5**

Question Source:
- Bank # _______
- Modified Bank # **EAP-212**
- New _______

Question History:
- Previous NRC Exam _______
- Previous Quiz / Test _______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis _____

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 _____

Comments:

1 POINT

QUESTION # 97

RO ONLY

Which ONE of the following describes a benefit of adding **CAUSTIC** to the LPI system after a LOCA?

A. Increases the acidity of the water in the RBES.

B. Decreases the amount of radioactive iodine produced.

C. Increases the amount radioactive iodine maintained in solution.

D. Decreases the formation of hydrogen from the Zirconium-water reaction.

1 POINT

QUESTION # 97

027000 K1.01

A. Incorrect – Addition decreases the acidity of the water in the RBES and neutralizes the pH. Primarily the pH is neutralized to decrease hydrogen formation in the RB following a LOCA.

B. Incorrect – The Iodine concentration is a fixed number for each core load. The fission process determines the amount of iodine produced.

C. Correct – Adding caustic increases the amount radioactive iodine maintained in solution. This addition increases the iodine partition factor.

D. Incorrect – Adding caustic lowers the hydrogen concentration following a LOCA by reducing the Zinc-boron (Zinc-corrosion) reaction but not the Zirc-water.

TERMINAL OBJECTIVE

Become better prepared to mitigate the consequences of possible environmental releases of fission products following a core-damaging accident by examining the probable behavior of those fission products that would likely be of most concern following an accident, and by examining some of the more likely escape routes for these nuclides. (T1)

ENABLING OBJECTIVES

1. List the four principal fission products that would likely be of most concern in regards to environmental releases following a core damaging accident. (R1)

2. List the two basic categories of events that can lead to core damage and describe the four stages of the loss of core cooling event category. (R2)

3. Recognize that the major off-site dose consequences resulting from core-damaging accidents would be from short-lived, gaseous nuclides released to the atmosphere. (R3)

4. Explain the expected behavior of the noble gas nuclides, such as Kr-85, Kr-88, Xe-131, and Xe-133, if they were released from the fuel following a core-damaging accident. (R4)

5. Describe how gaseous, elemental iodine concentrations in the RB can be reduced following a core-damaging accident; explain the relationship that the pH of the RBS water has on this process, and how RBS water pH may be controlled. (R5)

6. Explain how the reaction between elemental iodine and cesium can prove to be beneficial following a core-damaging accident. (R6)

7. List five of the more probable release paths for gaseous activity from the RB into the Auxiliary Building which can generally be readily isolated by the operator. (R7)

8. List five of the possible release paths for gaseous activity from the RB directly to the environment. (R8)

9. Describe the leak pathway to the environment for noble gases that was the most likely contributor to the offsite doses recorded for the TMI-2 accident, and explain what made this the most likely source. (R9)

10. Explain why the fission product activity in the fuel gap region consists of, for the most part, the longer-lived nuclides such as Kr-85 and Cs-137, and why this can be significant for reactor accidents. (R10)

a) This results from the zinc-boric acid reaction, the zinc being found in galvanized grating in the RB and the boric acid contained in the RB spray water.

b) The EOP directs the operator to have the Chemistry group add caustic to the RBES to reduce the spray water acidity, thereby minimizing this reaction. Chemistry procedures require that the caustic addition be commenced within 30 minutes of taking suction on the RBES and complete within 24 hours. Caustic addition has two functions:

1) Minimize hydrogen production from the boric acid reaction with zinc and aluminum in RB materials

2) Maintain Iodine in solution to minimize dose from iodine in the RB atmosphere (possible releases)

C. Recirculation to the BWST using PT/0204/007

1. Purposes

a) Each RBS pump is run, one at a time, on a periodic basis for Performance testing, to:

1) Demonstrate operability of the RBS Pumps.

2) Identify problem areas as early as possible.

3) Cycle:

(a) BS-11 ('A' RBS Pump Discharge Check)

(b) BS-16 ('B' RBS Pump Discharge Check)

(c) LP-29 (BWST 'A' Header Check)

(d) LP-30 (BWST 'B' Header Check)

b) The operating procedure also has an enclosure to align each RBS train for recirculation to the BWST.

1) Performance does not use this for their testing.

2) This could be used, if the BW recirc. pump is unavailable, to recirc. the BWST for sampling.

2. Test Performance

a) RBS Pumps are operated in recirc from the BWST. BS-21 is throttled as required to obtain the desired flow rate. Various parameters are monitored to determine operability.

b) The 'C' LPIP is operated with each RBS Pump to verify LP-29 and LP-30 fully open.

c) If ES actuation occurs during the performance of the test, an enclosure is provided to realign the system for ES.

3. Acceptance Criteria

3. Protection of the environment from iodine releases following an accident, like that for the noble gases, is dependent upon the ability to contain the nuclides within the RB until the concentrations are reduced to acceptable levels. Unlike the noble gases, though, the iodines are very volatile, meaning, that in addition to decay, steps can be taken to reduce the concentrations in the RB through additional means, too.
4. Gaseous, elemental iodine quickly reacts with water in the RB environment following an accident, greatly reducing the inventory of gaseous iodine present. This reaction occurs even more readily if the water has a high pH (i.e. it is basic); sodium hydroxide would be added to the RBS water following an accident, raising the pH and helping the RBS to "scrub" gaseous iodine from the RB atmosphere. (This higher pH also would help to reduce the acidic corrosion of the metal in the RB from the boric acid spray).
5. Hypoiodous acid is formed when elemental iodine combines with water; a small amount of hypoiodous acid may become airborne after it is formed, and the problem with this is that the RBS entering the RB will no longer be effective in scrubbing this iodine that "got away." The small amount formed, though, will be effectively handled by the charcoal filters in the PRV System.
6. Organic iodide is believed to be formed when elemental iodine combines with organic compounds in the containment building, such as methane or ethylene, to form methyl or ethyl iodide.
   a) Organic iodides are difficult to remove from the containment environment because they are not extremely reactive; while RBS is relatively ineffective, charcoal filtering can be acceptable if the charcoal is fresh and dry. If the charcoal is old, or if a high humidity exists, the effectiveness is greatly reduced.
   b) Fortunately, only about 2% of the total iodine released from the fuel will manifest itself as organic iodide.
7. Elemental iodine produced by the fission process can quickly react with another fission product, cesium, to form cesium iodide (CsI).
   a) At high temperatures (2200N F) CsI forms in the vapor state, but then quickly condenses as cesium oxide particles.
   b) CsI is extremely soluble in water, so that following a LOCA, almost all of the CsI formed will be retained in solution.
   c) This high solubility should prove to be very beneficial in limiting the amount of gaseous iodine released to the environment following an accident.

Exam Question Report

27-Jan-99

Question ID:	EAP212	Revision No:	0	Revision Date	10/29/1999
Question Description:	EAP212				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-E31 - Cooldown Following a Large LOCA
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 6; SRO = 6	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Which ONE of the following describes the benefit of adding caustic to the LPI system after a LOCA? (.25)

A) increases the acidity of the water in the emergency sump.

B) decreases the formation of hydrogen pockets in the Containment.

C) decreases the amount of radioactive iodines released due to piping breaks.

D) increases the amount of radioactive iodines maintained in solution.

Answer

D

Lessons

ID	Description
EAP-E31	Cooldown Following Large LOCA (EAP-E31)

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

QUESTION # 98

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	____
Group #	**3**	____
K/A #	**028000 A4.01**	____
Importance Rating	**4.0**	____

Technical Reference(s): **PNS-HDC**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-HDC #3 & #12**

Question Source:
Bank # **PNS-637**
Modified Bank # ____
New ____

Question History:
Previous NRC Exam ____
Previous Quiz / Test __X__

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ____

Comments:

1 POINT

QUESTION # 98

RO ONLY

Unit 3 plant conditions:

- ES actuation has occurred
- The Reactor Building Hydrogen Analyzer has been placed in service
- Heat tracing has been lost on Reactor Building Hydrogen Analyzer Channel 3A

Which ONE of the following is correct?

Channel 3A…

must

A. ~~will~~ be manually tripped immediately due to loss of heat tracing.

B. will automatically trip ~~immediately~~ due to loss of heat tracing.

C. hydrogen indication will read higher than Channel 3B.

D. hydrogen indication will read lower than Channel 3B.

1 POINT

QUESTION # 98

028000 A4.01 RO ONLY

A. Incorrect - see B. A manual trip is not required.

B. Incorrect - the moisture separator level switch automatically trips the channel.

C. Correct - condensation from the sample line will cause Channel A to indicate conservative (higher) than channel B, which is indicating RB H2.

D. Incorrect - see C. Hydrogen concentration will indicate higher.

TRAINING OBJECTIVES

Terminal Objectives

1. Discuss the systems, which are installed at Oconee to detect and control the H^2 concentration in containment. Given a procedure, place the RBHA and the CHRS is service. (T1)

2. Discuss the production of Hydrogen in the containment building following a LOCA, and those systems, which are installed at Oconee to detect and control the H^2 concentration in containment. Operate the RBHA and CHRS as required following a LOCA or MHA.(T2)

Enabling Objectives

1. Concerning H^2 production in the containment following a LOCA, list: (R1, R2)
   1.1 Major sources
   1.2 Order of magnitude of production
   1.3 Factors which affect the production rate
2. Discuss the basis for and the value of the maximum allowable H^2 concentration inside the RB. (R3)
3. Concerning the RBHA system, discuss the following: (R5, R6 & R7)
   3.1 Why the hydrogen sample line to the Local Hydrogen Analyzer Panel is heat traced.
   3.2 What will occur if excessive moisture develops in the Hydrogen Analyzer moisture separator during operation.
   3.3 Where the sample flow through the Hydrogen Analyzer discharges to.
4. List the control room indications fed from the Hydrogen Analyzer Panel. (R8)
5. Given a drawing of the Remote Hydrogen Analyzer Panel, state the function performed by each of the following: (R9)
   5.1 "Remote Selector" push-button
   5.2 "Bypass To Post Accident Sample Panel" ("ON" and "OFF" positions)
   5.3 H^2 Analyzer Isolation Valves ("OPEN" and "SHUT" positions)

2. The CHRS should be placed into operation whenever the containment hydrogen concentration exceeds 0.5%, but no sooner than seven days after initiation of the event. The CHRS should be shut off and isolated when the indications on the PAM indicators from the two hydrogen analyzer trains reach 4.5% (average) if both trains of hydrogen analyzers are operable. If only one hydrogen analyzer train is operable, the recombiner must be stopped when the indication reaches 4.3%.
   - By placing the system in service at 0.5% minimum concentration or after seven days from the initiation of the LOCA, RB post-accident hydrogen concentration should be maintained below 4% (lower flammability limit).
   - The minimum concentration for rated efficiency (95%) of the recombiner is 0.5%
   - The recombiner maximum design concentration is 5.0%. The limit of 4.5% and 4.3% provides for instrument and sample error.
3. The CHRS should not be operated if the RB pressure > 23.0 psig or temperature > 180°F.
   - Recombiner to Containment hose system drains pressure / temperature limits would be exceeded.
4. The CHRS is designed to process 90 scfm of gas containing up to 5% hydrogen with the balance consisting of varying amounts of oxygen, nitrogen or water vapor.
5. Permanent shielding for radiation protection of equipment and personnel is considered unnecessary for the CHRS. The recombiner and control panel foundations are located to make maximum use of the BWST and existing walls as shielding. However, temporary shielding may be required to minimize radiation exposure to the operator or off site dose due to shine from the recombiner.
6. Use caution to avoid burns. Portions of the system develop high temperatures.
   - Gas temperatures in the reaction chamber exceed 1300°F.
7. Use caution around live electrical circuits.
   a) requires a 480 volt power supply / a portable step-down transformer is installed between the 600 volt load center and the recombiner and its control panel
   b) Power supplies:
      1) Units 1, 2 and 3 from XOD2A, XOD2B and XOD2C respectively. XOD2A,B,&C are powered from B3T and B4T.
      2) Units 1, 2, or 3 may also be supplied from 2XO or 3XO (cable must be field routed from the equipment room)

6. Concerning the RBHA system, recognize the following: (R10, R11)
   - 6.1 The RB Isolation valves can only be operated from the respective unit's control room.
   - 6.2 The Remote Hydrogen Analyzer Panel, rather than the Local Panel, is the one that will be used by the operator for containment hydrogen sampling following a LOCA.
7. Given conditions that would result in a valid ES actuation, state the: (R12, R13)
   - 7.1 Time requirement for placing the RBHA in service
   - 7.2 Required operator action if the Gaseous Post Accident Sample Panel is in operation when the ES actuation occurs.
8. Given a Limit and Precaution contained in the RBHA System operating procedure, discuss the basis. (R14)
9. For PT/0150/22H, H^2 Analyzer Valve Stroke Test, describe: (R20)
   - 9.1 The purpose
   - 9.2 How the test is performed
10. Given a copy of PT/0150/22H, H^2 Analyzer Valve Stroke Test, and a set of data, evaluate if the acceptance criteria is met. (R21)
11. State the purpose of PR-59 and PR-60. (R16)
12. Given a Limit and Precaution contained in the CHR System operating procedure, discuss the basis. (R4, R17)
13. Given a set of plant conditions, determine the proper operation / alignment of the RBHA and CHR systems and the basis for that specific operation / alignment. (R22)
14. Given a set of plant conditions, analyze RBHA and CHR system operation and determine system status and any required actions / corrective actions. (R19)
15. Given a copy of ITS / SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS / SLC LCO's. (R23)
16. Apply all ITS / SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R24)
17. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITS / SLC's. (R25)

5. Radio communication should be established between the Control Room and the applicable unit's Remote Panel when placing the RBHA system in service.

6. RB Hydrogen Analyzer system heat tracing is not required for system operability. Heat tracing improves the accuracy of the hydrogen sample by removing moisture from the system.

   a) The function of heat tracing is to maintain sample gases near the containment conditions. Excess condensation reaching the RB Hydrogen Analyzer cell assembly will result in conservative readings (H^2 Indication > Actual).

I. PT/0150/22H, H^2 Analyzer Valve Stroke Test

1. The purpose is to determine operability of H^2 Analyzer Sample Valves (PR-71 through 80).

2. Test Performance

   a) Valves are isolated and the area between the isolation and the valve being tested is pressurized to 30 psig with IA.

   b) Caution must be exercised to ensure the pressure does not exceed 35 psig as this could cause damage to the diaphragm in the H^2 Analyzer.

   c) The valve is opened and the pressure bleeds off to the RB.

   d) The valve is closed and pressure is verified to return to 30 psig.

3. Acceptance Criteria

   a) All valves cycle

   b) This valve stroke test is different in that there is no valve position indication. The method for verifying acceptance criteria is a pressure decrease / increase as the valve is cycled.

2.4 Post Accident Hydrogen Control (OC-PNS-HDC-5)

A. Hydrogen Production vs. removal

1. If no method of hydrogen removal mechanism is used during a DBA LOCA, then the maximum hydrogen concentration which can be expected in Containment over a 30 day post-accident time period is approximately 5.5 % (**OC-PNS-HDC-9**). **OC-PNS-HDC-8**) shows the hydrogen generation rates for each of the sources of hydrogen production at various times post-LOCA.

Exam Question Report

27-Jan-99

Question ID:	PNS637	Revision No:	0	Revision Date	10/29/1999
Question Description:	PNS637				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	PNS-HDC - Hydrogen Detection and Control
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	Reference: PNS-HDC	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

The following conditions exist on Unit #3:

• ES actuation has occurred.

• The Reactor Building Hydrogen Analyzer has been placed in service.

• Heat tracing has been lost on Reactor Building Hydrogen Analyzer Channel 3A.

Which ONE of the following is correct? (.25)

A) Channel 3A will automatically trip immediately due to loss of heat tracing.

B) Channel 3A will be manually tripped immediately due to loss of heat tracing.

C) Channel 3A hydrogen indication will read higher than Channel 3B.

D) Channel 3B hydrogen indication will read higher than Channel 3A.

Answer

C
A. Incorrect - the moisture separator level switch automatically trips the channel.

B. Incorrect - see A. above.

C. Correct - condensation from the sample line will cause channel A to indicate conservative (higher) than channel B which is indicating RB H2.

D. Incorrect - see C. above.

Lessons

ID	Description

1 POINT

QUESTION # 99

RO ONLY

Unit 3 plant conditions:

- Core refueling in progress
- The first assembly is being loaded into the core
- The first assembly being loaded has operated for 1 fuel cycle

Comparing the difference in NI indication for the first fuel assembly loaded, which ONE of the following is correct?

The (first) Fuel Assembly that is loaded in QUADRANT Y-Z near the ______ of the core and has a ______ rod installed will provide the **HIGHEST** 3NI-1 neutron indication.

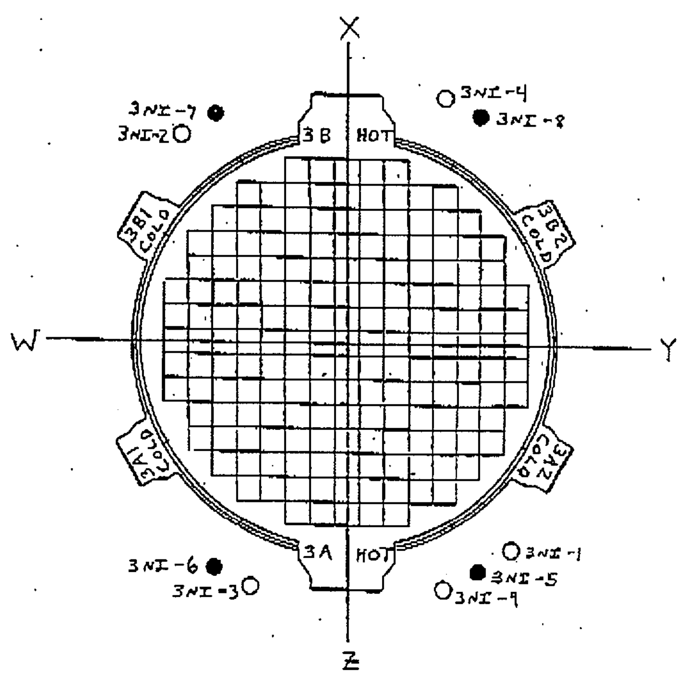
SEE ATTACHMENT

A. outer edge / control

B. outer edge / axial power shaping

C. center / control

D. center / axial power shaping



1 POINT

QUESTION # 96

008000 A2.07 (2.5/2.8) SRO ONLY

A. Incorrect – This is an option and the SRO would dispatch someone to open CC-8 immediately but this action requires ≈ 8 minutes to complete. A manual reactor trip is required within ≈ 4 minutes due to high CRD temperatures.

B. Incorrect – During a loss of CC the reactor is required to be tripped when CRD temperatures reach 180°F. This temperature will be reached within 4 minutes.

C. Incorrect – This is a correct action during a loss of CC event but it would not be required immediately. A unit shutdown is prompted by a high PZR level of >285 inches.

D. Correct – Loss of seal injection and CC requires the SRO to direct the crew immediately trip all RCPs and activate the SSF RCMUP to supply RCP seal injection to prevent seal damage and possible major seal leakage.

QUESTION # 99

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	____
Group #	**3**	____
K/A #	**034000 A4.02**	
Importance Rating	**3.5**	____

Technical Reference(s): **PNS-CED**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-CED # 1, 2**

Question Source:
Bank # ______
Modified Bank # ______
New ___X___

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 99

RO ONLY

Unit 3 plant conditions:

- Core refueling in progress
- The first assembly is being loaded into the core

Which ONE of the following results in the **HIGHEST** indicated Source Range neutron level?

SEE ATTACHMENT

The Fuel Assembly is loaded near the ______ of the core and has a ______ rod installed.

ASSUME the Fuel Assembly has been in the core for one cycle.

A. outer edge / control

B. outer edge / axial power shaping

C. center / control

D. center / axial power shaping

Attachment not necessary

1 POINT

QUESTION # 99

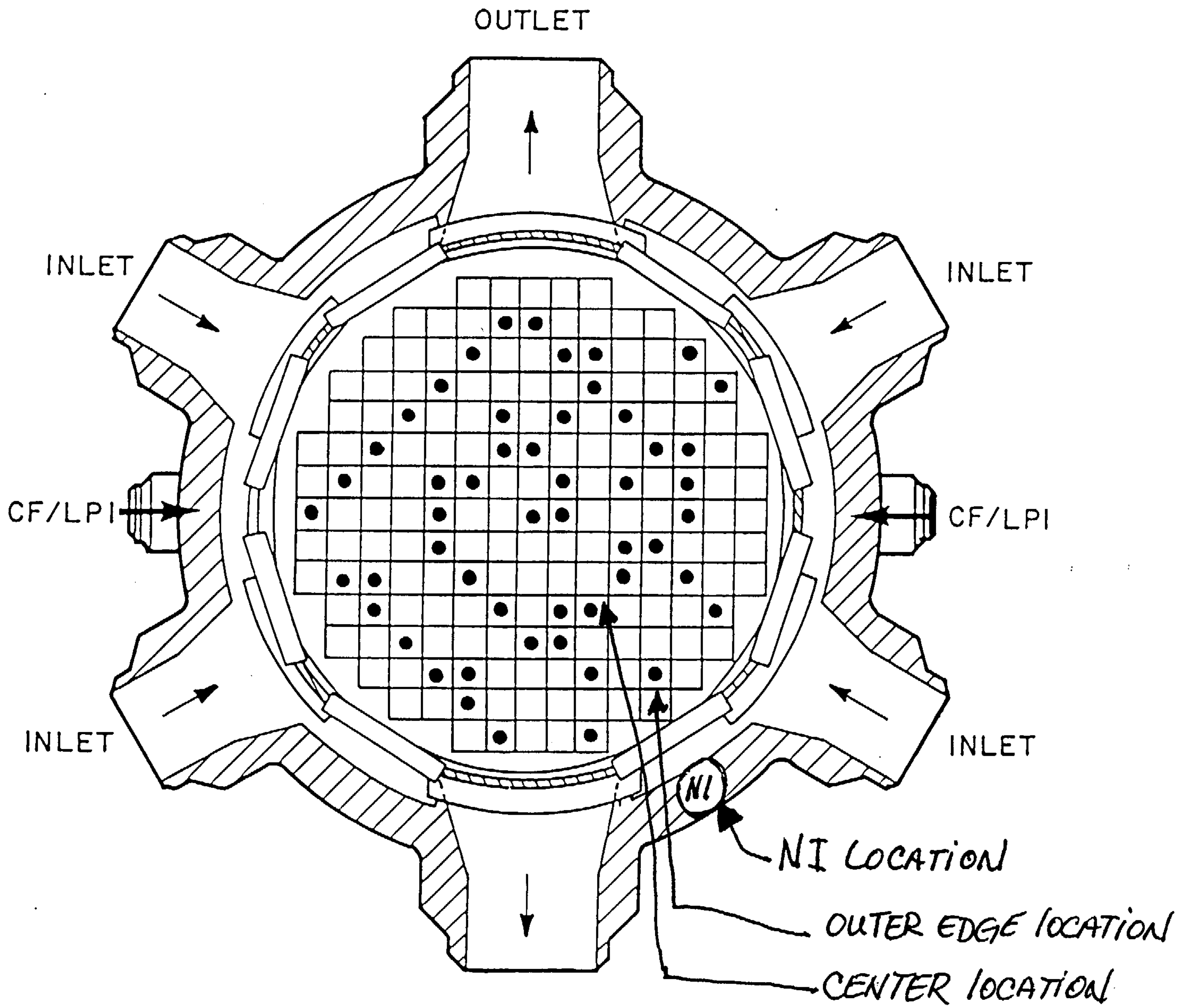
034000 A4.02 RO ONLY PRA 5-2-00

A. Incorrect – A control rod will absorb more neutrons than an APSR and provide a lower NI neutron indication. The second part is correct, the FA closest to the edge of the core will indicate the highest neutron level when compared to one loaded in the center.

B. Correct – The Fuel assembly loaded at the outer edge will cause the NI to have a greater response, as the FA is physically closer to the detector. The NI will detect more neutron leakage. APSR are made of a lower neutron absorbing material than a control rod.

C. Incorrect – A FA loaded in the center vs. the outer edge will indicate a lower NI response. The distance from the detector is greater if the FA is loaded in the center. A control rod will absorb more neutrons than the APSR.

D. Incorrect - A FA loaded in the center vs. the outer edge will indicate a lower NI response. The distance from the detector is greater if the FA is loaded in the center. An APSR will not absorb neutrons as great as a Control rod.


OUTLET
INLET
INLET
CF/LPI
CF/LPI
INLET
INLET
NI
NI LOCATION
OUTER EDGE LOCATION
CENTER LOCATION

TRAINING OBJECTIVES

Terminal Objectives

Describe the design, construction and use of the various types of control elements used at ONS, with emphasis on the differences between them.

Enabling Objectives

1.0 Concerning standard control rods: (R1)

1.1 State the purpose of control rods.

1.2 State the absorber material used and its active length.

1.3 Explain the difference between the standard control rod coupling and the APSR coupling dimension.

2.0 Concerning Axial Power Shaping Rods (APSRs): (R2)

2.1 State the purpose of APSRs.

2.2 State the absorber material used and its active length.

3.0 Concerning a burnable poison rod (BPRA): (R3)

3.1 State the purpose for using BPRAs.

3.2 State the absorber material used and its active length.

2. PRESENTATION

2.1 Control Rod Assembly

REFER TO OC-PNS-CED-1

A. Design and Construction

1. Control Rods are used to change reactivity quickly and/or compensate for parameters that change reactivity.
2. Each control rod assembly has :
   a) 16 rods
   b) A stainless steel spider
   c) A female coupling.

REFER TO OC PNS CED 2&2a

3. The neutron absorber material in the standard control rod is an alloy of silver - indium - cadmium. It is clad in cold-worked type 304 stainless steel tubing.
   a) Tubing forms a water-tight and pressure-tight container for the absorber material. The tube provides the structural strength of the control rods and prevents corrosion of the absorber material.
   b) A tube spacer is used to prevent absorber motion within the cladding during shipping and handling and to permit differential expansion in service.
4. There are also "Plant-life" CRAs which are the same as standard except Inconel clad, which provides added resistance to creep and corrosion.
5. Active absorber length is 134 inches in standard, 139 inches in plant-life design. Provides fine control of power level for small changes.
6. Control rods are designed to withstand all operating loads including those resulting from hydraulic forces, thermal gradients, and reactor trip deceleration.
   a) Because the Ag-In-Cd alloy poison does not yield a gaseous product under irradiation, internal pressure and swelling of the absorber material does not cause excessive stressing or stretching of the clad.

7. Because of their length and the possible lack of straightness over the entire length of the rod, some interference between the rods and the fuel assembly guide tubes is expected. However, the rods are flexible and only small friction drag loads result. Similarly, thermal distortions of the control rods are small because of the low heat generation and adequate cooling. The control rod assemblies do not encounter significant frictional resistance to their motion in the guide tubes.
   a) The same is true for the Axial Power Shaping Rods (APSRs) also.
8. Full length guidance of the rod assembly is provided by the control rod guide tube of the upper plenum assembly, and the fuel assembly guide tubes. At the fully withdrawn position the lower end of the rod remains in the fuel assembly guide tube.

B. Control Rod Spider

1. The control rod spider is a stainless steel assembly, which couples the sixteen (16) control rods to the control rod drive. The control rod spider, shown in Figure PNS-CED-1, retains the control rods by means of a nut threaded to the upper shank of each rod. After assembly, the nuts are lock welded.

REFER TO OC-PNS-CED-3

2. The spider allows for connection to the control rod drives by means of a female coupling. To prevent inadvertently connecting a control rod to an axial power shaping rod the female coupling and bayonet of control rods are slightly dimensionally different from the female coupling and bayonet of axial power shaping rods.

C. Control Rod interface with Fuel Assembly (FA)

REFER TO OC-PNS-CED 8 thru 12

1. The control rod elements move into the FA thru the guide tubes in the assembly.
   a) When fully inserted, the control rod spider assembly will rest on the top of the FA upper end fitting.
   b) Proper alignment is assured by the "legs" of the spider fitting into the slots on the FA upper end fitting ears.
2. All types of rods interface with the FA in this manner, however the BPSRs and Source rods are fixed in place in the FA.
   a) The rod spiders have feet (tabs) that extend out far enough that the upper plenum grid when lowered onto the core will rest on top of the spider feet to hold the rod in the assembly.
3. For more information refer to OP-OC-RT-GP05 & GP07.

2.2 Axial Power Shaping Rod Element

REFER TO OC-PNS-CED-4

A. Design and Construction

1. Purpose: Partial length rods provide axial flux shaping.
2. Each axial power shaping rod assembly has:
   a) 16 rods
   b) A stainless steel spider
   c) A female coupling

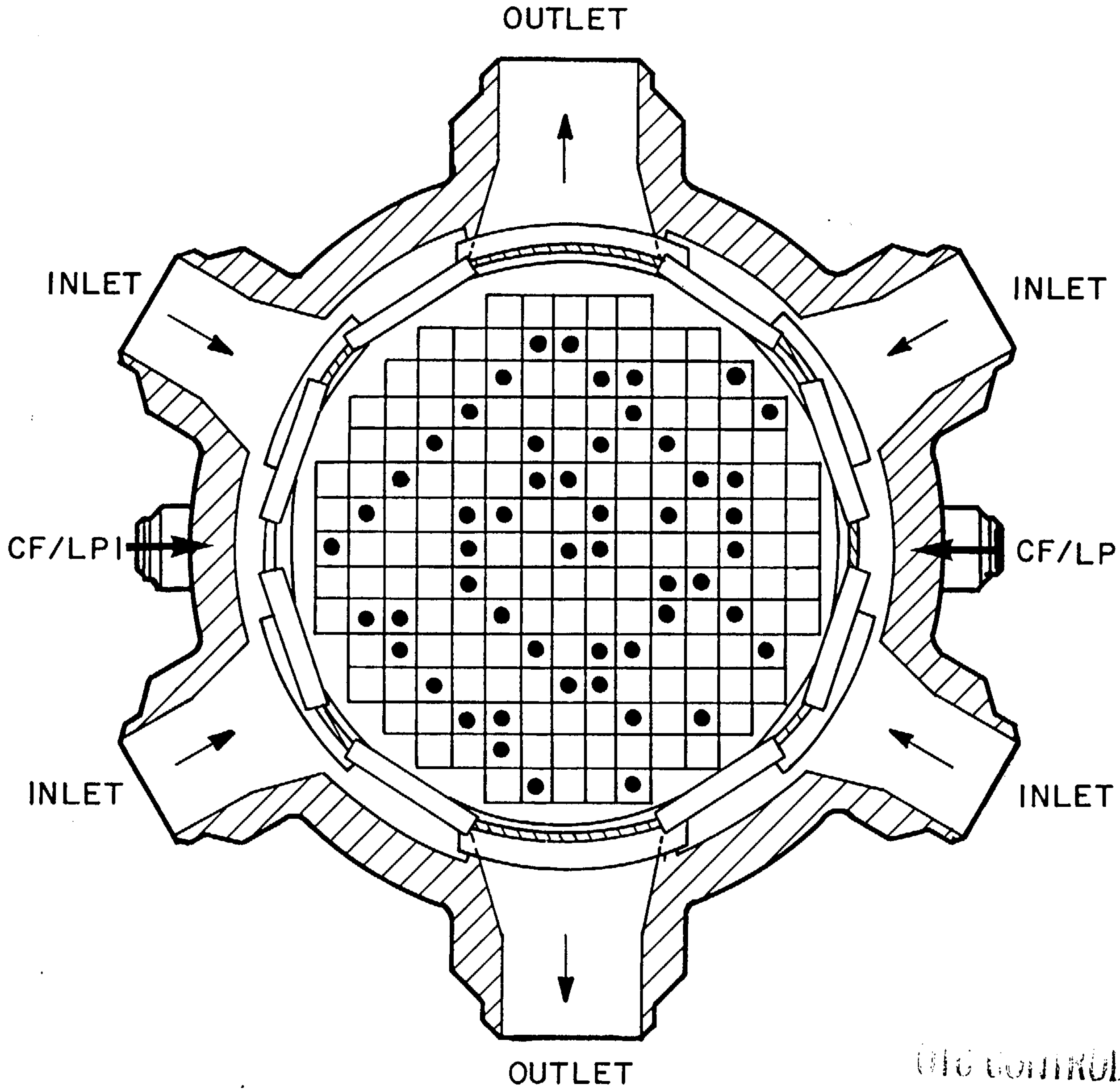
REFER TO OC-PNS-CED-5

3. Each axial power shaping rod has a 63 inch section of neutron absorber material.
   a) This absorber material is Inconel 600 and is clad in cold-worked, Type 304 stainless steel tubing.
      1) The tubing provides the structural strength of the axial power shaping rods and prevents corrosion of the absorber material.
   b) The area above the absorber material is hollow with holes in the cladding to equalize pressure across the hollow area.
   c) Inconel is used instead of a stronger absorber to reduce the power peaking effect of fuel shadowed and allows more even fuel burnup.
4. As with the control rods, the axial power shaping rods are designed to withstand all operating loads including those from hydraulic forces and thermal gradients.
   a) Because the Inconel does not yield gaseous products under irradiation, internal pressure is not generated within the clad.
   b) Swelling of the absorber material is negligible, and does not cause unacceptable clad strain.
5. Full length guidance of the axial power shaping rod assembly is provided by the control rod guide tube of the upper plenum assembly and the fuel assembly guide tube. At the fully withdrawn position the lower end of the APSRA remains within the fuel assembly guide tube to maintain the continuity of guidance throughout the rod travel.

B. APSR Spider

1. The APSR Spider throat is very similar to the Control Rod Spider Throat. The only difference being in the dimensions of the female coupling as discussed earlier.

C. For more information refer to OP-OC-RT-GP05 & GP07.



TITLE: REACTOR VESSEL DESIGN AND CONSTRUCTION	NOTES: 1) HOT LEG BREAK FLOW PATH 2) OUTLET FLOW IS OUT OF THE BROKEN LEG	ID.NO. OC-PNS-RVD-18 DATE: 1/21/87
		REF. FSAR, VOL. 4, SECT. 4, 1985
		DRN. BY: GDF / RWP APR. BY: RPO
		TRAINING USE ONLY

QUESTION # 100

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	_____
Group #	**3**	_____
K/A #	**103000 A2.03**	_____
Importance Rating	**3.5**	_____

Technical Reference(s): **PNS-PRV**

Proposed references to be provided to applicants during examination: NONE

Learning Objective: **PNS-PRV**

Question Source:
Bank # _______
Modified Bank # _______
New ___X___

Question History:
Previous NRC Exam _______
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 100

RO ONLY

Unit 1 plant conditions:

INITIAL CONDITIONS

- ES Digital Channel 5 in test

CURRENT CONDITIONS

- A LOCA is in progress
- ES has actuated

Which ONE of the following is correct concerning the PRV system?

The ES signal provides a ~~direct~~ CUPS? actuation of the …

A. "A" PRV Fan only.

B. "A" PRV Fan Discharge Valve only.

C. "A" and "B" PRV Fans.

D. "A" and "B" PRV Discharge Valves.

previous ES Digital channel in "TEST" Question

1 POINT

QUESTION # 100

103000 A2.03 (3.5/3.8) RO ONLY PRA – 4-6-00 (GTH)

A. Incorrect – Misconception that the ES Digital channel will not actuate while in test.

B. Incorrect – Operator misconception is the discharge opens first to establish a flow path (since the fans have no recirc flow path), then the fan starts when the valve is fully open. Misconception that the ES Digital channel will not actuate while in test.

C. Correct – The ES signal feeds the fan start logic then the fan start subsequently provides a signal to open the discharge valve. A channel in test will actuate.

D. Incorrect – Operator misconception is the discharge opens to establish a flow path since the fans have no recirc flow path, then the fan starts when the valve is fully open.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

T1 Describe the operation and purpose of the Penetration Room Ventilation System.

ENABLING OBJECTIVES:

1. Draw a basic flow diagram of the Penetration Room Ventilation System. (R1)
2. State the purpose of the Penetration Room Ventilation System. (R2)
3. List the types of filters used in the Penetration Room Ventilation System. (R3)
4. Explain the advantages of the external carbon sampling canisters. (R4)
5. State the purpose of the charcoal filter. (R5)
6. Explain the reason for the maximum flow limit through the Penetration Room Ventilation System. (R6)
7. State the purpose of PR-13 and PR-17, A & B filter outlet valves. (R7)
8. Explain the two purposes of PR-20, PRV fan suction cross-connect valve. (R8)
9. Describe the operation of the Penetration Room Ventilation System following Engineered Safeguards actuation. (R9)
10. State the reason for the high humidity limit in the penetration rooms and any associated actions. (R10)
11. Briefly describe the requirements for opening a Penetration Room floor drain while containment integrity is required. (R11)
12. Briefly describe the requirements for draining a system to the Penetration Room floor drains. (R12)
13. Describe the purpose of the travel stops installed on PR-13 and PR-17, including when they should be adjusted. (R13)
14. Describe two circumstances when it may be necessary to periodically adjust Penetration Room Ventilation flow. (R14)
15. Explain the various means available to the operator in the control room and locally to identify degraded Penetration Room Ventilation flow. (R15)

5. Refueling Transfer Tubes:

   Each refueling tube is equipped with a blind flange which is only opened during a refueling shutdown for transfer of fuel to the spent fuel pool.

2.3 Emergency Operations

Instructor Note:

CP-601, Cooldown Following Large LOCA, of EP/1,2,3/A/1800/01, Emergency Operating Procedure, gives guidance on operation after E.S. initiation.

A. Engineered Safeguards Operation

1. During normal operation, the PRV System is on standby with each fan aligned with a filter assembly.
2. Engineered Safeguards Channels 5 and 6 (High RB Pressure of 4 psig) will initiate the following sequence of events:
   a) PRV Fans A and B receive a signal to start.
   b) Valves PR-15 and PR-19 will open by a signal from the fan starts. The fan start is verified on the RZ modules and the "valve open position" should be verified in the Control Room.
3. Following the Engineered Safeguards Actuation the operator should:
   a) Verify PRVS is in operation and send an operator to adjust (1)(2)(3)PR-13 (Filter A Outlet) and (1)(2)(3)PR-17 (Filter B Outlet) as necessary to maintain 1000cfm flow through each filter train.
   b) If either Penetration Room fan fails to start on ES initiation, open (1)(2)(3) PR-20 (PR Fans Suction Tie) at its locally mounted switch to allow the operating fan to purge through both filters.

   NOTE: Do not exceed 1100 cfm flow through each filter train. Excessive flow decreases the efficiency of the filter packs.
4. Circuitry has been modified such that the fans do not stop upon reset of E.S. channel. To shut down the fans requires deliberate separate action. No new switches have been added. Fans will still be controlled at the RZ modules. However, after reset of E.S. Channel in order to turn off fans, the OFF buttons must be pushed.

B. Loss of Air Flow Through a Filter

1. Redundant fans, cross connected piping, and locked open filter inlet valves minimizes the possibility of a loss of cooling air flow to the filters. Analysis completed per PIR 4-090-0057 indicates that natural circulation around the filter will provide adequate cooling to prevent carbon ignition.

INITIAL SUBMITTAL

OCONEE EXAM 2000-301
50-269, 270, AND 287/2000-301

JULY 10 - 14, 18, 19, AND 20, 2000

INITIAL SUBMITTAL

COMMON - VOLUME 1
WRITTEN EXAM

COMMON

Volume 1

NRC Copy

QUESTION # 1

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**1**	**1**
	K/A #	**000005 K3.06**	
	Importance Rating	**3.9**	**4.2**

Technical Reference(s): **OP/1105/09, CRD**
IC-CRI

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-CRD #17.2**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

ok

QUESTION # 1

Unit 1 plant conditions:

INITIAL CONDITIONS:

- A reactor trip occurred 6 hours ago
- RCS temperature = 549°F and steady
- RCS pressure = 2155 psig and steady
- While withdrawing Group 1 control rods to 50%:
    - Group 1 Rods 2 through 9 withdrawn to 10%
    - Group 1 Rod 1 remained at 0%

CURRENT CONDITIONS:

- Group 1 has been inserted to 0%
- Group 1 relatch in progress

Which ONE of the following is correct concerning the process for latching Group1?

Latch Safety Group 1 in the _______ speed / ...

A. run / to preclude damage to the spider.

B. jog / which will energize the sync circuitry.

C. run / because the rod is not misaligned or stuck.

D. jog / to ensure the clamping contacts are energized.

1 POINT

QUESTION # 1

000005 K3.06 (PRA) 1-29-00
(1)

A. Incorrect – applies to the recovery speed of a stuck or jammed rod after the rod is latched in jog.

B. Correct – prerequisite to latching CRDs per OP/1105/09. Selecting jog activates the sync circuitry in order to select the Aux. power supplies to the CRD's. This is required for Safety Group movement

C. Incorrect – run is used to move CRDs that are jammed/stuck but jog is used for latching.

D. Incorrect – jog activates the sync circuitry. When latching clamping contacts are operated by the clamp/release function.

2.7 Operating limits have been established to assure that control rod drop is prohibited under conditions which would defeat the hydraulic snubbing action of the control rod drive mechanism. Two of the major concerns are gases in the CRDM "Torque-Taker Tube" and fluid vaporization above the "Torque-Taker" when a rod is dropped. The limits established for control rod operation are listed below:

2.7.1 Control rod operation is allowed when RCS pressure is above and to the left of the curve shown on Enclosure "Dissolved Gas Concentration Curve."

2.7.2 If control rods must be operated during an RCS cooldown when pressure is < 350 psig, maintain RCS temperature constant for 1/2 hour prior to going below 350 psig to allow CRDM temperatures to stabilize.

2.8 If the pressurizer level decreases to less than 184 inches with the RCS depressurized, all the CRDs must be vented.

2.9 Prior to a dropped or asymmetric rod recovery, contact the Duty Reactor Engineer to evaluate the effects of local power distribution and the possible necessity for special maneuvering limits.

2.10 If partially withdrawn or fully withdrawn control rod is stuck or jammed, do not operate the control rod in JOG speed. Operate in RUN speed only. The possibility of overloading the spider exists if the CRD is operated in JOG speed when the CRD is not free running. If a fully inserted control rod is stuck or jammed, the control rod may be operated in JOG speed only for the purpose of latching the CRDM to the lead screw.

2.11 Pulling any individual control rod with Group 1 withdrawn to 50% is an unanalyzed condition. With the reactor shut down, ensure all rods are inserted prior to withdrawing an individual control rod.

3. Procedure

3.1 Refer to Enclosures referenced in Section 4 for the specific items to be performed.

1. Initial Conditions

_____ 1.1 Control Rod Drive System energized per Enclosure "Control Rod Drive System Startup."

NOTE:	1. For initial withdrawal of Group 1 to 50%, Latch and PI alignment need <u>only</u> be performed on Group 1. 2. Latching of CRDs is necessary anytime the CRD Bkrs have been tripped and subsequently re-closed.

_____ 1.2 IF RCS temperature is < 250°F, Latch and PI alignment of Safety Group 1 has been performed per Enclosure "Latch and PI Alignment".

_____ 1.3 IF RCS pressure is > 1800 psig, Latch and PI alignment of ALL Safety and Regulating groups has been performed per Enclosure "Latch and PI Alignment".

_____ 1.4 Limits and Precautions have been reviewed.

2. Procedure

_____ 2.1 Transfer Group 1 to the Auxiliary Power Supply per Enclosure "Transfer Rods Between Normal and Auxiliary Power Supply".

_____ 2.2 Withdraw Group 1 to 50% to provide additional available shutdown margin.

Information Use

2.2 Latch and PI alignment of Regulating group as follows:

2.2.1 Press selector for GROUP.

2.2.2 Select CRD group on the Group Select Switch.

2.2.3 Press selector for SEQ OVERRIDE.

2.2.4 Select JOG on the Speed Selector.

2.2.5 Press selector for the LATCH switch and insert for approximately 15 seconds. Release the LATCH switch.

2.2.6 Compare absolute and relative readings on the PI panel.

2.2.7 If it is required to reset the relative PI, rotate the single select switch to the desired CRD and use the PI reset raise/lower switch.

13.13 Group Select Switch (Off, 1-8)

14. Explain the purpose for the Clamping Contactors associated with the CRD power supplies. (R14)

15. Explain the CRD Patch Panel including the associated S.L.C. requirement. (R15)

16. Given a Limit and Precaution from OP/O/A/1105/09, Control Rod Drive System, explain the basis of the limit or precaution. (R16)

17. Given the procedure, describe the bases of the steps involved in the following CRD system evolutions: (R17)

    17.1 Transferring between D.C. Hold, Auxiliary and Regulating power supplies for the CRDs.

    17.2 Latch and PI alignment of a safety group or any individual rod.

18. Describe the process for verifying the "A" and "CC" phases of Groups 1-4 stators operable. (R18)

19. Describe how the operator resets the Control Rod Drive Trip breakers from the Diamond Control panel. (R19)

20. Discuss the following concerning the Diverse Scram System (DSS): (R20)

    20.1 Operation and bases of DSS

    20.2 Signal inputs

    20.3 Actuation setpoints

    20.4 System reset

    20.5 Operability verification by the operator

21. Apply ITS/SLC's rules to determine applicable Conditions and Required actions for a given set of conditions. (R21)

22. Given a copy of Improved Technical Specifications, and associated Bases, analyze a given set of conditions for applicable ITS/SLC LCO's. (R22)

23. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITC/SLC's. (R23)

5. Manual Transfer/Sync and Transfer Confirm Pushbutton/Indicator
   a) The "Sync" lamp indicates that the auxiliary inverter power supply is in phase synchronism with the power supply being swapped.
      1) Placing the Diamond panel speed selector switch to "jog" will activate the sync circuit
      2) The programmer for the auxiliary power supply will sequence the motor phases to the appropriate two phases to match the parallel supply. (DC Hold or Group power)
   b) Manual Transfer pushbutton is used to transfer individual rods or groups to and from the auxiliary inverter power supply. It will not function unless "Clamp" is indicated.
   c) The "Transfer Confirm" lamp indicates that all transfer relays for the group being swapped have rotated into position transferring the selected control rod drive mechanisms to the auxiliary inverter power supply.
      1) When transferring "TO" the auxiliary source, the light will illuminate.
      2) When transferring "FROM" the auxiliary source, the light will extinguish.
   d) It is important that the operator know if one or more transfer switches have failed to reposition to the desired power source.
      1) As long as the two sources are connected and no rod movement is attempted, there is no problem
      2) However, if movement is attempted with two sources connected the rod(group) would drop when opposite phases between the two sources were energized canceling each other out.
   e) Verification from Normal to Auxiliary
      1) All transfer switches have successfully rotated to their correct position by indication of the "transfer confirm" lamp illuminated.
   f) Verification from Auxiliary to Normal
      1) All transfer switches successfully rotated when the "transfer confirm" light goes out.
      2) If the light were to burn out, an unsuccessful transfer would be indicated by all control "on" lamps remaining lit for that rod(group) on the PI panel.

d) This incident is covered in detail in the lesson OP-OC-RT-PD, Power Distribution.

10. If a partially withdrawn or fully withdrawn CR is stuck or jammed, do not operate the control rod in JOG speed. Operate in RUN only.

    a) The possibility of overloading the spider exists if the CRD is operated in JOG speed when the CRD is not free running.

    b) If a fully inserted CR is stuck or jammed, the CR may be operated in JOG speed only for the purpose of latching the CRDM to the lead screw.

11. Pulling any individual control rods with group 1 withdrawn to 50% is an unanalyzed condition. With the reactor S/D, ensure all rods are inserted prior to withdrawing an individual control rod.

B. Transferring Between Power Supplies

- (Refer to proper Enclosure of OP/0/A/1105/09)

1. Transfer of a group/rod from normal (DC Hold For Safety Rods) power supply to the auxiliary power supply.

   - **INFORMATION USE procedure**

   a) Diamond must be in manual.

   b) Select group desired on the Group Select Switch.

   c) Select ALL or desired rod on the Single Select Switch.

   d) Press selector for SEQ OVERRIDE.

   e) Press selector for AUXILIARY.

   f) Select JOG on the Speed Selector.

      - NOTE: Insure Manual Transfer Sync. Light is lit before pressing clamp.

   g) Press selector for CLAMP.

   h) Press selector for MANUAL TRANSFER switch until the TRANSFER CONFIRM lamp comes on followed by the CONTROL ON lamps on the PI panel.

      - All CONTROL ON lamps on the group/rod transferred should be on.
      - If light is NOT on, this indicates a switch has stuck with both power supplies energized.

   i) Press selector for CLAMP RELEASE pushbutton.

   j) Press selector for GROUP.

QUESTION # 2

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**000005 K 1.03**	
Importance Rating	**3.2**	**3.6**

Technical Reference(s): **RT-FPP, STG-ICS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **RT-FPP #5, STG-ICS #13**

Question Source:
Bank # ____
Modified Bank # ____
New __X__

Question History:
Previous NRC Exam ____
Previous Quiz / Test ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ____

Comments:

ok

1 POINT

QUESTION # 2

Unit 3 plant conditions:

INITIAL CONDITIONS:

- Power level 100%
- A unit shutdown is in progress
- Shutdown rate = 2%/minute
- ALL ICS stations in AUTOMATIC

CURRENT CONDITIONS:

- Power level = 85%
- Group 7 position = 85% and not moving
- Diamond CRD "insert" light ON
- Neutron error = (-) 2%
- The OATC places the Diamond to HAND

Which ONE of the following correctly predicts the RCS temperature and reactor power relationship four (4) hours later?

ASSUME no further operator action

Tave will be _______ / Reactor power will be _______

A. maintained at setpoint / lower.

B. maintained at setpoint / higher.

C. lower but remain within 1.2°F of setpoint / lower.

D. higher but remain within 1.2°F of setpoint / higher.

1 POINT

QUESTION # 2

000005 K1.03 (PRA) 1-29-00
(2)

QUESTION SETUP: As the unit is decreasing power the CRs stop inserting while all other ICS stations continue to receive a decrease demand signal. This will cause an increase in Tave as FDW decreases and rods remain constant (rods do not correct for the mismatch in actual Tave vs. Tave setpoint).
When the Diamond is placed into hand this will transfer Tave control to FDW. Tave will be higher than setpoint and FDW will initially increase to reduce Tave to setpoint.

Note: When the Diamond is placed into hand this will cause the ICS to TRACK flux.
As Xenon concentration increases reactor power will decrease (flux decrease) Tave will remain at setpoint via FDW. As reactor power decreases FDW will also decrease as they track flux and Tave will remain constant at setpoint.

A. Correct – The reactor cannot correct for Tave error when the Diamond is in Hand. Tave control will transfer to FDW and be maintained at setpoint. The affects of Xenon concentration changes (increasing) will decrease flux. As flux decreases the unit will decrease power from the affects of Xenon while FDW maintains Tave at setpoint.

B. Incorrect – If power was in an increase mode from a lower stable power level the affects of Xenon will cause a positive reactivity effect and this would be a correct answer.

C. Incorrect – This would be correct if the diamond remained in automatic. FDW would respond to Tave error of ≥1.2°F.

D. Incorrect – If power was in an increase mode from a lower stable power and the Diamond was in automatic Tave would be maintained within 1.2°F via FDW. Increasing power would burnout Xenon and cause a positive reactivity effect.

PWR	INSTRUCTOR GUIDE

VOLUME TITLE REACTOR THEORY **TIME** 4 Hrs

INSTRUCTOR GUIDE TITLE FISSION PRODUCT POISONS **REV** 1

OBJECTIVES

5. Describe the following processes and state their effect on reactor operation:
   a. equilibrium xenon
   b. xenon behavior following power changes
   c. xenon behavior following a reactor trip

6. Plot the curve and explain the reason for the reactivity insertion by xenon-135 versus time for the following:
   a. initial reactor startup and ascension to rated power
   b. reactor startup with xenon-135 already present in the core
   c. power changes from one steady state power level to another
   d. reactor trip
   e. reactor shutdown

7. Explain the process and reasons why the Reactor Operator compensates for the time dependent behavior of xenon-135 concentration in the reactor.

8. Describe the effects that xenon concentration has on neutron flux shape.

9. State the characteristics of samarium-149 as a fission product poison.

10. Describe the production of samarium-149.

11. Describe the removal of samarium-149.

12. Define equilibrium samarium.

13. Plot the curve and explain the reason for reactivity insertion by samarium-149 versus time for the following:
    a. initial reactor startup and ascension to power
    b. reactor shutdown

14. Describe the effects of power changes on samarium concentration.

15. Compare effects of samarium-149 on reactor operation with those of xenon-135.

STARTUP FROM A XENON FREE CONDITION

A XENON FREE CONDITION is any time in the life of the core when either Xe-135 has not been produced or the reactor has been shutdown long enough to allow all of the Xe-135 to decay away. Figure 6-5 shows the time required to reach equilibrium xenon concentration from a xenon free condition for three power levels.

Commencing power operations from a xenon-free condition results in the immediate production of Xe-135 directly from fission and a large amount of I-135. Additional Xe-135 is produced as a decay product of the iodine. As the Xe-135 starts building in, some of it is removed by burnout and decay. Equilibrium Xe-135 will be reached in about 40 hours when starting up to full power, in about 44 hours when at 50% power and about 48 hours at even lower power levels. The equilibrium level is reached sooner at the higher power level due to the faster production rate.

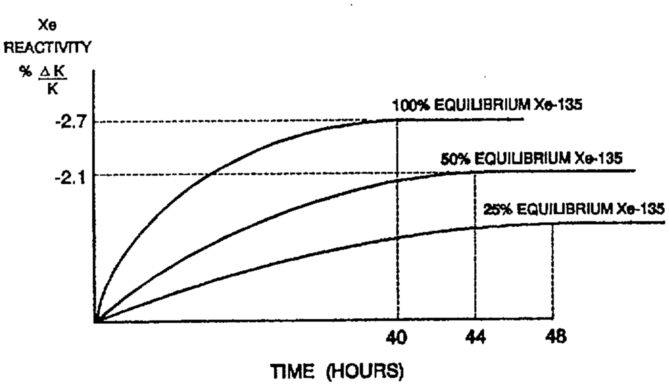


Figure 6-5 Clean Core Startup

XENON BEHAVIOR AFTER A REACTOR SHUTDOWN (PEAK XENON)

If a reactor is shutdown from an equilibrium xenon condition, the behavior of the concentration of Xe-135 or its reactivity worth is as shown in Figure 6-6. Approximately 6 to 10 hours after shutdown the amount of Xe-135 in the core increases to a peak value then decays away to a xenon free condition at a rate that eventually approaches the 9.2 hour half-life of Xe-135. A good rule of thumb for estimating the time to peak xenon after a scram is: "Time to peak xenon (hrs) equals the square root of % power after a shutdown". Using this rule a scram from 100% power will result in peak xenon about 10 hours later. The core can be considered to be xenon free approximately 70 to 80 hours after shutdown. Starting the shutdown from a higher power level will require more time to reach the xenon free condition.

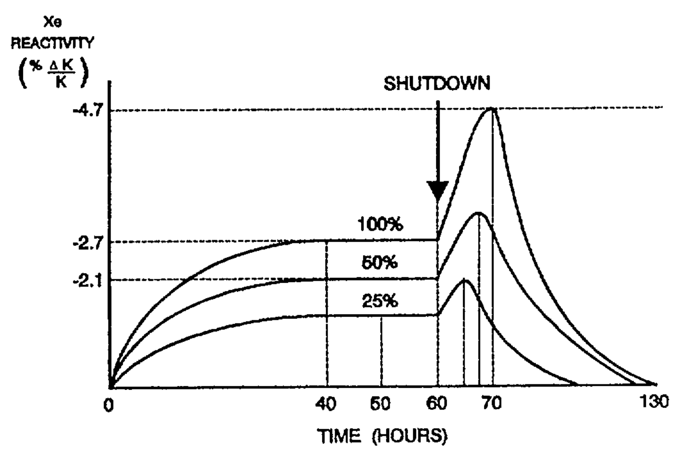


Figure 6-6 Xenon Peak After Shutdown

RULE

"TIME to REACH the PEAK is EQUAL to the SQUARE ROOT of the POWER"

When a downpower maneuver or trip occurs the time in hours to reach the xenon peak concentration is equal to the square root of the power before the downpower.

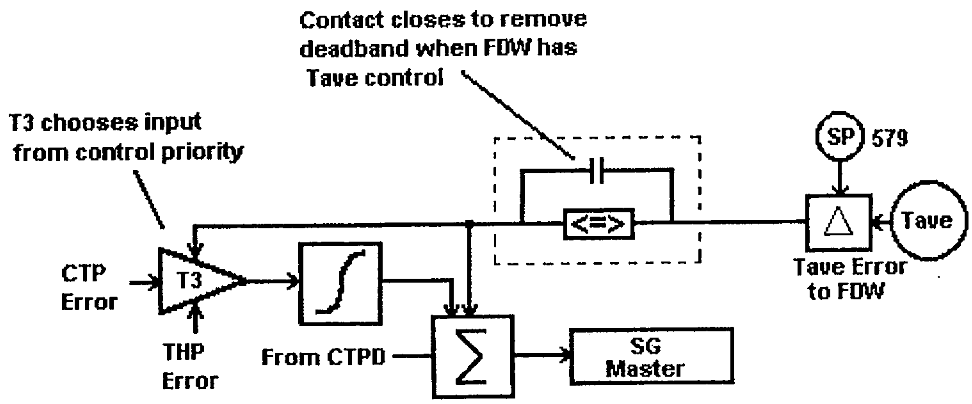
Figure 6-6a Time to Peak Xenon

9. Identify the conditions that would shift the Turbine Master to "hand" and those exceptions that would defeat the shift. (R9)
10. Identify the functions of the Turbine Bypass Valves in terms of the following: (R10)
    - 10.1 Setpoint control
    - 10.2 Setpoint bias application
    - 10.3 Independent overpressure protection
    - 10.4 Control interlocks
11. Describe the operation and limitations of the Turbine Load and Unload circuit. (R11)
12. List the inputs that affect total FDW demand and identify when each is utilized. (R12)
13. Identify the conditions that will remove the control deadband from Tave error to the feedwater subsystem. (R13)
14. Explain why feedwater temperature modification to feedwater demand is necessary and the effects it has on plant efficiency. (R14)
15. Describe how loop feedwater demands are generated and the factors (Loop Tcold ratio and RC Flow ratio) which affect the balance between the two demand signals. (R15)
16. List the conditions that block the temperature initiated delta Tc modification. (R16)
17. Identify the purpose and operation of the SG high and low level limits circuits including actuating conditions and Operator over-ride capabilities. (R17)
18. Given a set of conditions, identify the position response of the following: (R18)
    - 18.1 Main FDW Control Valves
    - 18.2 Main FDW Block Valves
    - 18.3 Startup Control Valves
19. Explain how a feedwater runback is accomplished in the FDW subsystem if some or all of the control stations are in hand. (R19)
20. Describe how ICS feedwater pump speed signal is processed from FDW loop demands and valve differential pressure. (R20)
21. Explain how the FDW valve delta P auctioneering circuitry can prevent a unit transient for any single delta P signal failure. (R21)

1) A proportional THP error term applied to the FDW summer with a limited maximum modification of ± 50#.

2) The same THP error will input to FDW integral when FDW responsible for THP control.

c) Tave Error



1) Normally a Tave proportional error term is applied via the FDW summer.

2) Helps balance FDW against reactor power to maintain heat balance by responding to gross Tave errors (> 1.2 degrees F).

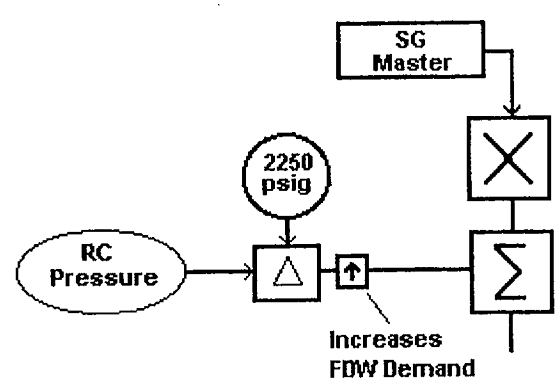
3) Deadband of ± 1.2 degrees is applied during normal operation when Reactor has ultimate Tave control

4) Deadband is removed when FDW has Tave control and the Tave error will also be applied to the FDW integral.

d) RCS Pressure Error

1) A proportional error term applied ABOVE 2250 psig.

2) Circuit increases FDW demand to correct for heat imbalance in effort to prevent RPS high pressure trip at 2345 psig.



(a) The same error term is applied to the Reactor to decrease demand and insert rods as well as and to the Turbine to increase demand and lower THP.

QUESTION # 3

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**0000 15/17**	**A2.08**
Importance Rating	**3.4**	**3.5**

Technical Reference(s): **ARM 2SA-9/D2, AP/2/A/1700/16, CASE "A" RCP Evaluation (Immediate RCP trip criteria)**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-CPM #15**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 __X__

Comments:
GTH

1 POINT

QUESTION # 3

Unit 2 plant conditions:

- Reactor power = 20% and steady
- 2SA-9/D2 (RCP VIBRATION HIGH) actuated
- 2SA-16/D2 (RC Pump Motor 2B1 Oil Pot Low Level) actuated
- All RCPs seal leakage flow = 0 gpm

- 2B1 RCP parameters:
  - <u>SEAL RETURN FLOW</u>
    - ➢ 4.0 gpm
  - <u>HIGHEST VIBRATIONS</u>
    - ➢ Motor shaft = 3.2 mils
    - ➢ Spool piece = 17.6 mils
    - ➢ Upper bearing = 17.3 mils
  - <u>SEAL RETURN TEMPERATURE</u>
    - ➢ 186°F increasing
  - <u>OIL POTS</u>
    - ➢ Upper - Level = +.22" steady and Temperature = 108°F steady
    - ➢ Lower - Level = -1.3" decreasing and Temperature = 113°F increasing
  - <u>MOTOR BEARING TEMPERATURE</u>
    - ➢ Upper Guide = 130°F decreasing
    - ➢ Lower Guide = 125°F increasing
    - ➢ Thrust = 140°F steady

Which ONE of the following operator actions is correct?

SEE ATTACHMENT

The 2B1 RCP should be immediately tripped due to ...

A. high seal return flow.

B. high sustained vibration.

C. seal return temperature increasing.

D. lower motor bearing temperature increasing.

1 POINT

QUESTION # 3

000015/17 A2.08 (PRA) 1-29-00 (GTH/PMS)
(2)

A. Incorrect – With the current conditions, high seal return flow does not warrant an immediate RCP trip although this answer would be correct if seal leakage was high and not 0 gpm.

B. Incorrect – With the current conditions, high vibration does not warrant an immediate RCP trip although if the vibration indication were an emergency high vibration condition this answer would be correct.

C. Incorrect – Seal temperature would be need to be > 200°F for this to meet the criteria for an immediate RCP trip.

D. Correct – Decreasing lower bearing oil pot level and lower bearing temperature increasing meets an immediate RCP trip condition of AP/1700/16, Abnormal RCP Operation.

13. Discuss bypassing the RCP starting interlocks, including the plant conditions when the interlocks would be bypassed and operator action required to bypass the interlocks. (R17)

14. Explain the bases behind the additional requirement for starting the fourth RCP associated with RCS flow rates prior to pump start. (R15)

15. Given a set of plant conditions utilize information provided in the ARGs, RCP Operating and Abnormal Operating procedures to ensure proper operation of all units RCPs during all modes of operation. (R16)

16. Given a copy of ITS/SLC and Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's. (R18)

17. Apply all ITS/SLC rules to determine applicable conditions and Required Actions for a given set of plant conditions (R18)

18. Compute the max. Completion time allowed for all applicable Required Actions to ensure compliance w/ ITS/SLC's. (R18)

D-2

RC

PUMP VIBRATION HIGH

NOTE: Due to Y2K upgrade this alarm no longer has reflash capability. If alarm locks in Operator should increase monitoring of RCP vibrations.

1. Alarm Setpoint

Parameter	2A1 RCP		2B1 RCP		2A2 RCP		2B2 RCP	
RCPM Upper Bearing Displacement (MOX)	15 mils	A2136	15 mils	A2154	15 mils	A2145	15 mils	A2163
RCPM Upper Bearing Displacement (MOY)	15 mils	A2137	15 mils	A2155	15 mils	A2146	15 mils	A2164
RCPM Motor Coupling Displacement (MIX)	OOS* 15 mils	A2138	OOS* 15 mils	A2156	OOS* 15 mils	A2147	OOS* 15 mils	A2165
RCPM Motor Coupling Displacement (MIY)	OOS* 15 mils	A2139	OOS* 15 mils	A2157	OOS* 15 mils	A2148	OOS* 15 mils	A2166
RCP Spool Piece Displacement (IPX)	16 mils	A2140	15 mils	A2158	15 mils	A2149	15 mils	A2167
RCP Spool Piece Displacement (IPY)	16 mils	A2141	15 mils	A2159	15 mils	A2150	15 mils	A2168
RCPM Motor Stand Velocity Probe (MIZX)	3.00 mils .186 in/sec	A2142	3 mils .186 in/sec	A2160	3 mils .186 in/sec	A2151	3 mils .186 in/sec	A2169
RCPM Motor Stand Velocity Probe (MIZY)	3.00 mils .186 in/sec	A2143	3 mils .186 in/sec	A2161	3 mils .186 in/sec	A2152	3 mils .186 in/sec	A2170

* These alarms have been disconnected by ONOE 12077

ABNORMAL REACTOR COOLANT PUMP OPERATION
AP/2/A/1700/16

CASE A

Reactor Coolant Pump Evaluation

4.0 Immediate Manual Actions

_____4.1 Monitor the Reactor Coolant Pump(s) for Immediate Trip criteria.

Parameter	Trip Limit
RCP Seal Return Flow Actual (computer points; A1648, A1649, A1650, A1651) plus Seal Leakage Flow	> 4.1 gpm
RCP Upper Seal Temperature (2TE-1707, 1709,1711, 1713)	> 200°F
RCP Control Bld Off TE (computer points; A1272, A1273, A1274, A1275)	> 200°F
RCP Seal Integrity (**2A1 Only**) RCP LOWER SEAL CAVITY PRESSURE (2PT-219)	Lower Seal Press ≈ RCS Pressure OR Lower Seal Press ≈ RB Pressure
RCP Seal Integrity (2B1, 2A2, 2B2) • RCP UPPER SEAL CAVITY PRESSURE (2PT-206, 207, 208) • RCP LOWER SEAL CAVITY PRESSURE (2PT-220, 221, 222)	Two of the three RCP seals stages fail as evidenced by d/p across the remaining stage approximately equal to RCS pressure (with Seal Return established).*
RCP Vibration	Sustained actual Emergency High Vibration as verified by Alarm Response Guide for (2SA9/E-2) "RCP VIBRATION EMERG. HIGH".
Low oil pot level	AND any RCP Motor Brg Temp Increasing
Loss of HPI Seal Injection	AND Component Cooling has been lost

* RCP seal d/p is determined as follows:

- 1st stage d/p = system pressure - RCP Lower Seal Cavity Pressure.
- 2nd stage d/p = RCP Lower Seal Cavity Pressure - RCP Upper Seal Cavity Pressure.
- 3rd stage d/p = RCP Upper Seal Cavity Pressure - RB atmospheric pressure.

chg #96 chg #96

QUESTION # 4

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**BW/E9 A1.2**	
Importance Rating	**3.2**	**3.5**

Technical Reference(s): **TA-AM1**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **TA-AM1, 3,4,8,9,14**

Question Source:
Bank # ______
Modified Bank # ______
New ___X___

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 4

Which ONE of the following provides the operator with indications of **INADEQUATE** natural circulation?

A. Incores increasing / OTSG pressure decreasing

B. Core ΔT increasing / OTSG pressure increasing

C. Incores decreasing / OTSG pressure decreasing

D. OTSG level decreasing / OTSG pressure increasing

1 POINT

QUESTION # 4

BW/E9 A1.2 (PRA) 1-30-00

A. Correct – Core temperatures increasing coupled with decreasing OTSG pressure indicates that OTSG heat transfer does not exist.

B. Incorrect – High ΔT across the core (Loop ΔT) is an indication of inadequate NC conditions or loss of coupling. OTSG pressure increasing is an indication that steam is being produced therefore primary to secondary heat transfer exist.

C. Incorrect – Core temperatures decreasing is an indication of adequate natural circ when coupled with SG pressure decreasing.

D. Incorrect – OTSG level decreasing and OTSG pressure increasing is an indication of steaming in the OTSG.

TRAINING OBJECTIVES

TERMINAL OBJECTIVES

1. Learn to recognize the symptoms/indications and requirements for proper natural circulation cooling modes of operation, so as to 1) be able to insure proper heat removal from the core if forced RCS flow (RCPs) is lost and 2) be able to determine if natural circulation flow is lost. (T1)
2. Become familiar with, and be able to describe, all of the possible modes of natural circulation operation, and explain the effect that different core decay heatlevels.has on natural circulation. (T2)
3. Become familiar with, and be able to explain, the operator actions in the control room that can affect natural circulation operations and describe the effects on natural circulation produced by degraded equipment and systems or inappropriate operation of this equipment. (T3)

ENABLING OBJECTIVES

1. State the four basic requirements necessary for natural circulation flow. (R1)
2. Describe how feedwater to the SGs is normally controlled to establish natural circulation cooling after the RCPs are tripped. (R2)
3. Describe the normal expected Tc andTh responses following a loss of forced RCS flow. (R3)
4. Explain the effect on natural circulation cooling of varying each of the four basic requirements necessary for natural circulation flow, and generally, how each requirement can be varied. (R4)
5. Explain how and why a plant cooldown performed via natural circulation cooling is different and more restrictive than a plant cooldown performed when forced flow is available. (R5)
6. Explain why the RV head vents must be maintained open during a plant cooldown in natural circulation. (R6)
7. Describe the limitations on RCS pressure control while operating in natural circulation and list the available options for controlling RCS pressure during this mode. (R7)

TRAINING OBJECTIVES (continued)

8. Explain how abnormal or incorrect SG pressure control during subcooled natural circulation will adversely affect system parameters. (R8)

9. Compare the responses of the system, during subcooled natural circulation, to properly controlled feedwater versus abnormal or improperly controlled feedwater. (R9)

10. Describe how the failure to properly control SG levels, Pressurizer level, and SG pressures resulted in an over-cooling transient during the RCP Switchgear fire on Oconee Unit 1 in January, 1989. (R10)

11. Recognize that cold leg temperature anomalies can occur, particularly at low decay heat levels during subcooled natural circulation and explain why these anomalies can occur. (R11)

12. Given a set of RCS temperature/pressure values during natural circulation operation, be able to determine the proper indication to be used for monitoring RCS cooldown rates. (R12)

13. Compare the indications and possible control differences between a normal two SG cooldown vs. a single SG cooldown while in subcooled natural circulation. (R13)

14. List the three general causes for interruption of natural circulation and describe the symptoms that would be seen by the operator in the control room if a loss of natural circulation occurs. (R14)

15. Describe the differences between subcooled and two-phase natural circulation. (R15)

16. Explain the process by which the Boiler-Condenser Mode (BCM) of natural circulation evolves following a Small Break LOCA (SBLOCA). (R16)

17. Explain how the rate of feedwater addition to the SGs affects the evolution to BCM and why SG levels must be maintained at the "Loss of Subcooled Margin Setpoint" for BCM operation. (R17)

18. Explain how degraded HPI System operation can have an adverse effect on BCM operation and, briefly, what operator actions can be taken if HPI flow is lost during a SBLOCA. (R18)

2. PRESENTATION

2.1 Subcooled Natural Circulation Cooling

A. Post-trip Decay Heat

1. Operator actions specified in the EOP are based on the primary goal of establishing and maintaining a successful method for removing post-trip decay heat from the core. The operator's task is to insure that this decay heat energy is transferred to an ultimate heat sink which in most cases is Lake Keowee (via the CCW System).

2. Normally this is accomplished by forcing RCS flow through the core, then through the SGs, where the heat is rejected to the Feedwater System, and ultimately to the CCW System.

B. Basic Requirements For Natural Circulation Flow

1. Without forced circulation (i.e., RCPs are tripped) the next best method for heat removal is via natural circulation of the RCS through the core to the SGs where the heat isrejeted to the Feedwater/CCW Systems. Natural circulation of the RCS water through the core is possible as long as the following four requirements are met:

a) A heat source is available to produce warm (low density) water.

b) A heat sink is available to produce cool (high density) water.

c) A flow path that couples the warm water to the cooler water is provided.

d) The elevation of the cooler water (heat sink) is higher, relative to the heat source, so that this heavier water "falls" into warm area as the lighter, warmer water is pushed into the heat sink.

C. Normal Feedwater Control For Natural Circulation

1. Natural circulation in the RCS will normally develop following a loss of forced circulation provided the SG levels are increased to, and maintained at, 50% OR so that the elevation of the heat sinks are higher than the core.

2. Following a reactor trip and flowcoastdown from a full power operating condition, SGs will automatically begin to fill to the 50% OR level.

3. If Main Feedwater is available, it should be feeding the SGs through the Auxiliary Feed Rings at the tops of the SGs because the ICS should have automatically swapped the Feedwater valves from their normal alignment through the lower Main feed rings to the Auxiliary headers (SG Emergency Header Blocks open, and the SG Normal SU Header Blocks close).

   a) The 50% OR level for the SGs is necessary to maintain natural circulation because this establishes the heat sinks elevation (SGs) above the heat source elevation (the core).

   b) By swapping feed to the Auxiliary Feed Rings, natural circulation flow is promoted during the fill to the 50% OR level by spraying the feedwater directly onto the SG tubes at a high elevation. Spraying directly onto the tubes (even a small portion of the total number of tubes) precludes having to actually establish a water level in the SGs, because as long as the water is spraying onto the tubes, this, in effect, raises the elevation of the heat sinks above the core.

   c) However, if this feed were to be completely terminated prior to establishing a proper water level, natural circulation would be severely hampered and would probably fail to evolve.

4. Of course, if Main Feedwater is not available, EFW automatically raises SG levels to ≈240" XSUR (equivalent to 50% OR if SG pressure = 1000 psig) by spraying through the Aux Feed Rings.

5. It should be noted that once the SGs have reached the appropriate levels, the Feedwater control valves must be adjusted to maintain this level. If the feed rate is not sufficient to maintain this level, levels will drop below the required heat sink level, and natural circulation flow will be interrupted.

   For subcooled natural circulation, this should not be a problem if Main Feedwater is used through the S/U Control Valves since they will automatically maintain the 50% OR setpoint. If EFW is used, FDW-315 and FDW-316 must be manually regulated.

D. Primary Temperature Indications

   1. Natural circulation requires approximately 10-15 minutes to become fully established under normal circumstances from a full power trip.

   2. During the transition from forced to natural circulation:

      a) The cold leg temperatures should remain near the saturation temperature for the existing SG pressure.

      b) The hot leg temperatures and CETCs will increase until a stable ΔT between the hot and cold legs is established, generally at 30-40°F.

E. Effectiveness of Natural Circulation Cooling

The effectiveness of natural circulation flow in providing cooling, and how quickly steady state flow develops, relates directly to the four requirements that were listed as being required for natural circulation:

1. The strength of the heat source:
    a) Since the heat source provides the "driving force" for natural circulation, if all other factors remain the same, a hotter heat source will produce greater natural circulation flow rates.
    b) For the core, this depends upon the amount of post trip decay heat present which, in turn, is dictated by the core's operating history (EPFDs) and the amount of time that has elapsed between the time of the reactor trip/shutdown and when a loss of forced circulation occurs.
    c) In summary, reactor trips from high power levels will produce more pronounced indications of natural circulation flow than trips from very low power levels.
2. The strength of the heat sink:
    a) As with the heat source, a stronger heat sink will produce greater natural circulation flow rates.
    b) Cooler FDW results in denser (heavier) RCS water at the higher heat sink elevation which, in turn, "falls" into the core more readily.
    c) The strength of the heat sink is related to the temperature of the water used to feed the SGs, and to the pressure at which the SGs are controlled (Psat).
    d) As a general rule, using EFW instead of MFW will produce greater natural circulation flow rates and, therefore, has more cooling effect because EFW is cooler than MFW.
    e) Similarly, lower SG pressures will produce cooler water as the Psat is reduced.
3. The resistance to flow:
    a) In order for natural circulation flow to occur the heat sink and heat source must be coupled.
    b) The resistance to flow caused by the connecting piping and components in the RCS impacts the effectiveness of the system in evolving into a steady flow condition. Resistance to flow is generally dictated by the design of the system itself and the operator will have no control over this factor.

c) However, there are certain situations where the system can evolve into a condition that inhibits natural circulation flow due to increased resistance to flow caused by steam or gas binding in the connecting piping (hot legs).

d) In these situations prompt operator actions are necessary to maintain or re-initiate natural circulation flow. These actions are addressed in a later section of this lesson.

4. The difference between the relative height of the water at the heat source and at the heat sink:

   a) If the height of the heat sink is increased relative to the source more driving force is obtained due to gravity effects on the denser, heavier water.

   b) increasing SG levels or spraying the feedwater in at the top of the SG versus into the downcomer region effectively increases the height of the heat sink.

F. Plant Cooldown via Subcooled Natural Circulation

1. EOP instructions regarding plant cooldown using natural circulation basically deem such an evolution as undesirable if it can be at all avoided.

2. Once steady-state natural circulation has been established, the plant should be maintained at this "hot shutdown" condition until RCPs can be restarted to effect the plant cooldown.

3. The basis for this approach is that a natural circulation cooldown is more difficult to regulate and must be controlled at a much slower rate than is possible under forced flow conditions.

4. Under ideal conditions, RCS flow rates during natural circulation will be on the order of 2-4% of nominal 4-RCP flow:

   a) Loop transit time (the time for a pound of water to circulate around the RCS) which is normally about 12 seconds during 4-RCP operation will increase to approximately 4-7 minutes during steady-state natural circulation. This relatively long transit time results in a much more sluggish response from the system during natural circulation operation making control of a cooldown rate more difficult.

   b) In addition, the severely reduced flow rates during natural circulation results in very limited flow in and around the very top of the dome of reactor vessel head limiting effective cooldown of the thick metal mass in this area.

c) If a system cooldown proceeded at a normal rate during natural circulation operations, the RV head would remain considerably hotter than the cold legs and, as the RCS wasdepressurized, the water in the dome would soon reach saturation pressure for its higher temperature and a void would form in the head.

d) The problem with this occurrence is that the steam void would eventually grow large enough to become the controlling RCS pressure regulator instead of thePressurizer while at the same time displacing any water that was in the area.

e) RCS pressure control would be lost and only ambient cooling would eventually collapse the head void if RCPs could not be run.

f) Also, steam voids in the system, external to thePressurizer, can result in sudden variations in RCS levels if the void were to suddenly collapse.

5. If a natural circulation cooldown<u>must</u> be performed, the maximum allowable cooldown rate is reduced to < 50°F/hr in order to provide more time for cooling in the RV head area.

6. In addition, the RV Head Vents<u>will be</u> opened prior to initiating the cooldown and will be maintained open throughout the cooldown:

   a) Opening the head vents helps prevent a stagnant region of water/steam from forming in the RV head area by venting off any steam formed and by providing a path for the hot water in this area to be displaced by cooler water entering the area.

   b) As the cooldown proceeds RCS pressure is maintained at 2155 psig until well into the cooldown so that the large ΔP across the vent valves maximizes the relief flow through them.

7. Analysis has shown that by using the head vents a natural circulation cooldown at less than 50°F/hr can be maintained all the way to decay heat removal conditions without causing significant head voiding.

8. Problems with natural circulationcooldowns have occurred in the industry. One such event occurred at St.Lucie Unit 1 in June of 1980. During a natural circulation cooldown, a steam void occurred in the reactor vessel head. This and the subsequent collapse of the void caused anomalies in Pressurizer level which masked an RCS leak.

G. RCS Pressure Control During Natural Circulation

1. With no RCPs operating, normal RCS pressure control is unavailable since normal PRZ spray is lost.

2. Ideally, this should not present undue problems since the plant will normally remain at steady state hot shutdown conditions until RCPs can be regained.

3. Auxiliary Pressurizer spray control is intended to be used during the latter stages of plant cooldown when RCPs become unavailable. Using Aux Spray at elevated RCS pressures (high pressurizer temperatures) can present complications because of the large ΔT that exists between the spray water and the Pressurizer. The limitations on the use of Aux Spray at high RCS pressures were highlighted during the Unit 1 RCP Bus fire in January, 1989:

   a) Tech Specs limits the allowable ΔT between the PZR and the spray fluid to 410°F in order to limit the thermal stress on the spray line.

   b) At nominal RCS pressures (2155 psig), the PZR temperature is on the order of 650°F. Since Aux spray is initiated from the HPI System, the spray fluid will be in the 100°F temperature range resulting in a 550°F ΔT between the PZR and spray fluid. Not until RCS pressure is reduced to around 750 psig will the ΔT be able to meet the 410°F limit.

   c) So, unless circumstances warrant extraordinary efforts to reduce RCS pressure, Auxiliary Pressurizer Spray is not a viable option at high RCS pressures.

4. Other options are available to reduce RCS pressure whenever RCPs are not running though:

   a) Turning off the Pressurizer heaters and allowing ambient losses to cool the Pressurizer. This, of course, will not result in a rapid reduction in RCS pressure.

   b) Lowering Pressurizer level, with the heaters deenergized, will result in a more rapid pressure reduction, as the Pressurizer bubble expands.

   c) Using the PORV, if a RAPID reduction in RCS pressure is required.

5. In emergency situations, if the PORV is not available, the Emergency Coordinator may give approval to exceed the 410°F ΔT limit and authorize use of Aux Spray to depressurize the RCS.

2.2 Low Decay Heat Load Effect OnSubcooled Natural Circulation

Past analyses of natural circulation operations have focused on reactor trips from full power operations where the initial post-trip decay heat source is fairly large. Recent operating events at B&W plants, including theSwitchgear fire at Oconee Unit 1 in January 1989, indicate that sensitivities not previously stressed, exist during natural circulation events initiated with low decay heat loads present.

A. SG Pressure Control

1. While nominal transitions to natural circulation should evolve within about 10 minutes following trips from full power, low power events can generally be expected to result in longer periods of time before stable natural circulation is reached because of the lower thermal driving head.

2. During this period of time the operator may be induced into taking compensatory actions for what he or she feels is a failure of the system to evolve properly into natural circulation. However, drastic action, such as SG depessurization, is not required at this point to induce natural circulation nor is it desirable.

   a) During any natural circulation event, but especially during low decay heat situations, SG pressure control is of paramount importance.

   b) Natural circulation is characterized by the ability of the system to promote and maintain near-normal post-trip SG pressures.

   c) Purposely depressurizing the SGs in an effort to quickly promote natural circulation flow will result in rapid overcooling of the RCS when natural circulation begins since, particularly at low decay heat loads, a reduction in cold leg temperatures will occur as the SGs are fed to the natural circulation level setpoint. Further depressurization of the SGs will only compound the overcooling problem.

   d) As long as the core exit SCM is maintained there is no threat to core cooling so, again, drastic action to try and force the system into a faster evolution into natural circulation is undesirable.

B. Feedwater Control

1. In order to maintain steady-state natural circulation in the RCS the SGs must be fed heat through the Aux. nozzles at a rate that is adequate to remove the decaydecay and/or a level of 50% OR must be established and maintained. Either of these options results in the heat sink elevation being above the core elevation.

2. While it is desirable to increase SG levels to the natural circulation setpoint in order to establish stable loop flow, the rate of fill is dependent upon the amount of decay heat available. Excessive fil! rates, especially at lower power levels, will cause a rapid cooldown of the cold legs.
   a) Following a trip from 100% power, the MFDW flow rate established automatically by ICS should not require operator intervention.
   b) However, a trip from 100% power requiring EFDW will require operator intervention to prevent overcooling due to the excessive EFDW flow rates established when FDW-315/316 fully open on low SG levels.
   c) Following a trip from low power (or low decay heat) levels, operator intervention will be required whether MFDW or EFDW is available.
3. At lower power levels it may also be necessary to throttle feedwater flow somewhat to limit SG depressurization that is caused by the condensation of the steam by the feedwater spraying into the SGs during the fill.
4. On the other hand, if feedwater flow is severely reduced before the SGs reach 50% OR natural circulation flow will likely interrupted due to a loss of the SG heat sink.
5. Therefore, unless violation of the RCS cooldown rate limit is imminent, some flow should be maintained to the SGs through the Auxiliary Feed Ring until the 50% OR (or 240 XSUR) level setpoint is reached.
6. Key parameters to be monitored during SG fill to the natural circulation setpoint are cold leg temperatures and SG pressures:
   a) Cold leg temperatures, as natural circulation flow develops, should be about equal to the saturation temperature in the SGs. For example:
      1) If SG pressures are controlled at or near normal post trip pressure (1000 psig), SG temperatures, and therefore RCS cold leg temperatures, should be about 550°F.
      2) If cold leg temperatures decrease much below the normal expected post-trip temperature, it is a good indication that overcooling may be occurring.
   b) Again, SG pressures are of paramount importance in monitoring and controlling subcooled natural circirculation:
      1) Decreasing SG pressure indicates excessive steam flow from the SG, or depressurization due to excessive feedwater flow (for the decay heat level), both of which are undesirable.

2) Reduced SG pressures may also reflect less steam generation, which is tied to RCS flow stagnation (i.e. interruption of natural circulation).

c) To minimize SG depressurization, particularly when decay heat levels are low:

1) it is important to isolate all loads from the SGs, such as the MSRHs, CSAEs, and the Aux Steam Header.

2) Control TBVs to limit SG depressurization, i.e., do not purposely depressurize the SGs in an effort to force the system more quickly into natural circulation (when the RCS is subcooled).

3) It may be necessary to throttle feedwater somewhat to limit SG depressurization, but <u>some flow must be maintained until the 50% OR level is reached</u>. Only if violation of the cooldown limit is imminent or has occurred should feedwater flow be completely stopped.

C. Oconee Unit 1 RCP Bus Fire - Lessons Learned

1. Several important lessons relating to natural circulation operations became evident during the fire that occurred in the 1TA Switchgear in January, 1989. Aside from the many equipment malfunctions that took place during the event, the evolution into natural circulation mode of cooling was unique in that it occurred from a low decay heat load condition (reactor power was at approximately 4% when the reactor was manually tripped and the RCPs secured).

2. The event demonstrated the sensitivities of the system, particularly at low decay heat loads, to excessive feedwater addition to the SGs and excessive depressurization of the SGs.

3. Abnormal SG pressure and feedwater control, and several control equipment failures, resulted in cold leg temperatures as low as 398°F within a two-hour period following the reactor trip, a cooldown of 120 °F in the first 30 minutes of the transient, and SG pressures as low as 430 psig.

4. In addition, when all RCPs were tripped, normal PZR spray control was lost and RCS pressure was increasing due to high HPI makeup flow:

a) An attempt was being made to limit RCS pressure increases by using the TBVs (i.e. by cooling off) rather than by limiting PZR level increases (i.e. throttling HPI flow).

QUESTION # 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**1**	**1**
	K/A #	**000051 G2.1.7**	
	Importance Rating	**3.7**	**4.4**

Technical Reference(s): **STG-MT, CF-FPT, STG-MS, STG-ICS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **MT-11 ICS-10 FDWPT-11**

Question Source:
- Bank # ________
- Modified Bank # ________
- New ___X___

Question History:
- Previous NRC Exam ________
- Previous Quiz / Test ________

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 __X__

Comments:

1 POINT

QUESTION # 5

Unit 1 plant conditions:

- Reactor power = 10%
- Turbine load 90 MWe
- Shutdown in progress following a startup from MODE 6
- The core is operating with a slightly <u>positive</u> α_{MT}
- "1A" Main FDWPT operating
- ICS Reactor Master in HAND

CURRENT CONDITIONS:

- Condenser vacuum rapidly decreases to 20" Hg and then stabilizes

Which ONE of the following is correct?

ASSUME NO OPERATOR ACTION

specific determiner

~~The reactor will ____ and~~ Unit 1 ~~core~~ power reactivity will **initially** ____ ~~due to the~~ and the positive α_{MT}. Rx will immediately ____

A. trip / increase

B. trip / decrease

C. NOT trip / increase

D. NOT trip / decrease

High Flux Trip due to uncontrollable power increase

core reactivity ≠ reactivity

1 POINT

QUESTION # 5

000051G2.1.7 BOTH PRA 2-2-00

A. Incorrect – The turbine will trip on low vacuum at 21.75" Hg. The reactor will not trip due to < 30% power turbine-reactor trip bypass in effect. The reactor would trip if vacuum reduced to 19" Hg due to the FDW pumps tripping.

B. Incorrect – The turbine will trip on low vacuum at 21.75" Hg. The reactor will not trip due to < 30% power turbine-reactor trip bypass in effect. The reactor would trip if vacuum reduced to 19" Hg due to the FDW pumps tripping.

C. Correct - The turbine will trip on low vacuum (setpoint = 21.75" Hg). The reactor will not trip because: 1) the turbine to reactor trip is bypassed < 30% 2) Main FDWPT will remain in operation due vacuum is remaining above it's trip setpoint of 19"Hg. When the turbine trips the TBVs will maintain MS pressure (since vacuum is > 7" Hg) via the ICS "at setpoint". RCS temperature will initially increase due MSSVs closing and increasing MS pressure then long term remain constant as the bypass valves maintains MS pressure at the same pressure as before the turbine trip.

D. Incorrect – The turbine will trip on low vacuum (setpoint = 21.75" Hg). The reactor will not trip because: 1) the turbine to reactor trip is bypassed < 30% 2) Main FDWPT will remain in operation due vacuum remaining above it's trip setpoint of 19"Hg. When the turbine trips and the MSSVs disc-dump close Tave will initially increase from the post turbine trip conditions. TBVs will open and reduce the elevated Tave while maintaining MS pressure (since vacuum is > 7" Hg) via the ICS "at setpoint". Long term RCS temperature will remain constant as the bypass valves maintains MS pressure at the same pressure as before the turbine trip.

PWR — INSTRUCTOR GUIDE

LESSON COURSE TITLE REACTOR THEORY **TIME** 12 Hrs

LESSON PLAN TITLE REACTIVITY COEFFICIENTS **REV** 1

REFERENCES

1. "Nuclear Reactor Engineering," Glasstone & Sesonske, 1981

2. "Reactor Core Control for Large Pressurized Water Reactors," Westinghouse, 1987

RESPONSIBILITY	SIGNATURE	TITLE	DATE
ORIGINATION			
REVIEW/ CONCURRENCE			
REVIEW/ CONCURRENCE			
APPROVAL/ OBJECTIVES			
APPROVAL/ FINAL			

OBJECTIVES

1. Define the moderator temperature coefficient of reactivity.

2 Define the fuel temperature coefficient of reactivity.

3. Describe the effect on the magnitude of the moderator temperature coefficient of reactivity from changes in moderator temperature and core age.

4. Explain resonance absorption.

5. Explain doppler broadening and self shielding.

6. Describe the effects of core age, moderator temperature and boron concentration on the moderator temperature coefficient.

7. Describe the effects of core age, fuel temperature and moderator temperature on the fuel temperature (Doppler) coefficient.

TRAINING OBJECTIVES

TERMINAL OBJECTIVES

At the conclusion of this lecture, the student will be able to:

1. Describe the purpose and operation of the main turbine and its related components.
2. Explain the function of and purpose for the various protective actions/devices associated with the main turbine.

ENABLING OBJECTIVES

1. Describe the steam flow path from entry into the high pressure turbine to the exit of the low pressure turbine. (R1)
2. Explain what is meant by "Double Flow" as it relates to the main turbines at Oconee Nuclear Station. (R2)
3. Explain the two purposes of the Main Steam Stop Valves. (R3)
4. Explain why the # 2 Main Steam Stop Valve has an internal bypass valve. (R4)
5. Explain the purpose of the Control Valve above seat drains. (R5)
6. Explain the purpose of the Reheat Stop Valves. (R8)
7. Explain the purpose of the Intercept Valves. (R9)
8. Explain the purpose of the exhaust hood spray on the Low Pressure Turbines. (R10)
9. Describe the conditions that could result in high exhaust hood temperatures. (R11)
10. Discuss the consequences of high exhaust hood temperatures. (R12)
11. Identify the trip setpoint for "Low Condenser Vacuum". (R13)
12. Explain why the turbine should be placed on turning gear when it is shutdown. (R14)
13. Describe three methods of placing the turbine on turning gear. (R15)
14. Identify the causes of an auto trip of the turning gear motor. (R16)
15. Discuss the purpose of the extraction check valves. (R17)
16. Concerning Main Turbine vibration indication, identify that from: 0-600 RPM indication is invalid, 600-780 RPM indication is approximate, and from 780-1800 RPM indication is accurate. (R18)

6. LP Turbine Journal Bearings
   a) Each LP turbine has two journal bearings to support the rotor.
   b) Same as journal bearings used in HP turbine.
   c) The bearings are numbered consecutively:
      1) #3 & 4 on "A" LP turbine
      2) #5 & 6 on "B" LP turbine
      3) #7 & 8 on "C" LP turbine
7. Exhaust Hood Water Sprays
   a) The exhaust hoods are designed for an operating temperature $\leq$ 300 °F.
      1) If the hoods are heated above the design limit severe damage could occur.
      2) Exhaust hood heating is caused by high backpressure (low vacuum) which increases windage losses or low steam flow past last stages of blades which reduces cooling at low loads.
   b) Water spray manifolds are provided just downstream of the last stage buckets, near the bucket tips, to maintain the exhaust hood temperatures.
      1) Water from the Condensate System is delivered to the manifold via a 2.5" header that taps off between the CSAE's and the Steam Seal Packing Exhauster.
      2) Controlled by an automatic regulator valve (C-304). The valve is regulated via a temperature sensing relay in each turbine exhaust hood.
      3) A bypass valve (C-524) can be used to manually regulate flow.
      4) C-304 will begin to supply flow if the temperature reaches 120° F and will be fully open if hood temperature reaches 180° F.
   c) Local indicating dial thermometers are located in each exhaust hood. Two thermometers are on each side of each exhaust hood.
   d) As added protection against exhaust hood overheating, the temperature sensing relays in each exhaust hood will initiate 2 turbine panel statalarms; at 225°F and 175° F.
   e) The main turbine low vacuum trip setpoint is **21.75 in. Hg**.

6. Describe the operation of the turning gear motor including any associated interlocks. (R5)

7. Concerning the Motor Speed Changer (MSC) and the Motor Gear Unit (MGU) changes the speed of the FDWPT. (R6)

   7.1 Recognize that the MSC has two speeds.

      A. The Fast Raise/Fast Lower speed will run the MSC from the Low Speed Stop (LSS) to the High Speed Stop (HSS) in ~6.5 seconds.

      B. The Raise/Lower speed will run the MSC from the LSS to the HSS in ~95 seconds.

   7.2 Recognize that with the MSC at HSS and the MGU controlling FDWPT speed, there is a "gap" between the MGU and MSC demand signals.

      1. Understand that it will take some time before the MSC demand is less than the MGU demand.

   7.3 Describe how operation of the MSC and MGU changes the speed of the FDWPT.

8. Explain the basic operation of the motor speed changer, motor gear unit and the HAND JACK switch. (R17)

9. Describe the effect on the FDWPT speed control circuitry when the HAND JACK switch is placed in the ON position and explain how the HAND JACK switch causes this effect. (R7)

10. Describe how to tell if the MSC or MGU is on its low speed or high speed stop from the FDWPT front standard. (R10)

11. Describe any trips associated with the FDWP's / FDWPT's (include any setpoints and logic associated with the trip). (R8)

2) When the "B" FDWPT trips, FDW flow and discharge pressure drop and the unit is forced into a loss of FDWP runback.

3) The "A" FDWPT is still at its HSS due to the instrument failure, but due to the FDW load at 65% power and the fact that only "A" FDWPT is running, the FDWP discharge pressure will be < 1275 psig.

4) Operational guidelines are to start "A" FDWPT first on a startup and stop it last on a shutdown since "A" FDWPT has the higher discharge pressure trip.

5. Suction Pressure Low - < 235 psig

   a) 2/3 logic

   b) Prevents cavitation.

6. MSLB circuitry

   a) ≤ 550 psig on 2 out of 3 transmitters on either MS line for 2 seconds will send a trip signal to both main FDWPT's

7. FDWPT Exhaust Vacuum Low - ≈ 19 in. Hg

   a) FDWPT exhausts to the Condenser.

   b) Loss of Condenser vacuum could cause damage to the turbine blades due to heating.

8. SG level High (also simultaneously trips Main Turbine.)

   a) Setpoints vary between units and are subject to change. (≈ 98% Operating Range – 2 out of 2 logic)

   b) Two level transmitters per SG monitor level to provide the High SG level trip/Steam Generator Overfill Protection.

   c) The circuit will trip the Main Turbine and both Main FDWP's if the correct logic is presented from either SG, since either SG can cause overfill.

   d) Overfill can cause RCS overcooling and can present Reactor Vessel PTS concerns.

9. Oil Fire Trip

   a) When "Turbine Oil Fire Trip" station in Control Room is pulled.

10. Thrust Wear

    a) Setpoint in both active and inactive directions.

11. Loss of ICS Power

    a) If ICS Hand and Auto power is lost, the MFDWP's will automatically trip due to SG High Level trip circuit.

are supplied from either SG outlet or Turbine Header pressure and

4. TBV setpoints and controls are discussed in the ICS lesson plan.

   a) TBVs are interlocked closed when Condenser vacuum lowers to ≤ 7" Hg. Vacuum. ADVs would then be required for S/G cooldown.

P&I 9-9-99

REFER to OC-STG-MS-7

F. Steam Traps

1. An "idle" steam line (one in which there is no flow due to a closed isolation valve) tends to collect condensation in the line where the steam is bottled up.

2. All steam lines in the plant have small drain lines on the upstream side of the isolation valve in the line, to keep the condensation from building up.

3. These small drain lines automatically drain condensate to the condenser to prevent the steam line from filling with water and/or to prevent water hammers from occurring. This is accomplished by the use of a steam trap.

   a) Thermostatic steam traps at Oconee are basically a valve with an expandable bellows acting as a valve disk.

REFER to OC-STG-MS-8

   b) When cool (from condensation in the line), the bellows will contract and allow flow through the valve.

REFER to OC-STG-MS-9

   c) When hot (from steam flow), the bellows will expand and shut off flow through the valve.

4. During a system heatup, reduced steam temperatures cause nearly constant condensation formation. To aid in its removal, motor operated trap bypass valves are opened. Later on these valves are closed per procedure once steam temperatures and flows have increased.

REFER to OC-STG-MS-3

2. The purpose of the main steam relief valves is to prevent the maximum pressure of 1155 psig from being exceeded when the turbine trips and the stop valves shut, by relieving the steam to atmosphere.
   a) 1155 psig = 10% above the 1050 psig design pressure limit per ASME II pressure vessel codes.
3. Normal Steam flow in each line at 100% load is » 5.5 million pounds per hour (total of 11 million pounds/Hr).
4. Relief valves are sized to relieve 13.1 million pounds per hour and is such that the steam load produced from a Reactor trip from 112% core thermal power (Tech. Spec. safety limit) can be dissipated through this system.
5. Relief valve settings are staggered so that upon a Turbine Trip from a 100% power, all MSRV's will open, but reseat in a pattern designed so that as the heat generated by the primary drops, the number of valves remaining open decreases, until all relief valves are closed and OTSG pressure is being controlled by the TBV's.
6. These staggered settings, while still protecting the lines from overpressure, also prevent excessive blowdown which would result in excessive cooldown of the primary.
7. The relief valves are designed for as much as 6% blowdown, meaning that they may reseat as low as 94% of lift pressure. This is accomplished by the use of a conical shaped valve disc which offers more surface area for the steam once it has opened. This design is necessary so that the valve is not constantly "popping" and reseating at lift pressure.
8. With this in mind one can see that the 2 relief valves with the lowest lift settings (1050 psig) may not reseat until pressures as low as 977 psig. (By procedure the acceptable lift setting range for the 3 lowest valves is 1039-1060 psig)
9. This is the cause for 2 (or more) relief valves remaining open at normal post trip steam pressures of » 1010 psig. Operators should realize that this is somewhat expected and that it is incorrect to immediately assume that these relief valves are "stuck" open, which would indicate a mechanical failure of the valve(s)

9. Identify the conditions that would shift the Turbine Master to "hand" and those exceptions that would defeat the shift. (R9)

10. Identify the functions of the Turbine Bypass Valves in terms of the following: (R10)

    10.1 Setpoint control

    10.2 Setpoint bias application

    10.3 Independent overpressure protection

    10.4 Control interlocks

11. Describe the operation and limitations of the Turbine Load and Unload circuit. (R11)

12. List the inputs that affect total FDW demand and identify when each is utilized. (R12)

13. Identify the conditions that will remove the control deadband from Tave error to the feedwater subsystem. (R13)

14. Explain why feedwater temperature modification to feedwater demand is necessary and the effects it has on plant efficiency. (R14)

15. Describe how loop feedwater demands are generated and the factors (Loop Tcold ratio and RC Flow ratio) which affect the balance between the two demand signals. (R15)

16. List the conditions that block the temperature initiated delta Tc modification. (R16)

17. Identify the purpose and operation of the SG high and low level limits circuits including actuating conditions and Operator over-ride capabilities. (R17)

18. Given a set of conditions, identify the position response of the following: (R18)

    18.1 Main FDW Control Valves

    18.2 Main FDW Block Valves

    18.3 Startup Control Valves

19. Explain how a feedwater runback is accomplished in the FDW subsystem if some or all of the control stations are in hand. (R19)

20. Describe how ICS feedwater pump speed signal is processed from FDW loop demands and valve differential pressure. (R20)

21. Explain how the FDW valve delta P auctioneering circuitry can prevent a unit transient for any single delta P signal failure. (R21)

b) Other causes of LRM movement which will trip the Turbine Master to manual:

1) Depressing the EHC Increase/Decrease pushbuttons.

2) EHC Load limit setting exceeded (Unit 3 only)

- The load limit signal bypasses the LRM to signal the control valves. However, on unit three there is a circuit that will cause the LRM to drive in an effort to remain balanced with the load limit signal. When this occurs, the LRM movement will cause the Turbine Master to trip to hand. This is not present on units 1 & 2.

J. Turbine Bypass Valves

1. There are three general functions for the Turbine Bypass Valves.

a) Pressure control at low loads before the turbine is capable of accepting pressure control.

1) The valves will automatically control at THP setpoint when the Turbine is NOT controlling THP.

b) High Pressure Relief (Two Types)

1) High pressure relief at THP setpoint +50 psig bias when the Turbine is controlling THP

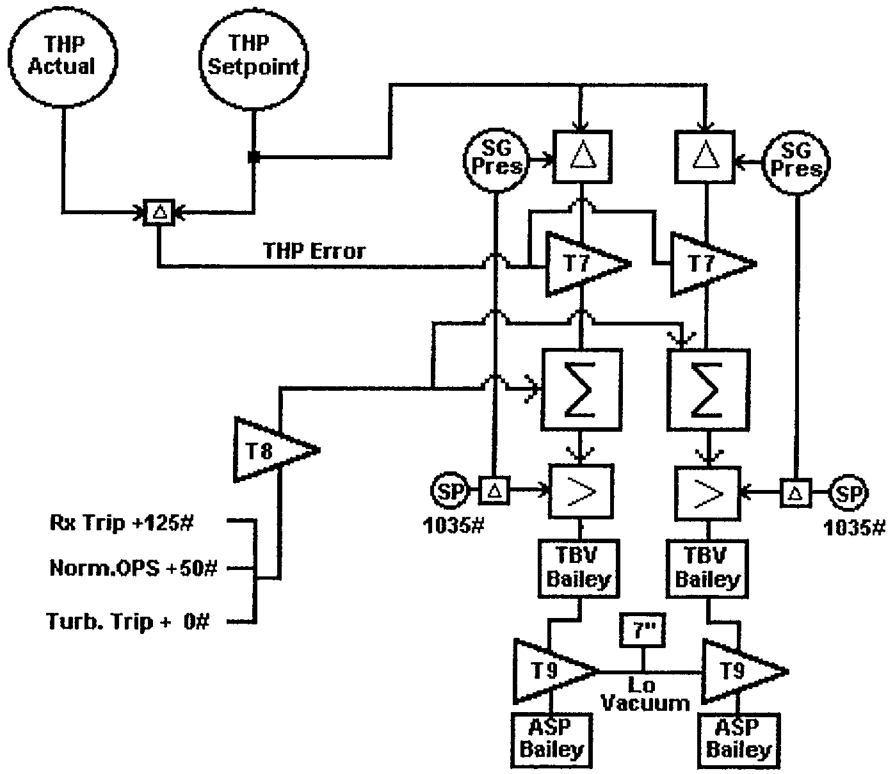
2) Independent high pressure relief that will operate proportionally to SG pressure.

(a) The by-pass valves will start open at 1035 psig and be fully open at 1055 psig.

c) Pressure control after a reactor trip at setpoint +125 psig to limit RCS cooldown when the Diamond control system receives a "trip confirm" signal from control rod drive breakers opening.

1) Diverse Scram System (DSS) will also apply the 125 psig bias to the TBVs during an Anticipated Transient Without a Scram (ATWS) to limit or minimize positive reactivity added due to RCS temperature decreasing.

2. Turbine Bypass Valve Controls and Setpoints



a) There are two separate signals that will be utilized to develop a position demand for the TBVs.

1) THP error:

(a) This signal will control TBV position demand at times when the Turbine is in automatic and controlling THP.

(b) This is the same signal that the turbine will be using to control THP and was added to allow a smooth shift of steam pressure control from/to the TBVs

2) OTSG Outlet pressure error:

(a) SG outlet pressure (not turbine header pressure) is used to determine TBVs pressure error (actual pressure versus setpoint) when the Turbine is in "manual".

3) The #7 transfer function processor (T7) will select the proper control signal based on turbine status.

(a) Turbine Master in **AUTO** = **THP** error signal to TBVs.

(b) Turbine Master in **HAND** = **OTSG** outlet error to TBVs.

b) In BOTH cases, the "setpoint" is established by the operator with the bias control knob on the Turbine Bailey station. (This is the same setpoint as THP setpoint)

c) The TBVs MUST be in "automatic" in the control room AND the Auxiliary Shutdown panel (ASDP) for the valves to respond to the ICS signals.

1) In "auto" their position is controlled by the control error plus bias unless condenser vacuum is < 7" at which point they can only be controlled at the ASDP.

(a) The #9 transfer function processor (T9) will input the vacuum interlock demand signal.

2) Placing the TBV controllers to "manual" at the ASDP will defeat all automatic and manual operation from the control room and the valves can only be controlled from the ASDP.

d) Manual Operation of TBVs

1) The bypass valves are normally controlled by their associated control room Bailey station.

(a) One controller for the "A" steam header TBVs (two valves) and one controller for the "B" steam header TBVs (two valves).

(b) If the station is in hand the operator controls actual TBV position via the Bailey toggle switch.

e) TBV Setpoint Shifts

1) With the TBVs in AUTO, they will control as follows:

(a) At setpoint + 125 psig, if the Reactor is tripped.

- Shift is actuated by "Trip Confirm" on the Diamond Control Panel which receives the signal from CRD breaker logic.
- Shift can also be actuated by the Diversified Scram System (DSS).
- This bias will shift setpoint to 1010 psig which corresponds to a saturation temperature of approx. 546°F for the steam generators.

  (885# + 125#bias = 1010#)
- RCS temperature will follow and be maintained slightly higher at approx. 555°F.
- This will limit the change in temperature of the RCS to the core on a reactor trip. (Tc = 555°F)

(b) At setpoint + 50 psig if the Turbine Load Status Flag is "TRUE". (See "Turbine Load and Unload" below)

(1) The 50# bias will allow the TBVs to act as overpressure relief.

(c) At setpoint with NO bias if the turbine is tripped or the Turbine Load Status Flag is "FALSE".

(1) The TBVs will assume pressure control because the turbine is off-line (MSSVs closed) and can no longer control pressure.

2) The #8 Transfer function processor (T8) will pass the appropriate biased signal to the TBVs.

3. Turbine Load and Unload

a) During a normal plant startup the reactor and steam generators will be producing 10-20% steam flow through the TBVs before the turbine is rolled and synchronized to the generating grid.

b) Once the turbine is on line an initial load of about 30 Mwe is placed on the turbine-generator.

c) **Manual Turbine Loading**

1) Load is manually increased by the operator using the turbine Bailey or the increase/decrease pushbuttons on the turbine panel which will control the main steam control valves.

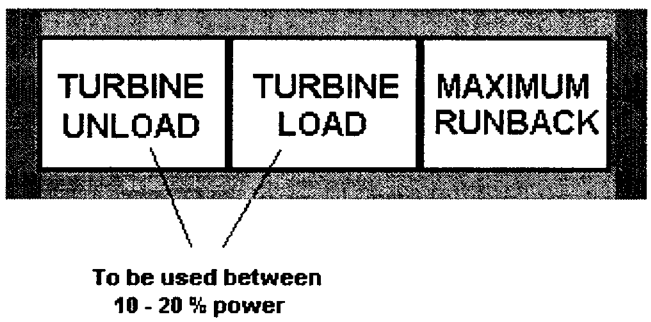
2) TBVs respond to maintain steam pressure at setpoint.

3) As more steam is directed to the turbine, THP decreases and the TBVs will close to maintain setpoint.

d) **Automatic Turbine Loading**

1) The turbine Load and Unload system enables the operator to smoothly introduce and remove the main turbine into the plant control process.

(a) This feature is necessary to provide a smooth transition of steam pressure control from the TBVs to the Turbine or vice versa.



2) Two back-lit buttons are provided on the LCP, TURBINE LOAD and TURBINE UNLOAD.

(a) Each button illuminates when the function is active.

(b) These buttons, along with additional logic, control a separate Turbine Load status flag that controls MWe tracking when the turbine is in manual.

(c) Turbine Load status flag is visible to the operator on the OAC at Point ID X2060, "TURBINE LOADING STATUS" (True/False).

3) The Turbine Load status flag exists in the ICS to signal to the system that MWe tracking is allowed, and to signal to the bypass valves to add the normal operation 50 psi bias.

(a) The Turbine Loading status is controlled by the operator by the TURBINE LOAD and TURBINE UNLOAD pushbuttons on the LCP.

(b) Certain CTPD demand conditions will also automatically set the state of this flag.

(1) If the turbine is in manual and the Turbine Load status flag is true, the CTP Demand will track MWe.

(2) If the Turbine Load status flag is not true, CTP Demand will be set either by the operator (target load on the LCP) or by one of the other tracking signals.

(3) The Turbine Master can be in automatic OR in manual and the status flag be false.

(4) The Turbine Load status flag is always true above 20% CTPD (unless BOTH Generator breakers are OPEN).

(i) During a load rejection when the Generator breakers open, the status flag immediately turns false to help lessen the THP transient by removing the TBV bias allowing the valves to control THP at setpoint.

(ii) The Turbine Master remains in automatic because this condition generates a runback to 20% CTP.

(iii) However if a Power Load Unbalance were to occur at the same time, the LRM would respond to the PLU condition and the Turbine would trip to manual. NO Tracking would occur because the status flag is false.

(5) The turbine LOAD status flag is always false below 10% CTPD.

This ensures that turbine will not force the reactor to severe subcriticality if tracking MWe at low power.

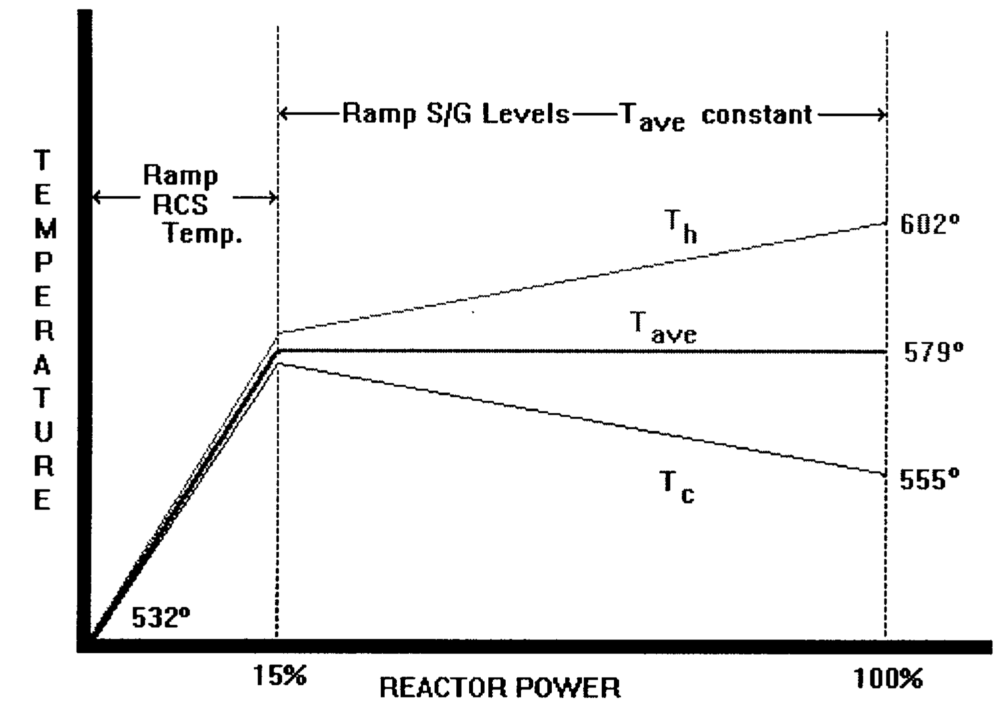
(c) If the Turbine Load status flag is false, the bypass valves and Main Turbine have exactly the same control set point (unless the reactor is tripped).

(d) When the Turbine Load status flag is true, the TBVs have a set point 50 psig greater than the turbine.

4) When the turbine is in automatic and CTPD is greater than 10%, pressing the TURBINE LOAD button will initiate automatic turbine loading.
   (a) This function causes steam flow to be smoothly transferred from the TBVs to the main turbine.
   (b) When both TBV demands are less than 0%, the Turbine Load status flag will be set "true", and the normal 50 psi bias is applied to the bypass valves.
5) When the turbine is in automatic and CTPD is less than 20%, pressing the TURBINE UNLOAD button will initiate automatic turbine unloading.
   (a) This feature causes steam flow to be smoothly transferred from the main turbine to the bypass valves.
   (b) Unloading will stop when electric load is less than 30 MWe.
6) If below 20% CTPD, pressing the TURBINE UNLOAD button will always reset the Turbine Load status flag to false. This action will stop MWe tracking if the turbine is in manual or automatic.
7) The automatic load or unload function can be stopped by pressing the TURBINE LOAD or TURBINE UNLOAD button again, or by placing the turbine in HAND.
8) The load or unload function can be reversed by pressing the opposite button.

<table>
<tr><td><10% CTPD</td><td rowspan="2">FALSE</td><td>ALWAYS</td><td rowspan="2">Turbine cannot take THP control
Don’t Track Mw
TBVs control at Setpoint</td></tr>
<tr><td rowspan="2">10% to 20% CTPD</td><td>UNLOAD Depressed</td></tr>
<tr><td rowspan="2">TRUE</td><td>LOAD and TBVs Zero demand</td><td rowspan="2">Turbine CAN take THP control
Track Mw
TBVs overpressure protection
Setpoint plus 50 psi</td></tr>
<tr><td>>20% CTPD</td><td>ALWAYS
(Except Load Rejection)</td></tr>
</table>

e) Increasing Load



1) The relationship of temperature with load involves OTSG level.
2) The level in the steam generators varies with load from about 15% to 100%.
3) Below 15% the level is maintained constant by the low level limits of the Feedwater Control Subsystem.
4) A commonly used heat transfer equation is used to describe this temperature versus power curve.

$Q = U A \Delta T$

Q = heat produced in the primary system

A = heat transfer area of OTSG

ΔT = Tave - Tsat(RCS Tave minus the saturation temperature of the secondary side of the OTSG)

(a) The heat transfer coefficient, U, is basically a constant and does not change appreciably.

(b) Tsat is held constant by the action of the turbine controls or the turbine bypass valves maintaining THP at setpoint.

(c) Therefore we have three variables to deal with:

(1) Q → the heat produced in the primary which is dependent upon reactor power.

(2) A → heat transfer area in the OTSG which is varied by changing the level on the secondary side of the SG.

(3) Tave → which will vary depending on the combination of the above mentioned variables.

5) As reactor power is increased from 0% to 15%, Q increases.

(a) Since the steam generators are on low level limits, the level in the OTSG is constant, thus A is constant.

(b) Tsat is constant because of the action of the turbine bypass valves.

(c) Therefore using the equation, Q = UA (Tave - Tsat), it is seen that Tave must rise as the primary side heat is increased by raising reactor power for the equation to remain constant and the heat balance to be maintained.

6) From 15% to 100% load, OTSG level will increase, thus increasing the Area.

(a) For this condition, increasing Q while increasing the Area results in a higher load with a constant Tave.

(b) Again, Tsat is held constant, but this time by the turbine.

QUESTION # 6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**1**	**1**
	K/A #	**000055 G 2.4.49**	
	Importance Rating	**4.0**	**4.0**

Technical Reference(s): **AP/1700/11, AP/1700/25, EAP-SSF**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-SSF #26 & #27**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 6

Which ONE of the following meets Oconee **Unit 1** design basis operator time critical actions to activate the SSF during a Station blackout and loss of ALL FDW?

Establish RCP seal flow with the SSF RC Makeup Pump in _____ minutes and fed the OTSG with the ASWP in _____ minutes.

A. 9 / 12

B. 19 / 12

C. 12 / 9

D. 12 / 19

1 POINT

QUESTION # 6

000055 G2.4.49 BOTH GCW 2/21/00 (PRA)

Question setup: RCP seal flow with the SSF RC Makeup Pump be established within 10 minutes on Unit 1 and 20 minutes on unit 2 and 3. If no source of FDW is available to the SGs then SSF Aux service water is required within 14 minutes.

A. Correct, establishing Unit 1 RCP seal flow with the SSF RC Makeup Pump within 10 minutes and SSF Aux service water within 14 minutes is required.

B. Incorrect, RCP seal are not established within 10 minutes required to establish RCP seal flow with the SSF RC Makeup Pump. It would be correct for unit 2 or 3, which require that RCP seal flow be established within 20 minutes.

C. Incorrect, 9 minutes to establish RCP seal flow with the SSF RC Makeup Pump is correct. ASWP should feed the OTSGs with 14 minutes

D. Incorrect, the 12 minutes to establish RCP seal flow with the SSF RC Makeup Pump is not correct. It would be correct for unit 2 or 3, which require that RCP seal flow be established within 20 minutes.

1 POINT

QUESTION # 6

Unit 1 plant conditions:

INITIAL CONDITIONS:

- TD EFDW PUMP (1MS-93) “PULL TO LOCK” (Red Tagged)

CURRENT CONDITIONS:

- Main Feeder Busses 1 and 2 “Locked out”

Which ONE of the following meets Oconee Unit 1 design basis operator time critical actions?

Establish RCP seal flow with the SSF RC Makeup Pump in...

A. 9 minutes and flow to the SGs with the SSF ASW Pump within 12 minutes.

B. 19 minutes and flow to the SGs with the SSF ASW Pump is <u>not</u> required.

C. 9 minutes and RCPs can remain in operation.

D. 19 minutes and trip all RCPs in 2 minutes.

1 POINT

QUESTION # 6

000055 G2.4.49 BOTH GCW 2/21/00 (PRA)

Question setup: MFB lockout will cause a loss of HPI and CC (due to a loss of power) which will require that the SSF to be manned and RCP seal flow with the SSF RC Makeup Pump be established within 10 minutes on Unit 1 and 20 minutes on unit 2 and 3. If no source of FDW is available to the SGs then SSF Aux service water is required within 14 minutes and all RCPs must be secured within 3 minutes. In this case the MD EFDW pumps and the TDE EFDW pump are not available. Credit cannot be taken for cross connecting with another unit because with a 14-minute time limit you do not have time to determine if it is available.

A. Correct, establishing RCP seal flow with the SSF RC Makeup Pump within 10 minutes and SSF Aux service water within 14 minutes is required.

B. Incorrect, 19 minutes is not within 10 minutes required to establish RCP seal flow with the SSF RC Makeup Pump. It would be correct for unit 2 or 3, which require that RCP seal flow be established within 20 minutes.

C. Incorrect, the 9 minutes to establish RCP seal flow with the SSF RC Makeup Pump is correct however RCP must be secured because all FDW is lost to the SGs.

D. Incorrect, the 19 minutes to establish RCP seal flow with the SSF RC Makeup Pump is not correct. It would be correct for unit 2 or 3, which require that RCP seal flow be established within 20 minutes. Securing RCPs in 2 minutes is correct because RCPs must be secured within 3 minutes because all FDW is lost to the SGs.

26. Explain the basis for the following requirements: (R29)

    26.1 The SSF RCMU System must be placed in service on Unit 1 WITHIN ten (10) minutes of a loss of HPI Seal Injection and CC to the RCPs.

    26.2 The SSF RCMU System must be placed in service on Units 2 and 3 WITHIN twenty (20) minutes of a loss of HPI Seal Injection and CC to the RCPs.

    26.3 The SSF ASW System must be placed in service on any unit WITHIN 14 minutes of a loss of all sources of water to the SGs if RCMU flow is required (assumes HPI is lost).

27. Given a set of parameters, determine any required actions for manning / activation of the SSF. (R30)

28. Explain why the RCP's must be secured within three (3) minutes of an event requiring the use of SSF ASW. (R31)

29. Describe the basic steps of powering the SSF from the diesel generator during an emergency. (R32)

30. Given a set of diesel generator parameters, determine the appropriate corrective action for supplying power to the SSF. (R33)

Loss of Power

4. Immediate Manual Actions

_____ 4.1 **IF** CC and HPI Seal Injection are lost to the RCPs,

THEN establish RCP seal flow with the SSF RC Makeup Pump within 10 minutes.

- **REFER TO** AP/0/A/1700/025 (Standby Shutdown Facility Emergency Operating Procedure).

_____ 4.2 **IF** IA Header pressure < 90 psig,

- "Aux Bldg IA Hdr Press"
- "Turb Bldg IA Hdr Press",

THEN direct **Unit 3** to send an Operator to emergency start the Diesel Air compressor per enclosure "Emergency Start Of The Diesel Air Compressor" of AP/3/A/1700/022 (Loss Of Instrument Air).

4. Immediate Manual Actions

4.1 Determine which SSF systems need to be activated:

_____ 4.1.1 **IF** CC and HPI Seal Injection are lost to the RCPs,

THEN establish RCP seal flow with the SSF RC Makeup Pump to:

- _____ Unit 1 within 10 minutes (within 8 minutes if SSF is being manned due to increased RCP seal leakage)
- _____ Unit 2 within 20 minutes
- _____ Unit 3 within 20 minutes.

Change #016A

_____ 4.1.2 **IF** SSF RC ~~Makeup flow is required,~~

AND all sources of water to the SGs are **NOT** available,

THEN perform the following:

- _____ Trip all the Reactor Coolant Pumps on the affected Unit(s) within 3 minutes.
- _____ Establish flow to the SGs with the SSF ASW Pump:
  - _____ Unit 1 within 14 minutes
  - _____ Unit 2 within 14 minutes
  - _____ Unit 3 within 14 minutes.

_____ 4.2 Dispatch the required Operators with a Security Medeco Key, Emergency Ingress Key, and a flashlight to activate the SSF Systems.

SSF EMERGENCY OPERATING PROCEDURE

AP/0/A/1700/25, Standby Shutdown Facility Emergency Operating Procedure, was developed for and provides guidance to the licensed and non-licensed operators for placing the SSF D/G, RCMU, and ASW systems in service for the worst case design scenario, a station blackout. However, based on the initiating event, not all of the systems may be required to be placed in operation.

Guidance is also provided for maintaining the affected unit(s) in MODE 3 with $T_C \approx 555°F$, as well as for replenishing the inventory in the CCW system and SFP.

AP/0/A/1700/25 requires that the SSF RCMU system be placed in operation within 10 minutes (20 minutes on Units 2 & 3) of the initiating event and that the SSF ASW system be placed in service within 14 minutes of the initiating event. Analysis has shown that if the systems are placed in service within this time frame, insufficient RCS inventory will be lost out the PORV to cause formation of steam voids in the hot leg of a magnitude that would interrupt natural circulation during the subsequent cooldown and RCS pressure will remain low enough for the SSF RCMU System to inject water into the RCP seals to prevent seal degradation.

Immediate Manual Actions:

1. If CC and HPI Seal Injection are lost to the RCP's, establish RCP seal flow with the SSF RCMUP on Unit 1 within 10 minutes and on Units 2 & 3 within 20 minutes.
   - If flow is not established within the required time, the RCP seals may be damaged. The concern for maintaining integrity of the RCP seals (with Unit 1 seals being the most critical) is that in approximately ten minutes, HOT RCS water would be at the #1 seal possibly causing seal damage/degradation (on Units 2 and 3, the time would be closer to 20 minutes before seal damage would occur).

2. If SSF RCMUP flow is required and all sources of water to SG's are unavailable;

    2.1 Trip <u>ALL</u> RCP's within 3 minutes.

    - With the RCP's remaining on during a loss of all feedwater event, the primary system heatup and resultant loss of primary inventory out of the PORV will be maximized.
    - Analysis has shown that if the RCP's are not tripped until SSF ASW is established (at 14 minutes), unacceptable hot leg voiding could occur during the subsequent RCS cooldown due to loss of RCS inventory and shrinkage.
    - The RCP's must be secured within 3 minutes of the initiating event to ensure that subcooled natural circulation flow can be established.

    2.2 Establish flow to the SG's with the SSF ASW PUMP within 14 minutes.

    - If flow is not established <u>within 14 minutes</u>, RCS inventory loss (due to high pressure) could create enough voiding to inhibit natural circulation, once the ASW PUMP is started. In addition, RC pressure may be too high for the SSF RCMUP to maintain adequate flow to the RCP seals.

NOTE: It is important to understand that if the decision is made to man the SSF, the instructions should be specific as to the unit and systems that must be placed in operation. Example: Man the SSF and place the Unit 1 RCMUP in service supplying the RCP seals per the SSF AP.

3. Dispatch the required operators with a Security Medico Key (operates the hard lock/deadbolt on the door), Emergency Ingress Key (overrides the card reader/magnetic lock) and a flashlight to activate the SSF per the AP.

    - The SSF key ring (contains the Security Medico Key and the Emergency Ingress Key) is kept in special key lockers mounted in storage / supply rooms in Unit 2 and Unit 3 Control Room areas (the old Unit 2 and Unit 3 Supervisor's offices). These keys provide access to the SSF when Security has locked the doors during a blackout.
    - A dedicated flashlight is available above the special key lockers.
    - One RO/SRO for each affected unit will be required to activate the SSF. If another person is available, he/she should be sent. The intent is to have some one available (for safety reasons) to attend to the RO/SRO and not to be able to activate the SSF. Guidance is given in OMP 1-2 (Rules of Practice) about who should man the SSF in a single or multiple unit event.

QUESTION # 7

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**000062 K3.01**	
Importance Rating	**3.2**	**3.5**

Technical Reference(s): **LPW-SSS**
ARG-1SA-16/E5

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **SSS-LPW #7**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 7

Unit 1 Plant conditions:

INITIAL CONDITIONS:

- LPI operating on normal decay heat removal
- LPSW flow to LPI cooler:
  - "A" = 2690 gpm

CURRENT CONDITIONS:

- The BOP adjusts LPSW-251 ("A" LPI Cooler Outlet Controller) setpoint and establishes LPSW flow to the "A" LPI cooler at **<u>5910 gpm</u>**
- 1SA-16/E-5 (Decay Heat Cooler 1A Flow High) actuates

Which ONE of the following is correct?

1LPSW-251 ("A" LPI Cooler Outlet Controller)...

A. will automatically decrease flow to 5200 gpm.

B. is limited to 50% open to prevent exceeding 6000 gpm.

C. will require manual throttling by the operator to achieve 5200-5900 gpm.

~~83% Limiter~~ ~~must~~

D. WAS ~~is~~ positioned to the "fail-open" position and LPSW-4 (LPI Cooler Outlet) will be manually throttled to achieve < 5200 gpm.

why would

1 POINT

QUESTION # 7

000062 K3.01 Both PRA 2-21-00
(2)

A. Correct – The operator has opened LPSW-251 ("A" LPI Cooler Outlet Controller) (too much) during normal DHR operations and provided excessive flow to the "A" Cooler. The Moore controller interlock will automatically reduce flow to 5200 gpm. Then the operator will reset the controller logic and can manually control the valve.

B. Incorrect – 1LPSW-251 (LPI Cooler Outlet Controller) is normally aligned with the valve position set at 50%. Once actual flow is established in the header the 50% limit is removed. Travel stops for the valve is set for 85% and will remain installed to prevent excessive flow and cooler damage.

C. Incorrect – See "A". This is an automatic interlock function for 1LPSW-251 (LPI Cooler Outlet Controller). Operator action to manually throttle flow below 5200 gpm is required after the controller is reset.

D. Incorrect – The "failed-open" mode for 1LPSW-251 (LPI Cooler Outlet Controller) is not required during normal decay heat removal and would not be operating in this mode. If the valve was failed open it would exceed the 5900 gpm control limit of the Moore controller.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

1. Describe the purpose, normal operation, Engineered Safeguards operation, and abnormal operation of the Low Pressure Service Water (LPSW) system.
2. Given a copy of Improved Technical Specification, Selected License Commitments, and stated conditions, determine Applicability, Conditions, Required Actions, and Completion Times.

ENABLING OBJECTIVES

1. Discuss the purpose of the LPSW system. (R1)
2. Describe the plant loads and systems supported by the LPSW system. (R2)
3. Differentiate between the LPSW System design for Unit 1&2 and Unit 3. (R3)
4. Discuss the suction sources for the LPSW pumps. (R4)
5. Draw the LPSW suction and supply header arrangement. (R5)
6. Discuss all that is monitored by each RIA associated with the LPSW system. (R7)
7. Describe the high flow interlock associated with the LPSW flow through the LPI coolers.(R8)
8. Concerning normal operation of the LPSW system: (R9, R10, R11)
   - 8.1 When given a Limit and Precaution contained in the Low Pressure Service Water system operating procedure,OP/*/A/1104/10,describe the concern addressed.
   - 8.2 Discuss the number of LPSW pumps required to supply loads during normal operation.
   - 8.3 Describe the arrangement of the redundant power supply to the 1 and 2 "B" LPSW pump.
9. Describe the operation of important LPSW system valves as covered in this lesson plan. (R13)

E-5

DECAY HEAT COOLER 1A

FLOW HIGH

1. Alarm Setpoint

1.1 5900 gpm increasing.

2. Automatic Action

2.1 Manual or Auto control reverts to Automatic Override Mode and setpoint becomes 5200 gpm.

3. Manual Action

3.1 Verify that cooler flowrate has been reduced to 5200 gpm and that annunciator alarm has cleared.

3.2 Push "1LPSW-251 HIGH FLOW CLOSURE RESET" to regain control.

3.3 If 1LPSW-251 (1A LPI COOLER OUTLET CONTROL) is in Auto, verify "S" is selected and adjust setpoint to the desired flowrate.

3.4 If 1LPSW-251 (1A LPI COOLER OUTLET CONTROL) is in MANUAL, adjust the controller to the desired flowrate.

3.5 If 1LPSW-251 (1A LPI COOLER OUTLET CONTROL) has failed open:

3.5.1 Verify 1LPSW-251 FAIL SWITCH selected to NORMAL.

3.5.2 Throttle 1LPSW-4 (1A LPI CLR SHELL OUTLET) to reduce cooler flowrate.

3.6 Verify RCS Cooldown rate is within limits of (Unit 1 RCS Heatup/Cooldown Curves) enclosure of OP/0/A/1108/001 (Curves and General Information).

4. Alarm Sources and References

4.1 LPSW Flow Transmitter 1FT-124.

4.2 IP/0/A/0250/001C (LPSW to RCP Motor Coolers, LPI Decay Heat Coolers and RB Component Coolers).

2. Normal System Function
   a) Normal operation requires one LPSW pump per unit. Normal flow is ≈ 10,000 gpm per unit.
   b) Design flow: 15,000 gpm per LPSW pump
3. Selected Valves' Function/operation
   a) 1,2,3 LPSW-251 & 252: LPI Cooler outlet controllers. These are Moore controllers on all three units. Their setpoint is 3000 gpm. This is set by the operator per OP/A/1104/04, LPI procedure, on unit startup. When no flow is present, there is an internal limiter in the Moore controller that limits valve position to 50% open. Once any flow is detected, the 50% limiter is automatically removed.

      On ES actuation, the valve will open and control at setpoint. There is a switch for each valve that will fail the valve open if necessary. These are two position switches - NORMAL and FAIL OPEN. They are positioned by the operator per the LPI procedure.

      LPSW flow to the LPI coolers will automatically runback at ≥ 5900 gpm to a flow rate of 5200 gpm. Automatic runback prevents exceeding the LPI cooler shell design flow of 6000 gpm. A manual Reset button is provided above the Moore controller. Depressing the Reset button will return control of the valve to the operator via the Moore controller.

      Mechanical travel stops have been added to the control valves that limit valve travel to ≈ 85% open. Travel stops ensure that flow will not damage the LPI Cooler.
   b) 1,2,3 LPSW-4 & 5: LPI cooler outlet valves. These valves have been modified to give them throttle capability. If LPSW-251, 252 were to fail open, LPSW-4 and/or 5 may be throttled to prevent robbing flow from the RBCUs. This will also protect the LPSW pumps from pump run-out. If the Moore controller were to fail during LPI decay heat removal, LPSW-4 and/or 5 may be throttled to prevent overcooling the RCS.
   c) 1,2,3 LPSW-7&8 9&10 11&12 13&14: RCP Motor bearing and air cooler inlet and outlet valves. One switch controls both the inlet and outlet valve for that RCP. Computer printout must be checked to verify that both valves have traveled to desired position.
   d) 1,2,3 LPSW-516 & 525: Motor Driven EFWPs cooling water outlet valves. These valves open automatically on pump start. They fail open on loss of instrument air or loss of power.
   e) 1,2,3 LPSW-137: TDEFWP Cooling Jacket Supply. Opens when MS-93 opens. Discharges to trench.

QUESTION # 8

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**000068 A1.03**	
Importance Rating	**4.1**	**4.3**

Technical Reference(s): **CF-EFW**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-EFW #34**

Question Source:
Bank # **B-309**
Modified Bank # ____
New ____

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 __X__

Comments:

1 POINT

QUESTION # 8

Unit 3 plant conditions:

- Unit 3 Control Room has been evacuated due to a fire in the control room
- Conditions permit actions prior to evacuation
- 4160v and 6900v busses have been de-energized
- KHU has re-energized the 4160v loads

Which ONE of the following describes the OTSG level and control method of feedwater?

3A and 3B OTSG level would be controlled at ______ level using _______.

A. 25" S/U Level / 3FDW-35 and 44 (A and B FDW Startup Control)

B. 30" XSUR / 3FDW-315 and 316 (A and B EFDW Control)

C. 50% OR / 3FDW-35 and 44 (A and B FDW Startup Control)

D. 240" XSUR / 3FDW-315 and 316 (A and B EFDW Control)

1 POINT

QUESTION # 8

000068 A1.03 Both PRA 2-7-00
(2)

A. Incorrect – This is a normal post trip FDW operation. The 4160v busses will de-energize the condensate system loads and the Main FDWP's will trip. The EFDW system will automatically actuate and automatically control OTSG levels at 240" XSUR via 315/316 w/o RCP which were lost when the 6900v busses de-energized. Although the control room crew would evacuate to the ASDP the operator would not control FDW with the startup FDW control via the ASDP controls.

B. Incorrect – If RCPs were operating (6900v busses not lost) this would be a correct answer, as EFDW would control OTSG levels at 30" XSUR.

C. Incorrect – If RCP were not operating and Main FDW was in operation this would be a correct answer.

D. Correct - The EFDW system will automatically actuate and automatically control OTSG levels at 240" XSUR via 315/316 w/o RCP which were lost when the 6900v busses de-energized.

29. Describe the MANUAL control available for the TDEFDWP, from the Control Room and locally. (R23)

30. Explain how to bypass the AUTO START feature of the TDEFDWP. (R28)

31. Describe or make a sketch of the logic/conditions that will AUTO START the TDEFDWP when its control switch is in AUTO, including a description of AMSAC. (R25)

32. Describe the affect a Main Steam OTSG Isolation actuation will have on the EFDW system. (R58)

33. Describe the additional action required to allow emergency feed through the alternate ICS flowpath, following actuation of the MS Line Isolation circuit (R59)

34. List the EFDW SG Level setpoints for the conditions when RCPs are running and when all RCPs are off. (R37)

35. Describe the SG level indicators used in the EFDW System, including whether or not they are temperature compensated, and how to select the PRIMARY/BACKUP indicators. (R30)

36. Detail by sketch the status of the solenoids associated with EFDW Level control (Train A & B) when MFDW is operating and normal power to the solenoids is available. (R31)

37. Explain the operation of the solenoids associated with EFDW Level Control (Train A & B) when MFDW is lost and normal power to the solenoids is available from a provided sketch or by making a sketch. (R32)

38. Explain the operation of the solenoids associated with the EFDW Level Control (Train A & B) when MFDW is lost and normal power to the solenoids is lost, from a provided sketch or by making a sketch (R33)

39. List the locations of FDW-315 & 316. (R27)

40. List which Level Train is the PRIMARY TRAIN for SG A level control and which is the PRIMARY TRAIN for SG B level control. (R35)

41. Explain why the Primary Level Train for SG A is not the same as the Primary Level Train for SG B. (R36)

42. Describe how to manually control FDW-315 & 316, after a loss of MFDW, from the Control Room. (R34)

2. Turbine Driven Emergency Feedwater Pump

2.1 Pump

- 8 stage centrifugal Bingham pump
- Design flow - 880 gpm @ 1212 psid when feeding one SG
- Design flow -1080 gpm @ 1125 psid when feeding two SGs
- Shutoff head - 1456 psid @ 3400 RPM
- Pump bearings
- Operates in a oil bath
- Bearing jacket cooled by LPSW (LPSW-137), which discharges to trench; LPSW-137 opens when MS-93 opens. HPSW backs up LPSW.

2.2 Pump Seals

- EFDWP discharge supplies seal injection
- 6 gpm @ 10 psi greater than suction pressure
    - Normal range is 13 - 30 psi > suction pressure

2.3 Suction Flowpath (Figure OC-CF-EF-3)

A. The TDEFDWP normally takes a suction from the UST via C-156 (TDEFDWP Normal Supply). A B/U Suction Line via C-160 (TDEFDWP Suct from UST Riser) is also available.

B. The TDEFDWP's alternate suction is from the Hotwell, through normally closed valve C-391 (TDEFDWP Suction from HW), but ONLY if condenser vacuum is broken.

- This is due to flow restrictions below the expected flow rates that would be necessary.

2.4 Discharge Flowpath (Figure OC-CF-EF-1)

A. The TDEFDWP discharges through FDW-368 (TDEFDWP discharge to "A" SG block) and FDW-369 (TDEFDWP discharge to "B" SG block) to FDW-315 and FDW-316 to the SGs. Per DBD, if either FDW-368 or 369 is not open, the TDEFDWP is inoperable.

B. If FDW-315/FDW-316 fails to control SG levels properly, the TDEFDWP can be aligned to feed the SGs through the Startup Feedwater Control Valves by procedure (Loss of Main Feedwater AP).

1. Flow will be through normally locked closed valves FDW-94 (TDEFDWP discharge to SG A" Normal-Emergency Header) and FDW-96 (TDEFDWP discharge to SG "B" Normal-Emergency Header) to the Startup Feedwater Control Valves to the SGs.

3. Only one train required to trip to actuate isolation

B. System Actuation

1. Any two out of three outlet pressure transmitters on either OTSG reaches 550# (± 50#) decreasing.

2. 2 seconds after circuit initiation prevents TDEFDWP from starting and stops the TDEFDWP if running in automatic.

   a) Placing the control switch to "RUN you can manually start pump.

3. 2 seconds after circuit initiation trips both MFDWPs and closes the following FDW valves:

   a) FDW-31 (A Main FDW Block Valve)

   b) FDW-33 (A S/U FDW Block Valve)

   c) FDW-40 (B Main FDW Block Valve)

   d) FDW-42 (B S/U FDW Block Valve)

4. 5 seconds after actuation, (to prevent water hammer because CVs close faster) closes the following:

   a) FDW-32 (A Main FDW Control Valve)

   b) FDW-41 (B Main FDW Control Valve)

   c) FDW-35 (A S/U FDW Control Valve)

   d) FDW-44 (B S/U FDW Control Valve)

C. The main steam line break circuit must be disabled, to regain control of the startup feedwater valves, then align and feed through the alternate flowpath.

4. Level Control System

4.1 Description

A. Auto selected: Level Control System signal is passed through to the valve.

- 30" XSUR – any RCP running
- 240" XSUR - loss of all RCPs (natural circulation)

Note: These levels may have to be adjusted manually by the reactor operator for degraded containment conditions following actuation to: [60" acc or 270" acc] as required.

B. Manual selected: Manual control signal is passed through to the valve.

C. Powered from KVIB and KVIC

D. Placing FDW-315 and/or FDW-316 in AUTO will cause the respective valve(s) to go on auto level control.

Exam Question Report

27-Jan-99

Question ID:	PART-B309	Revision No:	0	Revision Date	10/29/1999
Question Description:	PART-B309				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	Reference: AP/1700/08	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

A fire in the unit's control room has forced an evacuation of the area. Prior to evacuating, you noticed that when the unit tripped, a 4160V and 6900V electrical failure occurred, but the Keowee Units re-energized the 4160 Main Feeder Buses. Determine the proper steam generator level and method of control for this situation: (.25)

A) 25 inches SU level using SU control valves (FDW-35 and 44).

B) 240 inches XSUR level using SU control valves (FDW-35 and 44).

C) 25 inches SU level using EFDW control valves (FDW-315 and 316).

D) 240 inches XSUR level using EFDW control valves (FDW-315 and 316).

Answer

D

Lessons

ID	Description

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

QUESTION # 9

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**000068 G2.4.16**	
Importance Rating	**3.0**	**4.0**

Technical Reference(s): **AP/1700/08 CASE A**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-ARG OBJ. #3 & #4**

Question Source:
- Bank # ______
- Modified Bank # ______
- New ___X___

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 __X__

Comments:

1 POINT

QUESTION # 9

Unit 3 plant conditions:

INITIAL CONDITIONS:

- MODE 1, Power level = 100%

CURRENT CONDITIONS:

- A fire in the Unit 3 Cable Room has been reported and extinguished with NO significant damage to plant equipment or controls
- AP/3/A/1700/008, Loss of Control Room, Case A, Conditions Permit Action Prior to Evacuation has been implemented due to heavy smoke in the Control Room
- All immediate actions of AP/3/A/1700/008, Loss of Control Room, Case A have been performed
- The plant is being maintained in MODE 3 from the Auxiliary Shutdown Panel

Which ONE of the following conditions will require plant control to be shifted to the SSF?

A. RCS temperature cannot be maintained ~~above~~ ≥ 525°F.

B. Turbine Header Pressure is being maintained at 1010 psig.

C. RCS temperature cannot be maintained above the minimum Mode 2 temperature limit.

D. Condenser vacuum at 7 inches with TBV selector station in the "HAND" position.

1 POINT

QUESTION # 9

000068G2.4.16 (4.0/3.0) Both – PRA 4-6-00 (1)

A. Correct - AP/1/A/1700/008, section 5.3 requires the operator to shift control to the SSF if RCS T-ave cannot be maintained greater than or equal to 525°F.

B. Incorrect – If steam pressure is maintained above 1010 psig this is well above the 525°F required by AP/1/A/1700/008, section 5.3.

C. Incorrect - the minimum temperature for MODE 2 operations is 527°F and is above the 525°F required by AP/1/A/1700/008, section 5.3.

D. Incorrect - the 7-inch condenser vacuum interlock is defeated with the TBV's in HAND and has no effect on the location of plant control operations.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

At the conclusion of both this training and the student's self-study of the Abnormal Procedures, the student will be able to demonstrate a working knowledge of the Abnormal Procedures (AP).

ENABLING OBJECTIVES

1. State the purpose of the Abnormal Procedures. (R1)

2. Describe all Abnormal Procedures listed in OMP 2-1, for referral from memory, include the conditions that require this referral (R2)

3. Briefly describe how to properly use an Abnormal Procedure including: (R3)
   - 3.1 Describe the sequence the procedure is formatted to.
   - 3.2 Describe the correct use of logical statements including, IF/THEN, AND/OR, IF AT ANY TIME, and WHEN statements.
   - 3.3 Describe the proper use of Place Keeping Aids.
   - 3.4 Describe the purpose for NOTES and CAUTION statements and how they are used.

4. Given a copy of the AP, perform the following: (R4)
   - 4.1 Walk-through and discuss each procedure step.
   - 4.2 Locate all instrumentation and controls referred to in the AP including those devices outside the Control Room that would require manual operations should any automatic action fail.
   - 4.3 Briefly summarize the actions to be taken in the Subsequent Actions section of the AP.

1. INTRODUCTION

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 250
4	Hot Shutdown[b]	< 0.99	NA	$250 > T > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

Table 2: Saturated Steam: Pressure Table

Abs Press. Lb/Sq In. p	Temp Fahr t	Specific Volume Sat. Liquid v_f	Specific Volume Evap v_{fg}	Specific Volume Sat. Vapor v_g	Enthalpy Sat. Liquid h_f	Enthalpy Evap h_{fg}	Enthalpy Sat. Vapor h_g	Entropy Sat. Liquid s_f	Entropy Evap s_{fg}	Entropy Sat. Vapor s_g	Abs Press. Lb/Sq In. p
0.08865	32.018	0.016022	3302.4	3302.4	0.0003	1075.5	1075.5	0.0000	2.1872	2.1872	0.08865
0.25	59.323	0.016032	1235.5	1235.5	27.382	1060.1	1087.4	0.0542	2.0425	2.0967	0.25
0.50	79.586	0.016071	641.5	641.5	47.623	1048.6	1096.3	0.0925	1.9446	2.0370	0.50
1.0	101.74	0.016136	333.59	333.60	69.73	1036.1	1105.8	0.1326	1.8455	1.9781	1.0
5.0	162.24	0.016407	73.515	73.532	130.20	1000.9	1131.1	0.2349	1.6094	1.8443	5.0
10.0	193.21	0.016592	38.404	38.420	161.26	982.1	1143.3	0.2836	1.5043	1.7879	10.0
14.696	212.00	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	14.696
15.0	213.03	0.016726	26.274	26.290	181.21	969.7	1150.9	0.3137	1.4415	1.7552	15.0
20.0	227.96	0.016834	20.070	20.087	196.27	960.1	1156.3	0.3358	1.3962	1.7320	20.0
30.0	250.34	0.017009	13.7266	13.7436	218.9	945.2	1164.1	0.3682	1.3313	1.6995	30.0
40.0	267.25	0.017151	10.4794	10.4965	236.1	933.6	1169.8	0.3921	1.2844	1.6765	40.0
50.0	281.02	0.017274	8.4967	8.5140	250.2	923.9	1174.1	0.4112	1.2474	1.6586	50.0
60.0	292.71	0.017383	7.1562	7.1736	262.2	915.4	1177.6	0.4273	1.2167	1.6440	60.0
70.0	302.93	0.017482	6.1875	6.2050	272.7	907.8	1180.6	0.4411	1.1905	1.6316	70.0
80.0	312.04	0.017573	5.4536	5.4711	282.1	900.9	1183.1	0.4534	1.1675	1.6208	80.0
90.0	320.28	0.017659	4.8779	4.8953	290.7	894.6	1185.3	0.4643	1.1470	1.6113	90.0
100.0	327.82	0.017740	4.4133	4.4310	298.5	888.6	1187.2	0.4743	1.1284	1.6027	100.0
110.0	334.79	0.01782	4.0306	4.0484	305.8	883.1	1188.9	0.4834	1.1115	1.5950	110.0
120.0	341.27	0.01789	3.7097	3.7275	312.6	877.8	1190.4	0.4919	1.0960	1.5879	120.0
130.0	347.33	0.01796	3.4364	3.4544	319.0	872.8	1191.7	0.4998	1.0815	1.5813	130.0
140.0	353.04	0.01803	3.2010	3.2190	325.0	868.0	1193.0	0.5071	1.0681	1.5752	140.0
150.0	358.43	0.01809	2.9958	3.0139	330.6	863.4	1194.1	0.5141	1.0554	1.5695	150.0
160.0	363.55	0.01815	2.8155	2.8336	336.1	859.0	1195.1	0.5206	1.0435	1.5641	160.0
170.0	368.42	0.01821	2.6556	2.6738	341.2	854.8	1196.0	0.5269	1.0322	1.5591	170.0
180.0	373.08	0.01827	2.5129	2.5312	346.2	850.7	1196.9	0.5328	1.0215	1.5543	180.0
190.0	377.53	0.01833	2.3847	2.4030	350.9	846.7	1197.6	0.5384	1.0113	1.5498	190.0
200.0	381.80	0.01839	2.2689	2.2873	355.5	842.8	1198.3	0.5438	1.0016	1.5454	200.0
210.0	385.91	0.01844	2.16373	2.18217	359.9	839.1	1199.0	0.5490	0.9923	1.5413	210.0
220.0	389.88	0.01850	2.06779	2.08629	364.2	835.4	1199.6	0.5540	0.9834	1.5374	220.0
230.0	393.70	0.01855	1.97991	1.99846	368.3	831.8	1200.1	0.5588	0.9748	1.5336	230.0
240.0	397.39	0.01860	1.89909	1.91769	372.3	828.4	1200.6	0.5634	0.9665	1.5299	240.0
250.0	400.97	0.01865	1.82452	1.84317	376.1	825.0	1201.1	0.5679	0.9585	1.5264	250.0
260.0	404.44	0.01870	1.75548	1.77418	379.9	821.6	1201.5	0.5722	0.9508	1.5230	260.0
270.0	407.80	0.01875	1.69137	1.71013	383.6	818.3	1201.9	0.5764	0.9433	1.5197	270.0
280.0	411.07	0.01880	1.63169	1.65049	387.1	815.1	1202.3	0.5805	0.9361	1.5166	280.0
290.0	414.25	0.01885	1.57597	1.59482	390.6	812.0	1202.6	0.5844	0.9291	1.5135	290.0
300.0	417.35	0.01889	1.52384	1.54274	394.0	808.9	1202.9	0.5882	0.9223	1.5105	300.0
350.0	431.73	0.01912	1.30642	1.32554	409.8	794.2	1204.0	0.6059	0.8909	1.4968	350.0
400.0	444.60	0.01934	1.14162	1.16095	424.2	780.4	1204.6	0.6217	0.8630	1.4847	400.0
450.0	456.28	0.01954	1.01224	1.03179	437.3	767.5	1204.8	0.6360	0.8378	1.4738	450.0
500.0	467.01	0.01975	0.90787	0.92762	449.5	755.1	1204.7	0.6490	0.8148	1.4639	500.0
550.0	476.94	0.01994	0.82183	0.84177	460.9	743.3	1204.3	0.6611	0.7936	1.4547	550.0
600.0	486.20	0.02013	0.74962	0.76975	471.7	732.0	1203.7	0.6723	0.7738	1.4461	600.0
650.0	494.89	0.02032	0.68811	0.70843	481.9	720.9	1202.8	0.6828	0.7552	1.4381	650.0
700.0	503.08	0.02050	0.63505	0.65556	491.6	710.2	1201.8	0.6928	0.7377	1.4304	700.0
750.0	510.84	0.02069	0.58880	0.60949	500.9	699.8	1200.7	0.7022	0.7210	1.4232	750.0
800.0	518.21	0.02087	0.54809	0.56896	509.8	689.6	1199.4	0.7111	0.7051	1.4163	800.0
850.0	525.24	0.02105	0.51197	0.53302	518.4	679.5	1198.0	0.7197	0.6899	1.4096	850.0
900.0	531.95	0.02123	0.47968	0.50091	526.7	669.7	1196.4	0.7279	0.6753	1.4032	900.0
950.0	538.39	0.02141	0.45064	0.47205	534.7	660.0	1194.7	0.7358	0.6612	1.3970	950.0
1000.0	544.58	0.02159	0.42436	0.44596	542.6	650.4	1192.9	0.7434	0.6476	1.3910	1000.0
1050.0	550.53	0.02177	0.40047	0.42224	550.1	640.9	1191.0	0.7507	0.6344	1.3851	1050.0
1100.0	556.28	0.02195	0.37863	0.40058	557.5	631.5	1189.1	0.7578	0.6216	1.3794	1100.0
1150.0	561.82	0.02214	0.35859	0.38073	564.8	622.2	1187.0	0.7647	0.6091	1.3738	1150.0
1200.0	567.19	0.02232	0.34013	0.36245	571.9	613.0	1184.8	0.7714	0.5969	1.3683	1200.0
1250.0	572.38	0.02250	0.32306	0.34556	578.8	603.8	1182.6	0.7780	0.5850	1.3630	1250.0
1300.0	577.42	0.02269	0.30722	0.32991	585.6	594.6	1180.2	0.7843	0.5733	1.3577	1300.0
1350.0	582.32	0.02288	0.29250	0.31537	592.3	585.4	1177.8	0.7906	0.5620	1.3525	1350.0
1400.0	587.07	0.02307	0.27871	0.30178	598.8	576.5	1175.3	0.7966	0.5507	1.3474	1400.0
1450.0	591.70	0.02327	0.26584	0.28911	605.3	567.4	1172.8	0.8026	0.5397	1.3423	1450.0
1500.0	596.20	0.02346	0.25372	0.27719	611.7	558.4	1170.1	0.8085	0.5288	1.3373	1500.0
1550.0	600.59	0.02366	0.24235	0.26601	618.0	549.4	1167.4	0.8142	0.5182	1.3324	1550.0
1600.0	604.87	0.02387	0.23159	0.25545	624.2	540.3	1164.5	0.8199	0.5076	1.3274	1600.0
1650.0	609.05	0.02407	0.22143	0.24551	630.4	531.3	1161.6	0.8254	0.4971	1.3225	1650.0
1700.0	613.13	0.02428	0.21178	0.23607	636.5	522.2	1158.6	0.8309	0.4867	1.3176	1700.0
1750.0	617.12	0.02450	0.20263	0.22713	642.5	513.1	1155.6	0.8363	0.4765	1.3128	1750.0
1800.0	621.02	0.02472	0.19390	0.21861	648.5	503.8	1152.3	0.8417	0.4662	1.3079	1800.0
1850.0	624.83	0.02495	0.18558	0.21052	654.5	494.6	1149.0	0.8470	0.4561	1.3030	1850.0
1900.0	628.56	0.02517	0.17761	0.20278	660.4	485.2	1145.6	0.8522	0.4459	1.2981	1900.0
1950.0	632.22	0.02541	0.16999	0.19540	666.3	475.8	1142.0	0.8574	0.4358	1.2931	1950.0
2000.0	635.80	0.02565	0.16266	0.18831	672.1	466.2	1138.3	0.8625	0.4256	1.2881	2000.0
2100.0	642.76	0.02615	0.14885	0.17501	683.8	446.7	1130.5	0.8727	0.4053	1.2780	2100.0
2200.0	649.45	0.02669	0.13603	0.16272	695.5	426.7	1122.2	0.8828	0.3848	1.2676	2200.0
2300.0	655.89	0.02727	0.12406	0.15133	707.2	406.0	1113.2	0.8929	0.3640	1.2569	2300.0
2400.0	662.11	0.02790	0.11287	0.14076	719.0	384.8	1103.7	0.9031	0.3430	1.2460	2400.0
2500.0	668.11	0.02859	0.10209	0.13068	731.7	361.6	1093.3	0.9139	0.3206	1.2345	2500.0
2600.0	673.91	0.02938	0.09172	0.12110	744.5	337.6	1082.0	0.9247	0.2977	1.2225	2600.0
2700.0	679.53	0.03029	0.08165	0.11194	757.3	312.3	1069.7	0.9356	0.2741	1.2097	2700.0
2800.0	684.96	0.03134	0.07171	0.10305	770.7	285.1	1055.8	0.9468	0.2491	1.1958	2800.0
2900.0	690.22	0.03262	0.06158	0.09420	785.1	254.7	1039.8	0.9588	0.2215	1.1803	2900.0
3000.0	695.33	0.03428	0.05073	0.08500	801.8	218.4	1020.3	0.9728	0.1891	1.1619	3000.0
3100.0	700.28	0.03681	0.03771	0.07452	824.0	169.3	993.3	0.9914	0.1460	1.1373	3100.0
3200.0	705.08	0.04472	0.01191	0.05663	875.5	56.1	931.6	1.0351	0.0482	1.0832	3200.0
3208.2*	705.47	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	3208.2*

*Critical pressure

Case A

Conditions Permit Action Prior To Evacuation

5. Subsequent Actions

CAUTION 5.1:	• With the TBV Loop A and B selector stations in the "HAND" position, the 7 inch condenser vacuum interlock is defeated. • The associated TBVs will immediately go to the position demanded from the Aux Shutdown Panel station when the TBV selector station is placed in "HAND". To ensure the TBVs do not change position, check the Measured Variable and adjust the "HAND" knob accordingly to zero out any error prior to placing the TBV Loop A and B selector stations to the "HAND" position.

_____ 5.1 Maintain Mode 3 with the average RCS temperature ≥ 525°F:

- Turbine header pressure ≈ 1025 psig
- Steam Generator levels at ≈ 25" SU Level
- Reactor Coolant pressure between 1800-2200 psig
- PZR level at ≈ 220 inches.

_____ 5.2 Maintain LDST level between 40-100 inches by manually cycling 1HP-24 (1A HPI BWST SUCTION) as required.

Location: (AB-1 N end of HPI hatch area)

Loss Of Control Room

Case A

Conditions Permit Action Prior To Evacuation

_____ 5.3 **IF** the unit **CANNOT** be maintained in Mode 3 with the average RCS temperature ≥ 525°F from the Auxiliary Shutdown Panel,

THEN proceed to the SSF and maintain those conditions unless the initiating event causes the unit to be driven to a lower temperature:

- **REFER TO** AP/0/A/1700/025 (Standby Shutdown Facility Emergency Operating Procedure).

END

QUESTION # 10

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**000074 A1.13**	
Importance Rating	**4.3**	**4.8**

Technical Reference(s): **EAP-E25**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E25 OBJ. #1**

Question Source:
Bank # ______
Modified Bank # **EAP 200 & 199**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 10

Unit 2 plant conditions:

- ICCM/RVLIS:
  - RCS pressure 1650 psig
  - CETCs = 704 and steady
  - Core Subcooling Margin = (-)95°F
  - Loop Subcooling Margin = (+)10°F

Which ONE of the following is correct?

A. The core is completely uncovered

B. The core is at least partially uncovered.

C. The core has been re-flooded by HPI, LPI, and CFTs.

D. The Hot legs have returned to subcooled state and adequate core cooling is imminent.

1 POINT

QUESTION # 10

000074 A 1.13 Both PRA 2-14-00
(1)

A. Incorrect – although the core is superheated CETCs are stable. This would indicate that some water level exist in the core.

B. Correct – superheated conditions is an accurate indication that at least part of the core is uncovered.

C. Incorrect – If ECCS is operating properly the core will reflood but at this time with superheated conditions in the core (not the Loops) the core has not yet reflooded. Superheated condition in the core is an accurate indication that at least part of the core is uncovered.

D. Incorrect - The hot legs indicate subcooled but this does not indicate that core cooling is occurring or will occur.

OBJECTIVES

TERMINAL OBJECTIVE:

1. Describe the use of Section 505 (Inadequate Core Cooling) of the Emergency Operating Procedure in order to perform the required actions of a Nuclear Control Operator in restoring adequate core cooling during an ICC event.

ENABLING OBJECTIVES:

1. Recognize that for CETCs to indicate superheated the core must have become at least partially uncovered. (R1)

2. State when the ICC section of the EOP should be initiated. (R2)

3. Explain why RCPs should not be stopped whenever ICC conditions exist, if they are still running.(R3)

4. Explain why SG(s) pressure must be lowered below RCS pressure in order to help mitigate ICC conditions. (R4)

5. Explain why RCS pressure should be lowered using the PORV and RCS high point and head vents, if an ECCS injection source is available. (R5)

6. Discuss the basis for the verification of RB Auxiliary Fan operation in the ICC section.(R6)

7. Describe two possible benefits from starting a RCP if CETCs indicate > 700°F or if CETCs indicate increasing superheat. (R7)

1. INTRODUCTION

1.1 The ultimate goal of the operator during any abnormal event is to ensure that the core remains covered and cooled. Failure to do so can result in extensive core damage as the fuel and cladding temperatures increase rapidly. A zirc-water chemical reaction will also occur at high cladding temperatures and generate large quantities of hydrogen gas that can impede efforts to restore heat transfer or create containment integrity concerns.

1.2 Superheated coolant conditions, as indicated by the CETCs, means that the core has become at least partially uncovered and that inadequate core cooling exists. Core temperatures will continue to increase if the situation is not corrected. Section 505, Inadequate Core Cooling (ICC), must be implemented if CETCs indicate superheated.

1.3 The objective of Section 505, Inadequate Core Cooling, is to use all available means to restore saturated or subcooled core conditions as quickly as possible.

1.4 Once saturated or subcooled core cooling conditions are restored, a recovery from ICC mitigation is performed and a transfer to the appropriate cooldown procedure is made.

2. PRESENTATION

The major EOP step numbers are in parenthesis at the end of the lesson plan steps.

2.1 If at any time (IAAT) CETCs > 1200°F and the TSC is operational, instruct the TSC to enter OSAG (Oconee Severe Accident Guidelines). (1)

Do NOT operate any plant components unless directed by the TSC.

CETCs > 1200°F is a primary indication of core uncovery. High clad temperatures (≈ 1800°F) lead to rapid clad oxidation and severe core damage. High CETC temperatures indicate that plant conditions are outside the bounds of the EOP (due to multiple failures). Mitigation of the event is transferred to OSAG at this point. The EOP is no longer used once the transition to OSAG has been made.

CAUTION:	RCPs are to remain in operation unless directed otherwise by this procedure.

If any RCPs are still in operation when the ICC section is entered they must be left in operation. With superheated conditions present, core cooling can be provided by pumping steam through the core with RCPs. Analyses have shown that forced steam cooling will provide adequate core cooling if a level exists in either SG. If RCPs are tripped, all forced steam cooling will be lost and the resulting "natural circulation" steam mode of cooling is not an acceptable method.

2.2 Initiate Full HPI. (2)

Superheated core conditions can only occur if there is not enough ECCS flow to offset the coolant, which is being lost from the system. It is essential to initiate or verify maximum ECCS flow from any and all sources, one of which is full HPI flow (all available HPIPs/both headers).

Ensure adequate HPI header flow per curve provided.

2.3 Ensure Initiation of ES Channels 3 and 4. (3)

Generally, for ICC events, the LPI system will have been automatically initiated by low RCS pressure and/or high Reactor Building pressure ES channel actuation signals. If not automatically initiated, Channels 3&4 should be manually initiated and, as with HPI flow above, maximum flow should be verified to assure optimum operation (2 pumps/2 trains) while throttling flow to preclude LPIPs runout.

Exam Question Report

27-Jan-99

Question ID:	EAP199	Revision No:	0	Revision Date	10/29/1999
Question Description:	EAP199				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-E27 - Inadequate Core Cooling
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 1; SRO = 1	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Which ONE of the following is CORRECT if core subcooling monitors indicate superheated? (.25)

A) The core can still be covered with saturated liquid.

B) The core must be at least partially uncovered.

C) The core must be completely uncovered.

D) The core exit thermocouples are > 700°F.

Answer

B

Lessons

ID	Description
EAP-E27	Inadequate Core Cooling (EAP-E27)

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

QNUM	13
HNUM	New
QCHANGE	
ACHOICE	
BCHOICE	
CCHOICE	
DCHOICE	
ANSCHANGE	
DAREA	
EXAM TYPE	NRC
QDATE	9/16/98
FAC	269 Oconee 1, 2, & 3
RTYP	B&W 177
EXLEVEL	R
AUTHOR	Sonalysts, Inc.
REFKEY	
KA1	017K5.03
KA1RO	3.7
KA1SRO	4.1
KA2	
KA2RO	
KA2SRO	

QVALUE 1.0

QUESTION 3 R13

Unit 1 Plant Conditions:

- Reactor is tripped.
- LOCA is in progress.
- CETCs are 725°F increasing.
- ECCS operating in a degraded mode.

Which ONE of the following is the core condition?

A. The core is covered with saturated liquid.

B. The core is partially covered with a steam blanket.

C. The fuel clad is exposed to fuel melt temperatures.

D. The fuel clad is exposed to the boric acid-stainless steel (SS) reaction.

R13

ANSWER	B
COGNITIVE	Comprehension
REFSPECIFIC	
MODULE	Lesson Plan E27, (Item 2.11), pg. 9
OBJECTIVE	ELO-1

ABASIS Incorrect. The core is not fully covered and >700°F indicates superheated conditions. To indicate superheat the core must be partially uncovered.

BBASIS Correct answer. CETCs >700°F indicates inadequate core cooling due to partially uncovered core or inability to reflood the core.

CBASIS Incorrect. This temperature is below the fuel melt temperature.

DBASIS Incorrect. The boric acid-SS reaction occurs at higher core temperatures.

QUESTION # 11

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**BW/E03 K3.1**	
Importance Rating	**3.2**	**3.8**

Technical Reference(s): **SAE-L21**

Proposed references to be provided to applicants during examination: **N/A**

Learning Objective: **SAE-L21 OBJ. #3**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 11

Unit 1 plant conditions:

- Core SCM = 0°F
- PZR level = 0 inches
- “1A” OTSG pressure = 430 psig and decreasing
- “1A” RC Loop Tc = 452°F and decreasing
- 1RIA-40 ALERT alarm
- RCS pressure = 1550 psig and decreasing
- Reactor Building pressure = 1.7 psig and increasing

Which ONE of the following transients is in progress?

A. Large tube rupture on the 1A OTSG.

B. Small break LOCA on the 1A1 Tc leg.

C. Excessive heat transfer on the 1A OTSG.

D. Inadequate heat transfer on the 1A OTSG.

1 POINT

QUESTION # 11

BW/E03 K3.1 Both (PRA) 2-3-00 (GTH)
(2)

A. Incorrect – during a SGTR RB pressure would not be increasing. RIA-40 in ALERT alarm does indicate that a SGTL in the "A" OTSG may be occurring but it is not a large tube rupture as RIA -16 or 17 is not in alarm. If RB pressure was not increasing then A may be correct.

B. Incorrect – during a LOCA RCS pressure decreases and RCS temperature remains high. RB pressure will increase during a SBLOCA and a steam leak inside containment can also increases RB pressure.

C. Correct - RB pressure is increasing due to a steam leak inside containment. RCS pressure and temperature decreasing together is an indication of an overcooling event or excessive heat transfer.

D. Incorrect – OTSG pressure would be decreasing in an inadequate heat transfer situation but RCS temperature would not decreasing due to loss of OTSG heat transfer.

LESSON SPECIFIC OBJECTIVES

Terminal Objective

1. The Crew should perform the actions required to diagnose a MS Line Rupture inside containment, isolate the SG with the MS Line Rupture and stabilize the Unit once the SG is isolated. During the evolution, plant status and operator actions will be monitored by the Oversight SRO. (T1)

Enabling Objectives

1. Utilize control board indication and plant response to diagnose a MS Line Rupture in the "1A" MS Line, inside containment. (R1)

2. Demonstrate the ability to perform the correct actions to mitigate the MS Line Rupture. (R2/R3/R3a/R7/R10)

    2.1 Manually trip the Reactor if an automatic trip has not already taken place. (R2)

    2.2 Perform IMAs (R3)

    A. The OATC should first perform from memory and then verify with the SRO.

    2.3 Perform a symptoms check: (R3)

    A. If the RCS is saturated, run rule #2. (R7)

    B. If the RCS is subcooled, run rule #6. (R10)

    2.4 Stabilize RCS CETCs when the overcooling is isolated. (R3a)

3. Compare the Unit response to the expected response for a LBLOCA and verify the diagnosis of a MS Line Rupture and not a LOCA. (R5)

4. When rule #6 has been completed, run Rule #7. (R6)

    4.1 Restore RCP support systems.

    4.2 Provide manual control of HPI parameters to allow throttling of HPI when SCM is $\geq 5°$.

5. Use the Turbine Bypass Valves on the intact steam generator to maintain CETCs constant. (R9)

INSTRUCTOR NOTE: Make sure that the Crew understands that the MSLB Isolation Circuit does nothing to the MD EFDWPs or FDW-315&316. It is still the responsibility of the Operator to secure EFDW flow to a faulted generator within three minutes.

b) The SG Operating Range level will immediately increase to 100% and then begin to return to normal because of the instantaneous change in pressure within the SG. The level detector sees the ΔP change as a high level. The MFDWP trip circuit will see this increase in level and trips the MFDWPs when SG level indicates >98% on the Operating Range. Even if the MSLB circuit does not work, the MFDWPs will trip automatically.

c) The high Reactor Building High Pressure trip (3.5 psig) is a backup to the Low RCS Pressure trip (1810 psig) for high energy line breaks.

3. The affected steam generator is quickly identified after turbine trip due to separation of the MS headers. MS pressure will continue to decrease on the "A" side, and will begin to recover on the "B" side.

B. Instruct the students to monitor the plant for the symptoms just discussed and activate Timer #1 to cause a Main Steam Line break inside containment on the "1A" SG.

INSTRUCTOR NOTE: During the blowdown of the affected SG, water that is in the MFDW and EFDW lines up to the first isolation valve will be sucked into the SG (becomes low pressure area as steam is blown out). On the simulator this phenomenon is accomplished by the addition of mass to the SG inventory. This addition occurs in three distinct dumps of water into the SG. Because of this the blowdown, depending upon break size, appears to stop and linger at approximately 600, 400 and 200 psig. If the students ask about the blowdown pauses, this is the reason.

C. *Freeze* the simulator once the A MS Header Pressure has decreased to around 100 psi and walk through the following:

1. Discuss with the students that the event is different from a LOCA in the following ways:

a) RCS pressure and temperature decrease not just pressure like a LOCA.

b) RB pressure and temperature increase but no RIA alarms actuate in the RB that indicate a LOCA.

2. Relate the Pressure/Temperature (P/T) display to a Steam Line Rupture.

   a) RCS Pressure and Temperature both decreasing rapidly. On the PTID, the "A" side will show a larger ΔT between Th and Tc than the "B" side. The "B" side ΔT should come together in a normal Post Trip pattern.

   b) Saturation will be reached unless the operator takes action to stop the overcooling event. RCS pressure will decrease rapidly to saturation. RCS Pressure and Temperature will then decrease, following the saturation curve.

3. The ESG (1-8) actuation was due to RB high pressure caused by the MS line break in containment; the Operators may be required to perform the following.

   a) Run Rule 7 as a result of the symptom check performed by the BOP.

      1) If any RCPs are operating then RCP support systems will be reestablished.

         (a) Unisolate LPSW (open LPSW-6 & 15) to RCPs.

         (b) Unisolate CC (open CC-7 & 8) to RCPs.

D. Place the simulator in *RUN* and allow the event to progress.

1. With no operator action to throttle HPI flow (as soon as a core subcooling margin of 5°F is re-established) and to maintain Tave constant, the RCS will refill and repressurize. PTS conditions could develop requiring operating in the TSOR (1 hour hold). If RCPs are secured and Tc decreases to <500°F and HPI has operated in the injection mode; operation in the TSOR is required. If RCPs are running, and temperature is allowed to decrease 100°F at >50°F/1/2 hour then entry into the TSOR will still be required.

2. The Crew must understand the criteria for securing all RCPs.

   a) Subcooling margin = 0°F and reactor power ≤1%, trip all RCPs.

   b) If this occurs, the Crew will run rule #2.

INSTRUCTOR NOTE: **During this scenario with all HPI available and MSLB isolation circuitry performing as designed, SCM will not be lost. Pressurizer level will go off scale low, but due to level still existing in the surge line the RCS does not reach saturation.**

QUESTION # 12

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**1**	**1**
K/A #	**000076 G4.12**	
Importance Rating	**3.4**	**3.9**

Technical Reference(s): **AP/1700/21 p.#1 / encl. 6.1**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **ADM-SRG 8**

Question Source:
Bank # **1998 NRC EXAM**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam **X (1998 NOT SIGNIFICANTLY MODIFIED)**
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 _____
55.43 _____

Comments:

1 POINT

QUESTION # 12

Unit 1 plant conditions:

- AP/1/A/1700/21, High Activity in the RC System is in progress
- Dose Equivalent Iodine (DEI) = 1.2μCi/gm
- Failed fuel calculations = .55%
- "A" SGTL = .0022 gpd
- Unit shutdown in progress at 2.9%/hour

Which ONE of the following is correct?

AP/1/A/1700/21, High Activity in the RC System, direct the operator to reduce power <u>slowly</u> instead of immediately tripping the reactor because a reactor trip would cause...

A. a spike on RIA-40 and an inaccurate OAC calculation for SGTL rate.

B. an inaccurate sample of iodine concentration due to increasing fission products.

C. DEI activity to be masked by an increase in radioactive particulate due to a crud burst.

D. a decrease in RB temperature thus causing an increase in RB iodine absorption in the concrete.

1 POINT

QUESTION # 12

000076 G2.4.12 PRA Both 2-21-00 (GTH)

A. Incorrect – The SGTL rate will be compensated for by RCS total activity and can be backed up via inventory balance. With failed fuel events RIA-40 is amplified and indicate higher.

B. Correct – Iodine activity will rapidly spike due to the loss of flux/burnout term therefore cause inaccurate Iodine indications.

C. Incorrect – DEI (fission products) is actually <u>not</u> masked by a crud burst. Chemistry can determine iodine from other particles with the RCS.

D. Incorrect – RB temperature is decreased following a reactor trip and the AP does require RB temperatures to remain > 100°F to prevent Iodine absorption into the RB concrete.

1 POINT

QUESTION # 13

Unit 3 plant conditions:

- Pressurizer Level #2 is selected for Pressurizer level control
- I&E has completed repairs to Pressurizer Level #3 transmitter
- Unit 3 SASS panel indications:
  - ➢ Pressurizer Level - Green Auto light OFF
  - ➢ Pressurizer Level - Red Trip "B" light ON

Which ONE of the following is correct operator action to return the Pressurizer Level "SASS channel" to AUTOMATIC operation?

Operate the _____ and the controlling signal will be from Pressurizer Level #__.

A. Test toggle / "2"

B. RESET button / "2"

C. Test toggle / "3"

D. RESET button / "3"

1 POINT

QUESTION # 13

BW/A03 K2.2 (3.3/3.3)
(2)

A. Incorrect - The Test toggle does not reset the channel. Test is used for I&E functions only. If Lvl 2 were tested the PZR level would fail low.

B. Correct – The Reset button is used to place the Level #3 in service but the signal will remain selected to #2 because it has been selected on the control board UB1.

C. Incorrect - The Test toggle does not reset the channel. Test is used for an I&E function only. The selected level on control board UB1 (#2) remains the controlling signal.

D. Incorrect – The Reset button is used but the controlling signal will remain on PZR lvl #2 and will not swap to PZR lvl #3.

4. When given the applicable data be able to make correct parameter computations. (R4)

5. When given a set of plant conditions and/or reactor operator actions be able to predict plant/system/component response, or the effect on the same or other systems or components. (R5)

6. Demonstrate an understanding of the guidance or rules in procedures by locating the answer to specific RO related questions. (R6)

7. Be able to recite, from memory, required procedural or administrative items detailed in Operations Management Procedure 2-1 (OMP 2-1, Encl. 4.9). (R7)

8. For APs, OMPs, SDs, Tech Specs, SLCs, and the EOP, become familiar with the content of each so as to be able to answer, from memory, questions relating to general systems alignments, available operator controls and instrumentation, bases for specific actions, and in the case of the EOP, the order of priority assigned for mitigating simultaneous casualties. (R8)

. SR(13) DB ✓
NRC(2) 115 ✓
✓Sim(3)

Duke Power Company Trans. (1) ID No AP/1/A/1700/021

PROCEDURE PROCESS RECORD

Revision No 5

LAN Location: SAROS

PREPARATION

(2) Station OCONEE NUCLEAR STATION

(3) Procedure Title High Activity In RC System

(4) Prepared By Eric Lamps Date 3/15/99

(5) Requires 10CFR50.59 evaluation?

☒ Yes (New procedure or revision with major changes)
☐ No (Revision with minor changes)
☐ No (To incorporate previously approved changes)

(6) Reviewed By John L. Collins Jr. (QR) Date 3-16-99

Cross-Disciplinary Review By JE Sanders (QR)NA ____ Date 3/16/99 Rx Eng

Reactivity Mgmt. Review By John L. Collins Jr. (QR)NA ____ Date 3-16-99

(7) Additional Reviews

Reviewed By Doug Berkshire Date 3-18-99 RP

Reviewed By Dean Cantrell Date 3-17-99 Chem

(8) Temporary Approval *(if necessary)*

By ____________ (SRO/QR) Date ______

By ____________ (QR) Date ______

(9) Approved By [signature] Date 3/18/99

PERFORMANCE *(Compare with control copy every 14 calendar days while work is being performed.)*

(10) Compared with Control Copy ____________ Date ______

Compared with Control Copy ____________ Date ______

Compared with Control Copy ____________ Date ______

(11) Date(s) Performed ____________

Work Order Number (WO#) ____________

COMPLETION

(12) Procedure Completion Verification

☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
☐ Yes ☐ NA Listed enclosures attached?
☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
☐ Yes ☐ NA Charts, graphs, etc. attached, dated, identified, and marked?
☐ Yes ☐ NA Procedure requirements met?

Verified By ____________ Date ______

(13) Procedure Completion Approved ____________ Date ______

(14) Remarks *(Attach additional pages, if necessary)*

MAR 1999 Received Operator Training

Duke Power Company Oconee Nuclear Station **High Activity In RC System**	Procedure No. AP/**1**/A/1700/021
	Revision No. 005
Continuous Use **Reactivity Management Related**	Electronic Reference No. OX002RGY

High Activity In RC System

Reactivity Management Related

TABLE OF CONTENTS

OCONEE NUCLEAR STATION

High Activity In RC System

1. Purpose

This procedure provides the actions necessary to maintain the plant in a safe condition following the discovery of high activity in the Reactor Coolant System.

2. Symptoms

Chemistry analysis indicates high activity in the Reactor Coolant System.

3. Automatic Systems Actions

None.

4. Immediate Manual Actions

None

High Activity In RC System

5. Subsequent Actions

NOTE:	The units of "µCi/ml" may be used interchangeably with the units of "µCi/gm" when comparing Chemistry sample results with limits found in Improved Technical Specifications and this procedure. {1}

_____ 5.1 Evaluate the Reactor Coolant System activity against ITS 3.4.11.

_____ 5.2 **IF** the evaluation determines that RCS activity exceeds the following ITS 3.4.11 limit:

- Dose Equivalent Iodine (DEI) > 1.0 µCi/gm

THEN Notify the OSM to **REFER TO** RP/0/B/1000/001, (Emergency Classification).

_____ 5.3 **IF** Dose Equivalent Iodine (DEI) $\geq$ 0.25 µCi/gm,

THEN perform the following:

_____ Notify Operations Work Coordination Support Duty Person.

_____ Notify the System Operating Center (SOC) that Unit 1 is **NOT** available for load following.

_____ **REFER TO** Enclosure 6.1, "Operating With Failed Fuel".

_____ 5.4 **IF** DEI < 0.25µCi/ml,

THEN **NO** further action is required.

END

NOTE:	The rate of reactor power changes should be limited to ≤ 3% FP/hr. The rate of reactor power decrease may exceed 3% FP/hr if a power decrease is required by ITS, or any situation deemed necessary by the Operations Shift Manager.

_____ 1. RCS letdown flow should be increased as much as possible to meet the following limits:

- RCS DEI concentration ≤ 1.0 μCi/gm
- Gross Specific Activity ≤ 100/E μCi/gm
- RB Iodine concentration < 20 DAC
- I-131 concentration in Hotwell < 1.0 x 10^{-6} μCi/ml (only a concern if a SG tube leak exists). {2}

_____ 1.1 **IF** increased letdown flow is unable to maintain the RCS activity within the above limits,

THEN perform the following:

- Evaluate the need to decrease Reactor Power.
- **REFER TO** ITS 3.4.11.

NOTE 2: Steady State Power History means that the reactor power level has been steady ($\pm$5% FP) for at least 3 days prior to the time when the Chemistry sample was taken.

_____ 2. **IF** at Steady State Power History,

AND Dose Equivalent Iodine (DEI) $\geq$ 0.5 µCi/gm,

THEN perform the following:

_____ 2.1 Notify Operations Work Coordination Support Duty Person,

_____ 2.2 Reduce power by 10% FP.

- **REFER TO** OP/1/A/1102/004, (Operation At Power).

_____ 2.3 Evaluate further actions with the Duty Reactor Engineer.

_____ 3. **IF** a Reactor trip occurs,

THEN perform the following:

_____ 3.1 Notify Chemistry to pull a RCS sample between 2 and 6 hrs after the Reactor trip occurs to determine Dose Equivalent Iodine (DEI) concentration. {3}

_____ 3.2 Increase letdown flow as much as possible to minimize the magnitude of the associated iodine spike.

_____ 4. **IF** a Reactor Power change of >15% FP occurs in a one hour time period,

THEN Notify Chemistry to pull a RCS sample between 2 and 6 hours after the Reactor Power change occurs and determine Dose Equivalent Iodine (DEI) concentration. {3}

NOTE:	Maintaining Reactor Building air temperatures > 100°F will help keep iodine in the gaseous form and enable removal with the Reactor Building Purge System. This will prevent the iodine from being absorbed in the concrete of the Reactor Building.

_____ 5. **IF** Unit is shutdown,

AND RCS cooldown is to be performed,

THEN maintain all Reactor Building air temperatures > 100°F until Reactor Building iodine is < 1.0 DAC.

_____ 6. **IF** operating with a SG Tube Leak,

AND the calculated Turbine Building Sump Environmental Concentrations (EC) is > 1.0,

THEN increase letdown flow as much as possible.

_____ 7. **IF** operating with a SG Tube Leak,

AND calculated leak size > 72 gpd,

THEN perform the following:

- Reduce reactor power by 10% FP at ≤ 3% FP/hr.
- Recalculate the SG Tube Leak size:

_____ 7.1 **IF** the SG Tube Leakage increases,

THEN evaluate possible shutdown.

_____ 7.2 **IF** the SG Tube Leakage decreases,

THEN stabilize Reactor Power and evaluate continued operation.

NOTE 8:	Hotwell liquid sample should be taken on a daily basis. Once DEI values have stabilized, the frequency of the sample may be relaxed by the ONS Chemistry Manager (or his designee).

_____ 8. **IF** a SG Tube Leak exists,

THEN notify Chemistry to sample the Hotwell liquid for I-131.

NOTE 9: To minimize exposure, the ONS Radiation Protection Manager (or his designee) may relax the frequency of the radiological surveys once daily RCS DEI concentration values stabilize.

_____ 9. Notify Shift Radiation Protection personnel to establish a daily sampling/surveying program for the following:

- Top of LDST (remote probe)
- HPI pump room (general area)
- Seal Supply filter room (general area)
- Seal Return filters and coolers
- Reactor Building Iodine.

_____ 9.1 **IF** a SG Tube Leak exists,

THEN perform the following:

_____ 9.1.1 Notify Shift Radiation Protection personnel to establish a daily sampling/surveying program for the Powdex cells (dose rate survey).

_____ 9.1.2 Notify Chemistry personnel to establish a daily sampling/surveying program for the following:

- TBS (dilution flow rate determination)
- Chemical Treatment Pond (activity release monitoring).

_____ 10. Record the following RIA counts daily in the Unit Log:

- 1RIA-40 (CSAE)
- 1RIA-47 (RB Particulate)
- 1RIA-48 (RB Iodine)
- 1RIA-49 (RB Gas)

END.

High Activity In RC System
Appendix

1. The 10CFR50.59 evaluation performed for Chemistry Sample Manual 3.10 Rev 15 states that the units of μCi/ml may be used interchangeably with μCi/gm.
2. This value comes from 10CFR20 Appendix "B" Table 2 Column 2 (Dose to the Public).
3. This step is performed to meet the requirement of ITS SR 3.4.11.2.

END

QNUM	37
HNUM	
QCHANGE	NEW
ACHOICE	NEW
BCHOICE	NEW
CCHOICE	NEW
DCHOICE	NEW
ANSCHANGE	NEW
DAREA	
EXAM TYPE	NRC
QDATE	9/18/98
FAC	269 Oconee 1, 2, & 3
RTYP	B&W 177
EXLEVEL	B
AUTHOR	Sonalysts, Inc.
REFKEY	
KA1	000076K3.06
KA1RO	3.2
KA1SRO	3.8
KA2	
KA2RO	
KA2SRO	

QVALUE 1.0

QUESTION 49 B37

Unit 1 Plant Conditions:

- AP/1/A/1700/21, High Activity in RC System is being implemented.
- Reactor Engineering has calculated that 0.75% of the fuel has failed.
- The reactor is being shutdown by reducing power at 3% FP/hour.
- A small SG 1A tube leak exists.

AP/1700/21 directs the operator to reduce power slowly instead of immediately tripping the reactor because a reactor trip would cause __________.

A. an inaccurate calculation of the SG tube leak rate.

B. the iodine activity in the reactor coolant to rapidly increase during fission product buildup.

C. Dose Equivalent Iodine (DEI) activity to be masked by crud burst radioactive particulates.

D. a required delay of more than 8 hours before initiating sampling activities for RCS Dose Equivalent Iodine (DEI) activity.

B37

ANSWER	B
COGNITIVE	Comprehension
REFSPECIFIC	AP 1700/21, Step 4.2
MODULE	
OBJECTIVE	ADM-SRG - 8

ABASIS	Incorrect. The tube leak rate calculation should be compensated for RCS activity and backed up by an inventory balance. With failed fuel in the RCS and a SGTL the RIA indications is elevated to a higher then normal indication as activity of the RCS is elevated.
BBASIS	Correct answer. The iodine activity will rapidly spike to the loss of neutron flux burnout of iodine in the fuel.
CBASIS	Incorrect. DEI (fission products) would not be masked by a crud burst. Chemistry can determine iodine from other radioactive constituents that are present in the reactor coolant.
DBASIS	Incorrect. Chemistry is required to take samples between 2 and 6 hours after a trip or a plant shutdown.

QUESTION # 13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**1**	**1**
	K/A #	**BW/A03 K2.2**	
	Importance Rating	**3.3**	**3.3**

Technical Reference(s): **IC-RCI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-RCI #3**

Question Source:
- Bank # **IC-70**
- Modified Bank # ______
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test **X**

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis **X**

10 CFR Part 55 Content:
- 55.41 **X**
- 55.43 ______

Comments:

1 POINT

QUESTION # 13

Unit 3 plant conditions:

- I&E has completed repairs to Pressurizer Level #3 transmitter
- Pressurizer Level #2 is selected for Pressurizer level control
- Unit 3 SASS panel indications:
    - ➢ Pressurizer Level - Green Auto light OFF
    - ➢ Pressurizer Level - Red Trip "B" light ON

Which ONE of the following is correct operator action to return the Pressurizer Level "SASS channel" to AUTOMATIC operation? (.25)

Operate the _____ and the controlling signal will be from Pressurizer Level #__.

A. Test ~~toggle~~ switch / "2"

B. RESET button / "2"

C. Test ~~toggle~~ switch / "3"

D. RESET button / "3"

1 POINT

QUESTION # 13

BW/A03 K2.2 (3.3/3.3)
(2)

A. Incorrect - The Test toggle does not reset the channel. Test is used for I&E functions only. If Lvl 2 were tested the PZR level would fail low.

B. Correct – The Reset button is used to place the Level #3 in service but the signal will remain selected to #2 because it has been selected on the control board UB1.

C. Incorrect - The Test toggle does not reset the channel. Test is used for an I&E function only. The selected level on control board UB1 (#2) remains the controlling signal.

D. Incorrect – The Reset button is used but the controlling signal will remain on PZR lvl #2 and will not swap to PZR lvl #3.

TRAINING OBJECTIVES

TERMINAL OBJECTIVES:

1. Describe how RCS temperature, pressure, level and flow measurement signals are generated, how these signals are processed, and how the indications and control functions are applied for unit operation. Be able to analyze various RCS indications and determine how changes in RCS temperature, pressure, level and flow measurement signals will affect plant control systems and operator indications. (T1, 2)

2. Describe the operation of the Reactor Vessel Level Indicating System (RVLIS) and Inadequate Core Cooling Monitor System (ICCM); how the signals used are generated, the various displays available to the operator, analyze when the indications are valid or invalid, and how to operate the system during all modes of unit operation. (T3, 4)

ENABLING OBJECTIVES

1. During all modes of operation, describe the basic operation and failure modes for the detectors used to generate control room indications for RCS temperature, pressure, level, and flow, including the following: (R1)

   1.1 Instrument location in the RCS and range of the instrument

2. During all modes of operation, explain how the parameter controlling signal is derived via the ICS median select circuitry: (R61)

   2.1 Given a set of parameters analyze how failures of input signals to median select will affect plant operation.

3. During all operating modes when SASS is required explain the operation of SASS (Smart Automatic Signal Selector) system including the following: (R22)

   3.1 List the signals that are monitored by SASS. (R23)

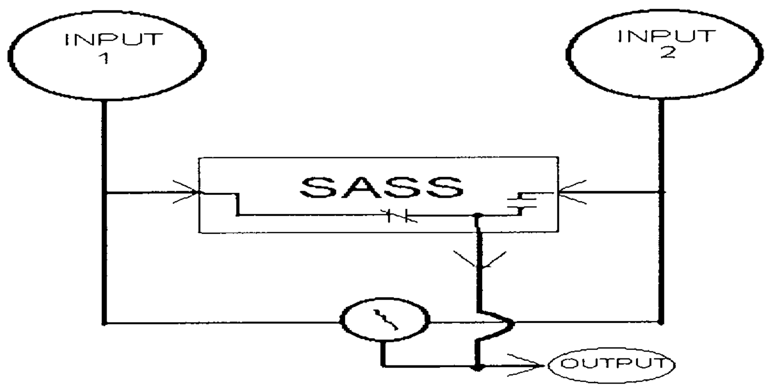
   3.2 Differentiate SASS operation and response for an AUTO trip and a MISMATCH condition. (R24, 25, 26, 27, 28)

   3.3 Given a set of conditions with SASS in AUTO OR MANUAL, explain the operator indications and actions that are necessary to swap/reset the controlling signals. (R29, 30)

   3.4 During all modes of operation describe how the operator monitors proper SASS operation from the control room. (R31)

B. SASS

SMART AUTOMATIC SIGNAL SELECTOR (SASS), provides protection for selected plant parameters against instrument failure by detecting the failure then automatically selecting an operable alternate instrument. If the operator has the operable instrument selected, SASS will not select the failed instrument. The OAC provides signal mismatch alarms that will alert the operator to controlling signal problems.



1. Signals monitored by SASS
   a) OTSG A Operating Range Level channel 1 and 2
      1) Key selectable (UB1)
   b) OTSG B Operating Range Level channel 1 and 2
      1) Key selectable (UB1)
   c) Pressurizer Level 1 or 2* and 3
      1) Push-button controlled (VB1)

* If the operator has PZR level #1 or 2 selected then SASS input will be from that selected level channel and the second SASS input from level #3. If the operator has level #3 selected then the second SASS input **defaults** to level #1. Level #3 is always the second SASS input.

2. SASS Panel location
   a) ICS cabinet #8 (The location for SASS controls are different on the simulator than the actual SASS panels in the control room).
3. SASS Operations:
   a) When in AUTO, SASS monitors the two input process signals and when there is a **RAPID FAILURE** of one of the signals it (SASS) will automatically select the "good" signal (the one not changing); independent of the operator selector switch position.
      - At the SASS panel, the red **TRIP** light for the signal that has failed will be on.
      - The green **AUTOMATIC** light will be off.
      - The amber **MISMATCH** light will be on.

- The OAC Alarm Video will display the **MISMATCH** message for the affected signal.
- The channel will stay in **MANUAL** until the signal is good and the channel is manually reset.
- When in **AUTO**, as long as one of the signals has not failed, the normal select switch can be used to change signals.

b) When in AUTO, if a **mismatch** between the signals occurs slowly, SASS will revert to **MANUAL**. (NO controlling signal swap)

- At the SASS panel, the green **AUTOMATIC** light will be off.
- The amber **MISMATCH** light will be on.
- The Alarm Video will display the **MISMATCH** message for the affected signal.
- When the **MISMATCH** clears, the channel will automatically reset and return to automatic operation.

c) When in **MANUAL**, if a failure of a signal occurs, SASS will not **AUTO TRIP**. The operator must manually switch the controlling signal from the selector in the Control Room.

d) SASS must be manually reset following a **AUTO TRIP** occurs and the instrument is repaired.

   1) Reset button is located next to the **AUTOMATIC** light on the SASS panel. The reset button is recessed in the cabinet. To reset, the button must be depressed. When reset the **AUTO** light should come on, the **TRIP** and **MISMATCH** light should go off.

e) On a loss of SASS power:

   1) For the channels in ICS Cabinet #8, whichever signal is selected (key or pushbutton) by the appropriate switch is fed thru.

f) SASS is powered from KI power panelboard.

g) The **TEST** switch simulates a rapid rate of change between the inputs and a >3% mismatch.

   1) The switch can be moved in either direction.

   2) The direction it is moved to is the signal that it is simulating the failure.

   3) Before testing a channel with a failed signal ensure that the appropriate select switch is positioned to the "good" signal.

      (a) In this condition, if the "good" signal is tested, SASS will deenergize the relays and the signal that is passed will be determined by the select switch, as long as the test switch is held in either "test" position.

      (b) When the test switch is released, SASS will again feed the good control signal.

h) SASS operation is directed by PT/600/01, Periodic Instrument Surveillance

Exam Question Report

27-Jan-99

Question ID:	IC070	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC070				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-RCI - RCS Instrumentation
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 13; LRO = 30; SRO = 30 Reference: IC-RCI P28, OBJECTIVE 30; PT/600/01	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

I&E has completed repairs to Pressurizer Level #3 transmitter. Pressurizer Level #2 is selected for Pressurizer level control. The following conditions exist at the Unit 3 SASS panel:

- Pressurizer Level - Auto light OFF.
- Pressurizer Level - Red Trip "B" light ON.

Which ONE of the following is correct for returning the Pressurizer Level "SASS channel" to AUTOMATIC operation? (.25)

Operate the________ and the controlling signal will be from Pressurizer Level #__.

A) Test toggle / "2"

B) RESET button / "2"

C) Test toggle / "3"

D) RESET button / "3"

Answer

B

A. Incorrect, If the Test toggle does not reset the channel test is used for I&E functions only. If Lvl 2 was tested the PZR level would fail low.

B. Correct, Reset is used to place the Lvl 3 in service but the signal will remain selected to #2.

C. Incorrect, same as A

D. Incorrect, Reset is used but the controlling signal will remain on Lvl #2 and will not swap to #3.

Lessons

ID	Description
IC-RCI	Reactor Coolant System Instrumentation

QUESTION # 14

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**2**	**2**
K/A #	**BW/E02 K3.1**	
Importance Rating	**3.2**	**3.8**

Technical Reference(s): **PNS-HPI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-HPI #5**

Question Source:
- Bank # ____
- Modified Bank # ____
- New __X__

Question History:
- Previous NRC Exam ____
- Previous Quiz / Test ____

Question Cognitive Level:
- Memory or Fundamental Knowledge __**X**__
- Comprehension or Analysis ____

10 CFR Part 55 Content:
- 55.41 __**X**__
- 55.43 ____

Comments:

1 POINT

QUESTION # 14

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Power = 100%

CURRENT CONDITIONS:

- Reactor trip
- Green “OFF” lights are illuminated on both CC pumps

Which ONE of the following is correct?

was designed to

1HP-5 (Letdown Isolation) will automatically close as letdown temperature increases to ______ °F which ~~will~~ prevent a release of ____________ from the purification demineralizer into the RCS.

A. 130 / Boron

B. 135 / Boron

C. 130 / Sulfur

D. 135 / Sulfur

Both correct

1 POINT

QUESTION # 14

BW/E02 K 3.1 (GCW) NEW
(1)
Question setup:
CC supplies cooling water to the letdown coolers. With both CC pumps OFF letdown temperature will increase. When Letdown temperature reaches 130°F a statalarm is received. When it reaches 135°F, 1HP-5 closes to isolate letdown. This is to protect the demineralizer resin. Temperatures greater than 135°F breaks down the resin resulting in a release of various collected ions and sulfur into the RCS, that can increase the corrosion process in the RCS.

A. Incorrect, 130°F is the alarm setpoint and although Boron is released as letdown temperature increases it is not the reason for the interlock.

B. Incorrect, 135°F is correct the second part is not. Although Boron is released as letdown temperature increases it is not the reason for the interlock.

C. Incorrect, 130°F is the alarm setpoint to warn operators of increasing LD temperature. The second part is correct as Sulfur will be released.

D. Correct, 135°F is the correct setpoint for 1HP-5 interlock and Sulfur is released when water going through the demins is elevated.

Question did not ask the reason why? only what will be released.

E. emergency injection

5. Explain the purpose for each HPI system interlock and when given plant conditions: (R5, R8, R11, R12, R39, R16, R20, R22)

   - Predict system/component/indicationresponse to HPI system interlock actuation.

   - Describe necessary actions and/or plant status required to return system/component/indicationto normal operating status.

6. Determine when HP-42 must be opened. (R6)

7. State what action may have to be taken when increasing letdown flow significantly > 70 gpm. (R7)

8. Describe manual operation of HP-5 and requirement to ensure proper response following manual operation. (R9)

9. Predict plant response (control rod, etc.) to placing a demineralizer in service which has not been boron saturated and explain the concern. (R10)

10. State the purpose of maintaining hydrogen overpressure on the LDST. (R13)

11. List the four possible sources of highly borated water that can be used for normal makeup. (R14)

12. List the two possible sources of low borated water that can be used for normal makeup. (R15)

13. Briefly explain why the Seal Return Coolers are designed for 250 gpm flow rate while actual seal return flow is much lower. (R17)

14. Summarize the effects of not maintaining 3 gpm warming line flow on the "A" loop injection nozzles. (R18)

15. Explain why the use of the "B" HPI injection nozzles with RCS temp. > 250°F must be documented in the Unit log. (R19)

16. Summarize the effects of restarting an HPIP following a total loss of HPI flow, prior to closing HP-31. (R21)

17. State two purposes for aligning pressurizer auxiliary spray during a normal shutdown. (R24)

145 psig at the letdown relief. Another variable is the deborating demineralizer. They have shown to have as much as a 30 psig ΔP. So 70+30+30= 130 psig. Using this operating experience thumb rule, the operator should be able to calculate allowable letdown and makeup flows to prevent lifting any letdown or demineralizer relief.

- The computer alarm setpoint for High Pressure in the Letdown line has recently been changed on all 3 units to alarm at 130 psi vs. the previous setpoint of 145 psi. This was done to give the operator time to respond to the alarm and take action to prevent lifting the letdown line relief valves.

c) *During low pressure conditions, such as unit S/U or S/D, it may be necessary to use the manual bypass (HP-42) located in the seal supply filter room, if HP-7 is unable to pass the required flow. Key is required for lock. Do not exceed 120 psig at local gauge.*

d) Normal letdown flow is approximately 70 gpm. If increased significantly above this value, an additional CC pump may be placed in service. Procedures have been changes to run both CC pumps during heat-up. When Unit is at normal operating temperature and pressure the heat removal capabilities are checked and the second CC Pump stopped.

1) When removing a L/D cooler from service, adjust HP-7 to maintain L/D cooler CC outlet temperature <225°F

2) events have resulted in flashing of the CC system as a result of inadequate flow balancing of CC, at a letdown flow of 88 gpm. Procedure revisions to enhance the flow balancing instructions should alleviate the problem.

3) Letdown flow should be limited to 120 gpm with 1 letdown cooler in service

e) *Interlock:*

1) *Letdown temperature is monitored downstream of the block orifice and its bypass. If the letdown temperature reaches 130°F a high temperature stat-alarm will sound and at 135°F the letdown isolation valve, HP-5, will be interlocked closed to protect the demineralizer resin.*

2) *To clear the high temperature once the problem has been corrected or if letdown is required, the demineralizer should be bypassed and isolated. The high temperature interlock can be bypassed by a switch on UB1 so that HP-5 can be opened. Refer to enclosure in OP/1,2,3/A/1104/02, HPI System.*

3) *Both the purification and deborating demineralizer resin beds have a 135°F temp. limit for protection of the resin. Temp. > 135°F breaks down the resin resulting in a release of various collected ions and sulfur, which can increase the corrosion process in the RCS.*

4) Letdown temperature must also be limited to prevent over-pressurization of the HPIPs. Accordingly, the Limits and Precautions section of OP/1,2,3/A/1104/02 includes a limit of 130°F for the LDST.

5) **Changing letdown temperature affects reactivity management due to the temperature effect on IX resin. Increasing letdown temperature increases RCS Boron; and decreasing letdown temperature decreases RCS Boron.**

f) Letdown flow is isolated by low RCS pressure (1600 psi) or high Reactor Building pressure (3 psi) actuation of ES Channels 1 and/or 2.

1) Channel 1 Closes HP-3 and HP-4

2) Channel 2 Closes HP-5

g) *HP-5 is an air operated valve with a spring to fail closed on loss of air. The valve has a manual handwheel, which may be used to open the valve.* ***Handwheel must be fully clockwise to allow normal pneumatic operation.*** AP/*/A/1700/014 directs the operator to manually open HP-5 if failed closed.

1) Continuous communications should be established with the Control Room if the valve is manually opened.

h) HP-6 operator has been replaced with the air operated spring to fail closed operator also.

4. *Demineralizers* (IX)

a) Purification Demineralizers

1) Each unit has a purification demineralizer and a spare purification demineralizer available. Units 1 and 2 share a spare purification demineralizer.

2) Each purification IX consists of a 50 ft^3 mixed resin bed which purifies the RCS by ion exchange and is not intended to be used as a filter.

QUESTION # 15

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**2**	**2**
	K/A #	**BW/A01 K1.3**	
	Importance Rating	**3.7**	**3.7**

Technical Reference(s): **PNS-PZR**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PZR OBJ. #7**

Question Source:
Bank # **PNS-653**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 15

Unit 1 plant conditions:

- A Feedwater transient has caused a runback to 65% power
- RCS pressure peaked at 2241 psig and returned to the following values:
  - RPS Channel A NR Pressure indicates 2154 psig
  - RPS Channel B NR Pressure indicates 2157 psig
  - RPS Channel E NR Pressure indicates 2158 psig
- Pressurizer temperature is 644°F

Which ONE of the following correctly completes the below statement?

1RC-1(PZR Spray) should be ______and PZR heater bank #2 should be ________.

A. open / off

B. open / on

C. closed / off

D. closed / on

1 POINT

QUESTION # 15

BW/A01 K1.3 (3.7/3.7) Both T1-G2/T1-G2 #55 RSI/PRA 5-2-00
(2)

A. Incorrect - PZR heater bank #2 should be energized. See answer b. explanation below.

B. Correct - The median selected controlling RC pressure is 2157 psig, which is above the RC-1 closing setpoint. The saturation pressure for 644°F is 2118 psia or 2103 psig, which is less than the median, selected controlling RC pressure of 2157 psig. Therefore, the PZR saturation recovery circuit would have the bank #2 heaters energized.

C. Incorrect - RC-1 should be open and heater bank #2 should be energized. See b. above.

D. Incorrect - RC-1 should be open. The second portion is correct, heater bank #2 should be on.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

Upon completion of this lesson, the student will demonstrate an understanding of the components, indications, controls and operation of the Pressurizer. The student will be able to assess the status of the Pressurizer during normal, abnormal and emergency conditions and determine corrective actions for improper system operation. The student will also be able to apply any ITS/SLC Conditions and Required Actions associated with the Pressurizer.(T1)

ENABLING OBJECTIVES

1. Explain the design basis of the pressurizer. (R21)
2. Describe pressurizer response during load or RCS temperature changes. (R1)(R2)(R3)
3. Given a set of conditions, calculate the change in pressurizer level for a change in RCS temperature. (R33)
4. Explain what is meant by a "subcooled" pressurizer and how to determine if the pressurizer is in a subcooled condition.(R22)(R27)
5. Explain what is meant by a pressurizer "hard bubble" and describe the adverse effects of a "hard bubble" on plant operation, (R23)
6. Identify the source of pressurizer spray for each unit. (R4)
7. Discuss the automatic setpoints and any interlocks associated with pressurizer instrumentation. (R5)
8. Explain the operation of the ICS RC pressure signal median select function as it relates to RC pressure control including: (R28)
    8.1 How median select chooses the controlling signal
    8.2 Which pressurizer components receive a median selected RC pressure signal.
9. Given a set of conditions, determine which RC pressure signal has been selected for control by the ICS RC pressure signal median select function. (R36)
10. Discuss the reasons for bypass flow around the pressurizer spray valve during normal operation. (R6)
11. Evaluate plant response to a failed open pressurizer spray valve without operator action. (R20)
12. Explain the operation of the Pressurizer Water Space Saturation Recovery Circuit. (R29)

E. Since all sources of heat in the system, i.e., core, pressurizer heaters, and reactor coolant pumps, are interconnected by the reactor coolant piping with no intervening isolation valves, system pressure relief protection is provided on the pressurizer. Overpressure protection consists of two code safety valves and one electromatic relief valve

2.4 Component Description

(Instructor Note: The non-nuclear instruments (NNI) inputs that are used in the control of pressurizer level, temperature and pressure are also inputs to the digital Integrated Control System (ICS). NNI inputs to pressurizer level, temperature and pressure control may be modified by ICS. As such, pressurizer level, temperature and pressure control can be considered to be a sub-function of ICS. Operation of NNI is addressed in the Reactor Coolant Instrumentation (RCI) lesson plan and operation of the ICS is addressed in the Integrated Control System lesson plan.)

A. Pressurizer Spray (PNS-PZR-1, 3, 4 & 15)

1. The pressurizer spray line originates at the discharge of the A1 RCP for Unit 1, and B1 RCP on Units 2 & 3.
2. Spray flow is caused by the difference in pressure between RCP discharge and vessel outlet due to head losses as the coolant flows through the vessel.
3. Pressurizer spray flow is controlled by a DC solenoid operated valve, RC-1, which responds to a manual open/close signal from the operator or automatically from the opening and closing pressure set points.
   a) RC-1 opens at 2205 psig increasing pressure and closes at 2155 psig decreasing pressure.
   b) RC-1 is controlled by the ICS median selected narrow range (NR) RCS pressure signal.
      1) The inputs to the ICS RC pressure signal median select function are:
         (a) RC pressure #1 on RCS loop A (input to RPS chan. A)
         (b) RC pressure #2 on RCS loop B (input to RPS chan. E)
         (c) RC pressure #3 on RCS loop A (input to RPS chan. B)
      2) Median select refers to the mathematical technique of selecting the middle of three signals as an output.
      3) This process adds a degree of redundancy and reliability to the system. For example, if RC pressure #1 was the controlling signal and it failed high or low, it would no longer be the median or controlling signal.

8. The pressurizer spray provided by the reactor coolant pump accomplishes reduction of pressure during a Reactor Coolant System cooldown.

9. Just prior to securing the last RCP, auxiliary spray is lined up to the pressurizer. This alignment supplies a flowpath from the discharge of the HPIPs, through a portion of LPI piping to the pressurizer spray line, to provide for further RCS pressure reduction.

   a) Auxiliary spray flow is controlled via HP-355, a pneumatic control valve located in the east penetration room.

   b) A controller for HP-355 is located on UB1 in the control room.

10. Below a system temperature of approximately 250° F, the Low Pressure Injection System is used for system heat removal and the steam generators and reactor coolant pumps are removed from service

B. Pressurizer Heaters (OP-PNS-PZR-7, 8, 9, 10 & 15)

1. The pressurizer heaters:

   a) replace heat lost during normal steady state operation

   b) raise the pressure to normal operating pressure during Reactor Coolant System heatup from the cooled down condition

   c) restore system pressure following transients.

2. Eleven groups of electric heaters, divided into four banks, are assembled into three removable horizontal heater assemblies. The ICS median selected narrow range (NR) RCS pressure signal controls these four banks of heaters.

3. Based on B&W calculations and startup testing experience, a conservative value for the total heat loss from the pressurizer, under normal hot standby conditions, is 107 kilowatts.

   a) By design, the first bank (126 kW) of heaters utilizes proportional control and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at the set point.

   b) Spay valve leakage and bypass flow effect actual heating requirements.

   Instructors Note: *Engineering is in the process of determining exactly what the heat loss is with the Pressurizer lagging installed. When the initial calculation was performed the Pressurizer was not fully lagged. It has been determined that 70kW will allow natural circulation in MODE 3 based on SSF calculations.*

   c) Auto/On/Off control is used for the remaining three banks.

d) The same analysis determined that this minimum of 126 kilowatts, which corresponds to the smallest single bank of pressurizer heaters, should be available from an assured power source within two hours after loss of off-site power in order to establish and maintain natural circulation in MODE 3.

Instructors Note: *During LOOP events no pressurizer spray will be present therefore the normal heating requirements are reduced.*

1) All pressurizer heater banks are capable of supplying the 126kW required by ITS from the control room. Although some of the heater groups are load shed, all heaters will be available within one minute.

2) Pressurizer Heater Bank 2 Group B can also be controlled in the SSF and the Aux. Shutdown Panel to achieve this function.

3) This heater bank is powered from MCC 1, 2, 3XSF, which is normally fed from load center 1, 2, 3X8.

4. The pressurizer heaters for each unit are normally supplied from non-safety related motor control centers (MCCs) XH, XI, XJ, XK. The pressurizer heaters are divided among the three 4160 volt ES buses such that the loss of one entire 4160 volt bus will not preclude the capability to supply sufficient pressurizer heaters to maintain natural circulation in MODE 3.

5. At 80" a low level interlock deenergizes the heaters to prevent damage while they are uncovered.

6. Pressurizer Water Space Saturation Recovery Circuit

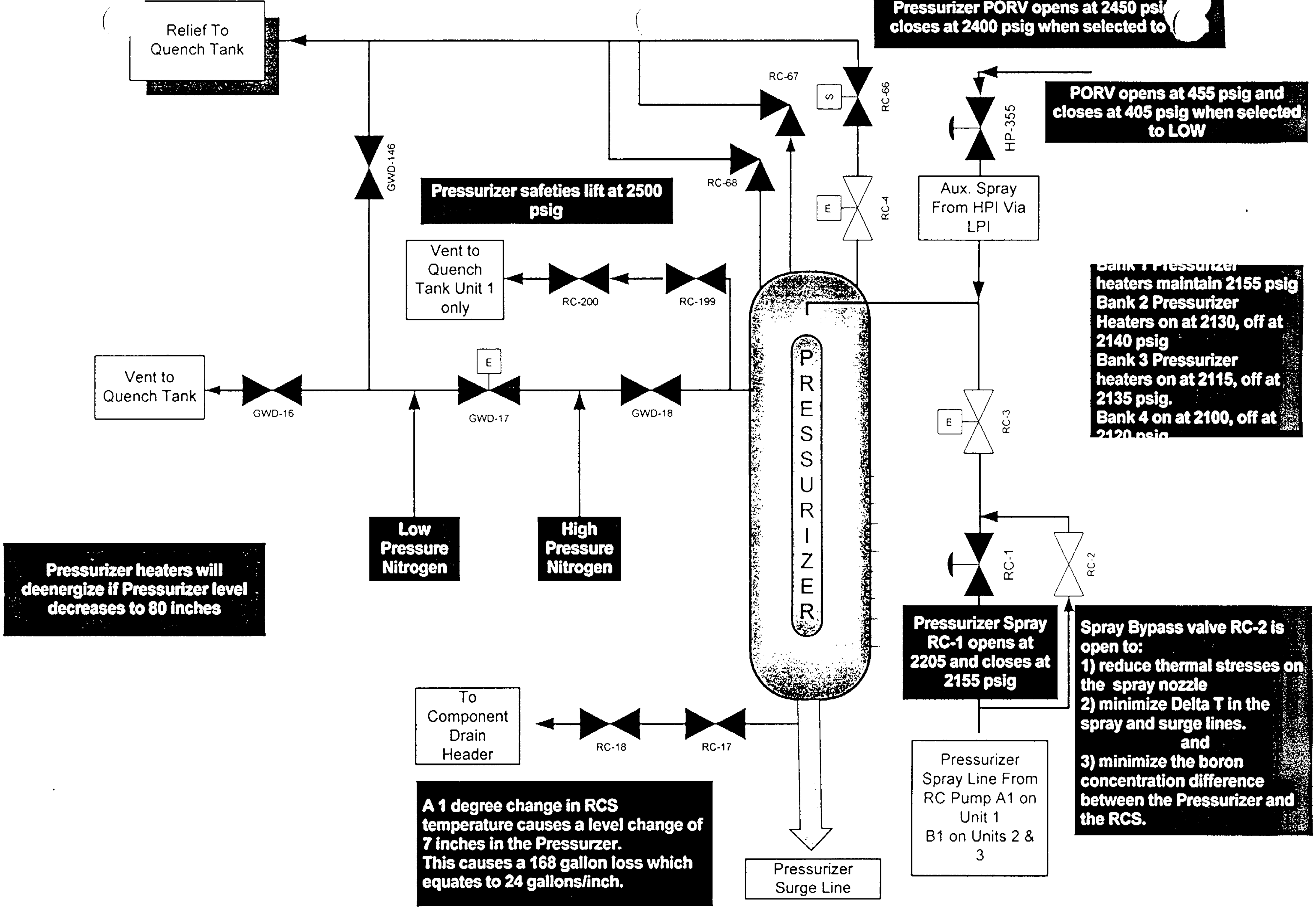
   In addition to being controlled by the ICS median selected narrow range (NR) RCS pressure signal, pressurizer heater bank #2 also receives a controlling signal from the Pressurizer Water Space Saturation Recovery Circuit.

   a) The purpose of this circuit is to automatically detect subcooled conditions in the pressurizer and energize a limited number of heater assemblies in order to reestablish saturated conditions (for the current RCS pressure).

   b) Pressurizer temperature 'C' is applied to a function generator to predict the corresponding saturation pressure (i.e., the predicted RCS saturation pressure for the current pressurizer temperature).

   c) If the predicted saturation pressure for the current pressurizer temperature is significantly below the actual RCS pressure, heater bank 2 is in AUTO and control is from the control room, the circuit will energize heater bank #2 in order to reestablish saturated conditions for the current RCS pressure.

   d) A 20 psig deadband is applied to minimize cycling of the heater bank.



TRAINING USE ONLY	
PRESSURIZER SYSTEM	
DRAWING #	OP-OC-PZR-1
DRAWN BY DER	DATE 7/8/99
REFERENCE	OFD -100A-1 2
APPROVED BY	

BW/A01K1.30
PNS653

Unit #1 is stable at 73% power following a runback from 100% power due to a feedwater transient. RCS pressure peaked at 2241 psig.

- RPS Channel A NR Pressure indicates 2154 psig
- RPS Channel B NR Pressure indicates 2157 psig
- RPS Channel C NR Pressure indicates 2153 psig
- RPS Channel D NR Pressure indicates 2154 psig
- RPS Channel E NR Pressure indicates 2158 psig
- Pressurizer temperature is 646□ F

Based on these conditions, 1RC-1, PZR Spray, should be _ and PZR heater bank #2 should be _.
(Choose ONE) (.25)

A) Open / Off

B) Open / On

C) Closed / Off

D) Closed / On

Answer

B
A. Incorrect - PZR heater bank #2 should be energized. See answer B. explanation below.
B. Correct - The median selected controlling RC pressure is 2157 psig which is above the RC-1 closing setpoint. The saturation pressure for 646□ F is □ 2150 psia or 2135 psig which is less than the median selected controlling RC pressure of 2157 psig. Therefore, the PZR saturation recovery circuit would have the bank #2 heaters energized.
C. Incorrect - RC-1 should be open and heater bank #2 should be energized. See B. above.
D. Incorrect - RC-1 should be open.

KA Number BW/A01K1.3 *RO* 3.7 *SRO* 3.7

Exam Level Both

RO Tier and Group T1-G2

SRO Tier and Group T1-G2

Question *ID:* 55

Plant conditions are as follows:

- Unit #1 is stable at 73% power following a runback from 100% power due to a feedwater transient.
- RCS pressure peaked at 2241 psig.
- RPS Channel A NR Pressure indicates 2154 psig
- RPS Channel B NR Pressure indicates 2157 psig
- RPS Channel C NR Pressure indicates 2153 psig
- RPS Channel D NR Pressure indicates 2154 psig
- RPS Channel E NR Pressure indicates 2158 psig
- Pressurizer temperature is 646 degrees F

Which ONE of the following correctly completes the below statement?

Based on these conditions, 1RC-1, PZR Spray, should be _________ and PZR heater bank #2 should be _________

a. Open / Off

b. Open / On

c. Closed / Off

d. Closed / On

Correct Answer b.

Answer and Distractor Justification

a. Incorrect - PZR heater bank #2 should be energized. See answer B. explanation below.

b. Correct - The median selected controlling RC pressure is 2157 psig which is above the RC-1 closing setpoint. The saturation pressure for 646□ F is □ 2150 psia or 2135 psig which is less than the median selected controlling RC pressure of 2157 psig. Therefore, the PZR saturation recovery circuit would have the bank #2 heaters energized.

c. Incorrect - RC-1 should be open and heater bank #2 should be energized. See B. above.

d. Incorrect - RC-1 should be open.

Question/Answer Source Reference OP-OC-PNS-PZR, REV. 10, Page 14, 16 - 17 of 35, OP-OC-PZR-1

Refence Provided with Question None

Learning Objective OP-OC-PNS-PZR, REV. 10, Enabling Objective 7

Question Source Bank - PNS653

Previous NRC Exam

Previous Quiz / Test

Cognitive Level 1

10 CFR Part 55 Content 41.8/41.10/45.3

Exam Question Report

27-Jan-99

Question ID:	PNS653	Revision No:	0	Revision Date	10/29/1999
Question Description:	PNS653				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	PNS-PZR - Pressurizer
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 29; SRO = 29 Reference: PNS-PZR	Max. Point Value:	0.25
		Passing Point Value:	0.25

27-Jan-99

Question

Unit #1 is stable at 73% power following a runback from 100% power due to a feedwater transient. RCS pressure peaked at 2241 psig.

- RPS Channel A NR Pressure indicates 2154 psig
- RPS Channel B NR Pressure indicates 2157 psig
- RPS Channel C NR Pressure indicates 2153 psig
- RPS Channel D NR Pressure indicates 2154 psig
- RPS Channel E NR Pressure indicates 2158 psig
- Pressurizer temperature is 646° F

Based on these conditions, 1RC-1, PZR Spray, should be _ and PZR heater bank #2 should be _.
(Choose ONE) (.25)

A) Open / Off

B) Open / On

C) Closed / Off

D) Closed / On

Answer

B
A. Incorrect - PZR heater bank #2 should be energized. See answer B. explanation below.
B. Correct - The median selected controlling RC pressure is 2157 psig which is above the RC-1 closing setpoint. The saturation pressure for 646° F is ≈ 2150 psia or 2135 psig which is less than the median selected controlling RC pressure of 2157 psig. Therefore, the PZR saturation recovery circuit would have the bank #2 heaters energized.
C. Incorrect - RC-1 should be open and heater bank #2 should be energized. See B. above.
D. Incorrect - RC-1 should be open.

QUESTION # 16

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	**1**	**1**
Group #	**2**	**2**
K/A #	**BW/A04 K2.1**	
Importance Rating	**3.5**	**3.3**

Technical Reference(s): **STG-MT, STG-EHC**
EOP

Proposed references to be provided to applicants during examination: NONE

Learning Objective: **STG-EHC #10 & #11**

Question Source:
Bank # **DB NRC #054**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test __X__

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 16

Unit 1 plant conditions:

- Reactor power = 24%
- The Turbine Generator has a HIGH vibration condition on Bearings 5 and 6
- The BOP has depressed the EHC-Turbine TRIP pushbutton
- A TURBINE TRIP did NOT occur and the reactor remains at 24% power

Which ONE of the following IMMEDIATE operator actions is correct?

A. Manually trip the reactor.

B. Manually trip the turbine locally.

C. Open BOTH generator output breakers.

D. Place BOTH EHC pump switches in “PULL-to-LOCK”.

1 POINT

QUESTION # 16

BW/A04 K2.1
(1)

A. Incorrect - Tripping the reactor is not a contingency to a failed turbine trip. This action could cause overcooling if the turbine did not trip. The RPS Turbine to Reactor trip is bypassed at this power level. This is not an UNPP event.

B. Incorrect, - Tripping the Turbine locally is not an action that can be performed immediately. This action requires sending an NLO out to the Turbine front standard (outside the Control Room).

C. Incorrect, - Opening both generator output breakers allows the potential of Turbine overspeed and damage. This is a correct operator action if the generator fails to trip following a Main Turbine trip.

D. Correct, - This action is an immediate contingency action to trip the turbine if the trip pushbutton does not work.

7. Describe the purpose and operation of the Operating Speed Governor. (R7)

8. Describe the purpose and operation of the Power Load Unbalance Circuit. (R8)

9. Describe how the Low Pressure Turbines are protected against overpressure. (R9)

10. Identify the automatic turbine trips including setpoint and type of protection provided. (R10)

11. Identify the automatic turbine trips and type of protection provided. (R23)

12. Describe the normal meter reading for the six DC power supplies located in the Cable Room EHC Cabinets and the indication that would be received due to an abnormal reading. (R11)

13. Descirbe the purpose of the First Stage Pressure Feedback signal. (R12)

    13.1 Describe why the First Stage Pressure Feedback signal is used in the Control Valve test and not Turbine Header pressure.

14. Explain why the 420 HZ Malfunction alarm must always be reset following a turbine startup. (R13)

2) Trips turbine at **98% level on either** steam generator's operating range.

3) Also trips the Main Feedwater Pumps

c) High Moisture Separator Level

1) Prevents carry over from damaging the Low PressureTurbine blading.

2) Trips turbine if either (first or second stage) of the Moisture Separator Drain Tanks back up to **approximately the bottom of any one of the four Moisture Separator Reheaters**.

3) The Turbine Trip Reduction Program has changed this circuitry to a **2 out of 3** trip logic on each MSRH. This is accomplished:

(a) Through the installation of three new level switches on each MSRH.

(1) The level switches for the Turbine Trip circuitry feed a 2 out of 3 logic for each MSRH such that any MSRH individually satisfying the Turbine Trip logic will cause a turbine trip.

(b) These level switches feed local alarm panels (one for the A side MSRHs and another panel for the B side MSRHs)

(1) These local alarm panels have individual alarm lights for the individual level switches that have a seal in feature to require a manual reset to clear the alarm indication.

(i) These local panels send an alarm signal if any level switch local alarm is actuated to the Heater Panel to actuate an alarm there.

d) Turbine Oil Fire

1) Prevents damage to the turbine bearings and shaft.

2) When the **trip lever is pulled** it shuts off the oil pumps (which supply oil to the turbine) to prevent them from feeding the fire and also trips the turbine.

5. Miscellaneous Trips

a) Low Hydraulic Oil Pressure

1) Normal operating pressure for the Control Pacs is 1600 psig. Should this pressure fall to a low value, the overall control of the valves would be lost.

2) Trips the turbine at **1100 psig** decreasing hydraulic pressure.

b) Generator Lockout

1) Prevents damage to the Main Generator.

2) **86A Generator Lockout Relay** will trip the turbine.

3) This trip is generated from protective relaying which monitors for electrical faults and abnormal conditions within the Main Generator.

c) Reactor Trip

1) Prevents overcooling the primary plant.

2) **Two independent reactor tripped confirmed signals from the control rod drive system** will each send a turbine trip signal.

(a) "A" Channel energizes the fast acting solenoids to trip the Main Stop Valves closed.

(b) "B" Channel energizes test circuits to ramp the Main Stop Valves closed (ramp requires ≈ 12 secs).

d) Low Feedwater Pump Discharge Pressure

1) Anticipatory trip of turbine prior to reactor trip due to either high temperature or high pressure.

2) Trips turbine at **800 psig** feedwater pump discharge header pressure.

e) AMSAC portion of AMSAC/DSS

1) Provided to mitigate the consequences of an anticipated transient without SCRAM

2) AMSAC will trip the Main Turbine and start all operable EFWPs

3) Setpoint: need **both channels of AMSAC/DSS** to be enabled (2/2 logic) **AND**: either

Both MFPs have low hydraulic oil pressure (<75 psig)

OR

Both MFPs have low discharge pressure (<770 psig)

f) Loss of 24V DC

1) Loss of 24V DC will render the EHC electrical trip system inoperable.

2) A loss of 24V DC will trip the turbine by deenergizing the pilot solenoids and repositioning the master trip solenoid.

g) Loss of 125V DC

1) Loss of 125V DC will render this EHC electrical trip system inoperable.

2) A loss 125V DC will trip the turbine by energizing the 24V DC trip bus.

h) Manual Trip Button

i) Manual Trip Handle

6. Summary of Turbine Trips

a) Mechanical Overspeed - ≈ 1980 RPM / 110% of rated speed

b) Backup Overspeed - ≈ 2003 RPM / 111.25% of rated speed

c) Loss of Both Speed Feedback signals to the turbine speed control circuitry

d) Low Condenser Vacuum - ≈ 21.75 inches Hg.

e) Loss of Stator Coolant - If runback does not reduce load below 740 MWe within 2 minutes and below 256 MWe within an additional 1½ minutes if condition has not cleared when first plateau is reached.

f) Low Bearing Oil Pressure - Originates from old thrust bearing wear detector pressure switches at <8 psig

1) This trip was changed to incorporate 3 pressure switches and a 2 out of 3 trip logic.

g) High Steam Generator Level Trips -

1) 98% Level on Operating Range on either SG

2) SG Overfill Protection - An additional auxiliary relay was added to the OTSG level control system circuitry. Existing hi level contacts feed this relay. When both level signal monitors for A or B SGs sense a high level, a redundant trip signal is sent to both MFDWPS and the Main Turbine.

h) High Moisture Separator Level Trip – level at bottom of MSRH

1) This trip incorporates 3 new level switches on each MSRH to provide a 2 out of 3 Turbine Trip logic.

i) Reactor Trip

j) Low Feedwater Pump Discharge Header Pressure - ≈ 800 psig

k) Low EHC Discharge Header Pressure - 1100 psig decreasing

l) Loss of 24V DC

m) Loss of 125V DC - 2 out of 3 relays which monitor 125V DC at three locations in the EHC cabinet

n) Generator Lockout - 86 GA

o) Manually initiated from:

1) Turbine Oil Fire Trip

2) Manual Trip Handle on Front Standard

3) Master Trip Button in Control Room

E. Control systems

1. Speed Control

a) Six different speed sets can be selected by the operator:

1) Close Valves

2) 100 RPM - Used to sound out turbine bearings

3) 500 RPM - Not used at Oconee

4) 1500 RPM - Not used at Oconee

5) 1800 RPM - Used to bring turbine to rated speed after 100 RPM checks are complete.

6) Overspeed Test - This pushbutton can be used to test either the mechanical overspeed device or the back-up overspeed trip circuit.

b) Three different acceleration rates can be selected by the operator.

1) Slow - 60 RPM/MIN

2) Medium - 90 RPM/MIN

3) Fast - 180 RPM/MIN

2. Load Control

a) The Load Control circuits in the EHC system develop the electrical signals that are sent to the control pacs and ultimately determine how far the valves open or close.

b) Develops output based on input from:

2.2 Component Description

A. Main Steam Stop Valves (OC-STG-MT-3)

1. Four Main Steam Stop Valves (MS-102, MS-103, MS-104 & MS-105) are located in the four Main Steam inlet lines to the High Pressure Turbine.

2. Each MSSV outlet is welded directly into the inlet of a Main Control Valve casing.

3. The four MSSV's are physically located on the Mezzanine Floor (3rd floor) of the Turbine Building at the front end of each unit's High Pressure Turbine.

4. The four MSSV's are also welded together to form a common casing below the stop valve seats. This common connection forms the Main Steam Chest.

5. The primary purpose of the MSSV's is to quickly shut off steam flow to the turbine under emergency conditions. MSSV closure is initiated by a reactor trip signal. To keep from rapidly cooling off the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. From the CRDI system, a Channel "A" trip circuit will close all the MSSVs in ≤ 1 second. A Channel "B" trip circuit will close all MSSVs in ≤ 15 seconds.

6. The MSSV's also serve as the isolation valves between the OTSG's in the RB and the Main Turbine. All four MSSVs are needed. If for example a MSLB occurred. Both S/Gs would discharge through the break. Once the MSSVs closed, then steam would be isolated to just the single affected steam generator.

   a) ONS ITS 3.7.2 – Turbine Stop Valves states that both MSSVs in each main steam line shall be operable in Modes 1, 2, and 3. ITS Bases states: an operable MSSV is one that closes within its required actuation signal time limits. The MSSVs must be Operable or closed. A Closed MSSV is performing its safety function.

7. MSSV Construction (Valves 1, 3 & 4)

   a) Basically consists of a casing that holds a spherical-shaped hardened valve seat that is ~ 24" in diameter.

   b) A valve stem passes through the bottom of the casing and attaches to a conical-shaped valve disc that mates with the valve seat when the stop valve is closed.

   c) Any steam leakage down the stem is routed to the Steam Seal Header.

d) A steam drain line and control valve are located above the stop valve seat to drain condensation from the casing that collects when the stop valve is closed.

e) An EHC Control Pack pushes the valve stem upwards and carries the valve disc with it to open the stop valve. Full stroke of the stem is ~ 8.5 inches.

8. #2 MSSV Construction (OC-STG-MT-4)

a) Basic construction is the same as the other MSSV's except that the #2 MSSV has an additional internal valve disc to regulate turbine warming prior to placing the turbine in operation.

b) The normal valve disc is modified:

1) The top portion of the disc is fabricated into an integral valve seat.

2) Orifices have been milled from the integral valve seat through the bottom portion of the main valve disc.

c) A smaller valve disc is attached to the normal valve stem of the #2 MSSV to mate with the integral valve seat formed into the main valve disc.

1) This smaller valve disc is called the #2 Main Stop Valve Bypass Valve.

2) When the proper logic (Shell Warming or Chest Warming) is selected, the operator can position the Bypass Valve disc from a fully shut to a fully open position from the EHC Control Panel in the Control Room.

3) With the Bypass Valve open, steam will flow through the main valve disc and through the orifices in the disc.

4) The Bypass Valve is used to:

(a) Slowly warm the HP Turbine Shell prior to placing the MT in service.

(b) Slowly warm the Main Control Valve components and the below seat metal of the MSSV's before admitting full steam through these valves.

(c) Reduce the ΔP across the MSSV's to allow the valves to open. For this reason, MSSV #2 is referred to as a "balanced" valve.

B. Main Control Valves

1. Four Main Control Valves (MS-107, MS-106, MS-109 & MS-108) are located just downstream of the four MSSV's.

2. Physically, the CV's are at the same location as the MSSV's.
3. The outlet of one MSSV is welded directly to the inlet of a CV.
   a) MSSV #4 connects to CV #1.
   b) MSSV #1 connects to CV #2.
   c) MSSV #3 connects to CV #3.
   d) MSSV #2 connects to CV #4.
4. Since the four MSSV's are interconnected below their seats, any stop valve will supply any control valve with steam, although at a more reduced rate than if the corresponding stop valve is open.
5. The CV's control the amount of steam supplied to the turbine to control turbine speed when the generator is not tied to the system grid.
6. The CV's control the amount of steam supplied to the turbine to regulate the Megawatt load when the generator is tied to the system grid.
7. Main Control Valve Construction
   a) Consists of a casing that holds a spherical machined and hardened valve seat that is ~ 20" in diameter.
   b) A valve stem passes through the top of the casing and attaches to a spherical shaped "poppet" type valve disc that mates with the valve seat when the CV is closed.
   c) An integral part of the lower portion of the valve stem is an internal pilot valve which mates with a seat machined in the main valve disc.
      1) The internal pilot valve disc lifts before the CV main disc does and begins to "blowdown" the steam pressure trapped behind the closed main disc in a balance chamber.
      2) Relieving this pressure in the balance chamber reduces the force against which the main disc must open.
      3) Only the #1, 2 & 3 CV's have this internal pilot. The #4 CV does not need it since it is the last CV to sequence open and the pressures should be balanced.
   d) Steam leakage past the stem to atmosphere is prevented by a stuffing gland at the top of the upper bushing of each Control Valve which directs the steam leakage through leakoff lines to either the High Pressure Turbine exhaust line or to the Steam Seal Header.
   e) A steam drain line and control valve is provided for each CV.
      1) Located just above the valve seat.

2) Each line from the four CV's goes to a common drain header back to the Main Condenser.

3) The common drain line has a motor operated valve (SD-273) operated from the Control Room.

4) Purpose - Drain any condensation that may form.

f) The CV control linkage pulls the stem upward against spring force to open the valve.

C. Steam Leads

1. The steam leads are the individual steam lines that connect the discharge of the control valves to the High Pressure Turbine.

2. Each steam lead leaves its respective CV, drops downward toward the TB Basement, then makes a turn upward to the turbine.

3. Each steam lead enters the High Pressure Turbine at a separate connection at the center span of the turbine. These connections are arranged such that OTSG steam flows to the turbine are balanced.

4. Each steam lead is provided with a steam drain at its low point.

a) #1 & 2 steam lead drains are piped through valves SD-285 & 284 (controlled from the Control Room).

b) #3 & 4 steam lead drains are piped through strainers and a 3/16" orifice to pneumatic valves SD-254 & 258.

1) Limit switches on the respective control valve control SD-254 & 258.

(a) When #3 CV opens, SD-254 closes.

(b) When #4 CV opens, SD-258 closes.

2) The orifice is in the line to limit the amount of steam blowdown during low load conditions when CV #4 and possibly #3 are closed.

3) The strainers keep the orifices from being clogged.

c) #2 steam lead drains to the MS pumping trap (TBV pumping trap on Unit #1).

d) #1, #3, and #4 steam leads drain to the condenser.

D. High Pressure Turbine (OC-STG-MT-5 & 6)

1. High Pressure Turbine Shell

5. Lube Oil Gages and Manual Trip Handle
    a) Lube Oil Gages
        1) Three oil pressure gauges are located on the very front of the standard for monitoring operation of the TLO System.
            (a) Pump suction gage
            (b) Bearing Header Pressure gage
            (c) Operating Oil Pressure gage
        2) Just below the oil pressure gages are two local temperature gages for monitoring bearing header oil temperature and oil drain temperature from #1 Bearing.
    b) A manual turbine trip handle is located to the right of the oil pressure gages. Turning the handle counter-clockwise and pulling it will initiate a turbine trip.
6. TSI Instruments

    The Differential Expansion Detector and the Eccentricity Detector are located in the oil bath inside the Front Standard enclosure.
7. "Dry Pocket"
    a) A section of the Front Standard is separated from the "oil bath" area where the Control Rotor Stub Shaft and Main Shaft Oil Pump are located.
    b) Located on the right side of the standard and can be accessed by a door.
    c) Houses pressure switches and valves associated with the EHC System.
    d) The Extraction Relay Dump Valve is located here. This removes the air from the Extraction Check Valves' power actuators.

4. Immediate Manual Actions

VERIFICATION/ACTION	CONTINGENCY ACTION
4.1 Verify a reactor trip has occurred or should have occurred,	**GO TO** Section 504, SG Tube Leak.
4.1.1 **Manually trip the Reactor.**	
___ Rx Manually Tripped ___ All Power Range NIs < 5% FP and decreasing.	Perform the following • Rule #1 "ATWS Actions" • **GO TO** Section 506, Unanticipated Nuclear Power Production.
4.1.2 **Manually trip the Turbine-Generator.**	
___ Turbine Manually Tripped	
___ All Turbine Stop Valves closed	Place both EHC pumps in the "PULL TO LOCK" position
___ Both Generator Output breakers open	Open PCB(s) from Control Room switches.
___ Turbine Bypass Valves (TBVs) controlling as expected.	Manually control TBVs.
4.1.3 **Verify RCP seal injection is available.**	
___ RCP Seal Injection available	Verify CC available. IF CC is NOT available, THEN perform the following: ___ Immediately trip all RCPs. ___ Notify CR SRO the SSF must be activated per AP/0/A/1700/025. (Standby Shutdown Facility Emergency Operating Procedure).
4.1.4 GO TO Section 5, Subsequent Actions.	

BW/A04K2.10
DB_NRC054

In an attempt to trip the turbine from 24% power due to high vibration, the CR operator depressed the EHC-Turbine TRIP pushbutton. The turbine TRIP did NOT occur and the reactor is still producing 24% power.

Which ONE of the following is the correct action the CR operator should perform IMMEDIATELY? (0.25)

A) Manually trip the reactor.

B) Have the Turbine locally tripped.

C) Place EHC pump switches in PULL-to-LOCK.

D) Open BOTH generator output breakers.

Answer

C
C is the fastest method to trip the turbine if the trip pushbutton does not work.
B will work but requires sending an NLO out to the front standard, wasting time.
C Tripping the reactor will not guarantee the turbine will trip. This could cause overcooling if the turbine did not trip. Also power is below the RPS trip setpoint therefore the reactor is not endangered. This is not an UNPP.
D The generator is not motoring. It is producing power. Motoring could occur if "C" is done and the TG did not trip.

QUESTION # 17

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**2**	**2**
	K/A #	**000008 A2.12**	
	Importance Rating	**3.4**	**3.2**

Technical Reference(s): **PNS-PZR**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PZR #25**

Question Source:
Bank # ____
Modified Bank # ____
New __X__

Question History:
Previous NRC Exam ____
Previous Quiz / Test ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ____
55.43 ~~__X__~~

Comments:

1 POINT

QUESTION # 17

Unit 2 plant conditions:

- Power level = 100% power
- 8 of 10 lights are lit on 2RC-67 (PZR RV) flow monitor
- Quench Tank temperature increasing

Which ONE of the following is the **INITIAL** transient response?

RCS Pressure is decreasing ________ and Pressurizer Level is decreasing ________.

A. rapidly / rapidly

B. rapidly / slowly

C. slowly / rapidly

D. slowly / slowly

Hinges on 8/10 lites, lits indicate
large steam release

1 POINT

QUESTION # 17

000008 A2.12 (3.4/3.7) Both T1-G2/T1-G2 #46 RSI/PRA 5-2-00
(2)

A. Incorrect - RCS pressure will decrease rapidly as indicated on the PZR RV Flow Monitors. Pressurizer level will not decrease rapidly due to the higher volumetric flow rate vs. mass flow rate due to steam exiting the PZR.

B. Correct - RCS pressure will decrease rapidly and Pressurizer level will initially decrease slowly and then have a slight in-surge.

C. Incorrect - RCS pressure will not increase slowly due to the indication from the 8 of 10 PZR RV Flow Monitors indicating a large amount of steam flow through the PZR RV.

D. Incorrect - Both RCS pressure and Pressurizer level decreasing slowly is inconsistent with either a water or steam space leak in PZR.

ENABLING OBJECTIVES (continued)

13. Discuss the operation of the pressurizer heaters including: (R7)
    13.1 Three purposes of pressurizer heaters.
    13.2 Purpose and level of interlock associated with pressurizer heaters.
    13.3 On/off setpoints for pressurizer heater banks 2, 3and 4.
14. Describe the physical operation of the PORV including what causes the Pilot Valve to operate and how this causes the PORV to open or close. (R8)
15. Explain the purpose of the two opening setpoints associated with the PORV. (R9)
16. Explain how to manually operate the PORV. (R37)
17. Given a set of conditions, determine operability of the PORV following a loss of power. (R30)
18. Discuss the reason for the pressurizer safeties and their setpoint. (R12)
19. Given a set of plant conditions, determine the response of Pressurizer level. (R14)(R15)
20. Explain the operation of SASS as it relates to pressurizer level control. (R31)
21. Given a set of conditions, determine how pressurizer level control/indication is affected by a loss of SASS and/or ICCM. (R35)
22. Discuss the use of Pressurizer Saturation Pressure Indication by the operator. (R16)
23. Discuss the forming of a pressurizer steam bubble including any precautions to be taken during the evolution. (R17)
24. Given a completed copy of PT/0/A/201/04 PORV Operability Test apply compare data taken to acceptance criteria to determine PORV operability. (R10)(R11)
25. Differentiate between a pressurizer steam space leak and a water space leak. (R32)
26. Given a set of plant conditions, determine the position of the PORV. (R13)
27. Given a set of conditions, calculate the expected PORV discharge temperature. (R34)
28. Given a copy of a Limit and Precaution from OP/A/1103/05, Pressurizer Operation, be able to state the reason for that limit and precaution. (R18)
29. Apply ITS/SLC's rules to determine applicable Conditions and Required actions for a given set of Pressurizer conditions. (R24)

D. Abnormal Pressurizer Operations

1. System Response to a Pressurizer Steam Space Leak

Instructor Note: This is a generic description. Actual plant response may vary due to plant conditions. The magnitude of plant response will be dictated by the size of the steam leak. System response description also assumes no operator action.

a) RC pressure and subcooling margin will decrease.

b) Pressurizer level will remain constant.

c) Pressurizer heaters will energize as their setpoints are reached. Depending on leak size, additional steam generation may reduce or terminate RC pressure decrease.

d) If steam leak is small, boron concentration in pressurizer will increase relative to RCS over time.

e) Increase in RC makeup flow (and decrease in LDST) will be negligible due to the difference in the specific volume of steam vs. a liquid, i.e., less mass will be lost if leak is in the form of steam.

f) If steam leak size is in excess of capacity of pressurizer heaters, RC pressure and subcooling margin will continue to decrease, eventually resulting in a reactor trip on low RCS pressure or variable low pressure.

g) Following the reactor trip, RC pressure and subcooling margin will continue to decrease. If saturated conditions occur in the hot legs or reactor vessel head prior to ES actuation, the voiding and subsequent expansion of RCS will cause an increase in pressurizer level.

h) If the pressurizer level increases above the steam leak, the RC pressure decrease may be reduced or stopped. Otherwise the pressurizer will continue to fill until completely full.

i) If ES channels 1&2 actuate prior to a loss of subcooling margin, a similar response will take place. HPI injection will cause pressurizer level to increase, which will, in turn, compress the existing bubble. Eventually, RC pressure will reach the PORV setpoint. Ultimately, the pressurizer will fill completely full.

2. System Response to a Pressurizer Water Space Leak

Instructor Note: This is a generic description. Actual plant response may vary due to plant conditions. The magnitude of plant response will be dictated by the size of the steam leak. System response description also assumes no operator action.

a) System response will be similar to a RCS leak anywhere on the system.

b) Makeup flow will increase (and LDST level will decrease) in response to the loss of RCS inventory.

c) Pressurizer level may remain constant or decrease depending upon the magnitude of the leak.

d) Again, RC pressure response will be dependent upon leak size. For small leaks, RCS pressure will remain constant. As pressurizer level drops, the steam bubble will expand to fill the space causing RC pressure to decrease.

e) Pressurizer heaters will energize as RCS pressure continues to drop.

f) If leak is of such a magnitude that RCS makeup flow cannot maintain level, RCS pressure will continue to drop until the reactor trips and ES actuates.

3. Identifying an Open or Leaking Pressurizer Relief Valve.

a) Pressurizer Relief Valve Flow Monitor will alert the operator to an open relief valve via alarm/indication.

b) RC-66 valve indication.

c) Pressurizer boron concentration increasing relative to RCS.

d) Pressurizer relief valve tailpipe temperature

1) Pressurizer relief valve tailpipe temperature is a function of pressurizer temperature (or pressure since saturated) and Quench Tank pressure.

2) Since flow through a relief valve is a constant enthalpy throttling process, the temperature downstream of a relief valve can be found by:

(a) Finding the point where the temperature (or *absolute* pressure) of the pressurizer intersects the saturation curve on the Mollier diagram.

INSTRUCTORS NOTE: Work a couple of examples with the students using the Mollier Diagram.

See handout #1 and #2

(b) Cross the horizontal constant enthalpy line from this intersection point to the point where the constant enthalpy line intersects the *absolute* pressure of the Quench Tank.

(c) Follow the *absolute* Quench Tank pressure line up to the intersection of the saturation curve.

(d) The constant temperature line that intersects this same point on the saturation curve is the temperature of the steam downstream of the relief valve.

3) Quench tank level, temperature and pressure

(a) Steam flow past a leaking relief valve will enter the quench tank below water level.

(b) The contents of the quench tank will condense the steam.

(c) The condensed steam will increase quench tank level, temperature and pressure.

(d) The same process occurs for an open relief valve except that the effluent rapidly heats the contents of the quench tank to saturated conditions, a steam bubble forms in the quench tank and the quench tank rupture disk ruptures.

4. Emergency Operating Procedure Evolutions

a) During a shutdown due to a steam generator tube leak/rupture which requires entry into the emergency operating procedure, the pressurizer heaters are manually secured and pressurizer spray (or the PORV if spray is unavailable) is manually initiated. This allows the operator to purposely lower RCS pressure to approximately 3-5° F subcooling to minimize the subcooling margin. This minimizes the differential pressure between the RCS and the steam generator, thus decreasing the tube leak rate.

b) Following a loss of feedwater event where no sources of feedwater are available and RCS pressure is approaching 2300 psig, an HPI pump is operated. Flow is established in each injection header through HP-26 and HP-27 and the PORV manually opened to provide a flowpath for cooling water from the BWST, through the core, to the basement of the reactor building.

2.7 System Interlocks/Automatic Actions

A. In automatic, HP-120 maintains pressurizer level at setpoint (normally 220 inches) under normal conditions.

B. In automatic, the pressurizer heater banks cycle as necessary to maintain RCS pressure as follows:

1. Pressurizer heater bank no. 1 will maintain RCS pressure at setpoint (normally 2155 psig).
2. Pressurizer heater bank no. 2 energizes at 2130 psig decreasing and de-energizes at 2140 psig increasing.
3. Pressurizer heater bank no. 3 energizes at 2115 psig decreasing and de-energizes at 2135 psig increasing.
4. Pressurizer heater bank no. 4 energizes at 2100 psig decreasing and de-energizes at 2120 psig increasing.

C. A 80-inch low pressurizer level interlock prevents the heaters from being energized while they are uncovered.

QUESTION # 18

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**2**	**2**
K/A #	**000009 K3.24**	
Importance Rating	**4.1**	**4.6**

Technical Reference(s): **EAP-E20**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **E-20 OBJ. #23**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 18

Unit 1 plant conditions:

INITIAL CONDITIONS:

- A SBLOCA has occurred
- ES 1 and 2 has actuated

CURRENT CONDITIONS:

- PZR level = 42 inches and increasing
- RCS pressure = 1430 psig and increasing
- Core SCM = 0°F
- "A" Loop SCM = 2°F
- "B" Loop SCM = 12°F

Which ONE of the following is correct?

HPI can only ______ be throttled when / ______.

A. ~~can / when~~ PZR level is ≥ 100 inches

B. ~~can / when~~ core SCM increases to ≥ 5°F

C. cannot ~~/ until~~ when the core and both Loops SCM are ≥ 5°F

D. cannot / ~~until~~ when RCS pressure is above the ES 1 and 2 actuation setpoint

what is the difference between

can when

cannot until

1 POINT

QUESTION # 18

000009 K3.24 Both PRA 2-8-00
(2)

A. Incorrect – PZR level = 100 inches is a normal throttling level for inventory control but not for ES throttling criteria.

B. Correct – ≥ 5°F is the criteria for throttling HPI following ES actuation.

C. Incorrect – Only core SCM is required when throttling HPI following ES actuation.

D. Incorrect – HPI cannot be throttled if RCS is > ES actuation pressure unless core SCM is ≥ 5°F.

21. Explain why and at what point during the normal post trip recovery phase normal RCS letdown should be re-established. (R17)

22. Explain why PZR level should be controlled at ≈100 inches following normal post trip recovery of PZR level. (R18)

23. Given a set of conditions determine if it is permissible to throttle HPI flow following a reactor trip, and explain why it is important to throttle HPI flow as soon as it is permissible. (R19)

24. Explain why several P/T curves are provided in the EOP and describe when each should be used. (R20)

25. State Station management's expectation concerning exiting the EOP while in natural circulation (36).

5) If core SCM > 5°F, throttle HPI to maintain P/T stable and REFER to enclosure 7.1 P/T Curves:

   (a) With the core at least 5°F subcooled, there should be no danger that SCM will be lost if HPI cooling flow is reduced.

   (b) It is important to throttle HPI as soon as allowable, for reasons listed above.

   (c) Two different sets of P/T curves in the EOP address different possible operating conditions:

      (1) If RB pressure is less than 3 psig it is assumed that the RB environment is non-hostile and instrument readings are accurate. ICCM or OAC instrumentation can be used to determine the P/T relationship.

      (2) If RB pressure is greater than 3 psig it is assumed that the containment environment is hostile and that instrument accuracy will be degraded. Encl. 7.1.A compensation for high RB temperature is completed by using ONLY the ICCM Instrumentation.

      (3) Curve 7.1.B is for use when RCS pressure has decreased below 600 psig and the LR Cooldown instrument is in service.

6) RCS Pressure Is Being Controlled (RCS P/T is Stable) The adequacy of RCS pressure control should be verified to identify potential failures in the pressurizer spray, heaters or relief valves or their corresponding control systems.

7) If PZR level is > 375 inches normal control of PZR operation has failed and the RCS may be at or is in danger of being water solid. Transferring to CP-604, Solid Plant Cooldown will:

   (a) Address some of the symptoms that could cause high PZR level indication, such as, a saturated RCS, PZR relief or RCS vents open.

   (b) If the RCS is not actually water solid (i.e. a PZR bubble exists) PZR level should quickly return to below 300" and a transfer out of CP-604 will be quickly made.

   (c) On the other hand, if the RCS is solid, CP-604 will provide the necessary directions for reestablishing a PZR bubble. It also provides instructions for starting RCPs with a solid plant if this becomes necessary.

QUESTION # 19

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**2**	**2**
K/A #	**BW/0E08 A1.2**	
Importance Rating	**3.1**	**3.1**

Technical Reference(s): **PNS-PRV**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PRV #2 & #9**

Question Source:
Bank # **PNS-280**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

3 distractors hinge on this

QUESTION # 19

Unit 2 plant conditions:

- A LOCA has occurred
- ES 1-6 has actuated properly
- All RB Purge Isolation Valves are "leaking-by"

Which ONE of the following will MINIMIZE the effects of a Reactor Building release to the environment?

~~RB Purge inlet prefilter~~

A. Automatic actions resulting from RIA-45, Vent Stack Gas monitor in Alert and/or Alarm.

B. Triple isolation valves in both intake/exhaust ductwork.

C. ~~Positive closing dampers in purge system ductwork.~~ shifting RBCU to high

D. Penetration Room ventilation filters.

~~RB purge inlet prefilter~~

shift RBCU to high speed

1 POINT

QUESTION # 19

BW/E08 A1.2
(1)

A. Incorrect, - The ventilation valves associated with RIA-45 will already be shut under the given conditions.

B. Incorrect, - The stem of the question states that the Purge Isolation Valves are leaking by which implies that all valves are leaking and allowing flow.

C. Incorrect, - The stem of the question states that the Purge Isolation Valves are leaking by which implies that all valves are leaking and allowing flow.

D. Correct, - The penetration room ventilation filters are designed to eliminate and minimize the passage of contaminants.

REACTOR BUILDING PURGE

Normal Ops:

Operation allowed in MODEs 5, 6, or NO MODE
PR-1-6 are "sealed" closed in MODEs 1, 2, 3, & 4
RIA 43, 44, 45, & 46 monitor exhaust air
Flow adjusted via PR-3 for Purge
GWR must be completed for Purge

Flowpath:

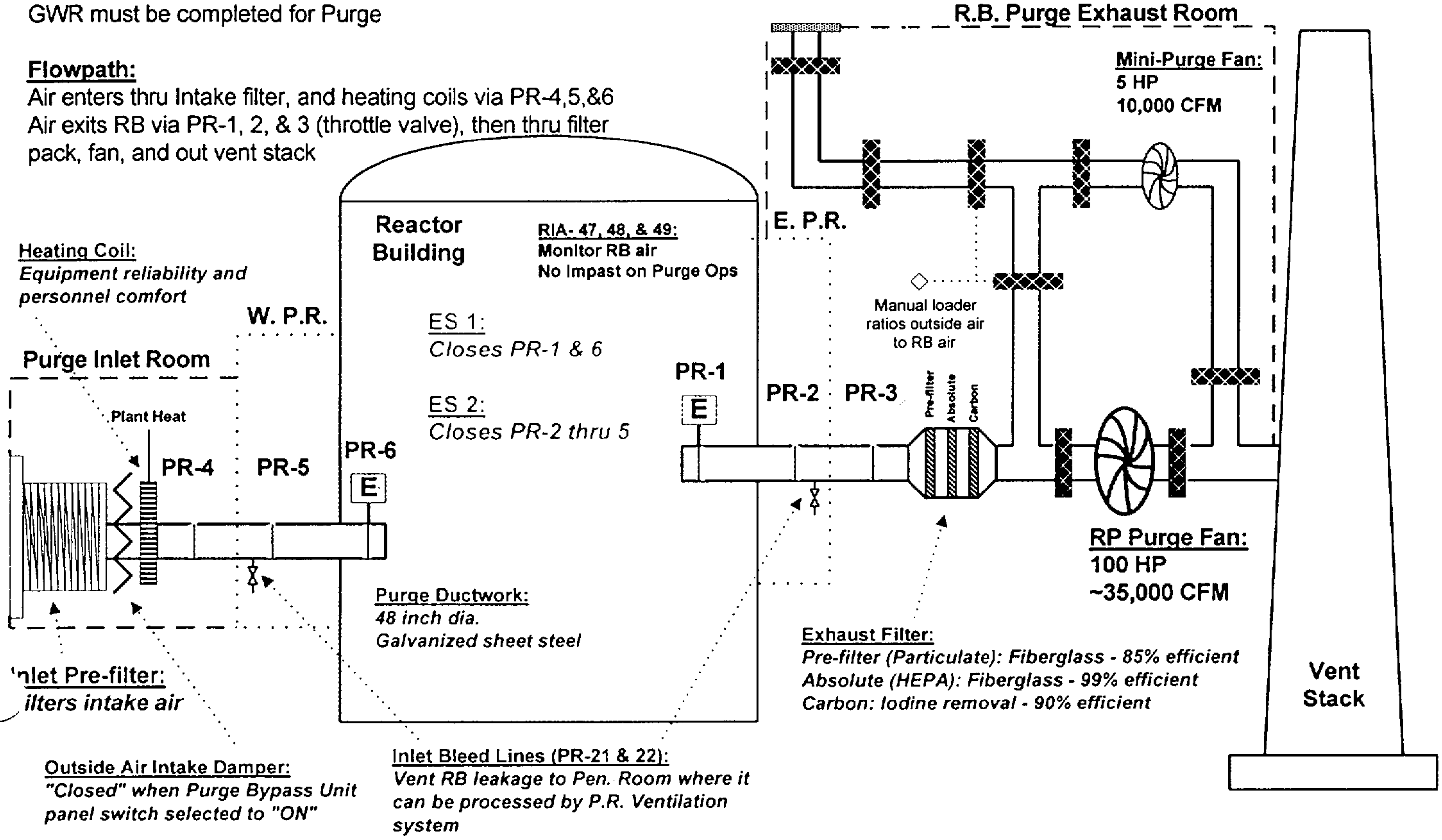
Air enters thru Intake filter, and heating coils via PR-4,5,&6
Air exits RB via PR-1, 2, & 3 (throttle valve), then thru filter pack, fan, and out vent stack

Interlocks:

- RIA-45 or 46 HIGH alarm - Trips Main & Mini fans and CLOSES PR-2 thru 5
- Closing any valve (PR-1 thru 6) trips Main Purge fan
- Closing Main Purge Fan Inlet &/or Outlet dampers trips fan (20 second time delay to allow fan start)
- Ductwork Vacuum ≥ 9 " trips Main fan
- Mini Purge Fan trips if Main Purge started
- SFP Vent. Exhaust Fans trip if Main Fan started (units 2 & 3 only)

Mini Purge Operation:

Started locally
Manual loader adjusted for 100% outside air prior to fan start
Loader rotated following fan start to ratio RB air for GWR release rate flow. (Mix of RB and outside)



TERMINAL OBJECTIVE

T1 Describe the operation and purpose of the Penetration Room Ventilation System.

ENABLING OBJECTIVES:

1. Draw a basic flow diagram of the Penetration Room Ventilation System. (R1)
2. State the purpose of the Penetration Room Ventilation System. (R2)
3. List the types of filters used in the Penetration Room Ventilation System. (R3)
4. Explain the advantages of the external carbon sampling canisters. (R4)
5. State the purpose of the charcoal filter. (R5)
6. Explain the reason for the maximum flow limit through the Penetration Room Ventilation System. (R6)
7. State the purpose of PR-13 and PR-17, A & B filter outlet valves. (R7)
8. Explain the two purposes of PR-20, PRV fan suction cross-connect valve. (R8)
9. Describe the operation of the Penetration Room Ventilation System following Engineered Safeguards actuation. (R9)
10. State the reason for the high humidity limit in the penetration rooms and any associated actions. (R10)
11. Briefly describe the requirements for opening a Penetration Room floor drain while containment integrity is required. (R11)
12. Briefly describe the requirements for draining a system to the Penetration Room floor drains. (R12)
13. Describe the purpose of the travel stops installed on PR-13 and PR-17, including when they should be adjusted. (R13)
14. Describe two circumstances when it may be necessary to periodically adjust Penetration Room Ventilation flow. (R14)
15. Explain the various means available to the operator in the control room and locally to identify degraded Penetration Room Ventilation flow. (R15)

16. For PT/0110/010, PRVS Vacuum Test, describe: (R18)

1. INTRODUCTION

1.1 Purpose of the PRV System:

A. The Reactor Building Penetration Room Ventilation System (PRVS) is designed to collect and process potential post-accident Reactor Building penetration leakage to minimize environmental activity levels.

1.2 General System Description:

A. The Penetration Room Ventilation System consists of two, 100% capacity redundant trains, each containing a fan and filter assembly. Both fans, discharging through a common line to the unit vent, are controlled from the main control room. **REFER to OC-PNS-PRV-1**.

B. During normal operation this system is aligned in a standby mode with each fan aligned to take a suction on it's associated filter assembly. The engineered safeguards signal from the Reactor Building pressure will actuate the system by starting both fans. Control room instrumentation monitors operation.

C. A motor power operated butterfly valve located at the discharge of each fan opens when it's respective fan starts. This valve remains closed if the fan fails to start and will go closed on a subsequent fan failure to prevent recirculation. A check valve is also provided at the discharge of each fan to prevent recirculation on a failure of the discharge valve to close. The normally closed pneumatic suction cross tie valve (PR-20) may be throttled open from its remote manual station to maintain adequate cooling air through an idle filter train.

D. The design flow rate from the penetration room far exceeds the maximum anticipated Reactor Building leakage. The design leak rate of ½ percent per day amounts to approximately 15 scfm compared to a design evacuation rate of 1000 scfm for each train of the system.

E. The system utilizes remote manual pneumatic control valves PR-13 and PR-17 in conjunction with constant speed fans to provide the proper negative pressure in the penetration room.

F. If during operation the leakage increases causing a decrease in negative pressure below 0.06 inches H_2O. (with respect to the outside), the remote manual control valve may be adjusted or leaks will be repaired to bring the negative pressure to .06 inches H_2O. or greater.

G. The remote manual control valve is also used to compensate for filter loading. Initially, it will be partially closed; and as the filter loads up causing a decrease in flow and negative penetration room pressure, the valve will gradually be opened so that the pressure drop across the filter/valve combination remains constant. By periodically adjusting the remote manual control valve to offset the effect of increased leakage and filter loading, the system characteristic remains constant.

H. The communicative paths between various parts of the penetration room are very large in comparison with the minute leakage that might exist. It therefore can be assumed that no pressure differentials exist in the room so that an instrument string sensing pressure at a single point can be used. Penetration Room pressure is displayed in the control room, and excessive and insufficient Penetration Room vacuum is annunciated by statalarms.

I. Fan status and radiation levels of filter effluent are displayed in the control room and excessive radiation is annunciated. Filter ΔP is displayed locally. Filter flow is displayed remotely, adjacent to the PR-13 and PR-17 remote control stations.

J. The system may be actuated by an operator during normal plant operation for testing.

2. PRESENTATION

2.1 Component Description - **Refer to: OC-PNS-PRV-1**

A. Penetration Room Ventilation Filters

1. The design basis for filtration was a requirement to remove 25%of the core iodine inventory. The 25% was derived using the standard assumption that during a Maximum Hypothetical Accident 50 percent of the halogens are released from the core and that 50 percent of the iodine released plates out within the Reactor Building.

2. To satisfy that requirement the filter train consists of three different type filters. A medium efficiency pre-filter and a high efficiency (HEPA) filter achieve particle filtration. An activated charcoal filter accomplishes absorption filtration.

   a) Prefilter:

   The prefilter consists of multiple horizontal tubular bags attached to a vertical metal plate header. The bags are made of ultra fine glass fibers and are supported so that adjacent bags do not touch and reduce the flow area. At the filter train design flow of 1000 cfm, the prefilter is operating at one half its rated flow.

   b) HEPA filter:

   The HEPA filter will intercept any particulates that pass through the prefilter. The filter consists of a single cell of fiberglass media mounted in a metal frame. The cell has face dimensions of 24 x 24 inches and a depth of 112 inches and is rated at 1150 scfm.

   c) Charcoal filter:

      1) Absorption filtration is accomplished by an activated charcoal filter. The filter consists of three horizontal removable type double tray carbon cells. Flow through the trays is essentially vertical. Each tray has a face area of 4.2 sq ft and a bed depth of 2 inches.

      2) At rated flow, the average face velocity is 40 ft/min and the residence time is 0.25 seconds. Each tray contains 40 lbs of carbon. The carbon is impregnated so that it will absorb methyl iodide as well as elemental iodine.

   d) External Carbon Sample Canisters:

      1) These 4 external sample canisters bypass the main carbon cells but the sample canisters contain the same thickness of absorbent carbon. Each sample canister line contains isolation valves to seal the line when the canister is removed for testing.

Exam Question Report

27-Jan-99

Question ID:	PNS280	Revision No:	0	Revision Date	10/29/1999
Question Description:	PNS280				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	PNS-RBP - Reactor Building Purge
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 14; LRO = 4; SRO = 4 Reference: OP/1102/14	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Following a LOCA, potential leakage past the purge system reactor building isolation valves is prevented from being released directly to the environment by which ONE of the following? (.25)

A) positive closing dampers in purge system ductwork.

B) I&E stroke testing PR valves and dampers each outage.

C) triple isolation valves in both intake/exhaust ductwork.

D) One (1) inch diameter bleed lines open-ended to the Penetration Room.

Answer

D

Lessons

ID	Description
PNS-RBP	REACTOR BUILDING PURGE SYSTEM (PNS-RBP)

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

BW/E08A1.2
PNS280 (Modified)

Plant conditions are as follows:

- A LOCA has occurred on Unit 1.
- RB pressure is 6 psig and decreasing.
- Purge Isolation Valves are leaking by.

Which ONE of the following will prevent release or RB contaminates to the atmosphere? (.25)

a. Automatic actions resulting from RIA-45, Vent Stack Gas monitor in Alert and/or Alarm.

b. Triple isolation valves in both intake/exhaust ductwork.

c. Positive closing dampers in purge system ductwork.

d. Penetration Room ventilation filters.

Answer

d.

Lessons

ID	Description
PNS-RBP	REACTOR BUILDING PURGE SYSTEM (PNS-RBP)

QUESTION # 20

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**2**	**2**
	K/A #	**000025 A1.02**	
	Importance Rating	**3.8**	**3.9**

Technical Reference(s): **AP/1700/26 Encl. 6.5**
TA-DHR p.#28_

Proposed references to be provided to applicants during examination: **AP/1700/26 Encl. 6.5**

Learning Objective: **TA-DHR OBJ. #14**

Question Source:
Bank # ________
Modified Bank # **TA-75**
New ________

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 20

Unit 1 plant conditions:

INITIAL CONDITIONS:

- MODE 1 at 100% power for 200 days
- Unit shutdown to MODE 6 to be performed

CURRENT CONDITIONS:

- Unit 1 has been in MODE 5 for 20 days
- Unit 1 experiences a "BLACKOUT"
- LT-5 = 50"

Which ONE of the following is the correct?

AP/1700/26, Loss of DHR indicates ______ hours to CORE UNCOVERY.

SEE ATTACHMENT

A. ~~4.8~~ 8.7

B. 6.9

C. 9.9

D. ~~13.2~~ 12.3

Use core Damage curve vice picking a wrong values core point on the core uncovery curve. @ LT-5 = 50"

1 POINT

QUESTION # 20

000025 A1.02 Both PRA 2-8-00
(2)

Using AP/1700/26, Loss of DHR.

A. Incorrect – Using Figure 2 CORE UNCOVERY (prior to RF). Using the lower value of LT-5 level at 50 inches indicates 4.8 hours

B. Correct - Using Figure 2 CORE UNCOVERY (prior to RF). Using the LT-5 level (50 inch curve) indicates 6.9 hours

C. Incorrect - Using Figure 5 CORE UNCOVERY (after RF). Using the LT-5 level 50" curve indicates 9.9 hours

D. Incorrect - Using Figure 5 CORE UNCOVERY (after RF). Using the LT-5 level 80" curve indicates 13.2 hours

LOSS OF DECAY HEAT REMOVAL
AP/1/A/1700/26

ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Assumptions

- Initial RCS/LPI Temperature = 140°F
- Upper SG Primary Handholes Removed To Vent RCS
- Worst Case Decay Heat (EOC)
- No Operator Action

Notes

1) "Prior To Refueling" curves assume all fuel assemblies in the core have experienced operation at power.

2) "After Refueling" curves assume approximately one third of the core is new fuel.

3) Curves for "LT-5=-18" are applicable to incidents where reactor vessel level has been reduced to the bottom of the hot leg. Example: LPI line break.

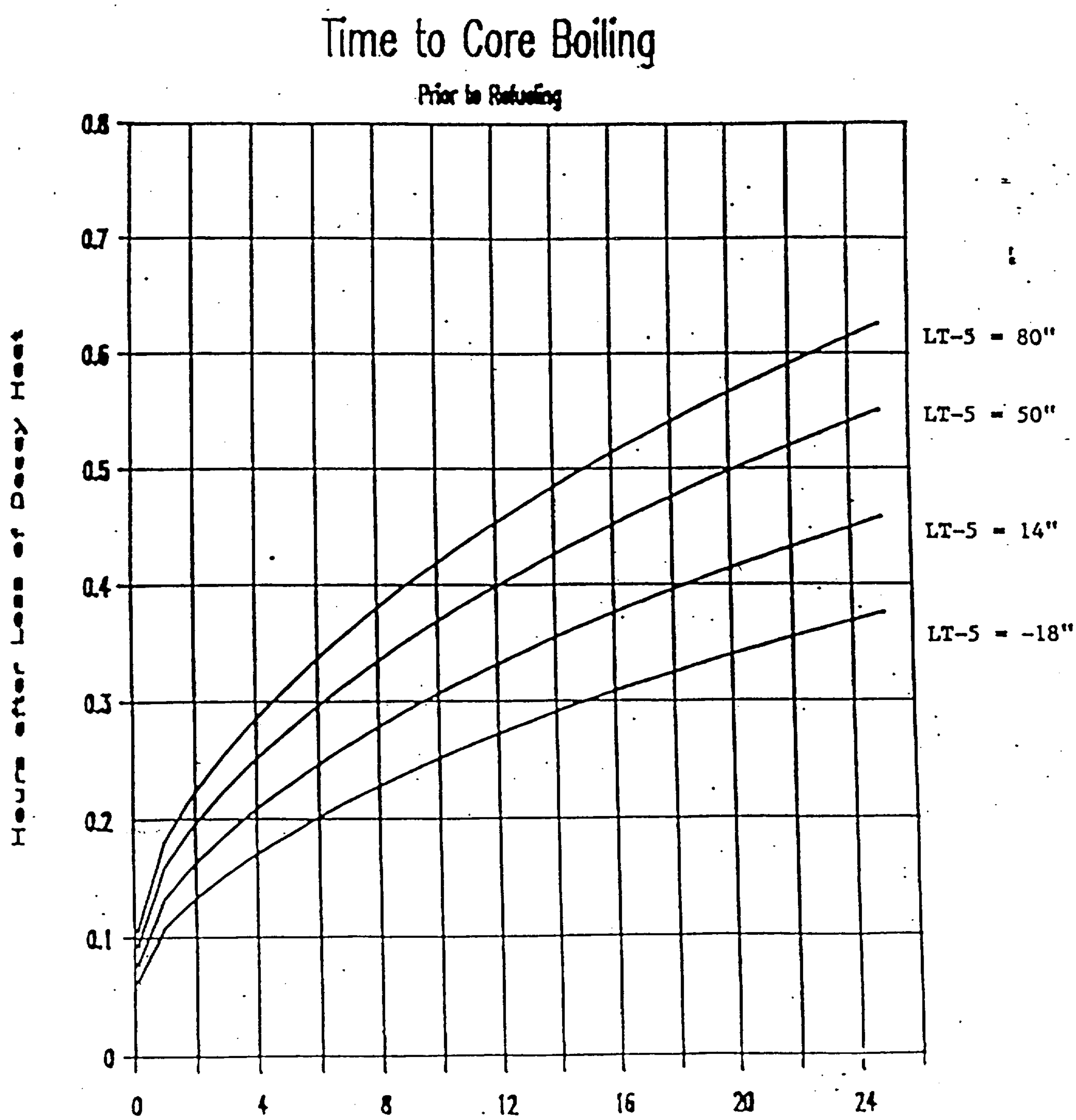
LOSS OF DECAY HEAT REMOVAL
AP/1/A/1700/26

ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Curves

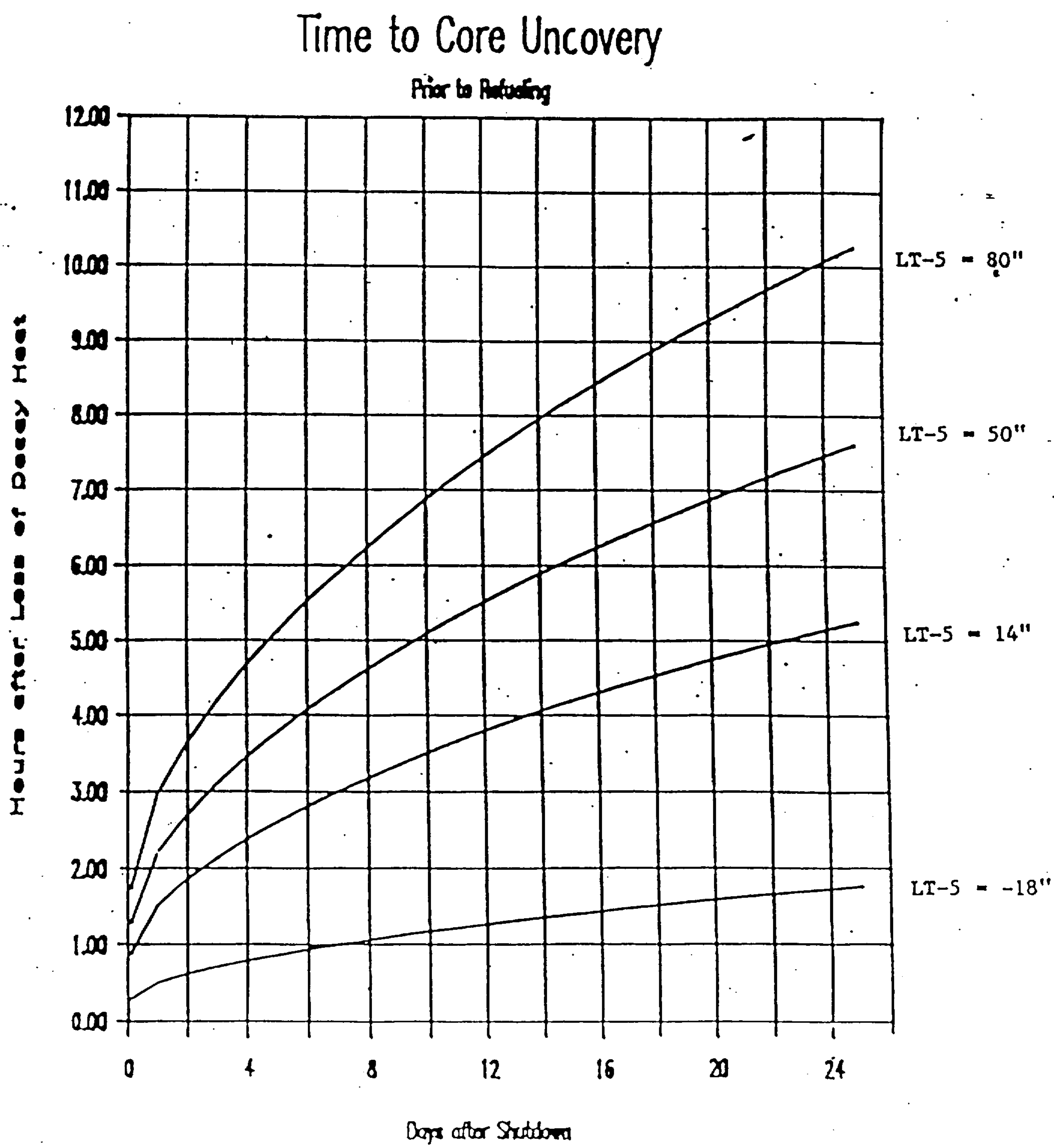
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 1



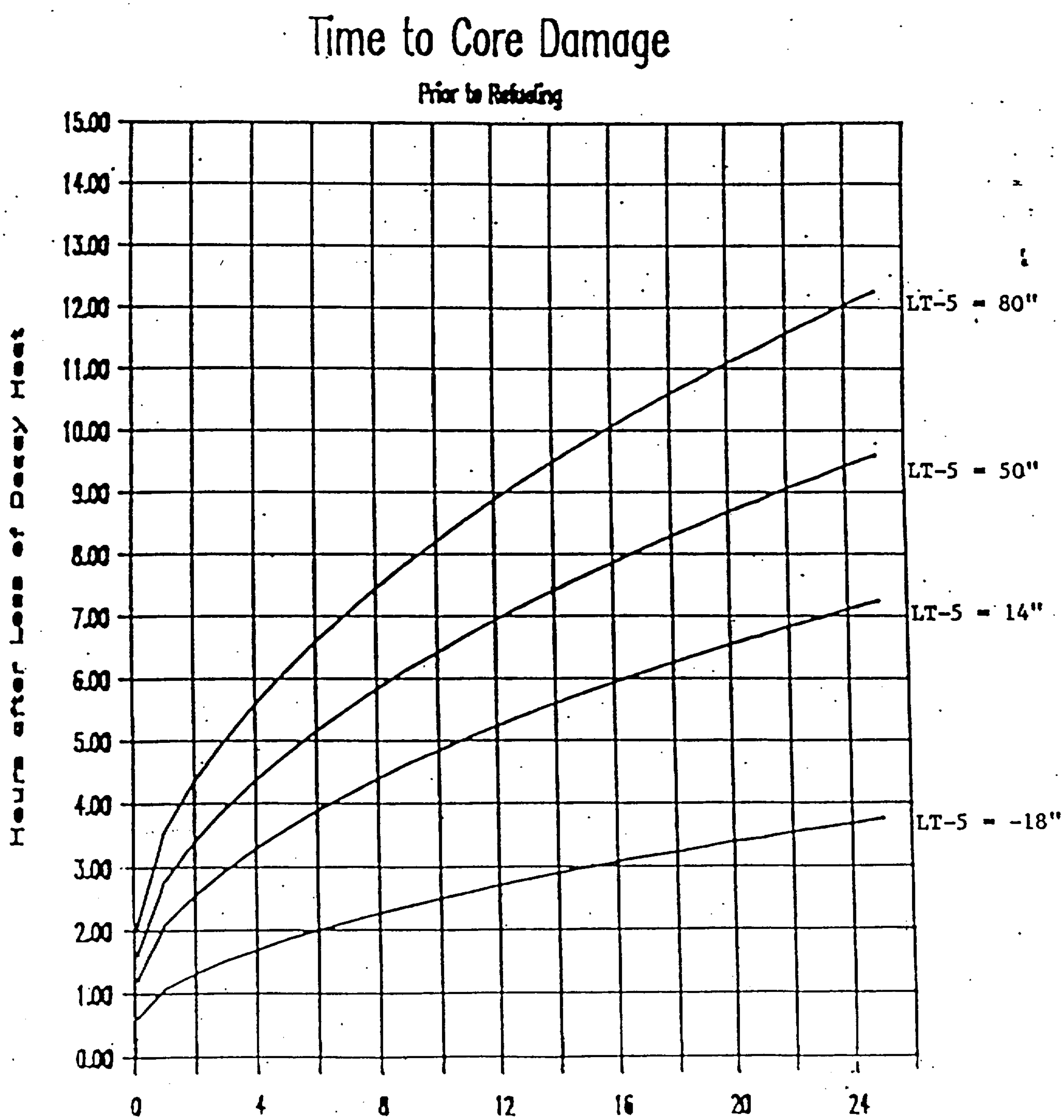
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 2



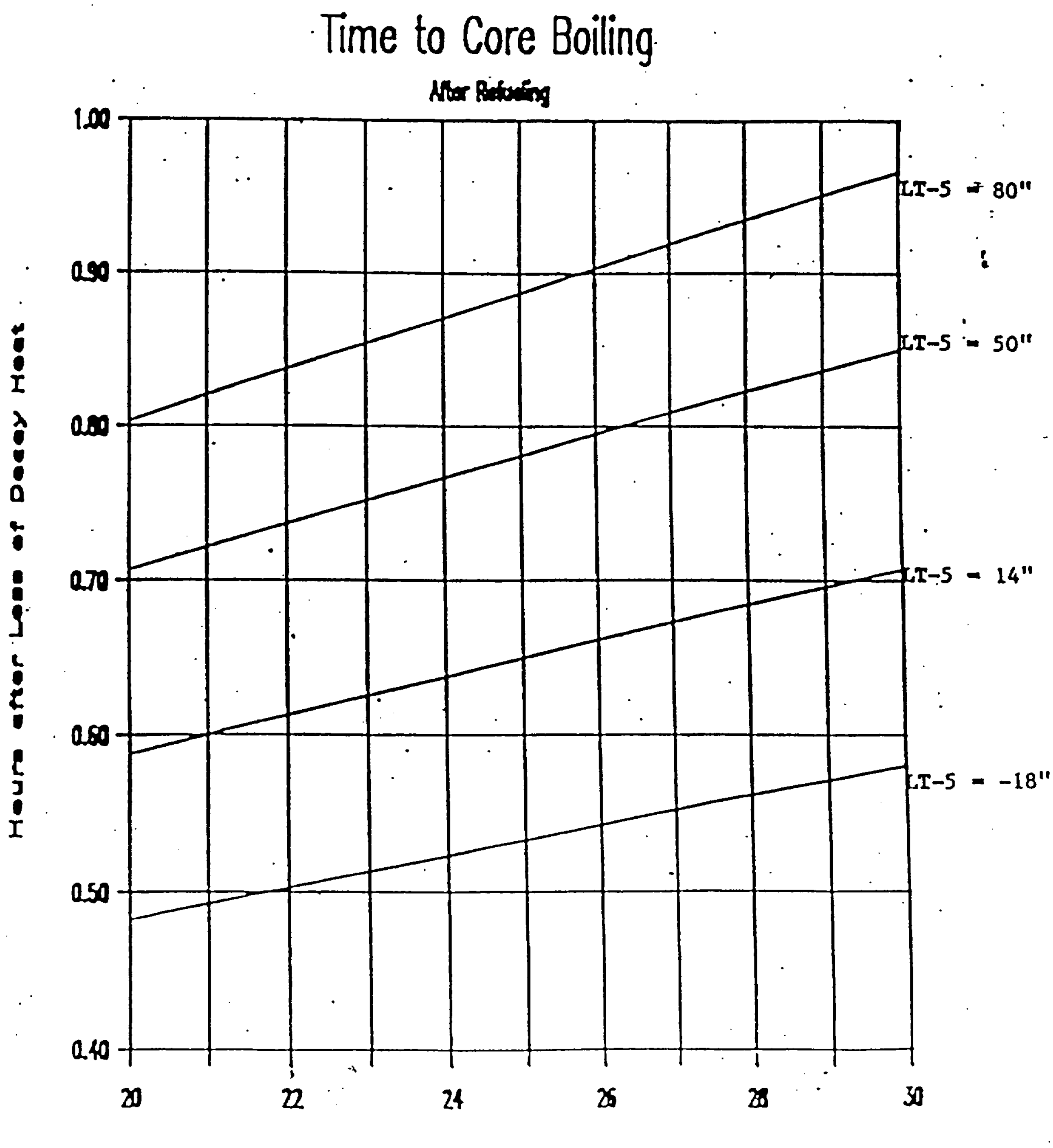
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 3



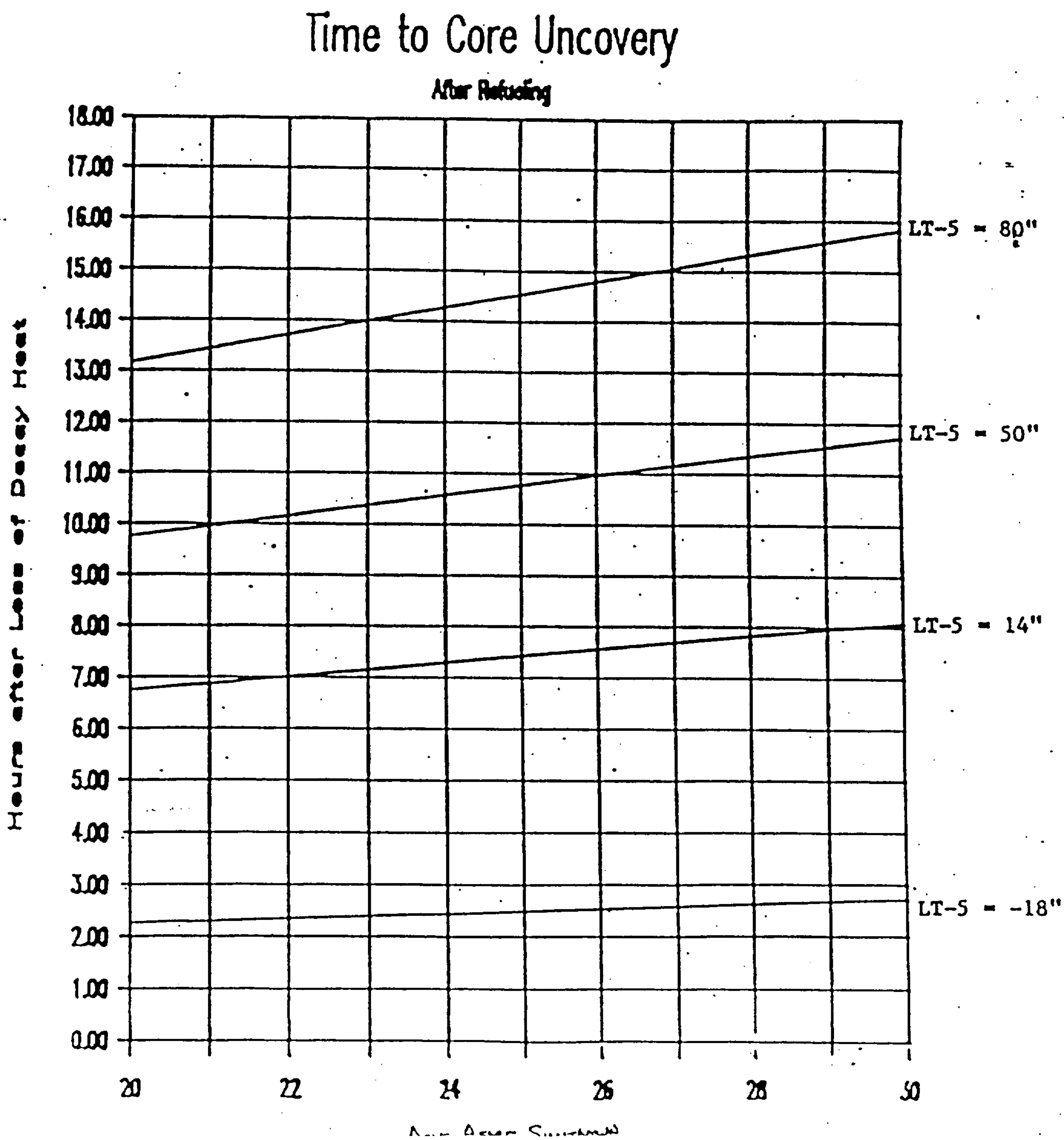
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 4



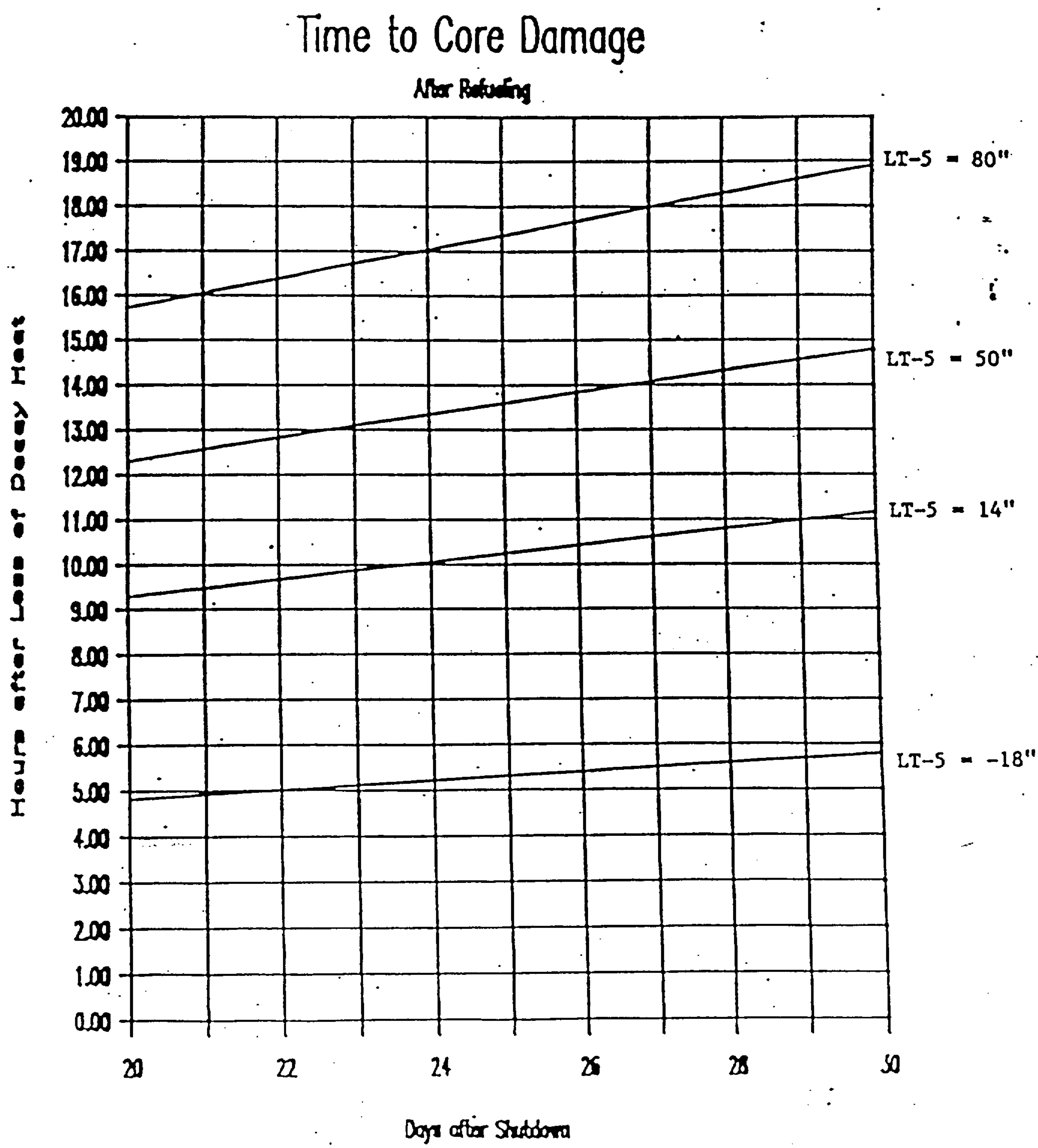
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 5



ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 6



11. Recognize that additional requirements must be satisfied in order to reduce RCS level below 50". (R11)

12. Recognize that PT/600/01 is the primary means that the operator uses to verify that DHR requirements are met. (R12)

13. Explain briefly why the 1LP-19 flange is installed prior to draining below 100" in the PZR on Unit 1. (R13)

14. Examine a given set of plant conditions, and use available operating procedures to properly mitigate an event involving the loss or degradation of decay heat removal. (R14)

15. For the Diablo Canyon, Oconee and Catawba events discussed in this lesson plan, be able to discuss the root cause for each event which led to a loss of decay heat removal situation. (R15)

3) If the LPI Pumps have been secured for reasons other than cavitation, the operator is directed to start a pump.

4) If RCS inventory is being lost, the operator is directed to identify the leak, either by reviewing evolutions in progress or by dispatching non-licensed operators, initiate makeup, and repair the leak when located.

5) If LPSW flow to the LPI Coolers is inadequate, the operator is referred to the Loss of LPSW AP and instructed to verify the LPSW lineup.

c) If these basic steps do not restore decay heat removal, then the operator is directed to provide other means of heat removal, depending on the shutdown configuration the plant is in. The following is a very general overview of the heat removal methods used in the various AP Cases:

1) Case A: the steam generators are fed from any available source. If this is not effective, then the RCS is allowed to heat up and repressurize, HPI is placed in service to control inventory, and the EOP is entered.

2) Case B: the RC loops are refilled and the RCS is repressurized. Natural circulation cooling is established. If DHR is ineffective, HPI and LPI may have to be operated in the ES mode, along with entry into the EOP.

3) Case C: Gravity feed from the BWST may be used to keep the core covered. LPI injection cooling may also be required.

4) Case D: SF Cooling Pump may be used to provide DHR. If inventory is lost, the LPI System may be run in recirc mode from the RBES.

2.4 Containment

A. Prior to reducing RCS level to < 50", the ability to achieve containment closure within calculated time must be ensured. AP/1700/26 (Loss of Decay Heat Removal) gives the necessary guidance to achieve containment closure. Enclosure 6.5 (Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability) may be used to ensure appropriate resources are allocated to attaining containment closure. It is the Operations Department's expectation that these curves <u>NOT</u> be interpolated. If a point falls between two curves, the curve denoting the next lower RCS level shall be used. This will ensure conservatism, by producing a result with a shorter calculated time.

Note: "Calculated time" will be determined by the Shift Work Manager. It will normally be the same as "Time to Core Boil" and will be listed on the

Plant Configuration Sheet. Actions required to be done inside the RB must be performed prior to reaching 110°F RB temperature.

B. RB Containment closure required due to a Loss of DHR capability.

1. Shift Manager notifies the following to perform required actions within the "calculated time":
   a) Mechanical Maintenance to close
      1) RB equipment hatch
         (a) If calculated time had been < 45 minutes, two personnel would have had to be stationed outside the Hatch ready to close it if necessary.
         (b) If calculated time is < 20 minutes, the equipment hatch should NOT be removed.
      2) All temporary RB penetration
      3) All SG secondary side penetrations within containment, unless SG secondary side isolation is provided outside the RB on MS lines and MFDW lines.
   b) I&E to close all RB instrument penetrations.
   c) Verify with Performance that all RB electrical penetrations are secure.
2. Evacuate RB of all non-essential personnel.
3. Secure RB purge
4. Send operator to close
   a) at least one door of RB personnel hatch.
   b) at least one door of RB emergency hatch.
5. Send operators to establish RB containment closure.
   a) Checklists for: East Pen. Rm, West Pen. Rm and Miscellaneous Valves
   b) If work is being performed on a valve on this checklist, the Operator can verify Containment Closure by verifying "inside Rx. Building" valves for that penetration are tagged per OP/1/A/1502/09 (Containment Closure Control).
6. Establish Containment Closure from the Control Room using Checklist in Enclosure 6.1 of AP/1700/26. (Loss of Decay Heat Removal)
7. Perform additional isolations as required per Enclosure 6.1 of AP/1700/26 (Loss of Decay Heat Removal)

Exam Question Report

27-Jan-99

Question ID:	TA075	Revision No:	0	Revision Date	10/29/1999
Question Description:	TA075				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	TA-DHR - Loss of Decay Heat Removal
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 14; SRO = 14 Reference: AP/1700/26	Max. Point Value:	0.25
		Passing Point Value:	0.25

27-Jan-99

Question

Plant conditions on Unit 1 are as follows:

• The reactor has been shutdown for 22 days following a 200 day run at full power to repair a S/G Tube Leak.

• The reactor is in Cold Shutdown.

• Both loops of LPI decay heat removal have just become inoperable.

• No other means of decay heat removal is available.

• LT-5 indicates 50".

Which ONE of the following correctly states the amount of time until Core Uncovery is expected to occur. (.25)

A) 5.0 hours

B) 7.2 hours

C) 9.1 hours

D) 10.2 hours

Answer

B
A. Incorrect - Uses figure 2 (Time to Core Uncovery - Prior to Refueling) at LT-5 14"
B. Correct - Use Figure 2 (Time to Core Uncovery, Prior to Refueling) at 50"
C. Incorrect - Uses figure 3 (Time To Core Damage - Prior to Refueling) at 50"
D. Incorrect - Uses figure 5 (Time to Core Uncovery - After Refueling) at 50"

Lessons

QUESTION # 21

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**2**	**2**
	K/A #	**000025 K3.02**	
	Importance Rating	**3.3**	**3.7**

Technical Reference(s): **PNS-LPI**
OP/1104/04 L/P_

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-LPI TI,#16**

Question Source:
Bank # ______
Modified Bank # **PNS-483**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 21

Which ONE of the following correctly completes the below statement?

The LPI system design pressure interlock ~~for LP-1 (~~LPI Return Block from RCS) is designed to. . .

A. automatically open LP-1 (LPI Return Block from RCS) when RC pressure is less than 400 psig.

B. ensure LP-1 (LPI Return Block from RCS) is closed to prevent over-pressurizing the LPI system during normal operation.

C. ensure LP-1 (LPI Return Block from RCS) and LP-2 (LPI Return) are NOT opened during LOCA.

D. prevent thermal pressurization between LP-1 (LPI Return Block from RCS) and LP-2 (LPI Return) after normal LPI DHR operations are complete.

1 POINT

QUESTION # 21

000025K3.02
(1)

A. Incorrect, - No automatic function to open LP-1. This is a manual operation when the LPI system is placed into the DHR mode of operation.

B. Correct, - The interlock prevents over-pressurization of down stream LPI system (Decay Heat Drop Line) piping when placing LPI in DHR mode of operation.

C. Incorrect, - The need to open LP-1 and LP-2 following a LOCA may be necessary to allow long term cooling.

D. Incorrect, - There is no provision nor design consideration for preventing a thermal pressure lock between LP-1 and LP-2 although this has actually occurred at Oconee. The space between these two valves is manually drained following isolation of normal DHR to prevent the thermal pressurization.

7. Compare the major design differences between the LPI systems at Oconee, to include the reasons for the differences. (R7) (PNSLPI007)
8. Explain what the High Pressure Mode of operation is, and what its purpose is, for Units 1 and 2. (R8) (PNSLPI008)
9. Explain when the High Pressure mode of operation is used on Units 1 and 2. (R9) (PNSLPI009)
10. State the purpose of the High Pressure Mode of operation on Units 1 and 2 and when it is used. (R10) (PNSLPI010)
11. Explain how overpressurization of the "B" LPI Header is prevented when Unit 1 or 2 is in the High Pressure Mode of Decay Heat Removal. (R11) (PNSLPI011)
12. Summarize the method by which overpressurization of the "B" LPI Header is prevented when Unit 1 or 2 is in the High Pressure Mode of Decay Heat Removal. (R12) (PNSLPI012)
13. Explain what the Switchover Mode of Operation is and why this mode is used on Units 1 and 2, but not Unit 3. (R13) (PNSLPI013)
14. Determine if the LPI System is properly aligned for Engineered Safeguards Standby Mode, when given various operating lineups. (R14) (PNSLPI014)
15. Describe the purpose and method used to vent the LPIPs.(R15) (PNSLPI015)
16. List several events which could lead to LPI System overpressurization during LPI System operation in the DHR mode. (R17) (PNSLPI017)
17. Given a list of malfunctions, determine which ones could lead to LPI System overpressurization during LPI System operation in the DHR mode. (R18) (PNSLPI018)
18. Recognize that with the LPI System in operation in the Normal Decay Heat Removal Mode, and with the HPI System secured, the normal purification demineralizers can be placed in service through the LPI System for RCS chemistry control. (R19) (PNSLPI019)
19. Identify the method used to provide RCS chemistry control when the LPI System is in the Normal Decay Heat Removal Mode and the HPI System is secured. (R20) (PNSLPI020)
20. Describe briefly how RV level can be raised or lowered using the LPI system. (R21) (PNSLPI021)

5. Vent the "A" and "B" LPI pumps.

   Associated PT - LPI Pump Vent - PT/0203/011

   a) Purpose: To vent the casings of non-operating LPI pumps to insure that these pumps can fulfill their intended function in case of a loss of coolant accident.

   b) Test Method: Vent the LPI pumps through their vent valves until all air is removed.

      NOTE: The suction lines to the LPI Pumps also be vented when the Pumps are vented to ensure that air-binding of the pumps does not occur.

6. Place the Low Flow Alarm Block Switch to the "Blocked" position.

7. Remove the "White Tags" from and close the breakers to LP-19 and LP-20. LPI is aligned for ES Standby Mode.

2.3 LPI System Interlocks

A. BWST Heaters will interlock off at 7 feet decreasing in the BWST.

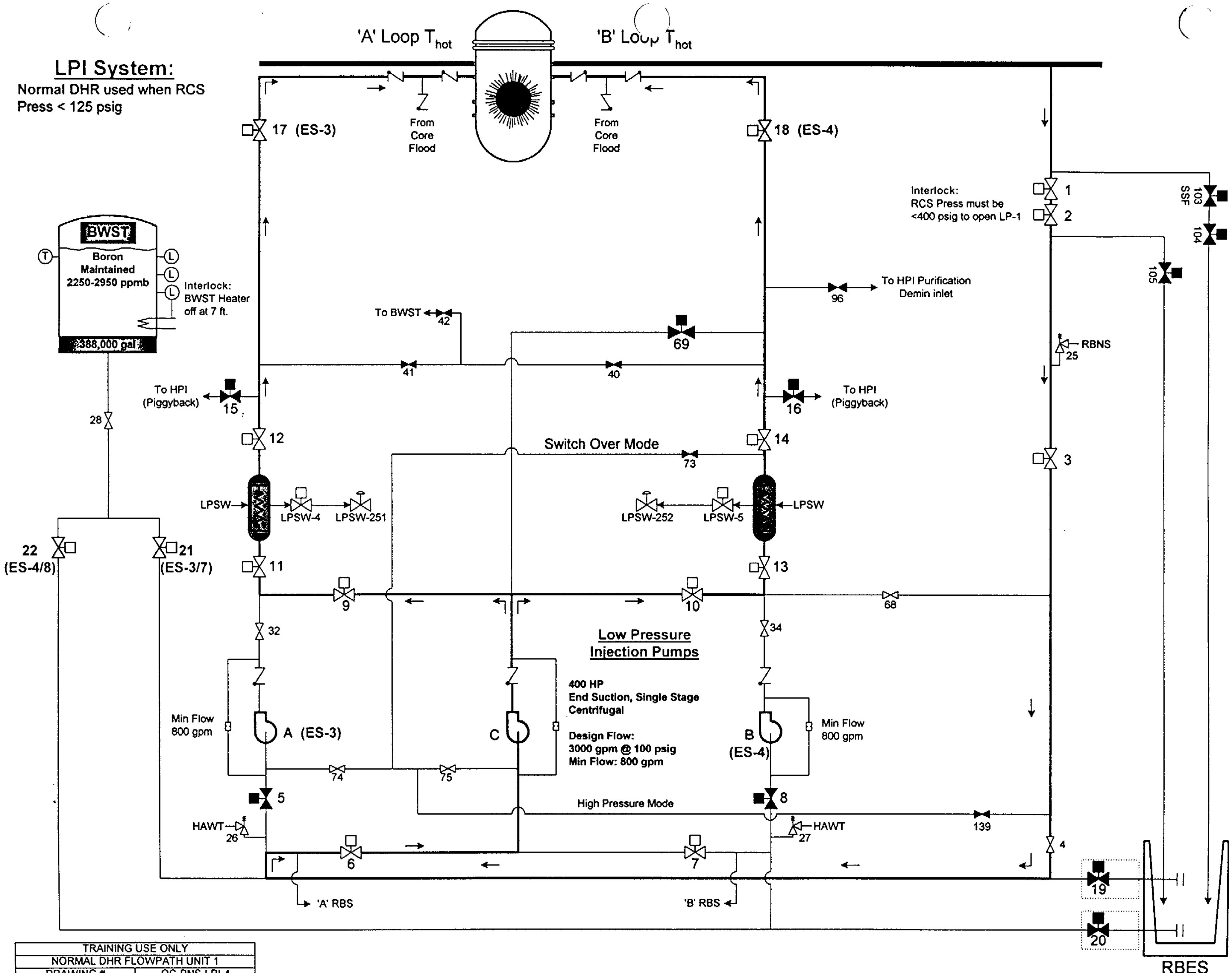
B. The interlock between RCS WR Pressure and LP-1 and LP-2 has been removed. It will be reinstalled at a later date when safety-grade pressure switches are obtained. Until that time, the opening of LP-1 and LP-2 will be administratively controlled to prevent overpressurization of the LPI System.

2.4 LPI System Overpressure Protection Design Basis

A. During LPI system operation in the DHR mode, the following events could lead to LPI system overpressurization:

   1. Erroneous actuation of HPI.
   2. Erroneous opening of CFT discharge valve.
   3. Erroneous addition of N2 to PZR.
   4. HP-120 fails open.
   5. All PZR heaters erroneously energized.
   6. Loss of DHR.
   7. Thermal expansion of RCS after starting a RCP as a result of stored thermal energy in the OTSG.

B. Currently (1-18-90), the design and licensing basis for LPI system overpressure protection while in the DHR mode is manual operator action. An evaluation has not been documented which demonstrates the acceptability of manual operator action to provide LPI overpressure protection for the above scenarios.



TRAINING USE ONLY	
NORMAL DHR FLOWPATH UNIT 1	
DRAWING #	OC-PNS-LPI-4
DRAWN BY: RJL	DATE: 7/28/99
REFERENCE:	OFD-102A-1.1, 2, 3
APPROVED BY:	Signature On File

2.12 Do <u>NOT</u> remove an LPI pump from service unless two operable LPI pumps are available. This prevents single pump failure causing loss of decay heat removal capability.

2.13 1LP-20 (1B RX BLDG SUCTION) can be cycled only during one of the following conditions:

- Defueled Maintenance Work Window.

  or

- LPI System in ES lineup and 1LP-20 is being cycled by an approved Procedure.

  or

- LPI System in LPI NORMAL MODE (1 LPI pump & ≈ 1500 gpm LPI flow to both LPI Coolers) and 1LP-20 is being cycled by an approved Procedure.

2.14 1LP-19 (1A RX BLDG SUCTION) can be cycled only during one of the following conditions:

- Defueled Maintenance Work Window.

  or

- LPI System in ES lineup and 1LP-19 is being cycled by an approved Procedure.

  or

- 1LP-19 may be opened to supply alternate LPI Pump suction path through 1LP-19/1LP-105.

2.15 Water addition to Spent Fuel Pool, 1 BWST, or 2 BWST shall <u>NOT</u> be made when BWST is in recirc through purification loop. This is to prevent BWST overflow.

2.16 1LP-1 (LPI RETURN BLOCK FROM RCS) is interlocked to prevent opening until RCS pressure < 400 psig.

2.17 If LPSW flow exceeds ≈ 5900 gpm in decay heat cooler, 1LPSW-251/252 (LPI COOLER OUTLET CONTROL) will automatically throttle LPSW flow to ≈ 5200 gpm to prevent cooler damage.

2.18 BWST boron concentration shall be maintained to comply with COLR.

2.19 Prior to and during draining of tube side (LPI) of LPI Coolers, shell side (LPSW) drains on affected cooler should be white tagged closed, to prevent an inadvertent release to Lake Keowee. {1}

27-Jan-99

Exam Question Report

Question ID:	PNS483	Revision No:	0	Revision Date	10/29/1999
Question Description:	PNS483				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	PNS-LPI - Low Pressure Injection System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 16; SRO = 16	Max. Point Value:	0.25
		Passing Point Value:	0.25

Question

The interlock between RCS pressure and LP-1 (LPI Return Block from RCS) and LP-2 (LPI Return): (Choose ONE.)(.25)

A) Opens LP-2 when pressure is less than 450 psig, and opens LP-1 when pressure is less than 400 psig.

B) Prevents overpressurizing the LPI system during normal operation.

C) Ensures LP-1 and LP-2 do <u>NOT</u> open, following a LOCA.

D) Ensures LP-2 is opened before LP-1.

Answer

B

Lessons

ID	Description
PNS-LPI	Low Pressure Injection System (PNS-LPI)

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

QUESTION # 22

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**2**	**2**
K/A #	**000032 K3.01**	
Importance Rating	**3.2**	**3.6**

Technical Reference(s): **IC-NI p.#16**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-NI OBJ. #11**

Question Source:
- Bank # ______
- Modified Bank # **IC-468**
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 22

Unit 3 plant conditions:

- Unit startup in progress
- Operator is withdrawing Group 6 control rod bank
- Diamond is in MANUAL

- Time = 1000:00
  - WR Counts:

    NI-1 = 2.0 e2
    NI-2 = 2.2 e2
    NI-3 = 2.1 e2
    NI-4 = 2.0 e2

- Time = 1001:00
  - WR Counts:

    NI-1 = 3.0 e4
    NI-2 = 4.7 e2
    NI-3 = 4.5 e2
    NI-4 = 3.9 e2

Which ONE of the following is correct?

Group 6 Control Rod withdrawal will…

A. not stop because the Diamond is in MANUAL.

Automatically stop
B. ~~stop due to the High Startup Rate Rod Withdrawal Inhibit.~~

because only
C. not stop ~~as~~ NI-1 has failed to mid-scale ~~and the High Startup Rate Rod Withdrawal Inhibit circuitry will not respond to the failed signal.~~

??
D. stop only if the operator positions the Diamond Control ("joy stick") to "neutral" ~~because at least two WR NI's are required to actuate the High Startup Rate Rod Withdrawal Inhibit.~~

1 POINT

QUESTION # 22

000032 K3.01 Both PRA 2-8-00
(2)

A. Incorrect – the High Startup Rate Rod Withdrawal Inhibit will stop rod motion if the Daimond is in automatic or manual.

B. Correct – NI-1 has increased (spiked or failed) to 30000cps which exceeds the 2 dpm setpoint of the High Startup Rate Rod Withdrawal Inhibit. Only 1 NI is required to actuate the High Startup Rate Rod Withdrawal Inhibit circuitry.

C. Incorrect - The High Startup Rate Rod Withdrawal Inhibit will respond to high SUR even if it is caused by a failing NI signal.

D. Incorrect – Placing the Diamond Control ("joy stick") to neutral will stop rod motion, but high SUR will also stop rod motion. Only 1 of the 4 NI's is required to actuate the High Startup Rate Rod Withdrawal Inhibit.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

At the completion of this lecture the student will be able to explain the principles of operation, the output functions and the components used in operating the out of core Nuclear Instruments.

ENABLING OBJECTIVES

1. Describe the purposes of Nuclear Instrumentation. (R1)
2. Explain the TWO neutron reactions utilized at ONS for detection of thermal neutrons, and why these reactions with "target nuclei" are necessary in order to monitor neutron flux. (R2)
3. Identify the TWO adverse environmental conditions the Gamma-Metrics NIs are designed to withstand. (R3)
4. Discuss the ranges of the Source and Wide Range detectors and the location of the detectors with respect to the core. (R4)
5. Explain the type of detector utilized by the Source and Wide Range, including how these detectors operate. (R5)
6. Explain how sensitivity is increased in the Source Range Instruments. (R6)
7. Identify the fact that at ≈ 4E-3% power, the Wide Range circuitry "swaps over" from a "pulse counting" mode to the Campbelling mode of operation in order to derive Reactor power and explain why this is necessary. (R7)
8. Explain why the Source range detector is NOT de-energized during power operation when the Source Range is overranged. (R8)
9. Discuss the fact that alpha, gamma, and neutron pulses are produced in an Uncompensated Fission chamber and explain how and why the circuitry distinguishes between these sources. (R9)
10. Identify the outputs from the Source and Wide Range Signal Processor. (R10)
11. Concerning the High Startup Rate Rod Withdrawal Inhibit:
    - 11.1 List the source(s) which may provide input to the Inhibit circuitry. (R11)
    - 11.2 The setpoint associated with the Inhibit, including the point at which the signal resets. (R12)
    - 11.3 The function provided by the Inhibit. (R13)

3) An auctioneer circuit in the Signal Processor selects between the two Wide Range Output signals (pulse-counting vs. Campbelling Mode) at the appropriate time ($\approx 4 \times 10^{-3}$ % power).

4) Provides output (CPS, % power, DPM) to the following :

   (a) Digital indicator on front of Wide Range Monitor section of respective RPS cabinet (A1, B1,C1,D1). Reads whatever is selected by selector switch.

   (b) Computer

   (c) Dixon Indicators located on UB1 (frontboard in Control Room). (Refer to Handout OC-IC-NI-6)

      (1) Dixons will indicate power level and Rate of Change (-1 to +7 DPM) for NI-1,2,3,& 4 SR and WR.

      (2) NI-1,2,3,& 4 SR meters are grouped together and NI-12,3,& 4 WR meters are grouped together.

      (3) NI-1,2,3,& 4 Wide Range meters are designated as PAM indication.

   (d) The signal from NI-3 WR is also fed to a new safety related chart recorder on VB1

   (e) Chessell Recorder on UB1

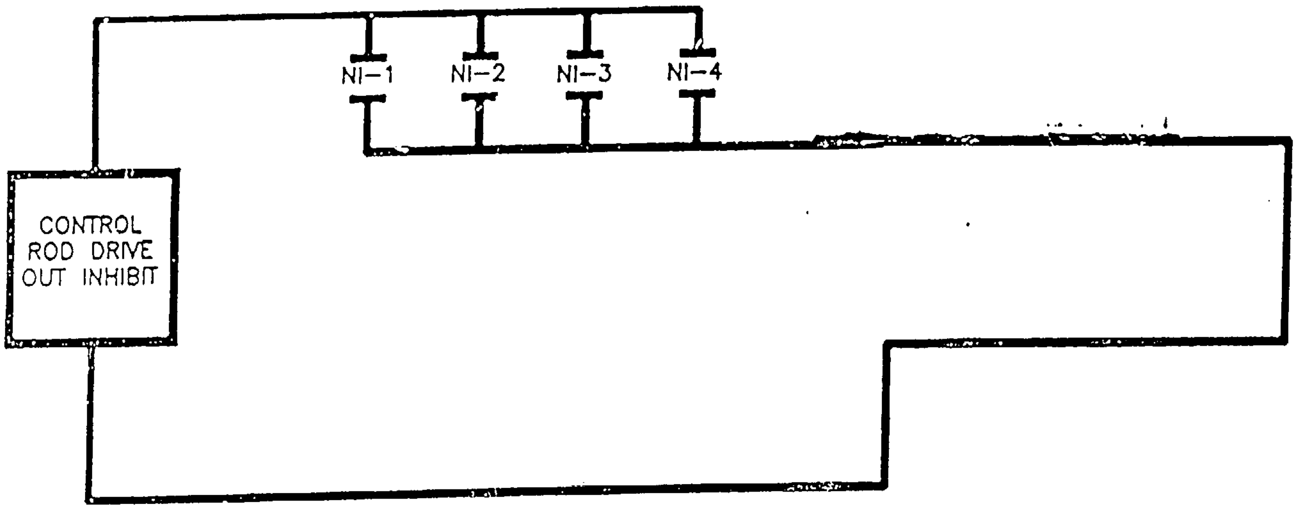
   (f) scalar output (random pulse) to Refueling Booth area

5) Provides for the following control functions:

   (a) High Startup Rate Rod Withdrawal Inhibit which stops outward rod motion in the event a SUR $\geq$ 2 DPM. (resets at .5 DPM.) (***Refer to Handout OC-IC-NI-7***)

   - this will occur anytime regardless of power level
   - fed from **any Wide Range SUR**
   - causes statalarm
   - gives red indicator light on Wide Range Monitor section of respective RPS cabinet (A1, B1, C1, D1)

   (b) Input to Rx Building Evacuation Alarm **(from all four SR *level* signals)**. Alarm is adjustable by I&E and is normally set at 1/2 decade above background. (Also gives red indicator light on Wide Range Monitor section of respective RPS cabinet (A1, B1,C1,D1)).



CONTACTS FOR NI-1 THRU NI-4 ARE NORMALLY OPEN. THEY WILL CLOSE IF A SUR OF 2 DPM OR GREATER OCCURS ON THE ASSOCIATED WIDE ~~OR SOURCE~~ RANGE CHANNEL. RESET AT .5 DPM.

Typo

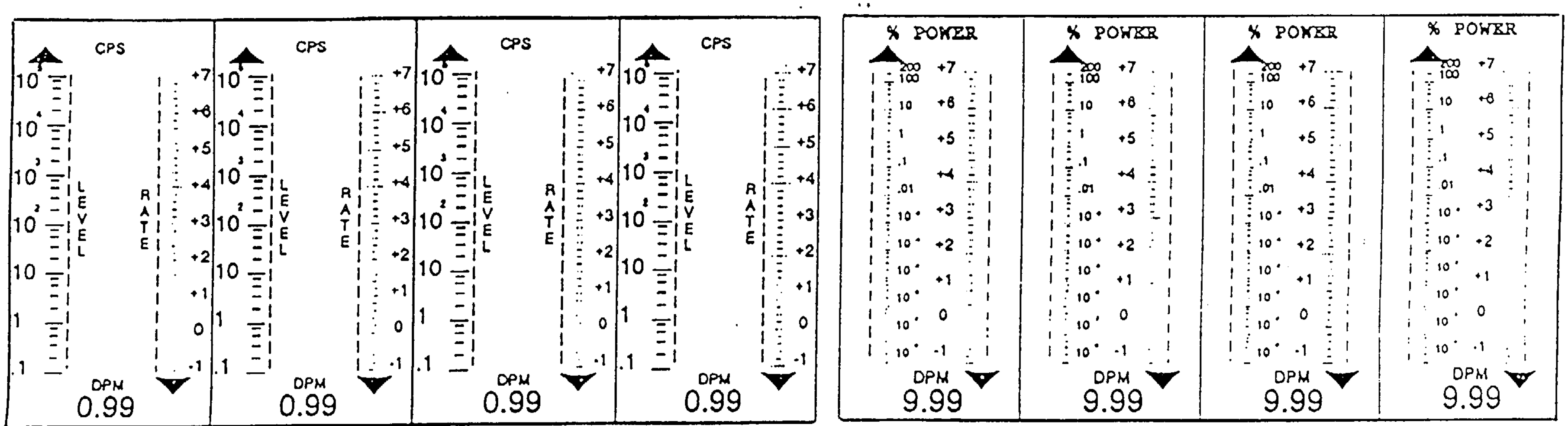
TITLE	NOTES	ID NO	DATE
NUCLEAR INSTRUMENTATION	HIGH STARTUP RATE CRD OUT INHIBIT (TEMPORARY DRAWING)	OC-IC-NI-7	8-15-95
		REF DPC I&C MAINT.	
		DRN BY JLC	APR BY
		TRAINING USE ONLY	

SOURCE RANGE

NI-1 NI-2 NI-3 NI-4

WIDE RANGE

NI-1 NI-2 NI-3 NI-4



PAM PAM PAM PAM

Exam Question Report

27-Jan-99

Question ID:	IC468	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC468				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-NI - Nuclear Instrumentation
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 11,12; SRO = 11, 12	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Which ONE of the following represents an input to the High Startup Rate Rod Withdrawal Inhibit circuitry, including the setpoint at which it actuates? (.25)

A) Wide Range NI-1 startup rate at 2 dpm.

B) Source Range NI-2 startup rate at 2 dpm.

C) Wide Range NI-3 startup rate at 3 dpm.

D) Source Range NI-4 startup rate at 3 dpm.

Answer

A
A. Correct
B. Incorrect- not SR SUR
C. Incorrect- stpt is 2 dpm
D. Incorrect- not SR SUR and stpt is 2 dpm

Lessons

ID	Description
IC-NI	NUCLEAR INSTRUMENTATION (OP-OC-IC-NI)

Enabling Objectives

ID	Description

Referenced Documents

QUESTION # 23

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**2**	**2**
K/A #	**000033 K3.01**	
Importance Rating	**3.2**	**3.6**

Technical Reference(s): **IC-NI, STG-ICS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-NI #26**

Question Source:
- Bank # **IC-618**
- Modified Bank # ______
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 23

Unit 1 plant conditions:

- Reactor power = 44%
- Increasing power at 3% per hour to 75% power
- ICS in Automatic

Which ONE of the following explains the response if the NI-5 Power Range **UPPER** detector FAILS LOW?

Indicated RPS Ch "A" Reactor Imbalance becomes...

A. negative and ICS does not respond to this failure.

B. positive and ICS does not respond to this failure.

C. negative and ICS withdraws control rods to compensate for failure.

D. positive and ICS withdraws control rods to compensate for failure.

1 POINT

QUESTION # 23

000033 K3.01

(2)

A. Correct - (NI-5 feeds "A" RPS) Imbalance = Top minus Bottom, a negative number will be the difference from the dif. circuitry. Median select will discriminate from NI-5 as a controlling NI.

B. Incorrect - If only the lower for NI-5 detector failed this would be correct.

C. Incorrect - This would be correct if ICS did not process median select from NI-5, 6, 9.

D. Incorrect – If the only lower detector for NI-5 failed imbalance would be positive. Input to ICS is NI-5, 6 or 9 that is median selected.

18. Describe the purpose for the STAR Module- Difference Amp. in the Power Range instrument channels section of the Reactor Protective System and list the outputs from this amp. (R20)

19. List the parameter that is used when calibrating the Delta Flux signal associated with the Power Range STAR Module -Difference amplifier. (R21)

20. Describe the purpose and operation of the STAR Module-Star Processor in the Power Range instrument channels section of the Reactor Protective System. (R22)

21. Recognize that the portions of the Reactor Protection System associated with the Flux/Flow/Imbalance Trip Bistable are now part of the STAR Module and explain the purpose behind this modification. (R23)

22. List the inputs to the Chessell 320 power recorder and explain how each input is selected . (R24)

23. Concerning SOER 90-3 (NI Miscalibration), discuss the problems associated with a miscalibrated NI. (R25)

24. Recognize that alternate indications of Reactor Power level are available to the Operator and be able to list several examples of these alternate indications. (R26)

25. When given a copy of Tech. Specs., be able to determine operability requirements for NIs. (R27)

26. Discuss the effects of a loss of power on:
    a. Nuclear Instruments (R28)

    b. Chessel Recorder (R29)

27. Discuss actions to take upon loss of any or all NIs. (R30)

a) NI-9 is located in the same approximate position as NI-5.

b) NI-9, NI-5, or NI-6 can supply inputs to the ICS. (***REFER to OC-IC-NI-9,10,11***).

1) (Unit 1 & 3) ICS "middle select" function automatically selects the middle signal of the three choices

2) (Unit 2) SASS monitors input from NI-9 and NI-5 to provide the ICS input automatically. Upon ICS upgrade, this goes away.

c) Axial location

1) The axial centerline of the top half of the core is aligned with the axial centerline of the top 70" detector. (***Refer to OC-IC-NI-12***)

2) The axial centerline of the bottom half of the core is aligned with the axial centerline of the bottom 70" detector.

3) <u>This arrangement allows for calculation of core imbalance.</u>

5. The detector is an uncompensated ion chamber(***REFER to OC-IC-NI-12)***

a) One electrode inside a single chamber.

b) Inside of chamber is lined with Boron 10.

1) B^{10} used for same reaction to produce ions.

$_5B^{10} + {_0N^1} \Rightarrow {_2a^4} + {_3Li^7} + \approx 2.30$ MeV

c) Chamber is filled with Nitrogen gas.

d) Ionization from neutron and gamma reactions results in current output proportional to reactor power.

e) <u>No gamma compensation is needed in the power range.</u>

1) Fission gammas are proportional to reactor power.

2) Gamma flux is relatively insignificant when compared to the neutron flux.

6. Power range uses two detectors per channel.

a) NI-9 uses three detectors on Unit One only.

7. Detector Power Supply

a) The detector is powered from the RPS cabinet ± 15V power supply (KVIA,KVIB,KVIC, or KVID).

b) The voltage supply to the detector is stepped up to 500V by a regulated DC to DC converter:

1) An oscillator and power amplifier provide a sinusoidal input (AC) to a step-up transformer.

5) The associated RPS trip bistables (NI-5, 6, 7, & 8)
   (a) High Flux trip bistable
   (b) Flux/Pump trip bistable
   (c) STAR Module-Flux/Flow/Imbalance trip bistable

10. The STAR Module-Difference Amplifier also receives a signal from both the upper and lower linear amplifiers.
    a) Determines the difference between the two signals (Power Imbalance)
       1) Top detector - bottom detector = delta flux.
       2) delta flux x % Rx. Power = imbalance
       3) This signal is calibrated to agree with incore imbalance from incore instruments, which are more accurate (at steady power level). This calibration includes a "fudge factor" to ensure the NI calculated imbalance is more conservative than the incore imbalance. This calibration is done at the same time the Total Flux is calibrated to Thermal Power Best.
    b) Provides a power imbalance signal which is sent to the following places:
       1) Imbalance meters in the Control Room
       2) Associated RPS cabinet
       3) Computer terminals
       4) Statalarm
       5) The STAR Module-Star Processor

11. The STAR Module-Star Processor has two inputs and one output.
    a) Inputs:
       1) Imbalance signal out of the difference amplifier.
       2) Reactor coolant system flow.
    b) Compares the reactor imbalance and the RCS flow:
       1) Using these inputs it generates the maximum allowable flux for these conditions.
    c) Signal is supplied to the STAR Module-Flux/Flow/Imbalance trip bistable where it is compared with total flux to provide the trip signal.

12. STAR Module (<u>All three units</u>)

   a) A Framatome digital based module trip string that replaces portions of the existing Analog power range RPS channel. The existing system has generated numerous Flux/Flow/Imbalance trips which occurred randomly and for which the cause could not be determined. The existing system is also hard to troubleshoot and since it is becoming obsolete, replacement parts are difficult to locate.

      1) The STAR modules utilize modern technology to increase the reliability of the RPS.

      2) The STAR modules also have the capability of storing instantaneous process parameter values at the moment of a trip, thus providing "records" for fault/root cause determination.

   b) The STAR module replaces the following components of the existing RPS:

      1) Flux/Flow/Imbalance Trip Bistable

      2) Square Root Extractors (RCS Flow)

      3) Difference Amplifiers

   c) The STAR modules are completely installed (done Nov. 97').

B. Wide Range NIs

1. At least two Wide Range NIs are required to be operable, unless at least two Power Range NIs are greater than 10%. (TS Table 3.5.1-1)
2. During startup when the Wide Range instruments come on scale, the overlap between the Wide Range and the Source Range instruments shall not be less than one decade. (TS 3.5.1.5)
3. A Tech. Spec. interpretation has been written due to the replacement of the SR and IR instruments with the Gamma-Metrics SR and WR instruments. This interpretation basically states that any Gamma-Metric SR instrument is equivalent to a SR instrument and any Gamma-Metric WR instrument is equivalent to an IR instrument. (NOTE: Exception is when refueling and NI-1 and 2 should be used).

C. Power Range NIs

1. At least three Power Range instruments shall be operable. Guidance is given for use of dummy bistables and for manual bypass of RPS channels fed by the Power Range instruments in Tech Specs. (TS Table 3.5.1-1)

2.5 Unusual Operating Conditions

A. Power supply failure to NIs

1. Causes complete loss of output signal.

B. Power supply failure to Chessell 320 recorder

1. Computer alarms fail high
2. Statalarms fail in NO alarm state

C. Erratic high levels seen in the source range or wide range while attempting to regain control after an accident could be indication of voiding the vessel downcomer.

1. Explained in detail in the Accident Mitigation Lesson Plan.

D. Operators are required to initiate a manual reactor trip if all Wide and Power Range instrument channels fail at power and the reactor has not tripped. (OMP 2-1)

Exam Question Report

27-Jan-99

Question ID:	IC618	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC618				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-NI - Nuclear Instrumentation
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = T1; SRO = T1	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Unit 1 is operating with the following conditions:

- Unit at 100% reactor power
- ICS in Automatic

Which ONE of the following explains the response if the 1NI-5 Power Range upper detector FAILS LOW? (.25)

Indicated RPS Ch "A" Reactor Imbalance becomes ...

A) positive and ICS does not respond.

B) negative and ICS does not respond.

C) positive and ICS withdraws control rods.

D) negative and ICS withdraws control rods.

Answer

B

B. Correct - (NI-5 feeds "A" RPS) Imbalance = Top minus bottom (0% - 50%) therefore a negative number.

A. Incorrect - See "B" above (not a positive number)

C. Incorrect - Input to ICS is NI-5, 6 or 9 that is medium select.

D. Incorrect - Same as "C"

Lessons

ID	Description
IC-NI	NUCLEAR INSTRUMENTATION (OP-OC-IC-NI)

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

QUESTION # 28

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**3**	**3**
K/A #	**BW/E14 K2.2**	
Importance Rating	**3.8**	**3.8**

Technical Reference(s): **EOP Encl. 7.6**
EAP-E22_

Proposed references to be provided to applicants during examination: **EOP Encl. 7.6**

Learning Objective: **EAP-E22 OBJ. #21**

Question Source:
Bank # ______
Modified Bank # __X__
New ______

Question History:
Previous NRC Exam ___X__**(1998 #38 Modified)**
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 28

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Time = 0915
- A loss of offsite power
- Rx trip occurs from 100% power
- MFBs are being supplied via CT-4
- HPI Cooling was initiated and the pressurizer is water solid
- EFDW has been aligned from Unit 1

CURRENT CONDITIONS:

- Time = 0945
- The operators are in the process of recovering from HPI cooling and have established EFDW flow at a rate of 190 gpm per SG.

Which ONE of the following is correct?

SEE ATTACHMENT WRONG CURVE

The RCS will...

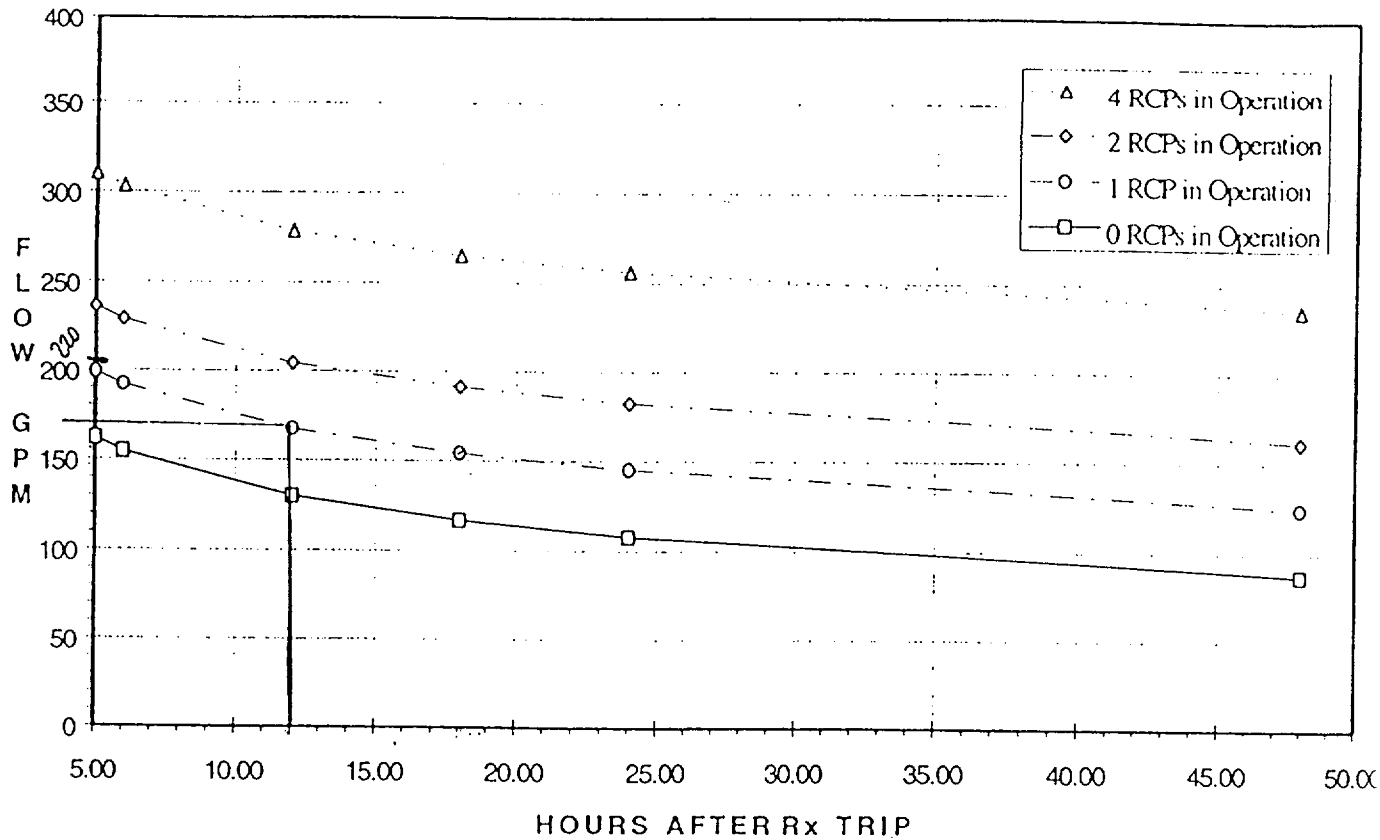
A. cooldown resulting in a decrease in SCM.

B. gradually heat up with a reduction in SCM.

C. continue to gradually cool with an increase in SCM.

D. remain at the same temperature, pressure, and SCM.

Total Feedwater Flow Required To Match NSSS Heat



100%
220 (2 RCP's)
170 (1 RCP)

70%
154
119

WRONG CURVE

1 POINT

QUESTION # 28

BW/E14 K2.2 Both PRA 05-11-00 (bank question Exam 7) (GTH)

A. Correct – Cooldown will increase along with a decrease in RC pressure and decrease SCM.

B. Incorrect – Heatup will not occur with the indicated EFDW flow

C. Incorrect – Cooldown will make pressure decrease, decreasing SCM.

D. Incorrect – RCS pressure and temperature will decrease with the indicated EFDW flow.

18. Briefly explain the CAUTION concerning restarting a RCP if the Boiler Condenser mode of SG heat transfer has taken place. (R17)

19. Explain the advantage of bumping a RCP in the loop with the higher Hot Leg level. (R14)

20. Explain why restarting a RCP regardless of subcooling margin is allowed, if one hour has elapsed since the reactor has tripped. (R15)

21. Given a set of plant conditions determine the correct actions to take using Section 502 of the EOP "Loss of Heat Transfer". (R22)

9. IAAT SCM ≥ 5°F, throttle HPI flow to maintain proper RCS P/T relationship. Adjust HP-7 (Letdown Control) as necessary to help control RCS pressure.
10. Energize all Pzr heaters.
11. Throttle FDW flow as necessary to prevent over-cooling.
12. When SG levels have reached LOSCM setpoint OR Core SCM margin > 0°F, go to step 16.4.

CAUTION:	Feeding a SG while in HPI cooling may cause excessive heat transfer.

C. If HPI Cooling is in progress and Main or Emergency FDW flow available, then recover from HPI cooling.

1. Ensure Main FDW or EFDW flow available to the SGs.
2. Prepare for normal letdown.

   Do not initiate at this time.

3. Disable both trains of MSLB isolation circuit.

   Activation of the MSLB circuit during solid plant operation could result in an undesired pressure transient.

NOTE:	HPI cooling recovery will be more difficult if these conditions are not met.

4. Determine plant conditions for HPI cooling recovery:

   Verification of these conditions should ease operator burden during the recovery process

   a) Refer to Encl. 7.6 (Total FDW Flow to Match NSSS Heat)
   b) EFDW flow capability verified
   c) Condenser vacuum and TBVs available

      Is vacuum available for controlling SG pressure/RCS temperature?

   d) RCP(s) operating (if available)
   e) Consequences of ES actuation evaluated

      Status of RB pressure, how close is the RB pressure to activating ES channels 1-6.

   f) Operator roles for recovery have been discussed

5. If RCS temperature ≤ 555°F, control THP to prevent RCS heatup by controlling SG pressure to match RCS Tsat.
6. Close RCS high point vent valves

QVALUE 1.0

QUESTION 38 B21

Unit 3 Plant Conditions:

- Reactor trip has occurred on Unit 3 at 0330 / 12-7-98 from **50% power**.
- At 2130 / 12-7-98, the control room is recovering from HPI cooling.
- EFDW has become available and is being restored to SG 1A and 1B.
- RCP 1B2 is operating.

Which ONE of the following is the feed flow rate that must be established to EACH INDIVIDUAL SG to match reactor decay heat?

(SEE ATTACHMENT)

A. 40 gpm

B. 80 gpm

C. 160 gpm

D. 190 gpm

B21

ANSWER	A
COGNITIVE	Analysis (Application)
REFSPECIFIC	EOP Encl. 7.6
MODULE	Lesson Plan E22
OBJECTIVE	ELO-15

ABASIS Correct answer. 18 hours after reactor trip from 100% power, approximately 160 gpm total FDW flow is required to match decay heat. A trip from 50% power will require approximately half of that amount, which is then divided by 2 for feed flow rate to each SG.

BBASIS Incorrect. Feed flow to each SG that equals total feed flow required at 18 hours after trip from 100% power.

CBASIS Incorrect. Total feed flow required at 18 hours after trip from 100% power.

DBASIS Incorrect, this is value for two RCP in operation after 18 hours following the reactor trip from 100%. The curve indicates diamonds and circles for the two different RCP combinations and these two identifiers are very close to the same shape on the curve.

QUESTION # 29

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**3**	**3**
	K/A #	**BW/A05 K3.2**	
	Importance Rating	**3.4**	**3.8**

Technical Reference(s): **EL-KHG**
AP/2000/02

Proposed references to be provided to applicants during examination: **AP/2000/02**
Encl. 6.3

Learning Objective: **EL-KHG #12**

Question Source:
Bank # **EL-266**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test __X__

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 29

Plant conditions:

INITIAL CONDITIONS:

- ONS Unit 1 is at 100% power
- KHU-1 is generating to the grid at 60 MW
- ACB-3 is closed

CURRENT CONDITIONS:

- RCS pressure on ONS Unit 1 rapidly decreases to 1000 psig

Which ONE of the following is correct one (1) minute after the RCS pressure decrease?

SEE ATTACHMENT

ACB- _____ will be or remain closed even if a Keowee _________________.

A. 5 / Main Transformer Lockout occurs.

B. 6 / Main Transformer Lockout occurs.

C. 7 / Emergency Lockout occurs on KHU-2.

D. 8 / Emergency Lockout occurs on KHU-2.

1 POINT

QUESTION # 29

BW/A05K3.2

A. Incorrect - ACB-5 opens on a Keowee Main Transformer Lockout

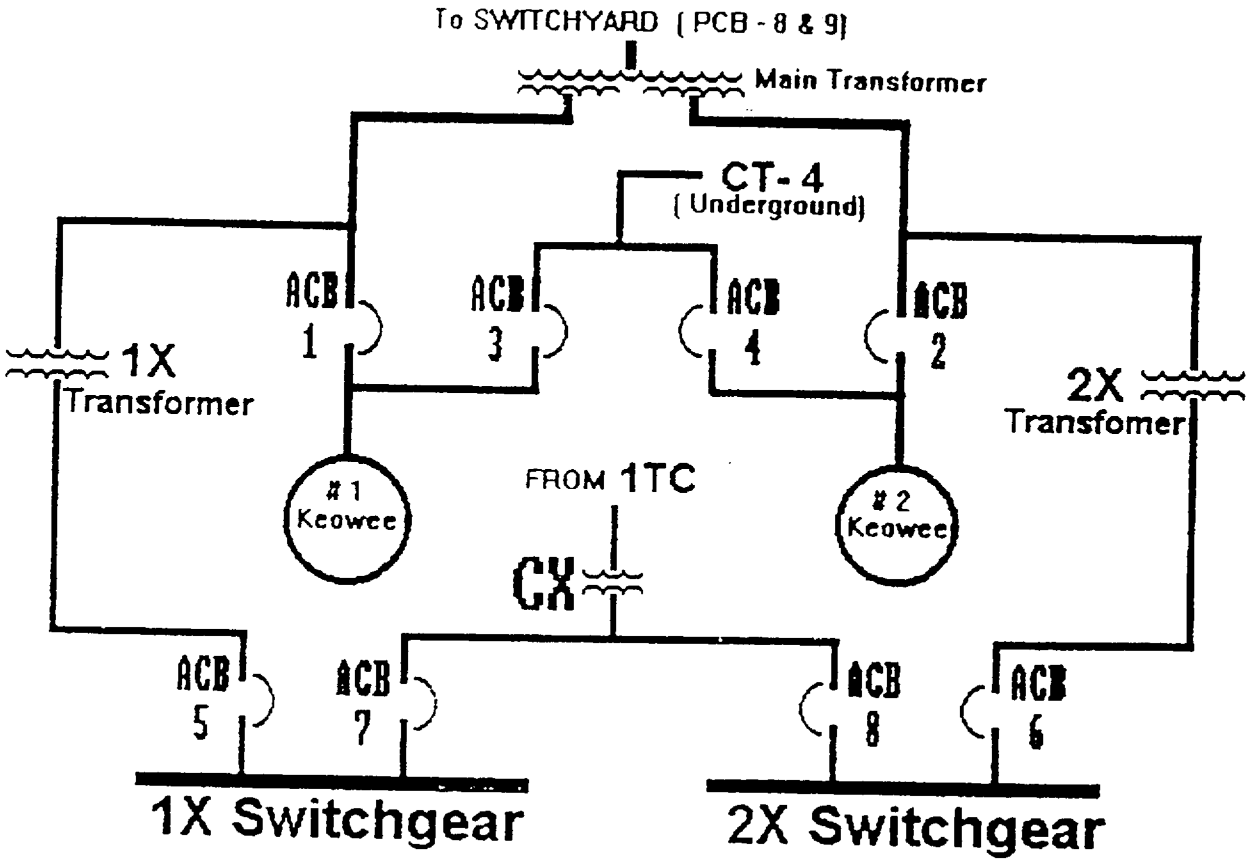
B. Incorrect - ACB-6 opens on a Keowee Main Transformer Lockout

C. Correct - ACB-7 is closed per initial conditions and would be unaffected by an Emergency Lockout on KHU-2.

D. Incorrect - ACB-8 is open and would not change positions based on KHU-2 Emergency Lockout.

why?

ACB-7 closed in stem


To SWITCHYARD (PCB - 8 & 9)
Main Transformer
CT- 4
(Underground)
ACB 1
ACB 3
ACB 4
ACB 2
1X
Transformer
2X
Transfomer
#1
Keowee
#2
Keowee
FROM 1TC
CX
ACB 5
ACB 7
ACB 8
ACB 6
1X Switchgear
2X Switchgear

TRAINING OBJECTIVES

Terminal Objective

T1. Demonstrate an understanding of the basic operation of the Keowee Hydro units during both normal and emergency operation.

T2. Assess the operation of the Keowee Hydro units during normal and emergency operations.

Enabling Objectives

1. State the purpose of the Keowee Hydro Generators. (R1)
2. Explain the basic operation of the Keowee Waterwheel Turbine. (R2)
3. Describe the basic operation of the Keowee CO2 Fire Protection System. (R3)
4. Given a set of conditions, determine when the Keowee CO2 Fire Protection System will automatically actuate and when manual operation is required. (R17)
5. Describe the purpose and function of Oconee control board switches associated with Keowee Hydro unit. (R7)
6. Interpret the response of the Keowee Hydro Units from operation of the KHU switches located in the ONS control room. (R19)
7. Describe the purpose and function of all panel board indications in the control room associated with Keowee Hydro Generators. (R8)
8. Given indications from available ONS control room instrumentation, assess the status of the KHUs. (R20)
9. Determine the sequencing of actions required to regain normal control of the Keowee Hydro unit following an emergency start signal. (R9)
10. Verify proper operation of ACB 1-4 during all modes of operation. (R11)
11. Evaluate the intent of any given limits and precautions associated with OP/0/A/1106/19, Keowee Hydro at Oconee. (R12)
12. For an emergency lockout (ELO) or normal lockout (NLO) of a KHU: (R10)

12.1 describe automatic actions that occur.

12.2 determine events that that will cause an ELO or NLO.

12.3 determine actions required following an ELO or NLO. (R21)

13. Given a copy of OP/0/A/1106/19, Keowee Hydro at Oconee, verify the proper sequence of actions have occurred for: (R4)

    13.1 an automatic start of a Keowee Hydro unit.

    13.2 a manual start of a Keowee Hydro unit.

    13.3 a manual shutdown for a Keowee Hydro unit.

    13.4 an Emergency Start of a Keowee Hydro unit.

14. State the locations of the manual emergency start controls. (R6)

15. List all signals that will initiate an emergency start of a Keowee Hydro unit. (R5)

16. Given a set of conditions, verify proper sequence of actions have occurred for an Emergency Start of the Keowee Hydro Units. (R18)

17. Given a copy of AP/O/A/2000/002, KHS Emergency Start, discuss the reason for the performance of specific steps. (R14)

18. Explain the basis for the critical action steps of the following NLO JPMs associated with the KHG: (R25)

    18.1 NLO-045, Restore power to the 600 volt switchgear 1X

19. Describe how various degraded conditions of this component could affect continued safe plant operation and the impact on accident mitigation, if any. (R16)

20. Draw and explain the Electrical Distribution System of the Keowee Hydro Station down to the 600V load centers. (R13)

21. Given a set of conditions, diagnose the status of the KHS 600V power supply system. (R22)

22. Given a copy of ITS/SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's. (R26)

23. Apply all ITS /SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R27)

1. One route is a 4000-foot, 13.8 KV underground feeder to transformer CT-4, which supplies the two redundant 4160-volt standby buses. The underground emergency power feeder is connected at all times to one Hydro electric generator through one of two air circuit breakers (ACB-3 or 4) and is energized along with CT-4 whenever the generator is in service. The underground feeder and associated transformer are rated to carry the full ES loads of one unit plus auxiliaries for maintaining safe hot S/D of the other two units.

2. The second route is a 230KV transmission line from Keowee through the isolated station 230 KV Yellow Bus to the startup transformer of each unit. This overhead path is aligned at all times to one KHU through one of two ACBs (ACB-1 or 2).

D. Emergency Start Signals are supplied from:

1. Engineered Safeguards Channel 1 & 2
2. Main Feeder Bus Monitor Panel
3. Switchyard Isolation
4. Manually from one of the start switches located in each of the Control Rooms and in each of the Cable Rooms

1.4 Normal and Emergency System Operation

A. Each Keowee unit is provided with its own automatic startup controls located in the Keowee control room <u>and</u> in the Oconee Unit #2 control room. The controls located at ONS are operational only when the Unit specific LOCAL/REMOTE switch at Keowee Hydro Station is placed in the REMOTE position.

B. For peak power generation, each unit is normally started and aligned by a Keowee operator to supply power to the Oconee 230KV switchyard through it's respective generator air circuit breaker (ACB-1 or 2) and the 13.8KV/230KV step up transformer.

C. Following receipt of an emergency start signal, both units automatically emergency start and operate in standby. If the units are already operating when they receive an emergency start signal, ACB-1 and/or 2 and 3 or 4 open, separating the Keowee Hydro Unit from the Duke Power System grid and the underground power path . They will continue to run in standby until needed. From a shut down condition, each unit's voltage regulator permits it to accept full emergency power loads as it accelerates from zero to full speed, which takes ≈ 18 seconds from receipt of the emergency start initiation signal. Per Tech Specs, this must be accomplished in 23 seconds.

1. Following an emergency start, the unit tied to the underground feeder supplies CT-4, and the other unit is available to supply the station 230KV switchyard Yellow Bus.
2. If a Switchyard Isolation occurs, the unit available to the overhead path is connected automatically to the 230KV Yellow Bus through PCB-9 after the bus has been automatically isolated from the system. This will supply power to each unit's main feeder buses through its respective startup transformer.
3. If the overhead power path is not available due to equipment malfunction (i.e., emergency lockout of Keowee Unit or startup transformer) <u>or</u> unit has experienced a LOCA (ES Channel 1&2 actuation) in conjunction with the loss of power, power will be supplied to the affected unit(s) main feeder buses from the Keowee Unit tied to the underground feeder via CT-4 and the standby buses.
4. If the Overhead unit is available but the overhead path is not and the Underground unit becomes inoperable, the underground ACBs (3 & 4) will automatically swap.

2.6 Power Supplies

A. Normal Power (See Figure OC-EL-KHG-18 and 19)

1. Unit Aligned to the Overhead

   a) From 230 KV Switchyard backcharging through Keowee main transformer

   b) To transformer 1X/2X through breaker 5/6 to 600 V Auxiliary Load Centers 1X/2X

   c) If the Keowee Unit is running power comes from output of Keowee Unit through transformer 1X/2X to breaker 5/6 to 600 V Aux. LC 1X/2X.

2. Unit Tied to the Underground

   a) Auxiliary power is supplied to the 600V Aux. LC through CX transformer through ACB7/8.

B. Emergency Power

1. With the AUTO/MANUAL Transfer switch associated with ACBs 5 & 7/6 & 8 in AUTO, the auxiliary switchgears are in a normal power alignment. The Normal Power alignment being determined by the position of the Underground breaker. If the underground breaker is CLOSED, then the Normal power is from CX transformer. If the underground breaker is OPEN, then the Normal power is from the units respective 600V transformer (either 1X or 2X). During a loss of power to a unit's 600V switchgear, a 4 second timer starts. If power is restored to the Normal source within this 4 seconds, the timer resets and no breaker action occurs. When the 4 second timer times out, the Normal power supply breaker OPENS and a 30 second timer starts. If power comes back to the Normal source during this 30 seconds, then the normal breaker will CLOSE back in. If the 30 second timer times out, and there is power available on the Alternate source, then the alternate breaker will close in. This breaker stays closed until manually opened, unless this Alternate source loses power and the Normal source has regained power. If this occurs, the alternate breaker opens after a four (4) second timer times out. If this occurs and there is power available on the Normal source, then the normal breaker closes in immediately. If all of these actions have occurred and the unit is back on its Normal power source, then the timers are all reset and the transfer scheme is ready to begin again.

2. With the transfer switch in MAN, no automatic transfers will occur. If power is lost to either units 600V Auxiliary LC, manual action must be taken by the operator to restore power per AP/0/A/2000/002, Keowee Hydro Station - Emergency Start.

3. If power is not restored to the 600V switchgears 1X &/or 2X, then two independent sets of batteries will supply control power to operate the units. Operation in this mode is limited to ≈ 1 hour.

Exam Question Report

27-Jan-99

Question ID:	EL266	Revision No:	0	Revision Date	10/29/1999
Question Description:	EL266				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EL-KHG - Keowee Hydro Generator
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 13; LRO = 13; SRO = 22 Reference: EL-KHG	Max. Point Value:	0.25
		Passing Point Value:	0.25

Question

The following conditions exist:

- KHU-1 is generating to the grid at 60 MW
- ACB-3 is closed
- ONS Unit 1 is at 100% power
- RCS pressure on ONS Unit 1 rapidly decreases to 1000 psig

Which ONE of the following is correct one (1) minute after the RCS pressure decrease? (0.25)

ACB- ___ will be closed even if /...

A) 5 / a Keowee Main Transformer Lockout occurs.

B) 6 / a Keowee Main Transformer Lockout occurs.

C) 7 / an Emergency Lockout occurs on KHU-2.

D) 8 / an Emergency Lockout occurs on KHU-2.

Answer

C
A. Incorrect - ACB-5 opens on a MTLO
B. Incorrect - ACB-6 opens on a MTLO.
C. Correct - ACB-7 is closed per initial conditions and would be unaffected by an ELO on KHU 2.
D. Incorrect - ACB-8 is open and would not change positions based on KHU-2 ELO.

Lessons

ID	Description
EL-KHG	Keowee Hydro Generators EL-KHG

QUESTION # 30

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**003000 A3.03**	
Importance Rating	**3.2**	**3.1**

Technical Reference(s): **PNS-CPS**

Proposed references to be provided to applicants during examination: **OP/1108/01 Encl. 3.31 curve RC P/T**

Learning Objective: **PNS-CPS #2.3**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 30

Unit 1 plant conditions:

- Heatup is in progress
- RCS pressure = 310 psig
- RCS temperature = 190°F
- 1B1 RCP is ready to be started
- #2 seal inlet = 115 psig

Which ONE of the following is correct?

SEE ATTACHMENT

The 1B1 RCP #1 Seal ΔP is...

A. high and RCS pressure needs to be increased.

B. high and RCS pressure needs to be decreased.

C. low and must be increased by reducing RCS pressure.

D. low and must be increased by reducing #2 seal inlet pressure.

RCS P ↑

what RCS pressure required to start 1B1 RCP.

1 POINT

QUESTION # 30

000030A3.03

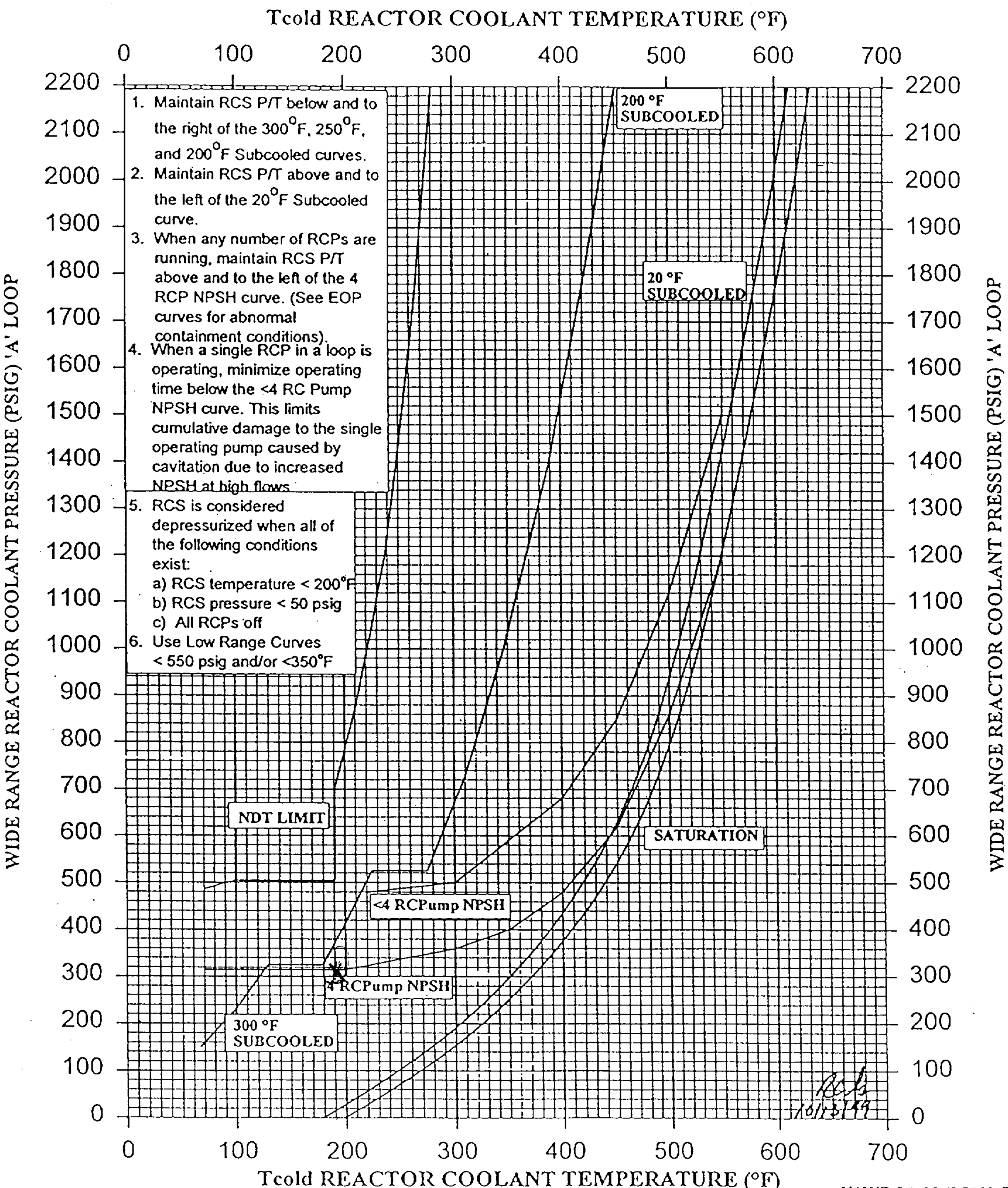
A. Incorrect - #1 Seal ΔP is low and needs to be increased. The second portion of this distracter is correct.

B. Incorrect, - #1 Seal ΔP is not high. Decreasing RCS pressure would violate the RCP minimum NPSH curve.

C. Incorrect - Decreasing RCS pressure would decrease #1 Seal ΔP. Lowering pressure would violate the RCP minimum NPSH curve. If #2 seal inlet pressure is lowered then this would be correct.

D. Correct - The minimum ΔP required across #1 seal is 205 psid. The actual ΔP = 195 psid. The ΔP is developed by the difference between RCS pressure and the #2 seal inlet pressure. A reduction in #2 seal inlet pressure via throttling 1HP-277 is required if seal ΔP is too low.

Unit 1 Wide Range Cooldown Curve



U1WRCD 33 EFPY, Rev. 0

RCS TEMPERATURE	MAX COOLDOWN RATE
T > 280°F	≤ 45°F in any 1/2 hour period
150°F < T ≤ 280°F	≤ 20°F in any 1/2 hour period
T ≤ 150°F	≤ 9°F in any 1 hour period
RCS depressurized	≤ 45°F in any 1 hour period

OBJECTIVES

TERMINAL OBJECTIVE:

1. The ISS and NLORQ student will be able to describe the major components which makeup the seal packages of the Westinghouse and Bingham RCPs as well as the normal operation of each seal type. (T1)

2. The student will be able to describe the major components which make up the seal packages of both the Westinghouse and Bingham RCPs and the normal and abnormal operation of each seal type. The student will also be able to diagnose various seal failure combinations. (T2)

ENABLING OBJECTIVES:

1. Explain the purpose of RCP seals. (R1)

2. For the Unit 1 Westinghouse Reactor Coolant Pump Seals:

   2.1 Briefly describe the operation and construction of the following components: (R2)

      A. The #1 seal.

      B. The #2 seal.

      C. The #3 seal.

      D. RCP radial bearing

   2.2 Describe the basis for the #1 seal leakoff flow normal operating curve. (R3)

   2.3 Explain how and why proper #2 seal inlet pressure is maintained during: (R4)

      A. Unit startup

      B. Normal operation

   2.4 Explain why the # 2 seal would be likely to fail following a failure of the Number 1 seal. (R5)

   2.5 Describe the condition that would result in the #3 seal leakoff being diverted to the reactor building normal sump instead of the Quench Tank. (R6)

   2.6 Describe how the radial bearing is cooled during: (R7)

      A. Unit startup

      B. Normal operation

      C. Loss of seal injection flow

B. This explanation of basic mechanical seals is intended to provide a basis for understanding the seal operation of the Westinghouse and Bingham RCP seal packages.

2.3 Component Description for Westinghouse Pumps (Unit 1) **(OC-PNS-CPS-1)**

A. Design Data

1. Vertical, single stage, centrifugal pump
2. Impeller:
    a) stainless steel
    b) 38" O.D.
    c) 7 vane
3. Turning Vane - Diffuser:
    a) encompasses impeller
    b) converts velocity head to pressure head
4. Design flow of ≈ 88,000 GPM (≈ 102,000 gpm actual)
5. Develops 350 ft. of head (@88,000gpm)
6. 1190 RPM

B. Flow (RCS)

1. Flow enters the bottom of the impeller and is discharged through passages in the turning vane-diffuser to the discharge in the side of the pump casing.

2.4 System Normal/Abnormal Operations of the Westinghouse Seals(OC-PNS-CPS-2)

A. Number one seal (#1 seal)

1. controlled leakage
2. film riding
3. The number one seal is the primary seal pressure boundary. It is composed of a runner which rotates with the shaft, and a non-rotating seal ring attached to the seal housing. Both seal faces are made of silicon nitride.
    a) Silicon Nitride increases the durability of the #1 seal to minimize seal face damage should the fluid film thickness be too thin and the faces contact momentarily.

4. During operation the two seal faces must be separated by ≈ .5 mils to prevent damage to the seal.

   a) This is accomplished by maintaining a minimum of 205 psid across the seal with the required minimum #1 seal leakoff flow (**OC-PNS-CPS-7**). During startup or shutdown when system pressure is low, it may be necessary to throttle open 1HP-277 (SEAL RETURN THROTTLE VALVE) to lower #2 seal inlet pressure in order to meet this requirement. NOTE: Throttling open 1HP-277 is now less of a concern since the #1 seal Δp requirement has been decreased from 275 psid to 205 psid. (Westinghouse recommendation ≥ 200 psid)

      1) It takes >200 psid to lift the stationary seal and ring support assembly and create the proper film thickness and consistency between the two sealing faces to ensure that the two faces will not make contact on pump start or operation. (At 200 psid the minimum #1 seal leakoff flow is .2 gpm)

      2) The required #1 seal Δp is a function of the RCS pressure. The actual Δp is developed by the difference between the RCS pressure (seal injection with HPI seals applied) and the #2 seal inlet pressure. HPI seal injection provides clean cool water to the seals for cooling and is filtered to protect the seals from wear/erosion.

      3) The seal film thickness is self regulating. As pressures within the seal cavity change the opposing forces to open and close the seal faces remain equal as long as the minimum 200 #Δp is available.

INSTRUCTOR NOTE: Discuss the use of HANDOUT #7 (No. 1 Seal Normal Operating Range) curve.

   b) #1 Seal Leakoff Normal Operating Curve (**OC-PNS-CPS-7**)

      1) The lower #1 seal leakoff flow limit on the bottom of the curve (**OC-PNS-CPS-7**) is established to provide for:

         (a) Minimum film thickness to separate the seal faces.

         (b) To provide for minimum flow through the faces in order to keep them cool. If seal faces overheat they can distort or crack.

         (c) The upper limit of seal leakoff flow (< 5gpm) is based on industry experience for expected #1 seal leakoff flow.

e) During startup operation, 1HP-277 is throttled to:

1) Ensure a minimum of 30 psid across the number two seal (normal 60 psid, upper limit 90 psid). This forces the small amount of flow past the #2 seal required for cooling and lubrication and on to the #3 seal.

2) To aid in attaining the proper #1 seal Δp (205 psid) the #2 seal inlet pressure can be reduced to 20#.

f) In the event of a complete failure of the #1 seal, the #2 seal is capable of withstanding full system pressure.

1) During operation as the primary seal (failure of #1 seal), the number two seal becomes a film riding seal similar to operation of the #1 seal.

2) When the #2 seal becomes a film riding seal, it passes sufficient flow to maintain a film thickness and consistency to keep the faces apart. This increased flow will show up as increased input to the standpipe. The increased flow (expected to be 4-12 gpm) will be in excess of the ability of the standpipe orifice to pass causing a high standpipe level alarm on the computer to alert the operator of an abnormal seal condition.

3) Fragments from the failed #1 seal can cause a failure of the number #2 seal. Therefore, operation in this mode should be closely monitored per the Abnormal RCP Operation AP/1/A/1700/16.

4) #2 Seal can be damaged due to:

(a) The #2 seal becoming a film riding seal the softer seal face (graphitar) is subject to erosion from the increased flow across the face.

(b) Damage to the #2 seal face can also occur from #1 seal debris.

C. Number three seal (#3 seal)

1. Rubbing face seal similar to number two seal

2. Is not designed to withstand full system pressure

3. Diverts leakage past the #2 seal to the Quench Tank or R.B. normal sump.

a) The #3 seal cavity closure has a bronze bushing on the shaft to restrict flow up the shaft and aid in directing #3 seal leakoff to the RBNS or QT.

QUESTION # 31

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**003000 A4.06**	
Importance Rating	**2.9**	**2.9**

Technical Reference(s): **PNS-CPS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-CPS OBJ. #4**

Question Source:
Bank # **PNS 697**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 31

Unit 3 plant conditions:

- Reactor power = 80%
- RCS pressure = 2150 psig
- RCP parameters:

CAVITY PRESS	LOWER	UPPER
3A1	2150	1075
3A2	1390	730
3B1	975	975
3B2	1100	50

Which ONE of the following correctly describes the condition of the RCP seals?

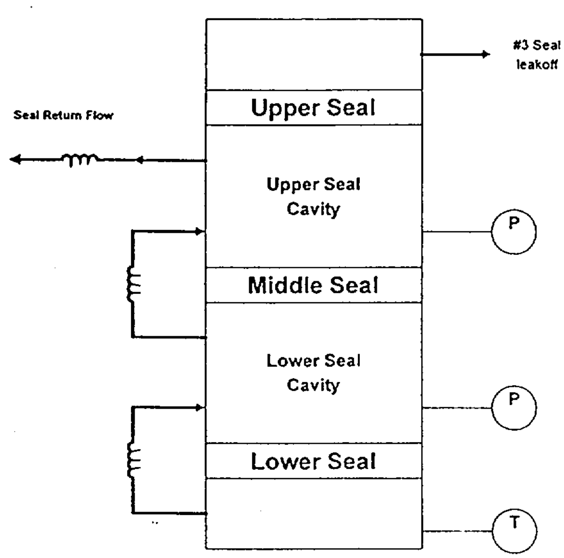
RCP 3A1 _________, 3A2_________, 3B1 _________, 3B2 _________.

A. Lower seal failed, All seals OK, Middle seal failed, Upper seal failed

B. All seals OK, Upper seal failed, Middle seal failed, Lower seal failed

C. Lower seal failed, Middle seal failed, All seals OK, Upper seal failed

D. Lower seal failed, Upper seal failed, Middle seal failed, All seals OK



1 POINT

QUESTION # 31

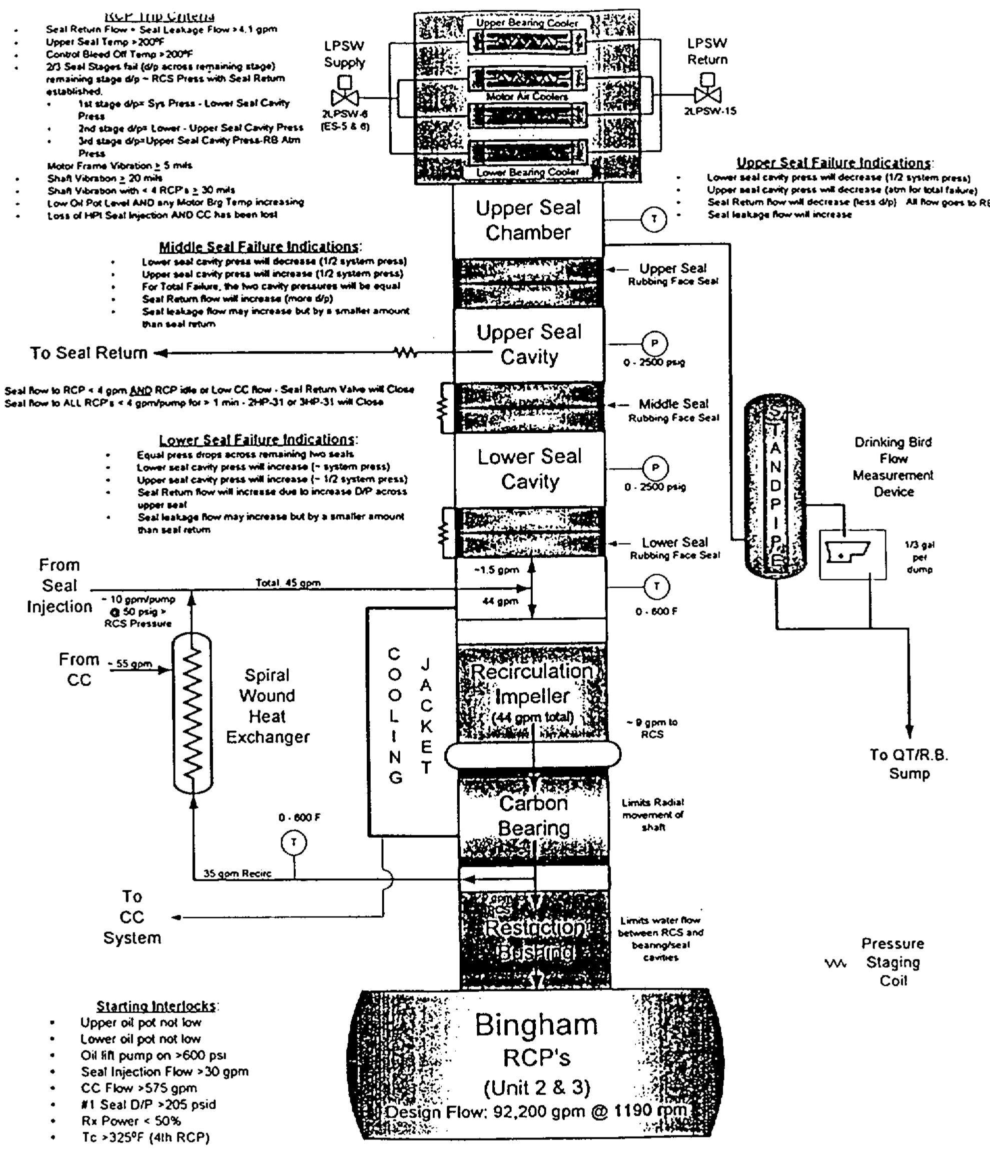
003000 A4.06 (2.9/2.9) Both PRA 2-9-00

A. Correct - See attachment drawing

B. Incorrect - See attachment drawing

C. Incorrect - See attachment drawing

D. Incorrect - See attachment drawing



TRAINING USE ONLY	
BINGHAM RCP SEALS (Unit 2 & 3)	
DRAWING #	OC-PNS-CPS-4
DRAWN BY RJL	DATE 6/24/99
REFERENCE	OFD-100A-2.3 & 3.3 REV 9
APPROVED BY	[illegible]

1 POINT

QUESTION # 31

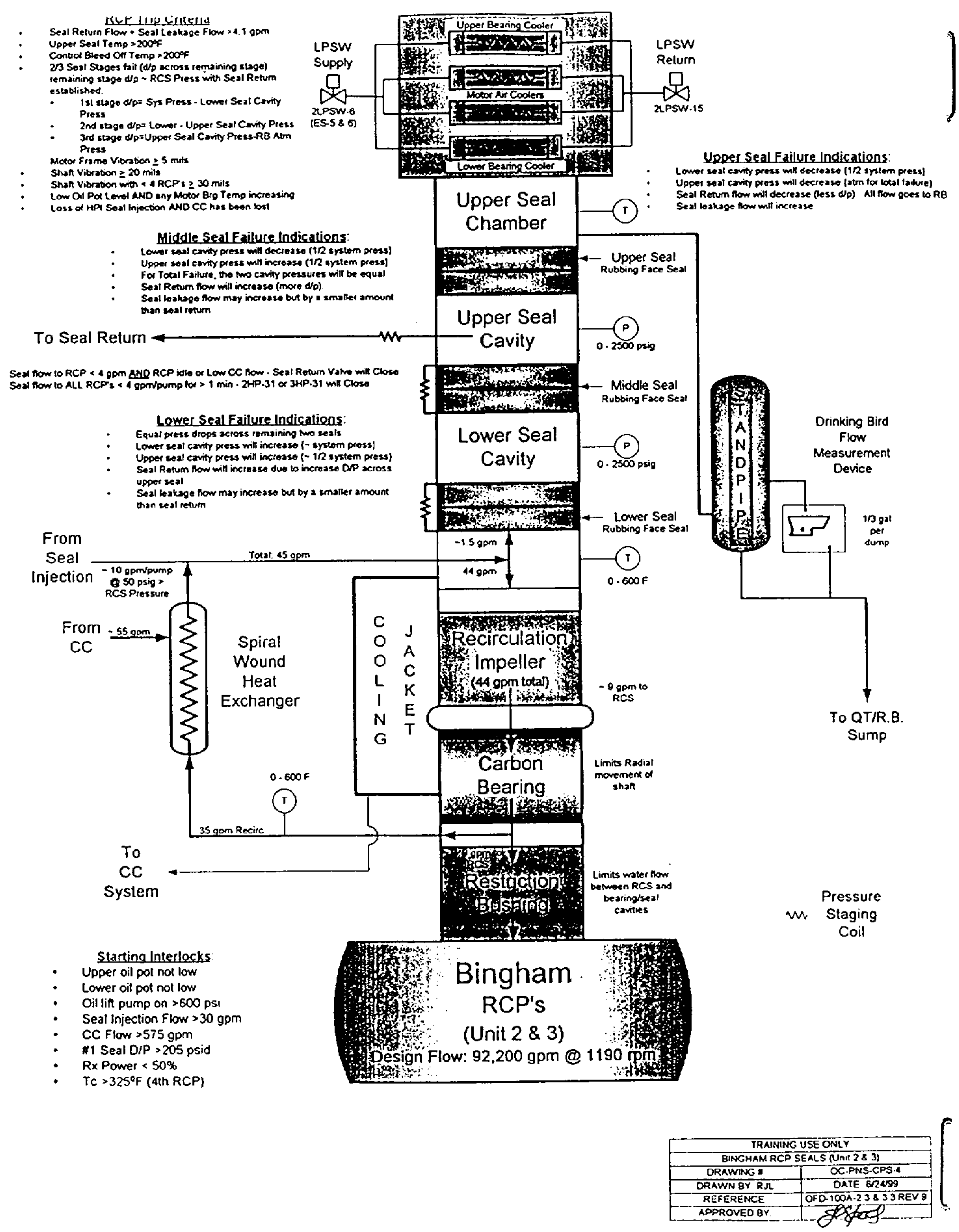
003000 A4.06 (2.9/2.9) Both PRA 2-9-00

A. Correct - See attachment drawing

B. Incorrect - See attachment drawing

C. Incorrect - See attachment drawing

D. Incorrect - See attachment drawing



4. When given a set of unit conditions analyze the operating status of the Bingham RCP seals during all modes of operation. (R20)

5. Draw a simplified drawing of the Unit's Reactor Coolant Pump including the following: (R22)

    5.1 Seal Package

    5.2 HPI seal flow path and approximate flows expected

    5.3 #1 Seal Bypass valve

    5.4 Standpipe fill valve

    5.5 Radial Bearing

    5.6 Recirc Impeller

    5.7 Heat exchangers and coolers

    5.8 Thermal barrier, restriction bushing

6. Given a set of conditions, evaluate the total RCP seal leakage and determine the appropriate operator action(s) utilizing the information and guidance provided in PT/1,2,3/A/600/10, Reactor Coolant Leakage, and OP/1,2,3/A/1103/06, Reactor Coolant Pump Operation. (R23)

7. Given a copy of ITS/SLCs and associated Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's. (R21)

8. Apply all ITS/SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R21)

9. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITS/SLC's. (R21)

1) 0 - 80 GPM

2) Pen recorder in control room (VB3) and gage on control board (UB1)

3) Computer Indication (unit 3)

b) Seal injection flow (individual)

1) 0 - 15 GPM

2) Gage in control room (VB3)

c) Seal return flow/seal leakage flow (individual)

1) 0 - 4 GPM Seal Return / 0 - 2 GPM Seal leakage

2) Dual Pen Chart in control room (VB3-unit 2, Unit board-unit 3)

(a) Chart value for seal return flow is a calculated value and should be verified by other indications

(b) Seal return calculations are based on seal cavity pressures

(c) Seal leakage is based on "Drinking bird" signal.

3) Seal leakage - digital indication (VB3-unit 2, Unit Board-unit 3)

4) Computer indication of seal return flow is actual flow and is generated by *rotometers* in the seal return line.

J. Seal Failures (**OC-PNS-CPS-8 to 11**)

1. General

a) Since there are no separate leakoff paths after each seal on a Bingham pump as there were on the Westinghouse pumps; we must look at cavity pressures, seal return, and seal leakage to evaluate seal conditions.

b) An increase in dp across the seal will increase flow through the seal staging coil by a larger amount than that through the seal.

1) Thus, a failure of either of the lower seals may not result in a noticeable increase in seal leakage - while giving a substantial increase in seal return.

c) A failure of any seal will cause a change in the Δp across the remaining seals. This increased Δp will cause an increase in the flow through the remaining staging coils due to the remaining coils sensing a higher Δp.

d) Thus, a significant increase in seal leakage to RB containment would indicate a problem with at least the upper seal.

2. Failure of **lower** seal

a) A failure of the lower seal will result in system pressure being evenly broken down across the remaining two seals.

b) Lower seal cavity pressure will increase (≈ system pressure for a total failure).

c) Upper seal cavity pressure will increase (≈ 1/2 system pressure for total failure).

d) Due to increased pressure in the upper seal cavity, Δp across the upper seal staging coil will be increased and thus seal return flow will increase.

e) Seal leakage may increase, *but* by a smaller amount than seal return.

3. Failure of **middle** seal

a) Lower seal cavity pressure will decrease (≈ 1/2 system pressure for total failure).

b) Upper seal cavity pressure will increase (≈ 1/2 system pressure for total failure).

c) For total failure the two cavity pressures will be ≈equal.

d) Seal return flow will increase due to increased Δp across remaining two staging coil.

e) Seal leakage may increase but by a smaller amount than seal return.

4. Failure of **upper** seal

a) Lower seal cavity pressure will decrease (≈1/2 system pressure)

b) Upper seal cavity pressure will decrease (atm. for total failure)

c) Seal return flow will decrease (less Δp). The seal return flow would actually drop to zero if the upper seal failure was total, all of the seal return would go to the RB.

d) Seal leakage flow will increase (less restriction to flow).

K. General Information

1. Seal flow to RCP < 4 gpm AND RCP idle OR low CC flow to a RCP.

a) The associated seal return valve will CLOSE.

1) This occurs to prevent hot RCS from going up the RCP shaft with no or inadequate cooling. Without limiting flow along the RCP shaft in this condition the RCP seals and other components are in jeopardy of damage due to over heating.

2. Seal flow to all RCPs < 4 gpm/RCP for > 1 min 2 or 3HP-31 closes.

RCP Trip Criteria:

- Seal Return Flow + Seal Leakage Flow >4.1 gpm
- Upper Seal Temp >200°F
- Control Bleed Off Temp >200°F
- 2/3 Seal Stages fail (d/p across remaining stage) remaining stage d/p ~ RCS Press with Seal Return established.
  - 1st stage d/p= Sys Press - Lower Seal Cavity Press
  - 2nd stage d/p= Lower - Upper Seal Cavity Press
  - 3rd stage d/p=Upper Seal Cavity Press-RB Atm Press
- Motor Frame Vibration ≥ 5 mils
- Shaft Vibration ≥ 20 mils
- Shaft Vibration with < 4 RCP's ≥ 30 mils
- Low Oil Pot Level AND any Motor Brg Temp increasing
- Loss of HPI Seal Injection AND CC has been lost

Upper Seal Failure Indications:

- Lower seal cavity press will decrease (1/2 system press)
- Upper seal cavity press will decrease (atm for total failure)
- Seal Return flow will decrease (less d/p). All flow goes to RB
- Seal leakage flow will increase

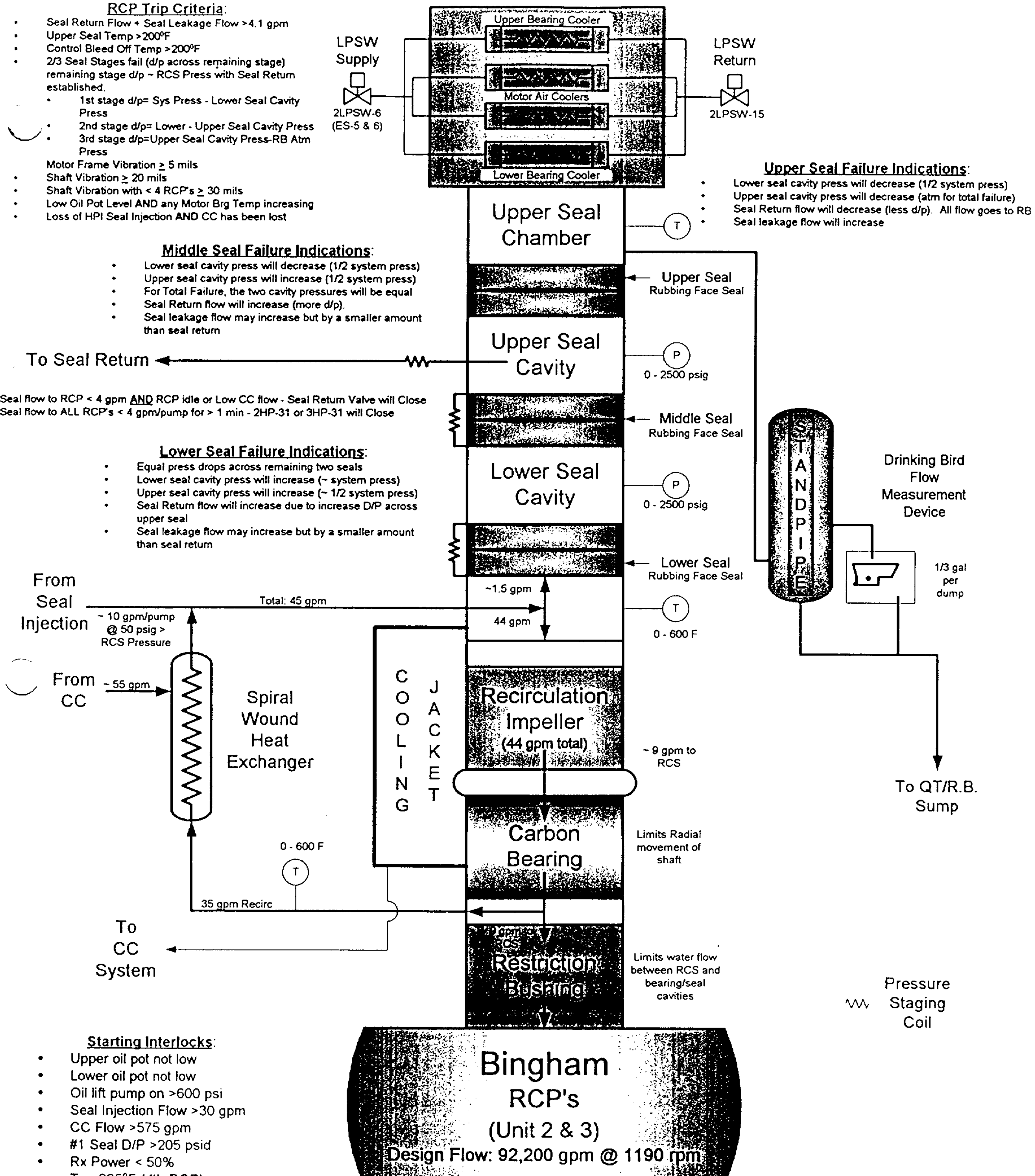
Middle Seal Failure Indications:

- Lower seal cavity press will decrease (1/2 system press)
- Upper seal cavity press will increase (1/2 system press)
- For Total Failure, the two cavity pressures will be equal
- Seal Return flow will increase (more d/p).
- Seal leakage flow may increase but by a smaller amount than seal return

Seal flow to RCP < 4 gpm AND RCP idle or Low CC flow - Seal Return Valve will Close
Seal flow to ALL RCP's < 4 gpm/pump for > 1 min - 2HP-31 or 3HP-31 will Close

Lower Seal Failure Indications:

- Equal press drops across remaining two seals
- Lower seal cavity press will increase (~ system press)
- Upper seal cavity press will increase (~ 1/2 system press)
- Seal Return flow will increase due to increase D/P across upper seal
- Seal leakage flow may increase but by a smaller amount than seal return



Starting Interlocks:

- Upper oil pot not low
- Lower oil pot not low
- Oil lift pump on >600 psi
- Seal Injection Flow >30 gpm
- CC Flow >575 gpm
- #1 Seal D/P >205 psid
- Rx Power < 50%
- Tc >325°F (4th RCP)

TRAINING USE ONLY	
BINGHAM RCP SEALS (Unit 2 & 3)	
DRAWING #	OC-PNS-CPS-4
DRAWN BY: RJL	DATE: 6/24/99
REFERENCE:	OFD-100A-2.3 & 3.3 REV 9
APPROVED BY:	[signature]

Exam Question Report

27-Jan-99

Question ID:	PNS697	Revision No:	0	Revision Date	10/29/1999
Question Description:	PNS697				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	PNS-CPS - Coolant Pump Seals
Last Used Date:	03/02/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	NLO = R20; LRO = N/A Reference: 003A3.03 [3.2/3.1]	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

RO ONLY

Unit 3 plant conditions:

- Operating in Mode 3
- RCS pressure = 2150 psig
- Quench Tank level increasing

RCP seal cavity pressures (PSIG) are:

	LOWER CAVITY PRESSURE	UPPER CAVITY PRESSURE
• RCP 3A1	2100	1050
• RCP 3A2	1400	700
• RCP 3B1	1050	1050
• RCP 3B2	1050	0

Which ONE of the following accurately describes the condition of the RCP seals? (.25)

A) RCP 3A1: Lower Seal - failed
RCP 3A2: All Seals - OK
RCP 3B1: Middle Seal - failed
RCP 3B2: Upper Seal - failed

B) RCP 3A1: All Seals - OK
RCP 3A2: Upper Seal - failed
RCP 3B1: Middle Seal - failed
RCP 3B2: Lower Seal - failed

C) RCP 3A1: Lower Seal - failed
RCP 3A2: All Seals - OK
RCP 3B1: Upper Seal - failed
RCP 3B2: Middle Seal - failed

D) RCP 3A1: Lower Seal - failed
RCP 3A2: Upper Seal - failed
RCP 3B1: Middle Seal - failed
RCP 3B2: All Seals - failed

Answer

A
Reference: AP/3/1700/16 Case A and B.

A. Correct. Each seal on Unit 3 breaks down pressure approximatly 1/3 of system pressure. Lesson plan page 24 (2.4). Based on working knowledge of the Bingham seal package.
B. Incorrect. RCP 3A1 lower seal is failed
C. Incorrect. RCP 3B1 Upper seal not failed
D. Incorrect. All seals are not failed

Lessons

ID	Description
PNS-CPS	COOLANT PUMP SEALS (CPS)

Enabling Objectives

ID	Description
PNSCPSR20	5. When given a set of unit conditions, determine the statu

Referenced Documents

ID	Description	Review Date	Ref Flag
AP/3/A/1700/016	Abnormal Reaction Coolant Pump Operation		

QUESTION # 32

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**004 A1.06**	
Importance Rating	**3.0**	**3.2**

Technical Reference(s): **OP/1108/01 Encl. 3.39**
LDST pressure vs level

Proposed references to be provided to applicants during examination: **Blanked out encl. 3.39**

Learning Objective: **PNS-HPI OBJ. #27**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 32

Given the Attached curve and the following . . .

Unit 2 plant conditions:

- Reactor power = 18%
- LDST pressure = 14 psig
- LDST level #1 = 53 inches
- LDST level #2 = 52 inches

Which ONE of the following is correct?

SEE ATTACHMENT

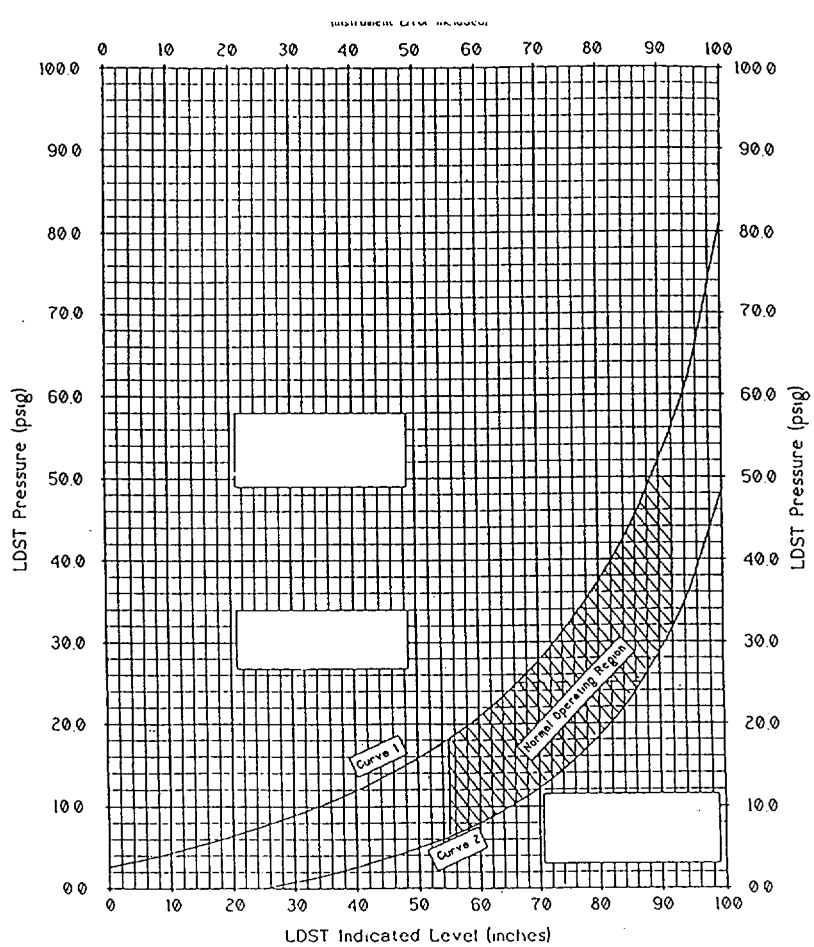
why 62"?

A. Increase LDST level to 62 inches.

B. Increase LDST hydrogen pressure to 26 psig.

C. Establish MODE 3 unit operation within the next 10 hours.

NP. D. Provide HPI suction from the BWST for transients requiring additional HPI flow.



1 POINT

QUESTION # 32

004000A1.06 BOTH GCW 02/12/00

A. Correct, Minimum LDST level when a HPIP is operating is 55" OR actions should be taken to increase LDST level ≥ 55".

B. Incorrect, LDST pressure should be > 25 psig for chemistry considerations, however in this case it would cause the LDST P/T relationship to be above and to the left of curve 1 which would cause the HPIPs to become inoperable.

C. Incorrect, this action is required when LDST P/T is above and to the Left ~~right~~ of curve 1. Both trains of HPI inoperable which would cause you to enter TS 3.03.

D. Incorrect, this action required when LDST P/T is below and to the Right ~~left~~ of curve 2.

15. Explain why the use of the "B" HPI injection nozzles with RCS temp. > 250°F must be documented in the Unit log. (R19)
16. Summarize the effects of restarting an HPIP following a total loss of HPI flow, prior to closing HP-31. (R21)
17. State two purposes for aligning pressurizer auxiliary spray during a normal shutdown. (R24)
18. Explain the thermal stress concern that can be involved with the use of pressurizer auxiliary spray. (R25)
19. Explain how to regain operator control of the HPI pumps after resetting ES channels 1 and 2. (R27)
20. Briefly explain quarter core cooling and how the use of HP-409/HP-410 can prevent it. (R28)
21. Briefly explain two reasons why suction to the HPI pumps might be supplied through LP-15 and LP-16. (R29)
22. Describe the reason and method for supplying HPI pump suction from the SFP. (R30)
23. Describe the purpose and method used to measure HPI System leakage. (R31)
24. Describe the purpose and method used to perform the HPI Full Flow Test. (R32)
25. Given a copy of PT/1,2,3/A/0600/010, RCS Leakage, determine if calculation is valid and correctly evaluate as required. (R33)
26. Recognize that the HPI Full Flow Test requires operators with no other duties for the duration of the test, due to the potential for RCS inventory loss and local monitoring required. (R34)
27. When given a copy of applicable portions of the High Pressure Injection System procedure, demonstrate an understanding of the procedure by locating the answer to specific questions on limits, cautions, notes, etc., within the procedure, or, explain the basis or reason for Limits and Precautions specifically related to or affected by the duties and tasks of the operator. (R35)
28. Concerning the Unit 3 loss of HPIPs event, (R38)

    28.1 Specify what the two primary causes were for this event.

    28.2 Explain the missed opportunity by the control room team to detect the level inaccuracy and the corrective action taken as a result of this event.

1. Minimum LDST level when a HPI Pump is operating is 55" **OR** actions should be take to increase LDST level ≥ 55". {3}

2. When HPI Pumps are operating: {3}

   2.1 LDST pressure and level should be within limits of "LDST Pressure Vs. Level" curves to prevent gas from entering HPI Pumps in event of HPI Emergency Injection.

   - Normal LDST operation pressure should **NOT** exceed 50 psig.
   - "LDST Pressure Vs. Level" curves are also located on OAC.

   OR

   2.2 (1)(2)(3)GWD-19 (LDST VENT) **AND** (1)(2)(3)GWD-20 (LDST Vent Blk) must be open.

3. If LDST Pressure Vs. Level is above and to the left of Curve 1, then declare **BOTH** trains of HPI INOPERABLE.

   3.1 Immediately depressurize LDST below Curve 1.

   3.2 Refer to ITS 3.0.3 for shutdown requirements.

   3.3 Notify NRC per OMP 1-10 (Usage and Testing of the NRC Emergency Notification System).

4. If LDST Pressure Vs. Level is below and the right of Curve 2, then perform the following:

   4.1 Pressurize LDST back into normal operating region of the "LDST Pressure Vs. Level" curve unless LDST is being depressurized intentionally by an approved procedure.

CAUTION:	If LDST Pressure Vs. Level is below and to the right of curve 2, it may be possible to draw a vacuum in LDST resulting in HPI Pump damage due to inadequate NPSH. This could occur even though sufficient LDST level exists.

   4.2 Carry a note on the RO Turnover Sheet to the effect that if a transient occurs which requires additional HPI flow, immediately open (1)(2)(3)HP-24 and (1)(2)(3)HP-25 to provide an adequate suction source to HPI Pumps.

- Separate high and low OAC level alarms for each have been added also: A1042 for level 1, A1043 for level 2. The high alarm setpoint is still 92 inches for Unit 2 & 3 and 90 inches for Unit 1.
- An additional non-safety grade level instrument is provided locally on Unit 3.
- OAC alarm O3P7461 indicates that Unit 3 LDST level transmitters are exceeding their 2% tolerance

c) *Interlock:*

1) Units 2 ONLY:
   - LDST at <18 inches (Lo-Lo Level alarm) will return 1,2HP-14 to the "NORMAL" position if it is in the "BLEED" position.
   - LDST Lo Level alarm is at 55 inches.
2) Unit 1 & 3 ONLY:
   - LDST at <40 inches will return HP-14 to the "NORMAL" position if it is in the "BLEED" position.
   - LDST at < 40 inches automatically opens HP-24 and HP-25 to align the BWST to the suction of the HPIPs. A DISABLE/ENABLE switch, LDST LEVEL INTERLOCK TO HP-24 AND HP-25, is provided on UB1 to control this interlock. Computer alarm indicates when initiate signal is present. Interlock utilizes a two out of two logic to open the valves, thereby preventing a single failure from actuating the interlock. This interlock being functional is **NOT** a requirement for HPI Pump operability.
3) Unit 1 Only
   - LDST Lo Level alarm is at 60 inches and the Lo Lo Level alarm is at 55 inches.
4) Unit 3 Only
   - LDST Lo Level alarm is at 57 inches and the Lo Lo Level alarm is at 40 inches.

d) Letdown storage tank pressure indication is provided over a range of 0-100 psi. *Normally at least 25 psig hydrogen overpressure is maintained in the letdown storage tank to scavenge any free oxygen and thereby help prevent corrosion on the primary side.*

- On Unit 2: 2 pressure transmitters are provided for redundant check for level/pressure relationship. The indications are the CR gage and OAC point A2191.

- Unit 1 & 3 pressure indication has been upgraded to QA-1 condition. Both pressure indications are displayed on the UB1 indicator.

e) A new non-safety grade pressure indication is provided locally on Unit 3.

Refer to OC-PNS-HPI-3

f) When adding Hydrogen to the LDST, pressure should be maintained in the normal operating envelope of the LDST Pressure vs. Level Curve. This will insure that the Chemist can draw a good sample and that the solubility of Hydrogen in water is maintained.

Refer to OC-PNS-HPI-3a See OP/0/A/1108/001

g) If LDST Pressure vs. Level is above and to the left of curve 1, then declare both trains of HPI INOPERABLE.

- Immediately depressurize the LDST below Curve 1.
- Refer to Tech. Spec. 3.0.3 for shutdown requirements.
- Notify the NRC per OMP 1-10 (Usage and Testing of the NRC Emergency Notification System).

h) If LDST Pressure vs. Indicated Level is below and to the right of curve 2, then perform the following:

- Pressurize the LDST back into the normal operating region of the LDST Pressure vs. Level curve unless the LDST is being depressurized intentionally by an approved procedure.
- If the LDST pressure vs. level is below and to the right of curve 2, then for some scenarios requiring additional HPI flow, it may be possible to draw a vacuum in the LDST causing the HPI pumps to be damaged due to inadequate NPSH. This could occur even though sufficient LDST level exists.
- Carry the following note on the RO Turnover Sheet :

  1) "If a transient occurs which requires additional HPI flow (RCS leak, LOCA, overcooling, Reactor Trip), immediately open HP-24 and HP-25 to provide an adequate suction source to the HPI Pumps.

i) OAC LDST Pressure/Level Graphic **Refer to OC-PNS HPI 17**

1) LDST Pressure vs. Level curve is provided on the OAC. Access from Trends pulldown menu, then select Prebuilt XY Plots, select WR or NR display.

2) The current Pressure vs. Level point is displayed on the grid, and an alarm is received when the point approaches the limits as indicated by the area around the white box..

Refer to OC-PNS-HPI-4

B. **Makeup**

1. The makeup line taps into the letdown line between HP-14 and the letdown filters. HP-16 serves as the makeup isolation valve and will isolate all flow through this line.

2. *Sources of Make Up*

a) Highly Borated Water (at or above RCS boron concentration):

1) A" Bleed Holdup Tank

2) Boric Acid Mix Tank

3) Concentrated Boric Acid Storage Tank

4) Quench Tank

b) Demin Water/Low Borated Water (below RCS boron concentration)

1) "B" Bleed Holdup Tank

2) *Deborating Demineralizer Effluent*

Refer to OC-PNS-HPI-5,6A,6B,6C

3. LDST Makeup Control

a) All flow through the makeup line flows through the makeup controller totalizer to provide the operator with an indication of the volume flow through the makeup line. It also provides control signals to the Continuous Boron Dilute portion of the feed and bleed interlock, which permits it to terminate flow through the Deborating Demin. when the pre-selected batch size is reached.

b) The makeup controller controls HP-15 which will isolate makeup flow from the BHUT's or the QT when the pre-selected volume is reached.

c) Basic operation of the batch controller is similar to the Moore controllers already in use at Oconee.

1) Display Window - Digital display at top of controller, displays whichever of the following is selected (lit) by pushing the display push-button (D) on the right.

5. Refueling Transfer Tubes:

   Each refueling tube is equipped with a blind flange which is only opened during a refueling shutdown for transfer of fuel to the spent fuel pool.

2.3 Emergency Operations

Instructor Note:

CP-601, Cooldown Following Large LOCA, of EP/1,2,3/A/1800/01, Emergency Operating Procedure, gives guidance on operation after E.S. initiation.

A. Engineered Safeguards Operation

1. During normal operation, the PRV System is on standby with each fan aligned with a filter assembly.
2. Engineered Safeguards Channels 5 and 6 (High RB Pressure of 4 psig) will initiate the following sequence of events:
   a) PRV Fans A and B receive a signal to start.
   b) Valves PR-15 and PR-19 will open by a signal from the fan starts. The fan start is verified on the RZ modules and the "valve open position" should be verified in the Control Room.
3. Following the Engineered Safeguards Actuation the operator should:
   a) Verify PRVS is in operation and send an operator to adjust (1)(2)(3)PR-13 (Filter A Outlet) and (1)(2)(3)PR-17 (Filter B Outlet) as necessary to maintain 1000cfm flow through each filter train.
   b) If either Penetration Room fan fails to start on ES initiation, open (1)(2)(3) PR-20 (PR Fans Suction Tie) at its locally mounted switch to allow the operating fan to purge through both filters.

   NOTE: Do not exceed 1100 cfm flow through each filter train. Excessive flow decreases the efficiency of the filter packs.
4. Circuitry has been modified such that the fans do not stop upon reset of E.S. channel. To shut down the fans requires deliberate separate action. No new switches have been added. Fans will still be controlled at the RZ modules. However, after reset of E.S. Channel in order to turn off fans, the OFF buttons must be pushed.

B. Loss of Air Flow Through a Filter

1. Redundant fans, cross connected piping, and locked open filter inlet valves minimizes the possibility of a loss of cooling air flow to the filters. Analysis completed per PIR 4-090-0057 indicates that natural circulation around the filter will provide adequate cooling to prevent carbon ignition.

1. Initial Conditions

1.1 Review Limits and Precautions.

1.2 Events are in progress that have caused Penetration Room Ventilation system to start.

1.3 EOP Reader has requested verification of proper operation of Penetration Room Ventilation System.

2. Procedure

CAUTION:
- High radiation levels may exist in RB Purge Equipment Room.
- Do **NOT** exceed 1100 cfm in either PRV filter train.

2.1 Establish ≈ 1000 cfm flow in each PRV filter train by adjusting for applicable Unit(s):

- Unit 1:
  - 1PR-13 (Penetration Room Filter 1A Outlet).
  - 1PR-17 (Penetration Room Filter 1B Outlet).

  Location: Rm 602, Ventilation Equipment Room, North wall.
- Unit 2:
  - 2PR-13 (Penetration Room Filter 2A Outlet).
  - 2PR-17 (Penetration Room Filter 2B Outlet).

  Location: Rm 610, Unit 1 & 2 SFP Change Room corridor, South wall.
- Unit 3:
  - 3PR-13 (Penetration Room Filter 3A Outlet).
  - 3PR-17 (Penetration Room Filter 3B Outlet).

  Location: Rm 612, Unit 3 SFP Change Room corridor, South wall.

Exam Question Report

27-Jan-99

Question ID:	CF024	Revision No:	0	Revision Date	10/29/1999
Question Description:	CF024				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	CF-FPT - Feedwater Pump Turbine
Last Used Date:		Question Type:	Essay
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 9; LRO = 8; SRO = 9	Max. Point Value:	0.8
		Passing Point Value:	0.8

Question

List eight (8) Main Feedwater Pump/Turbine trips that affect ALL three units. Include setpoints where applicable. (If a trip has a setpoint, the setpoint is part of that answer)(.8)

Answer

Any Eight

1) Manual /
2) Overspeed / ≈ 5200 rpm ± 200
3) FDWPT Exhaust Vacuum Low / ≈ 19 in. Hg
4) Oil Fire Trip
5) High Discharge Pressure / A ≈ 1275, / B ≈ 1240 psig
6) Low Bearing Oil Pressure / ≤ 4 psig
7) High SG Level / ≈ 98%
8) MSLB circuitry / ≤ 550 psig on either MS line (for 2 sec)
9) Thrust Wear
10) Low Suction Pressure / ≈ 235 psig ± 25

Lessons

ID	Description
CF-FPT	Main Feedwater Pump Turbines (CF-FPT)

Enabling Objectives

ID	Description

QUESTION # 33

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**1**	**1**
	K/A #	**013000 K2.01**	
	Importance Rating	**3.6**	**3.8**

Technical Reference(s): **EL-VPC**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EL-VPC #4.1**

Question Source:
- Bank # ______
- Modified Bank # **EL-219**
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test __X__

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 33

Which ONE of the following describes the designed backup power supply for the Oconee unit's DC distribution system?

________ DC busses on each unit are backed up from an alternate unit's associated DC busses via a/an ___________.

A. Vital / "make before break" ASCO switch

B. Vital / isolating diode assembly

C. Essential / "make before break" ASCO switch

D. Essential / isolating diode assembly

A. List three conditions that will cause a statalarm and identify the location of the alarms.

2.4 Explain the operation of the inverter fans and how their operation affects the inverter operation.

3. Describe, or draw, the power path from the DC power bus to the KU, KOAC, KI, and KX inverters, including the backup or AC Line source. (R3)

4. Discuss the Essential inverters. (R4)

4.1 Explain the difference between the vital and essential inverters.

4.2 Explain the difference between the KI&KX inverters and the KU inverter.

4.3 Demonstrate the ability to locate and explain all panel meters, lights, switches and breakers.

4.4 Discuss the operation of the essential inverters:

A. Explain the operation of the Manual Bypass switch.

B. Explain the operation of the Static Transfer switch.

C. Explain the function and operation of the ASCO Transfer switch.

D. Explain the operation and location of the Inverter Bypass switches.

1. Explain what would happen if both SW #2 and SW #3 Bypass switches were opened.

E. Explain the operation of the precharge switch.

1. Explain why the Precharge Light should be lit before closing the DC INPUT circuit breaker on an Essential inverter.

F. Describe the basic startup and shutdown of the inverters.

G. Describe the inverter status during normal operation.

4.5 Explain the statalarms associated with the essential inverters:

A. Identify three conditions that will cause a statalarm and identify the location of the alarms.

4.6 Explain the operation of the inverter fans and how their operation affects the inverter operation.

5. Explain the difference in the switching arrangements for Inverter and AC Line between the Vital Power panelboards and the Essential Power panelboards. (R5)

6. Explain how it is possible to interrupt power to the AC power panelboard if the three switches of the Static Inverter Bypass Switch are operated incorrectly. (R6)

t) **B4** - Alternate Source AC input (100 amp circuit breaker) - Used to connect the AC Line source (KRA) to the Manual Bypass switch. Normally closed during inverter operation.

u) **S1** - Manual Bypass Switch - Make before break switch that allows the operator to manually select either the Normal Source (Inverter Output) position "A" or the Alternate Source (AC Line - KRA) position "B" with no interruption of power to the panel. When the switch is in the Bypass position, the inverter is completely bypassed and the input as well as the output breaker can be opened to perform maintenance on the inverter.

B. **Vital Power System Operation** (Refer to OC-EL-VPC-1&3)

1. During normal system operation, each Vital Instrumentation Power Panelboard is energized via its associated Vital Bus inverter. Each inverter is energized from its associated 125VDC Instrumentation and Control Power Panelboard.

2. Each 125VDC Instrumentation Power Panelboard is energized from one of the unit's 125VDC Control Power buses and battery chargers through the associated set of isolating diodes. Should a problem develop with the battery charger or bus supplying a Vital Bus inverter, whereby this source is lost, an alternate unit's 125VDC supply will be auctioneered to supply the inverter through the isolating diode network without an interruption of power.

3. On the other hand, if a problem develops with a Vital Bus inverter and it fails to provide power, the associated 125VAC vital bus will lose power. There is no automatic transfer to the AC Line supply associated with the Vital Bus inverters.

   a) In order to re-energize the 125VAC vital bus, a manual transfer to AC Line must be made using the Manual Transfer switch.

4. Startup and shutdown:

   a) A simple startup procedure for the vital inverters would be as follows: (Assume that the 125VAC panelboard is being supplied by AC Line)

      1) Press the Precharge switch until the Precharge light is lit.

      2) Close the DC Input circuit breaker B1.

      3) Close the Inverter Output circuit breaker B2.

      4) Verify that the In Sync light is lit.

      5) Transfer the Manual Bypass switch to the Normal Source position.

2.3 **KOAC Computer Power System** (Refer to OC-EL-VPC-1,8,&9)

A. Purpose

1. The KOAC Computer Power System supplies power to various OAC equipment.

B. **KOAC Inverter**

1. The principles involved with the inversion of DC to AC by the KOAC Inverter are the same as those already discussed for every other inverter at the plant. The control panel for the KOAC Inverter is, of course, unique to itself; the various indications and controls on the panel, while sometimes using different nomenclature, serve identical purposes as the counterpart on the other plant inverters.

2. Where the other inverters in the plant are powered from the 125VDC I&C Buses, KOAC Inverters are powered from its unit's power battery bus, DP.

3. OP/1,2,3/1107/04, Operation of the Vital Bus, Computer, ICS, and Auxiliary Inverters, gives detailed operating instructions for startup, shutdown, and isolation of the KOAC Inverter.

2.4 **Regulated Power Supply** (Refer to 0C-EL-VPC-1,10, & 10A)

A. The Regulated Power System is a 120/240VAC, single-phase power system that serves as backup power for the equipment powered by the following systems:

1. 125VAC Vital Power (KVIA, KVIB, KVIC, and KVID).

2. Essential Power System (KI, KU, and KX).

3. KOAC Computer Power System

B. The system consists of:

1. two Regulated Power Panelboards, KRA and KRB

   a) KRA and KRB are permanently joined together by a jumper cable.

2. two redundant Voltage Regulators, A and B

3. two redundant 600/240/120V Transformers, A and B.

4. one ASCO Transfer Switch

NOTE: The KU inverter has exactly the same indications, lights and switches as the KI and KX inverters. However, it is a larger inverter, so the panel indications are spread out over two panels rather than one.

B. **Essential Power System Operation** (Refer to OC-EL-VPC 1&6)

1. During normal system operation, each Essential Power Panelboard is energized via its associated Essential inverter. Each inverter is energized from its associated 125VDC Instrumentation and Control Power Panelboard.

2. Each 125VDC Instrumentation and Control Power Panelboard is energized from one of the unit's 125VDC Control Power buses and battery chargers through the associated set of isolating diodes.

   a) Should a problem develop with the battery charger or bus supplying an Essential inverter, whereby this source is lost, the unit's other 125VDC Control Power bus will be auctioneered to supply the inverter through the isolating diode network without an interruption in power.

3. If a problem develops with an Essential inverter, and it fails to provide power, an automatic transfer to AC line should occur.

4. Transfer to Regulated Power (AC Line)

   a) Unlike the Vital inverters that can only be manually transferred to Regulated Power, the Essential inverters also have automatic transfers to Regulated Power. As a matter of fact, there are three possible manual transfers and two possible automatic transfers for each Essential inverter.

   b) Automatic transfers

      1) Static Transfer switch

         (a) The static transfer switch looks at the output voltage at two different locations. If the voltage at either one of these locations is lost, the static transfer switch will automatically swap the inverter to AC line within 1/4 cycle. The static transfer switch will also swap to AC line on high current.

            (1) If the inverter voltage is restored or the high current problem is resolved, the switch will swap back to the inverter in 30 seconds. This results in uninterrupted power to the panelboard.

2) ASCO Switch (Refer to OC-EL-VPC-7)

(a) Instances have occurred at Oconee where the inverter static transfer switch has failed to operate properly, generally due to blown fuses, resulting in a loss of power to the AC panelboard.

(b) To improve the reliability of the power supply to the panelboards, an ASCO automatic transfer switch has been incorporated to automatically swap the panelboard to the Regulated Power Supply if voltage to the panelboard is lost.

(c) To prevent competing with the static transfer switch operation, the ASCO backup switches are set to delay approximately 1/2 to 1 second before switching, after a power interruption.

(d) Several controls and indications on the backup transfer switch panel are use to monitor switch operation and to control switching action:

(1) TRANSFER TO EMERGENCY switch - This two-position toggle switch is used to force a transfer between Regulated Power (EMERGENCY) and Inverter (NORMAL).

(2) RESET TO NORMAL switch - Spring-loaded toggle switch used to return the backup transfer switch to a normal configuration (inverter supplying the panelboard) after an automatic transfer operation.

(3) LOAD CONNECTED TO NORMAL light - Will be lit if the backup transfer switch contacts have the inverter output connected to the AC panelboard.

(4) LOAD CONNECTED TO EMERGENCY light - Will be lit if the backup transfer switch contacts have the Regulated Power Supply connected to the AC panelboard.

c) Manual Transfers

1) Manual Bypass switch

(a) The Manual Bypass switch is internal to the inverter. It is a two position make before break switch that can be selected to either the Normal Source (inverter output) or the Alternate Source (AC Line).

2) Static Transfer switch

   (a) The Static Transfer switch can be forced to transfer using the pushbuttons on the front of the panel.

3) Inverter Bypass Switches

   (a) An enclosure that sits near the static inverter cabinet, and contains three breaker-type switches, SW-1, SW-2, and SW-3.

   (b) The switches are arranged so that the inverter can be completely isolated from the AC panelboard and the Regulated Power Supply, while the Regulated Power Supply is connected directly to the AC panelboard.

   (c) This arrangement allows for complete isolation of the inverter for maintenance.

   (d) From the diagram of the inverter circuit (OC-EL-VPC-6), it can be seen that :

      (1) Anytime SW-2 is closed, the AC panelboard is connected directly to the Regulated Power Supply.

      (2) In order for the inverter automatic transfer action to function, both SW-1 and SW-3 must be closed.

      (3) SW-3 must be closed to connect the output of the inverter to the AC panelboard.

      (4) If both SW-2 and SW-3 are open, the AC panelboard will lose power.

C. **Alarms**

1. Each Essential inverter (KI,KU&KX) has an Inverter System Trouble statalarm that will alarm in the Control Room.

2. These alarms are fed from local panels (1)(2)(3)SA12 and (!)(2)(3)SA13 which will alarm for the following conditions:

   a) Low input voltage

   b) Low output voltage

   c) Bypass voltage failure (Low AC Line voltage)

   d) ICS/Aux or Computer Inverter Bypass Switch/ASCOTransfer Switch

Exam Question Report

27-Jan-99

Question ID:	EL219	Revision No:	0	Revision Date	10/29/1999
Question Description:	EL219				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EL-DCD - DC Power Distribution
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	Reference: NRC #083	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Which ONE of the following describes the design of the Oconee DC distribution system for providing backup DC power? (.25)

A) DC Vital I&C buses on each unit are backed up from an alternate unit's Vital DC buses by a fast closing breaker.

B) DC Vital I&C buses on each unit are backed up from an alternate unit's Vital DC buses by an isolating diode assembly.

C) Essential DC buses on each unit are backed up from an alternate unit's Essential DC buses by a fast closing breaker.

D) Essential DC buses on each unit are backed up from an alternate unit's Essential DC buses by isolating diode assemblies.

Answer

B Ref: NRC #083

Lessons

ID	Description	
EL-DCD	DC Power Distribution	EL-DCD

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

QUESTION # 34

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**015000 A4.03**	
Importance Rating	**3.8**	**3.9**

Technical Reference(s): **IC-RPS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-RPS #10**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 34

Unit 2 plant conditions:

- Reactor startup is in progress
- Reactor power = 1% and steady

Which ONE of the following is correct concerning the status of the "MFP Trip Bypass" bistable?

The bistable OUTPUT STATE light is _____ to indicate that the bypassing action _____ in effect and the OUTPUT MEMORY light is __________.

A. bright / is / bright.

B. bright / is not / bright.

C. dim / is / dim.

D. dim / is not / dim.

1 POINT

QUESTION # 34

015000A4.03

A. Correct, - The "Both MFWP's Trip to Reactor Trip" contact buffer action is automatically bypassed during plant shutdown when reactor power has been reduced to 0.5%. Bypassing automatically occurs in each channel by tripping of the "MFP's Trip Bypass" bistable in each channel. The OUTPUT STATE light goes bright to indicate that the bistable has tripped and the bypassing action is in effect. If a bistable trips, output memory light goes bright until manually reset to indicate the bistable has tripped. The bistable does not automatically reset until reactor power is 1.75%.

B. Incorrect - The first part of the distracter is incorrect. At 1% power the bistable has not reset and bypassing action is still in effect.

C. Incorrect - If a dim light indicated a bypassed condition instead of a bright light this would be correct. See a. above.

D. Incorrect, - See a. above. This will be the status after power is increased to 1.75% and the output memory has been reset.

5.4 What administratively controlled action is required when SD Bypass is selected, and the basis for that action.

5.5 The basic operation required to place each RPS channel in SD Bypass, and the indications that alert the operator that a channel is in SD Bypass.

5.6 The consequence of selecting SD Bypass during full power operation.

6. Explain the following concerning the Manual Bypass (channel trip bypass) function in RPS: (R6)

6.1 The effect on an RPS channel of placing that channel in Manual Bypass.

6.2 The meaning of two-out-of four and two-out-of three logic in RPS.

6.3 When Manual Bypass is used.

6.4 The basic operation required to place an RPS channel in Manual Bypass, and the indications that alert the operator that a channel is in Manual Bypass.

6.5 What administrative limit is imposed on the use of Manual Bypass, and what safeguards are used to insure compliance with this limit.

7. Explain the following relative to a bistable: (R7)

7.1 basic electronic operation

7.2 two basic functions bistables serve in RPS

7.3 function and/or operation of each operator-related indication and control on a bistable module

8. Explain the following relative to a STAR Module (R29)

8.1 Inputs

8.2 Outputs

8.3 Normal operation

8.4 Trip conditions; indications when tripped and methods used to manually trip.

9. Explain the following relative to a dummy bistable: (R8)

9.1 purpose

9.2 restrictions placed upon use of dummy bistables and how those restrictions are enforced

9.3 how a dummy bistable is distinguished from a normal bistable module, and how the operator is alerted to the use of a dummy bistable in an RPS channel.

10. Explain the following concerning contact buffers in RPS: (R9)

10.1 Describe their basic electronic operation

10.2 Discuss the indications and controls located on the front of a contact buffer.

10.3 List the four uses of contact buffers in each RPS channel.

10.4 Discuss in detail the bypassing operation for the MT and FWP contact buffers, including purpose, setpoints, and all indications involved.

10.5 Discuss the consequences of operator error in contact buffer operation.

11. Describe the operation of the contact monitors used in the RPS, including the basic purpose, how signals are developed, and where these signals are used. (R11)

12. Explain the following concerning signal converters in the RPS: (R13)

    12.1 basic use

    12.2 purpose of the meter "mode" select toggle switch

13. Explain the following relative to RCS flow and RCS pressure signals in the RPS: (R14)

    13.1 How RCS pressure measurements are used in each RPS channel to provide reactor protection.

    13.2 The differences between the RPS channel 'A' and 'B' RCS flow and pressure circuits and those in channels C and D; and why those differences exist.

14. Explain or list the following relative to a Reactor Trip Module in the RPS: (R16)

    14.1 The basic purpose of the RT Module

    14.2 The operator-related indications and controls located on the front of each RT Module, and the purpose and operation of each.

15. Explain how the UV and Shunt Trip devices on each CRD breaker work. (R17)

16. List the power supplies used in each RPS cabinet and the consequences of the loss of each during full power operation. (R18)

17. Describe how to restore an RPS channel to normal operation following a loss of the AC vital power supply to it. (R19)

18. Identify which RPS channel and vital power supply is associated with each CRD breaker. (R20)

19. Given a one-line diagram of a typical RPS Trip String detail. (R21)

    19.1 Explain how AC power is delivered to the UV coils and ST relays.

    19.2 Describe how each component indicated on the diagram relates to the normal operation of the RPS channel.

because of noise or small variations present in the monitored analog signal.

(c) If a bistable is set to trip exactly at set point and reset exactly at set point, it would go through a rapid trip/reset series as the parameter neared the set point value.

(d) Dead band sets a small band between trip and reset points so that positive trip and reset actions can be performed.

7) Input, Dead band, and Setpoint jacks

(a) Used by I&E to connect digital voltmeter to read voltage corresponding to measured parameter value to the bistable, the dead band corresponding voltage, and the setpoint corresponding voltage.

4. STAR Processor Module **(OP-IC-RPS-34)**

a) The STAR Processor module is a microprocessor-based digital system that is designed to replace analog components in the B&W designed RPS. We are currently using the module for the Flux/Flow/Imbalance trip on all three units.

b) The module is capable of reading input signals from the plant and sending outputs that can be used to provide trips or actuations of safety system equipment, control a process, or provide alarms and indications.

1) Module has the capability to accept an input of seven (7) analog signals and output four (4). It can also input and output twelve (12) digital signals.

2) The inputs currently used are:

(a) Reactor Power in the upper portion of the core

(b) Reactor Power in the lower portion of the core

(c) Loop "A" Delta Pressure (flow)

(d) Loop "B" Delta Pressure (flow)

3) The outputs calculated by the module are:

(a) Loop "A" RCS Flow

(b) Loop "B" RCS Flow

(c) Total Flow

(d) Maximum Allowed Power

(e) Delta Flux

(f) Reactor Power

(g) Trip on Flux/Flow/Imbalance

(b) The bistable OUTPUT STATE light goes bright to indicate that the bistable has tripped and the bypassing action is in effect.

5) The contact buffer action is automatically re-inserted during SU when reactor power is increased to 1.75%, when the bistable automatically resets at this point.

(a) The bistable OUTPUT STATE light goes dim to indicate that the bistable has reset and bypassing action is no longer in effect. The OUTPUT MEMORY light will remain bright until manually reset.

6) CAUTION: None of the contact buffers for MT Trip or FWPs Trip automatically reset. They must be manually reset when the associated setpoints have returned above the trip points. If the contact buffers are still in the trip state when the bypassing bistables automatically reset, the associated RPS channels will trip.

j) Besides the indications on each contact buffer and bistable combination in the RPS cabinet, remote alarms in the form of annunciator alarms on alarm panel SA-18 in the Control Room will also sound to alert the operator to their status:

1) RPS FWPT/Reactor Trip P.S. Alert"

(a) Sounds if <u>any</u> of the eight <u>contact buffers</u> in the RPS channels monitoring MFWPs status trips.

(b) "P.S." refers to the pressure switches that monitor the 75 psig hydraulic oil pressure.

(c) "Alert" means that the operator is alerted to the fact that the RPS trip is imminent - if bypass is not in effect and the second contact buffer in that channel trips, the channel will trip.

(d) This alarm will not clear until <u>all</u> contact buffers monitoring the MFWPs have been manually reset.

2) "RPS FWPT/Reactor Channel Trip Bypass"

(a) Sounds when the <u>first</u> <u>bistable</u> in any of the four RPS channels trips to automatically bypass the MFWPs trip to Reactor Trip.

(b) Actuates when reactor power reaches 0.5% decreasing.

(c) Clears when the <u>last</u> bistable automatically resets as reactor power increases to 1.75%.

3) "RPS Turb/React Trip P.S. Alert"

(a) Sounds if <u>any</u> of the four <u>contact buffers</u> in the RPS channels monitoring EHC ETS pressure trip.

(b) Alarm will not clear until all four contact buffers have been manually reset.

4) "RPS Turb/React Channel Trip Bypass"

(a) Sounds when the first bistable in any of the four RPS channels trips to automatically bypass the MT Trip to Reactor Trip.

(b) Actuates when reactor power reaches 28% decreasing.

(c) Clears when the last bistable automatically resets as reactor power increases to 30%.

7. Auxiliary Relay **(OC-IC-RPS-13)**

a) Primary purpose of Auxiliary Relay module is to provide auxiliary contacts for external outputs to annunciators, computer points, etc.

b) Each Auxiliary Relay module houses four auxiliary relays and indicating lights.

c) The number of auxiliary relay contacts for a particular parameter will depend upon the number of annunciators or other devices output to.

d) Auxiliary relays are used in the RPS cabinets to provide isolated outputs to plant indications, computer, annunciators, etc..

1) Power Range Test Module not in "Operate" position.

2) Contact Monitor Test Module not in "Operate".

3) RCS Pressure Test Module not in "Operate".

4) RCS Temperature Test Module not in "Operate".

5) Contact Buffer monitoring RB pressure has tripped.

6) Shutdown Bypass (channel S/D Bypass Initiated and S/D Bypass Hi Press Trip Bistable Tripped).

7) Reactor Power high flux trip bistable tripped.

8) RCS high temp trip bistable tripped.

9) RCS high press trip bistable tripped.

10) STAR Module tripped.

11) RCS variable low press (press/temp) trip bistable tripped.

12) RCS low press trip bistable tripped.

13) RPS channel in Manual Bypass.

14) A dummy bistable installed in the RPS channel.

15) STAR Module jumpered out.

8. Contact Monitor **(OC-IC-RPS-14)**

QUESTION # 35

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**1**	**1**
	K/A #	**022000 A1.01**	
	Importance Rating	**3.6**	**3.7**

Technical Reference(s): **Steam Table**
PNS-RBS

Proposed references to be provided to applicants during examination: **Steam Table**

Learning Objective: **PNS-BS #2 & #6**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 35

Jnit 3 plant conditions:

- RCS pressure = 1100 psig decreasing
- ALL RB temperature indications and functions are inoperable
- Reactor Building Spray has just actuated
- Loss of SCM Setpoint calculation in progress

Which ONE of the following is correct?

Reactor Building temperature is ≈ _____°F.

A. 286

B. 240

C. 222

D. 150

1 POINT

QUESTION # 35

022000 A1.01 Both PRA 2-9-00

QUESTION SETUP: During a transient that has elevated RB temperatures above normal operating temperatures the operator must compensate RB level instrumentation due to the inaccurate level indications provided. Without RB temperature available the operator must use an alternate means for identifying the actual RB temperature. If a LOCA has occurred and RB pressure increases to the RBS actuation setpoint of 10 psig then the operator can obtain RB temperature by determining the saturation temperature for 10 psig.

A. Incorrect – This is the design temperature for the RB

B. Correct - This is the temperature ES RB Spray will occur at 10 psig. 14.7 psi + 10.0 psig = 25 psia and saturation for 25 = 240°F.

C. Incorrect – This is the temperature ES RB Cooling Unit and Essential RB isolation will occur at 3 psig 14.7 psi + 3.0 psig = 17.7 psia and saturation for 17.7 = 222°F.

D. Incorrect – This is the temperature that the RBCU drop-out plates release from the RBCU duct work to ensure proper cooling flow path.

1 POINT

QUESTION # 35

Unit 3 plant conditions:

- RCS pressure = 1100 psig decreasing
- RB pressure = 2.3 psig and increasing

Which ONE of the following is correct?

Reactor Building temperature will be ≈ _____°F when Reactor Building Spray **<u>actuates</u>**.

A. 286

B. 240

C. 222

D. 150

1 POINT

QUESTION # 35

022000 A1.01 Both PRA 2-9-00

A. Incorrect – This is the design temperature for the RB

B. Correct - This is the temperature ES RB Spray will occur at 10 psig. 14.7 psi + 10.0 psig = 25 psia and saturation for 25 = 240°F.

C. Incorrect – This is the temperature ES RB Cooling Unit and Essential RB isolation will occur at 3 psig 14.7 psi + 3.0 psig = 17.7 psia and saturation for 17.7 = 222°F.

D. Incorrect – This is the temperature that the RBCU drop-out plates release from the RBCU duct work to ensure proper cooling flow path.

Table 1. Saturated Steam: Temperature Table—*Continued*

Temp Fahr t	Abs Press Lb per Sq In p	Specific Volume Sat Liquid v_f	Specific Volume Evap v_{fg}	Specific Volume Sat Vapor v_g	Enthalpy Sat Liquid h_f	Enthalpy Evap h_{fg}	Enthalpy Sat Vapor h_g	Entropy Sat Liquid s_f	Entropy Evap s_{fg}	Entropy Sat Vapor s_g	Temp Fahr t
180.0	7.5110	0.016510	50.21	50.22	148.00	990.2	1138.2	0.2631	1.5480	1.8111	180.0
182.0	7.850	0.016522	48.172	18.189	150.01	989.0	1139.0	0.2662	1.5413	1.8075	182.0
184.0	8.203	0.016534	46.232	46.249	152.01	987.8	1139.8	0.2694	1.5346	1.8040	184.0
186.0	8.568	0.016547	44.383	44.400	154.02	986.5	1140.5	0.2725	1.5279	1.8004	186.0
188.0	8.947	0.016559	42.621	42.638	156.03	985.3	1141.3	0.2756	1.5213	1.7969	188.0
190.0	9.340	0.016572	40.941	40.957	158.04	984.1	1142.1	0.2787	1.5148	1.7934	190.0
192.0	9.747	0.016585	39.337	39.354	160.05	982.8	1142.9	0.2818	1.5082	1.7900	192.0
194.0	10.168	0.016598	37.808	37.824	162.05	981.6	1143.7	0.2848	1.5017	1.7865	194.0
196.0	10.605	0.016611	36.348	36.364	164.06	980.4	1144.4	0.2879	1.4952	1.7831	196.0
198.0	11.058	0.016624	34.954	34.970	166.08	979.1	1145.2	0.2910	1.4888	1.7798	198.0
200.0	11.526	0.016637	33.622	33.639	168.09	977.9	1146.0	0.2940	1.4824	1.7764	200.0
204.0	12.512	0.016664	31.135	31.151	172.11	975.4	1147.5	0.3001	1.4697	1.7698	204.0
208.0	13.568	0.016691	28.862	28.878	176.14	972.8	1149.0	0.3061	1.4571	1.7632	208.0
212.0	14.696	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	212.0
216.0	15.901	0.016747	24.878	24.894	184.20	967.8	1152.0	0.3181	1.4323	1.7505	216.0
220.0	17.186	0.016775	23.131	23.148	188.23	965.2	1153.4	0.3241	1.4201	1.7442	220.0
224.0	18.556	0.016805	21.529	21.545	192.27	962.6	1154.9	0.3300	1.4081	1.7380	224.0
228.0	20.015	0.016834	20.056	20.073	196.31	960.0	1156.3	0.3359	1.3961	1.7320	228.0
232.0	21.567	0.016864	18.701	18.718	200.35	957.4	1157.8	0.3417	1.3842	1.7260	232.0
236.0	23.216	0.016895	17.454	17.471	204.40	954.8	1159.2	0.3476	1.3725	1.7201	236.0
240.0	24.968	0.016926	16.304	16.321	208.45	952.1	1160.6	0.3533	1.3609	1.7142	240.0
244.0	26.826	0.016958	15.243	15.260	212.50	949.5	1162.0	0.3591	1.3494	1.7085	244.0
248.0	28.796	0.016990	14.264	14.281	216.56	946.8	1163.4	0.3649	1.3379	1.7028	248.0
252.0	30.883	0.017022	13.358	13.375	220.62	944.1	1164.7	0.3706	1.3266	1.6972	252.0
256.0	33.091	0.017055	12.520	12.538	224.69	941.4	1166.1	0.3763	1.3154	1.6917	256.0
260.0	35.427	0.017089	11.745	11.762	228.76	938.6	1167.4	0.3819	1.3043	1.6862	260.0
264.0	37.894	0.017123	11.025	11.042	232.83	935.9	1168.7	0.3876	1.2933	1.6808	264.0
268.0	40.500	0.017157	10.358	10.375	236.91	933.1	1170.0	0.3932	1.2823	1.6755	268.0
272.0	43.249	0.017193	9.738	9.755	240.99	930.3	1171.3	0.3987	1.2715	1.6702	272.0
276.0	46.147	0.017228	9.162	9.180	245.08	927.5	1172.5	0.4043	1.2607	1.6650	276.0
280.0	49.200	0.017264	8.627	8.644	249.17	924.6	1173.8	0.4098	1.2501	1.6599	280.0
284.0	52.414	0.01730	8.1280	8.1453	253.3	921.7	1175.0	0.4154	1.2395	1.6548	284.0
288.0	55.795	0.01734	7.6634	7.6807	257.4	918.8	1176.2	0.4208	1.2290	1.6498	288.0
292.0	59.350	0.01738	7.2301	7.2475	261.5	915.9	1177.4	0.4263	1.2186	1.6449	292.0
296.0	63.084	0.01741	6.8259	6.8433	265.6	913.0	1178.6	0.4317	1.2082	1.6400	296.0
300.0	67.005	0.01745	6.4483	6.4658	269.7	910.0	1179.7	0.4372	1.1979	1.6351	300.0
304.0	71.119	0.01749	6.0955	6.1130	273.8	907.0	1180.9	0.4426	1.1877	1.6303	304.0
308.0	75.433	0.01753	5.7655	5.7830	278.0	904.0	1182.0	0.4479	1.1776	1.6256	308.0
312.0	79.953	0.01757	5.4566	5.4742	282.1	901.0	1183.1	0.4533	1.1676	1.6209	312.0
316.0	84.688	0.01761	5.1673	5.1849	286.3	897.9	1184.1	0.4586	1.1576	1.6162	316.0
320.0	89.643	0.01766	4.8961	4.9138	290.4	894.8	1185.2	0.4640	1.1477	1.6116	320.0
324.0	94.826	0.01770	4.6418	4.6595	294.6	891.6	1186.2	0.4692	1.1378	1.6071	324.0
328.0	100.245	0.01774	4.4030	4.4208	298.7	888.5	1187.2	0.4745	1.1280	1.6025	328.0
332.0	105.907	0.01779	4.1788	4.1966	302.9	885.3	1188.2	0.4798	1.1183	1.5981	332.0
336.0	111.820	0.01783	3.9681	3.9859	307.1	882.1	1189.1	0.4850	1.1086	1.5936	336.0
340.0	117.992	0.01787	3.7699	3.7878	311.3	878.8	1190.1	0.4902	1.0990	1.5892	340.0
344.0	124.430	0.01792	3.5834	3.6013	315.5	875.5	1191.0	0.4954	1.0894	1.5849	344.0
348.0	131.142	0.01797	3.4078	3.4258	319.7	872.2	1191.1	0.5006	1.0799	1.5806	348.0
352.0	138.138	0.01801	3.2423	3.2603	323.9	868.9	1192.7	0.5058	1.0705	1.5763	352.0
356.0	145.424	0.01806	3.0863	3.1044	328.1	865.5	1193.6	0.5110	1.0611	1.5721	356.0
360.0	153.010	0.01811	2.9392	2.9573	332.3	862.1	1194.4	0.5161	1.0517	1.5678	360.0
364.0	160.903	0.01816	2.8002	2.8184	336.5	858.6	1195.2	0.5212	1.0424	1.5637	364.0
368.0	169.113	0.01821	2.6691	2.6873	340.8	855.1	1195.9	0.5263	1.0332	1.5595	368.0
372.0	177.648	0.01826	2.5451	2.5633	345.0	851.6	1196.7	0.5314	1.0240	1.5554	372.0
376.0	186.517	0.01831	2.4279	2.4462	349.3	848.1	1197.4	0.5365	1.0148	1.5513	376.0
380.0	195.729	0.01836	2.3170	2.3353	353.6	844.5	1198.0	0.5416	1.0057	1.5473	380.0
384.0	205.294	0.01842	2.2120	2.2304	357.9	840.8	1198.7	0.5466	0.9966	1.5432	384.0
388.0	215.220	0.01847	2.1126	2.1311	362.2	837.2	1199.3	0.5516	0.9876	1.5392	388.0
392.0	225.516	0.01853	2.0184	2.0369	366.5	833.4	1199.9	0.5567	0.9786	1.5352	392.0
396.0	236.193	0.01858	1.9291	1.9477	370.8	829.7	1200.4	0.5617	0.9696	1.5313	396.0
400.0	247.259	0.01864	1.8444	1.8630	375.1	825.9	1201.0	0.5667	0.9607	1.5274	400.0
404.0	258.725	0.01870	1.7640	1.7827	379.4	822.0	1201.5	0.5717	0.9518	1.5234	404.0
408.0	270.600	0.01875	1.6877	1.7064	383.8	818.2	1201.9	0.5766	0.9429	1.5195	408.0
412.0	282.894	0.01881	1.6152	1.6340	388.1	814.2	1202.4	0.5816	0.9341	1.5157	412.0
416.0	295.617	0.01887	1.5463	1.5651	392.5	810.2	1202.8	0.5866	0.9253	1.5118	416.0
420.0	308.780	0.01894	1.4808	1.4997	396.9	806.2	1203.1	0.5915	0.9165	1.5080	420.0
424.0	322.391	0.01900	1.4184	1.4374	401.3	802.2	1203.5	0.5964	0.9077	1.5042	424.0
428.0	336.463	0.01906	1.3591	1.3782	405.7	798.0	1203.7	0.6014	0.8990	1.5004	428.0
432.0	351.00	0.01913	1.30266	1.32179	410.1	793.9	1204.0	0.6063	0.8903	1.4966	432.0
436.0	366.03	0.01919	1.24887	1.26806	414.6	789.7	1204.2	0.6112	0.8816	1.4928	436.0
440.0	381.54	0.01926	1.19761	1.21687	419.0	785.4	1204.4	0.6161	0.8729	1.4890	440.0
444.0	397.56	0.01933	1.14874	1.16806	423.5	781.1	1204.6	0.6210	0.8643	1.4853	444.0
448.0	414.09	0.01940	1.10212	1.12152	428.0	776.7	1204.7	0.6259	0.8557	1.4815	448.0
452.0	431.14	0.01947	1.05764	1.07711	432.5	772.3	1204.8	0.6308	0.8471	1.4778	452.0
456.0	448.73	0.01954	1.01518	1.03472	437.0	767.8	1204.8	0.6356	0.8385	1.4741	456.0

OBJECTIVES

TERMINAL OBJECTIVES

1. Describe the purpose, location and modes of operation in regard to the RBS System. The student should also recognize important power supplies associated with the system. (T1)

2. Assess the status of the RBS system during various system conditions to verify proper operation and determine any required corrective actions.(T2)

ENABLING OBJECTIVES

1. State the two (2) purposes of the RBS System. (R1)

2. Given a set of conditions, determine if containment design pressure and temperature limits will be met. (R16)

3. List the power supplies for the RBS Pumps (R3)

4. List the following flow values for the RBS pumps. (R2, R7)

   4.1 Minimum flow requirement

   4.2 Normal ES flow when taking suction from BWST

   4.3 Normal flow when taking suction from RB Emergency Sump (RBES)

5. Draw the RBS System labeling all major components and valves. Include the following: (R5, R10)

   5.1 BWST

   5.2 RBES

   5.3 Recirc flowpath to BWST (for testing)

6. State the setpoint, statalarms armed, and equipment actuated by ES Channels 7 and 8. (R6)

7. For PT/0204/007, RBS Pump Test, describe: (R12)

   7.1 The purpose

   7.2 How the test is performed

1. INTRODUCTION

1.1 Purposes of the Reactor Building Spray (RBS) System

A. The Reactor Bldg. Spray system has no function during normal plant operation.

B. When actuated by high Reactor Building (RB) pressure, the system provides two major functions:

1. Removes sensible and latent heat from the containment atmosphere. This system, in conjunction with the RB Cooling System and the LPI System, is capable of removing sufficient heat from the containment atmosphere to maintain the Reactor Building post-accident conditions within design limits (59 psig, 286°F).

2. Operation of the RBS System also serves to entrain fission product iodine's (released into the RB during a LOCA) into the spray water, thereby reducing possible radio-iodine leakage to the environment (to meet 10CFR100 criteria concerning offsite dose limits).

1.2 General System Description

A. Equipment

1. The RBS System consists of two independent trains.

2. Each train consists of one RBS pump, which initially takes suction from the BWST via the LPI suction piping, and discharges through a discharge throttle valve to its RBS nozzle header at the RB dome following a high energy line break.

B. Cooling Capacity

1. This system is able (per the FSAR), independently of the RBC System, to furnish 100% of the design basis cooling in the RB atmosphere, after an accident, if <u>both</u> RBS trains are operating properly.

   Assuming single failure criteria on both the RBS and the RBC systems, one RBS train and two RBCU's would be available in an accident. During an accident, a minimum of two RBCU's and one RBS train are required to maintain containment pressure and temperature following a LOCA. Additionally, the one RBS train is also required to remove iodine from containment atmosphere and maintain concentrations below those assumed in the safety analysis.

2) Non-safety related outputs:

(a) BS Flow Train A and B flow gauge on VB2

(b) RBS Flow High/Low statalarm

(c) OAC Computer point (BS Line A Flow)

(d) OAC Computer point (BS Line B Flow)

2.2 Modes of Operation

A. ES Mode (Channels 7 and 8)

1. Setpoint

a) The RBS System automatically actuates if two of the three ESG RB pressure analog channels reach 10 **psig.**

1) The ITS required setpoint is $\leq$ 15 psig RB pressure.

2. The following actions occur if the RBS System actuates:

a) Both RBS pumps start.

b) BS-1 and BS-2 open.

c) LP-21 and LP-22 receive an open signal to supply RBS pumps from BWST.

1) These valves are normally open, but receive an ES signal in case they are closed.

d) Refer to OP-OC-BS-3 to explain the ES flowpath.

3. When ES-7 and 8 actuate, the "BS Header Flow High/Low" statalarm for each header is armed.

a) Low flow on a given header alarms at 1300 gpm.

1) The operator is directed to check RBS pump operation, BWST level, and suction and discharge valve position by the Alarm Response Guide.

b) High flow on a given header alarms at 1700 gpm.

1) The operator is referred to the EOP for guidance to throttle RBS flow to within operating limits.

4. Nominal RBS pump flow is 1500 gpm when suction to the pumps is from the BWST.

a) BS-1 and BS-2 should be throttled as required to maintain this flow rate as RB pressure changes occur following a LOCA.

b) Flow Indication and Recorders are located on VB2.

QUESTION # 24

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**2**	**2**
	K/A #	**000054 G2.4.11**	
	Importance Rating	**3.4**	**3.7**

Technical Reference(s): **CF-EF p.#23 & #26**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-EF #27**

Question Source:
- Bank # __X__
- Modified Bank # ______
- New ______

Question History:
- Previous NRC Exam **1998-#31**
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 __X__

Comments:

1 POINT

QUESTION # 24

Unit 3 plant conditions:

INITIAL CONDITIONS:

- TIME = 0845
- An automatic reactor trip from 50% power occurs

CURRENT CONDITIONS:

- TIME = 0900
- 3FDW-35 and 3FDW-44 (3A and 3B Startup FDW Control) failed closed
- 3FDW-31 and 3FDW-40 (3A and 3B Main FDW Block) failed closed
- 3FDW-315 (SG EFDW Control Valve to 3A SG) failed closed
- 3A Main FDWP operating

Which ONE of the following provides the levels at which OTSG 3A and 3B will stabilize?

ASSUME NO operator action

3A OTSG level _____ inches of the XSUR and 3B OTSG level _____ inches on the XSUR.

A. 30 / 25

B. 25 / 20

C. 14 / 20

D. 14 / 30

1 POINT

QUESTION # 24

000054 G2.4.11

Question setup:
Following the reactor trip the SGs will boil down to post trip level in ≈ 5-8 minutes. Normally ICS will control post trip SG level at (Low Level Limit LLL) 25 inches S/U Level via the Startup Control Valves (FDW-35 and 44). If the Main FDWPs trip the EFDW system will control post trip SG level at 30 inches XSUR. With 3FDW-31 and 3FDW-40 (3A and 3B Main FDW Block) failed closed and 3FDW-35 and 3FDW-44 (3A and 3B Startup FDW Control) failed closed Main FDW cannot feed either SG. Both SGs will boil down. When SG level reaches 21 inches for 30 second dry out protection will start the EFDW. EFDW will attempt to control both SG levels at 30 inches XSUR. FDW-315 failed closed will cause the "A" SG will remain dry (14 inches) and the "B" SG will control at 30 inches XSUR.

A. Incorrect – This would be correct if 3FDW-315 was operable. The second part of the distracter would be correct if ICS was maintaining the SG level.

B. Incorrect –This would be correct if FDW-35 ("A" SU FDW Control Valve) was operable. The second part of the distracter is incorrect as the SG level is not maintained at the dry out setpoint.

C. Incorrect – "A" SG will dry out and indicate 14 inches. The "B" SG will boil down to 20 inches (dry out setpoint) but will increase level as the EFDW system automatically controls 30 inches

D. Correct – EFDW will auto start on dry out protection and control 3B SG level at 30 inches XSUR. Failure of 3FDW-315 results in the inability to feed SG "A". Level will indicate 14 inches.

12. Describe the operation of the TDEFDWP bearing cooling jacket water system. (R10)
13. Describe the minimum recirculation flow path for the TDEFDWP. (R11)
14. List the sources of steam for the TDEFDWP. (R12)
15. Describe the purpose and operation of MS-93. (R24)
16. Describe how to manually open MS-93. (R43)
17. Explain the operation of the Primary Relay associated with the TDEFDWP including a description of the operation of the Speed Governor and Pilot Valve. (R13)
18. Explain how to use the Hand-Start lever of the Primary Relay to start the TDEFDWP in the event that the Auxiliary Oil Pump does not start when MS-93 opens. (R44)
19. List the two functions of the Trip Throttle valve and explain the operation of the Trip Throttle and Operating Valve associated with the TDEFDWP. (R14)
20. Describe how to reset the Trip Throttle Valve and what to look for to verify that it is reset. (R15)
21. Explain the operation of the Overspeed Governor and Emergency Relay associated with the TDEFDWP. (R16)
22. Describe how control oil and lube oil are supplied to the TDEFDWP during startup and operation. (R17)
23. List the normal and backup cooling medium for the TDEFDWP oil cooler. (R18)
24. Explain how the EFDW Systems can be cross-connected between units. (R19)
25. Describe the MANUAL and AUTOMATIC (including Auto 1 & AUTO 2) control available for the MDEFDWPs and their purposes. (R20)
26. Explain how to stop the MDEFDWPs following an the AUTO START or a MANUAL START.(R21)
27. Describe or make a sketch of the logic/conditions that will AUTO START the MDEFDWPs when the respective control switches are in AUTO, including a description of AMSAC and DRY OUT PROTECTION (R22)
28. Describe the purpose for AMSAC/DSS, including actuating setpoints and functions they provide following actuation. (R61)

Instrumentation and Controls

1. MDEFDWP

   1.1 Manual Control (Figure OC-CF-EF-13)

   - 4 position - OFF • AUTO 1 • AUTO 2 • RUN
   - No interlocks to prevent Manual start
   - Trip or OFF - prevents Auto start, secures pump after start

   1.2 Automatic Control (Figure-OC-CF-EF-13,21&22)

   A. AUTO 1 position (Dryout Protection)

      1. Purpose: Provide a diverse means of actuating MDEFDWPs as the SGs are approaching dryout conditions.
      2. Upon receiving a two out of two low XSUR level logic signal in either SG (21" for 30 seconds), both MDEFDWPs will start and the EFDW level control will initiate to control at the appropriate XSUR level depending upon operation of the RCPs. (30"/240")

   B. AUTO 2 position, MDEFDWPs will start when:

Both MFDWPs have low hydraulic oil pressure (< 75 psig)
OR
AMSAC/DSS (Diverse Scram System) enabled, AND
Both MFDWPs have low hydraulic oil pressure (< 75 psig)
(This must occur on both AMSAC Channels)
OR
AMSAC/DSS enabled, AND
Both MFDWPs have low discharge pressure (< 770 psig)
(both AMSAC Channels)

3. Only one train required to trip to actuate isolation

B. System Actuation

1. Any two out of three outlet pressure transmitters on either OTSG reaches 550# (± 50#) decreasing.

2. 2 seconds after circuit initiation prevents TDEFDWP from starting and stops the TDEFDWP if running in automatic.

   a) Placing the control switch to "RUN you can manually start pump.

3. 2 seconds after circuit initiation trips both MFDWPs and closes the following FDW valves:

   a) FDW-31 (A Main FDW Block Valve)

   b) FDW-33 (A S/U FDW Block Valve)

   c) FDW-40 (B Main FDW Block Valve)

   d) FDW-42 (B S/U FDW Block Valve)

4. 5 seconds after actuation, (to prevent water hammer because CVs close faster) closes the following:

   a) FDW-32 (A Main FDW Control Valve)

   b) FDW-41 (B Main FDW Control Valve)

   c) FDW-35 (A S/U FDW Control Valve)

   d) FDW-44 (B S/U FDW Control Valve)

C. The main steam line break circuit must be disabled, to regain control of the startup feedwater valves, then align and feed through the alternate flowpath.

4. Level Control System

4.1 Description

A. Auto selected: Level Control System signal is passed through to the valve.

- 30" XSUR – any RCP running
- 240" XSUR – loss of all RCPs (natural circulation)

Note: These levels may have to be adjusted manually by the reactor operator for degraded containment conditions following actuation to: [60" acc or 270" acc] as required.

B. Manual selected: Manual control signal is passed through to the valve.

C. Powered from KVIB and KVIC

D. Placing FDW-315 and/or FDW-316 in AUTO will cause the respective valve(s) to go on auto level control.

QNUM	11
HNUM	
QCHANGE	NEW
ACHOICE	
BCHOICE	
CCHOICE	
DCHOICE	
ANSCHANGE	
DAREA	
EXAM TYPE	NRC
QDATE	9/16/98
FAC	269 Oconee 1, 2, & 3
RTYP	B&W 177
EXLEVEL	B
AUTHOR	Sonalysts, Inc.
REFKEY	
KA1	BW/E02 A2.2
KA1RO	3.2
KA1SRO	3.8
KA2	
KA2RO	
KA2SRO	

QVALUE 1.0

QUESTION 31 B11

Unit 3 Initial Plant Conditions:

- Reactor startup and power increase in progress.
- Reactor power is 50%.

Unit 3 Current Plant Conditions:

- Reactor has tripped.
- Startup FDW control valves (3FDW-35 and 44) failed closed.
- Main FDW block valves (3FDW-31and 40) failed closed.
- 3FDW-315 (SG EFDW Control Valve to 3A SG) failed closed.
- Main FDW pump 3A operating (i.e., did not trip).

Which ONE of the following provides the levels at which SG 3A and 3B will stabilize? **(Assume no operator actions.)**

	SG A XSUR Level	SG B XSUR Level
A.	30 inches	25 inches
B.	25 inches	20 inches
C.	14 inches	20 inches
D.	14 inches	30 inches

QUESTION # 25

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**1**	**1**
Group #	**2**	**2**
K/A #	**BW/E04 K1.1**	
Importance Rating	**3.4**	**3.8**

Technical Reference(s): **EAP-E33**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-33 #9**

Question Source:
Bank # **EAP-134**
Modified Bank # _______
New _______

Question History:
Previous NRC Exam _______
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 __X__

Comments:

1 POINT

QUESTION # 25

Unit 1 plant conditions:

- Recovery from HPI Cooling is in progress
- EFDW has been restored
- ~~1A1 RCP is operating~~ 1A & 1B switchgear deenergized

Which ONE of the following describes the INITIAL operator actions when feedwater flow is established?

Initiate EFDW flow to the unaffected S/G(s) to...

A. establish a cooldown rate of approximately 45°F per 1/2 hour.

B. establish both OTSG levels at setpoint.

C. match decay heat and RCP heat.

D. match decay heat only.

1 POINT

QUESTION # 25

BW/E04K1.1
(1)

A. Incorrect - Cooldown should not be initiated until the SGs have adequate level established even if the pressurizer is saturated. After SG levels are established then this would be the proper cooldown.

B. Incorrect: Level should not be established until cooldown to 555° F or excessive RCS cooling will occur.

C. Correct: Directed to curve to match NSSS heat which includes decay and pump heat.

D. Incorrect: Not <u>only</u> decay heat, 1 RCP is operating.

8. Explain why a transfer to CP-602 to align LPI will be different if the RCS is saturated instead of subcooled.(R8)

9. Recognize the potential problems associated with recovery of SG heat transfer while in HPI forced cooling with a subcooled RCS. (R9)

10. Given a set of conditions, calculate pressurizer relief valve discharge temperature. (R13)

11. Explain the special precaution required when repressurizing SGs after HPI forced cooling when T_C > 555°F. (R10)

12. Discuss the criteria for determining which RCP to start or bump when directed by the EOP. (R14)

13. Recognize that if RCPs are started with any subcooled margin = 0°F, they should be monitored closely and manually tripped, if potential pump damage is indicated.(R11)

E. If no Main FDWPs or EFDWPs are available, then align EFDW from another unit per Loss of Main FDW AP.

F. If a SG has been isolated due to SGTL or MSLB and cannot maintain adequate heat transfer from the unaffected SG, then feed the affected SG.

If both SGs are affected then feed the least affected SG.

G. Establish SG(s) levels at the LOSCM setpoint.

H. Maintain SG pressure < RCS pressure.

I. Close the RCS high point vents.

J. Ensure PORV selected to correct position (based on T_C).

NOTE:	Thermal Shock conditions may develop if HPI is NOT throttled and RCS pressure controlled.

K. Regain Core SCM

1. Once core SCM $\geq$ 5°F, HPI can be throttled as needed to maintain proper RCS P/T relationships.
2. The PORV is isolated if it leaks past its seat.
3. Throttle EFDW as needed to prevent RCS overcooling.
4. Bypass to step 29.6 to regain proper RCS pressure control.

CAUTION:	Feeding SG(s) with HPI forced cooling may cause excessive heat transfer.

2.25 If core SCM > 0°F AND the ability to feed the SG(s) exists, then recover from HPI forced cooling. (29)

This is a very complicated process with many potential problems or pitfalls, some of which can occur rapidly.

Appropriate to this step would be an extensive pre-job briefing session to assure everyone knows the goal and their part in reaching it.

A. Prepare for letdown.

B. Open HP-5.

C. Disable both trains of MSLB Isolation circuit.

D. Determine plant conditions for HPI cooling recovery.

E. Close RCS high point vent valves.

F. Perform HPI forced cooling recovery.

Enclosure 7.6 of the EOP gives an initial reference value for the required FDW flow to match core decay heat.

G. Adjust HP-7 as necessary to help control RCS pressure.

H. Energize all Pzr heaters.

I. If leakage is indicated past PORV, then close RC-4.

Exam Question Report

27-Jan-99

Question ID:	EAP134	Revision No:	0	Revision Date	10/29/1999
Question Description:	EAP134				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-E22 - Loss of Heat Transfer
Last Used Date:	05/09/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	LRO = 4.2; SRO = 3.8	Max. Point Value:	0.25
		Passing Point Value:	0.25

27-Jan-99

Question

Unit 1 is in HPI Forced Cooling, both MDEFDWP's have been returned to service and EFDW flow has been verified. The following plant conditions exist:

- Core SCM = 11°F - the minimum SCM during the transient
- RCS pressure = 1485 psig
- Pressurizer temperature = 596°F
- Both Loop SCM's = 8°F

Which ONE of the following describes the intent of the INITIAL actions the NCO would take in regards to feedwater flow during the recovery from HPI cooling for these conditions? (.25)

Initiate EFDW flow to the unaffected S/G(s) to...

A) establish a cooldown rate of ≤ 45°F per 1/2 hour.

B) establish a S/G level of 240" XSUR.

C) match decay heat and RCP heat.

D) match decay heat only.

Answer

C

A. INCORRECT: Cooldown should not be initiated until pressurizer is saturated. Will not start at initial EFW flow.

B. INCORRECT: Level should not be established until cooldown to 555°F or overcooling will occur.

C. CORRECT: Directed to curve to match NSSS heat.

D. INCORRECT: Not only decay heat, RCP is on.

Lessons

ID	Description
EAP-E22	Loss of Heat Transfer EAP-E22

QUESTION # 26

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**2**	**2**
	K/A #	**000060**	**G2.1.28**
	Importance Rating	**3.2**	**3.3**

Technical Reference(s): **WE-GWD**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **WE-GWD #3.2**

Question Source:
Bank # ______
Modified Bank # **WE-081**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __**X**__
55.43 _____

Comments:

1 POINT

QUESTION # 26

Plant conditions:

- DW makeup to 2B BHUT in progress
- The GWD Vent Header is cross-connected with Unit 3 controlling the header
- 3GWD-1 (Vent Header Pressure Control) in auto
- 3A GWD Tank in service

Which ONE of the following explains how the GWD system will respond over the next thirty minutes?

3GWD-1 will ______, _______ the 3A GWD Tank.

A. open / depressurizing

B. close / depressurizing

C. open / pressurizing

D. close / pressurizing

1 POINT

QUESTION # 26

000060 G2.1.28

(2)

Question setup:
As pressure in the vent header increases above setpoint due to 2B BHUT fill, the controller for vent header pressure will cause GWD-1 to close. Closing GWD-1 will cause the in-service GWD tank pressure to increase.

A. Incorrect, - This would be correct if 2B BHUT level was decreasing. See above explanation.

B. Incorrect, - GWD-1 will close, but GWD tank pressure will increase as designed and described in the above explanation

C. Incorrect, - GWD-1 will not open, however GWD tank pressure will increase as described in the above explanation

D. Correct, - As 2B BHUT volume increases the tank is vented to the vent header. As pressure in the vent header increases above setpoint, the vent header pressure control system will cause 3GWD-1 valve position to be decreased/closing 3GWD-1 (compress mode). Closing 3GWD-1 will cause gas to be compressed into the in-service 3A GWD tank increasing the tank pressure.

OBJECTIVES

TERMINAL OBJECTIVE

1. The student will be able to describe the proper operation of the GWD components and GWD system during normal operation. The student will have a working knowledge of the major procedure steps in the transfer, sampling, and release of radioactive gas in the GWD system. (T1)

2. Describe the proper operation of the GWD system during normal operation and state the bases of selected steps from the GWD procedure. The student will also be able to perform the necessary calculations in order to determine if a particular release can be made without violating Tech. Spec. limits. (T2)

ENABLING OBJECTIVES

1. State the purpose of the GWD System. (R1)

2. List the potential sources of waste gas to the GWD System. (R2)

3. Describe for each of the following components: (R3)

   3.1 the purpose of the component

   3.2 how the component functions during normal system operations.

   A. vent header

   B. waste gas compressor

   C. waste gas separator

   D. unloader valve

   E. seal water cooler

   F. waste gas decay tank

   G. vent header pressure control valve (GWD-1)

   H. waste gas discharge filters

4. Describe the correct flowpath of gas through the GWD system for the following evolution's: (R4)

   4.1 Normal operation of the GWD system.

   4.2 Transfer of gas between two GWD tanks.

   4.3 Sampling of a waste gas decay tank.

   4.4 Release of an isolated GWD tank

5. Describe "cross-connecting" the vent header, and explain why this action would be done. (R5)

1. Introduction
   1.1 Purpose (Refer to OC-WE-GWD-1)
      A. Collect, holdup, and process potentially radioactive gaseous waste.
   1.2 General Description (Refer to OC-WE-GWD-2)
      A. Gases evolve from:
         1. venting components in Rx Building and Aux. Building
         2. release of soluble gases in stored liquids
         3. displacement of gases by liquids as tanks are filled
      B. All of the above sources:
         1. are subject to particulate carryover
         2. may contain activated noble gases
      C. Gases are:
         1. vented to vent header
         2. removed from vent header by GWD compressor
         3. discharged by compressor to GWD tanks
      D. After holdup for decay, tanks are:
         1. sampled
         2. released through GWD filters to unit vent stack
      E. If high radiation levels are detected during release, the release is automatically terminated by closing tank discharge valve.
   1.3 Two complete GWD systems at Oconee
      A. Oconee Units 1 & 2 share a system
         1. Oconee Unit 3 has a separate system
      B. Systems can be cross-connected and one system shutdown as needed.
   1.4 Review objectives

3. Auxiliary Building

a) Quantity	4 (two per system)
b) Volume each, cu ft.	1,100
c) Material	Carbon Steel
d) Design pressure, psig	100

4. Interim Rad Waste Building

a) Quantity:	3 (two for Units 1&2 System, one for Unit 3 System)
b) Volume each, cu ft.	1052
c) Design Pressure, psig	100
d) Design Temperature, °F	208
e) Material	Carbon Steel

F. Vent Header Pressure Control Valve (GWD-1) (Refer to OC-WE-GWD-2)

1. Allows bleed-back (recirc) of gas from in-service GWD tank back to vent header to maintain preset vent header pressure.
   - Necessary, since GWD compressor has excess capacity over what is required to maintain vent header negative during normal operation.
2. Controlled by Bailey Station in Control Room.
3. Usually in auto, vent header pressure slightly negative.
   a) If setpoint too low - air will be drawn into vent header., wasting gas tank space, and minimizing holdup time.
   b) If setpoint too high - gas will be allowed to escape into Aux. Bldg., NOT ALARA.
   c) Correct setting is best learned by experience.

G. Waste Gas Filters (Refer to OC-WE-GWD-2, 7)

1. Reduce particulate matter and radio-iodine released to vent stack.
2. All GWD tank releases pass through these filters.
   - Rating - 200 scfm
3. Comprised of:
   a) Pre-filter - removes particulates
   b) Absolute filter - removes particulates
   c) Carbon filter - removes iodine
4. One set per system

1. The quantity of radioactivity contained in each waste gas holdup tank shall be limited to ≤ 3.8E5 curies noble gases (considered as Xe-133).
   a) Local area or process monitors would alarm long before this limit could be reached
   b) Tech Spec requires the GWD tank(s) activity be determined daily, when gas is being added to the tank(s), to ensure that the limit of 3.8E5 curies is not exceeded.
      1) RP is tasked with this requirement. By monitoring RCS activity (samples and instrument readings) daily, RP is ensured that if the RCS Xe-133 equivalent activity is below 50 uci/ml on all three units, that the GWD tank activity will not reach its limit. When RCS activity exceeds 50 uci/ml on any unit, RP begins to perform daily surveys of the tanks for activity.
2. Limits WB exposure of individual at site boundary to less than .5 Rem in the event of an uncontrolled tank release.

C. Samples for the RB Purge are valid for 24 hours. Samples taken for Gas Tanks Release are valid for 72 Hours.
   1. This will allow Operations to HOLD a Gas Tank Release for favorable meteorological conditions to exist without having to resample. Sample results will be conservative as no additional radioactive gas would have been added to the tank, and some of the radioactive material will have decayed.

D. Maximum GWD Tank pressure shall not exceed 85 psig.

2.5 System Operation

Utilize OP/1&2 or 3/A/1104/18 "Gaseous Waste Disposal System" to perform all GWD System operations described in this section, except Depressurization of Reactor Building.

A. Normal Operation (Refer to OC-WE-GWD-2)
   1. Vent Header Split (Isolation Valves Closed)
   2. For each system:
      a) one compressor on
      b) one tank in service
      c) GWD-1 in auto and set slightly negative
      d) Compressor removes gases from vent header, discharging to tank.
      e) GWD-1 allows "recirc" to vent header of enough gas to maintain preset vent header pressure.

1) As pressure in the vent header increases above setpoint, the controller for vent header pressure will cause GWD-1 valve position to be decreased. With less gas being recirced to the vent header, then the vent header pressure should decrease and return header pressure to setpoint. The closing down of GWD -1 will cause gas to be retained in the inservice GWD tank and the GWD tank pressure will increase.

2) As pressure in the Vent header decreases below setpoint, the controller for vent header pressure will cause GWD-1 valve position to open up. With more gas being recirced to the vent header, then the vent header pressure should increase and return header pressure to setpoint. The opening up of GWD -1 will cause gas to be released from the inservice gas tank, returning the gas to the vent header, and the gas tank pressure will decrease.

B. Tank Isolation (Refer to OC-WE-GWD-2)

1. Tank should be isolated prior to high pressure alarm (70 psig).
2. Tank should be ≈ 5 psig prior to placing in service, add N if required to increase tank pressure
3. Procedure:

a) Close inlet valve for tank to be isolated <u>and</u> open inlet to tank to be placed in service.

b) Close "recirc" valve for isolated tank <u>and</u> open "recirc" valve for tank placed in service.

c) Verify that isolated tank pressure is not decreasing and vent header is controlling normally.

d) Sample isolated tank for hydrogen within 24 hours and purge with nitrogen (if > 3% hydrogen), as follows:

1) If hydrogen is > 3%, lower tank to 50 psig by transferring some for the gas to another tank.

2) Add 20 psig nitrogen to tank

3) Resample for hydrogen

4) Continue until hydrogen < 3%

C. Transferring Gas Between Tanks (Refer to OC-WE-GWD-2)

1. Normally done <u>from</u> in-service tank <u>to</u> another tank, this adds operational flexibility to determine which tank is used for in-service work and which tank(s) are used for isolation and decay.
2. Procedure:

a) Open inlet for tank receiving gas

Exam Question Report

27-Jan-99

Question ID:	WE081	Revision No:	0	Revision Date	10/29/1999
Question Description:	WE081				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	WE-GWD - Gaseous Waste Systems
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 3; SRO = 3 Reference: OP/12&3/1104/18	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

With 1GWD-1 (vent header pressure control) in auto, and Operations transferring water from the "1B" BHUT to Radwaste, which ONE of the following explains how the GWD system will respond? (.25)

1GWD-1 will...

A) Open, releasing gas into the vent header.

B) Close, releasing gas into the vent header.

C) Open, pressurizing the in-service GWD Tank.

D) Close, pressurizing the in-service GWD Tank.

Answer

A

Lessons

ID	Description
WE-GWD	GASEOUS WASTE DISPOSAL SYSTEM (WE-GWD)

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

QUESTION # 27

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**1**	**1**
	Group #	**3**	**3**
	K/A #	**000056 A2.01**	
	Importance Rating	**3.3**	**3.4**

Technical Reference(s): **PNS-PZR**

Proposed references to be provided to applicants during examination: **(Steam Tables)**

Learning Objective: **PNS-PZR #27**

Question Source: Bank # **PNS-700**
Modified Bank # ______
New ______

Question History: Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 ______
55.43 __X__

Comments:

1 POINT

QUESTION # 27

Unit 2 plant conditions:

INITIAL CONDITIONS:

- A loss of power (BLACKOUT) event has occurred
- 2RC-66 (PZR PORV) cycling

CURRENT CONDITIONS:

- Power has been restored
- RCS WR pressure = 885 psig
- PZR saturation pressure = 885 psig
- PZR level = 120 inches
- Quench Tank Pressure = 45 psig
- PZR RELIEF VALVE MONITOR (RC-66) indicates 3 LEDs lit

Which ONE of the following is the expected PORV tailpipe temperature °(F)?

(ASSUME 100% steam quality)

A. 532

B. 360

C. 325

D. 274

why did operator not close Block valve. Rewrite from original question with new @ statement.

1 POINT

QUESTION # 27

000056A2.01 (2)

A. Incorrect - 532°F is saturation for 885 psig.

B. Incorrect – constant enthalpy process from 900 psia to saturation line.

C. Correct – constant enthalpy throttle process 885 psig + 15 psi = 900 psia to 45 psig + 15 psi = 60 psia

D. Incorrect – Saturation temperature for 45 psia

12. Explain the operation of the Pressurizer Water Space Saturation Recovery Circuit. (R29)

ENABLING OBJECTIVES (continued)

13. Discuss the operation of the pressurizer heaters including: (R7)
    13.1 Three purposes of pressurizer heaters.
    13.2 Purpose and level of interlock associated with pressurizer heaters.
    13.3 On/off setpoints for pressurizer heater banks 2, 3and 4.
14. Describe the physical operation of the PORV including what causes the Pilot Valve to operate and how this causes the PORV to open or close. (R8)
15. Explain the purpose of the two opening setpoints associated with the PORV. (R9)
16. Explain how to manually operate the PORV. (R37)
17. Given a set of conditions, determine operability of the PORV following a loss of power. (R30)
18. Discuss the reason for the pressurizer safeties and their setpoint. (R12)
19. Given a set of plant conditions, determine the response of Pressurizer level. (R14)(R15)
20. Explain the operation of SASS as it relates to pressurizer level control. (R31)
21. Given a set of conditions, determine how pressurizer level control/indication is affected by a loss of SASS and/or ICCM. (R35)
22. Discuss the use of Pressurizer Saturation Pressure Indication by the operator. (R16)
23. Discuss the forming of a pressurizer steam bubble including any precautions to be taken during the evolution. (R17)
24. Given a completed copy of PT/0/A/201/04 PORV Operability Test apply compare data taken to acceptance criteria to determine PORV operability. (R10)(R11)
25. Differentiate between a pressurizer steam space leak and a water space leak. (R32)
26. Given a set of plant conditions, determine the position of the PORV. (R13)
27. Given a set of conditions, calculate the expected PORV discharge temperature. (R34)
28. Given a copy of a Limit and Precaution from OP/A/1103/05, Pressurizer Operation, be able to state the reason for that limit and precaution. (R18)

c) Pressurizer level may remain constant or decrease depending upon the magnitude of the leak.

d) Again, RC pressure response will be dependent upon leak size. For small leaks, RCS pressure will remain constant. As pressurizer level drops, the steam bubble will expand to fill the space causing RC pressure to decrease.

e) Pressurizer heaters will energize as RCS pressure continues to drop.

f) If leak is of such a magnitude that RCS makeup flow cannot maintain level, RCS pressure will continue to drop until the reactor trips and ES actuates.

3. Identifying an Open or Leaking Pressurizer Relief Valve.

a) Pressurizer Relief Valve Flow Monitor will alert the operator to an open relief valve via alarm/indication.

b) RC-66 valve indication.

c) Pressurizer boron concentration increasing relative to RCS.

d) Pressurizer relief valve tailpipe temperature

1) Pressurizer relief valve tailpipe temperature is a function of pressurizer temperature (or pressure since saturated) and Quench Tank pressure.

2) Since flow through a relief valve is a constant enthalpy throttling process, the temperature downstream of a relief valve can be found by:

(a) Finding the point where the temperature (or *absolute* pressure) of the pressurizer intersects the saturation curve on the Mollier diagram.

INSTRUCTORS NOTE: Work a couple of examples with the students using the Mollier Diagram.

See handout #1 and #2

(b) Cross the horizontal constant enthalpy line from this intersection point to the point where the constant enthalpy line intersects the *absolute* pressure of the Quench Tank.

(c) Follow the *absolute* Quench Tank pressure line up to the intersection of the saturation curve.

(d) The constant temperature line that intersects this same point on the saturation curve is the temperature of the steam downstream of the relief valve.

3) Quench tank level, temperature and pressure

(a) Steam flow past a leaking relief valve will enter the quench tank below water level.

(b) The contents of the quench tank will condense the steam.

(c) The condensed steam will increase quench tank level, temperature and pressure.

(d) The same process occurs for an open relief valve except that the effluent rapidly heats the contents of the quench tank to saturated conditions, a steam bubble forms in the quench tank and the quench tank rupture disk ruptures.

4. Emergency Operating Procedure Evolutions

a) During a shutdown due to a steam generator tube leak/rupture which requires entry into the emergency operating procedure, the pressurizer heaters are manually secured and pressurizer spray (or the PORV if spray is unavailable) is manually initiated. This allows the operator to purposely lower RCS pressure to approximately 3-5° F subcooling to minimize the subcooling margin. This minimizes the differential pressure between the RCS and the steam generator, thus decreasing the tube leak rate.

b) Following a loss of feedwater event where no sources of feedwater are available and RCS pressure is approaching 2300 psig, an HPI pump is operated. Flow is established in each injection header through HP-26 and HP-27 and the PORV manually opened to provide a flowpath for cooling water from the BWST, through the core, to the basement of the reactor building.

2.7 System Interlocks/Automatic Actions

A. In automatic, HP-120 maintains pressurizer level at setpoint (normally 220 inches) under normal conditions.

B. In automatic, the pressurizer heater banks cycle as necessary to maintain RCS pressure as follows:

1. Pressurizer heater bank no. 1 will maintain RCS pressure at setpoint (normally 2155 psig).
2. Pressurizer heater bank no. 2 energizes at 2130 psig decreasing and de-energizes at 2140 psig increasing.
3. Pressurizer heater bank no. 3 energizes at 2115 psig decreasing and de-energizes at 2135 psig increasing.
4. Pressurizer heater bank no. 4 energizes at 2100 psig decreasing and de-energizes at 2120 psig increasing.

C. A 80-inch low pressurizer level interlock prevents the heaters from being energized while they are uncovered.

Exam Question Report

27-Jan-99

Question ID:	PNS700	Revision No:	0	Revision Date	10/29/1999
Question Description:	PNS700				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	PNS-CPS - Coolant Pump Seals
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	NLO = R2; LRO = AP/1/A/1700/16 Reference: K/A: OO3A2.02 (3.5/3.9)	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Unit 2 startup in progress:

- RCS WR pressure = 885 psig and slowly increasing
- PZR saturation pressure = 885 psig and slowly decreasing
- PZR level = 120 inches and slowly decreasing
- Quench Tank Pressure = 45 psig and slowly increasing
- PZR RELIEF VALVE MONITOR indicates 3 LEDs ON for 2RC-66

Which ONE of the following is the expected PORV tailpipe temperature (□F)? (.25)

(ASSUME 100% steam quality)

A. 532

B. 360

C. 325

D. 300

Answer

C

A. Incorrect - 532 is saturation for 532.
B. Incorrect - isoenthlic process from 900 psia to saturation line.
C. Correct - isoenthlic throttle process 885 psig + 15 psi = 900 psia to 45 psig + 15 psi = 60 psia
D. Incorrect - 900 to 45 psi

Lessons

ID	Description
PNS-CPS	COOLANT PUMP SEALS (CPS)

Enabling Objectives

ID	Description
PNSCPSR02	2. For the Unit 1 Westinghouse Reactor Coolant Pump Seals:

Referenced Documents

ID	Description	Review Date	Ref Flag
AP/1/A/1700/016	Abnormal Reactor Coolant Pump Operation		

KA'S

INITIAL SUBMITTAL

OCONEE EXAM 2000-301
50-269, 270, AND 287/2000-301

JULY 10 - 14, 18, 19, AND 20, 2000

INITIAL SUBMITTAL

COMMON - VOLUME 2
WRITTEN EXAM

COMMON

Volume 2

NRC Copy

QUESTION # 36

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**0220000**	**A1.04**
Importance Rating	**3.2**	**3.3**

Technical Reference(s): **PNS-RBC**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-RBC #12 & #13**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 36

Unit 1 plant conditions:

- Mode 3
- 1A and 1C RBCU's in HIGH
- "A" RB Spray Train is inoperable
- Statalarm 1SA-9/C-9, RBCU Cooler Rupture, alarm actuated
- LPSW inlet flow to "1A" RBCU is 560 gpm
- RBNS level = 12" and steady

Which ONE of the following is correct?

These symptoms indicate the "1A" RBCU ________ a cooler rupture and ________.

to what

dilution of RBES follows a LOCA

A. has / could result in inadequate ~~Boron~~ concentration if subsequently suction were aligned ~~to the RBES.~~ IP

B. has / RB design pressure could be exceeded during a subsequent accident.

C. does not have / "1A" RBCU LPSW flow parameters should be checked to validate alarm condition.

D. does not have / "1A" RBCU should be isolated to determine LPSW leak location.

what R[illegible] at the LPSW leak

1 POINT

QUESTION # 36

022000A1.04 common rsi/gcw 05/11/00

A. Incorrect, a cooler rupture is not indicated because the RBNS is not increasing. If a rupture occurred the second part would be true.

B. Incorrect, a cooler rupture is not indicated because the RBNS is not increasing. If a rupture occurred the second part would be true. RB design pressure could be exceeded as “A” RBS Train is OOS.

C. Correct, a cooler rupture is not indicated because the RBNS is not increasing. However the ARG directs the operator to verify the alarm conditions by looking at cooler inlet and delta flow.

D. Incorrect, a cooler rupture is not indicated because the RBNS is not increasing. The cooler should not be isolated until a cooler leak is verified. If a cooler leak existed the second part would be true.

12. Describe two conditions that will activate a RBCU "Cooler Rupture" alarm. (R13)
13. Given a set of conditions, determine the proper operation / alignment of the RBC System and the basis for that specific operation / alignment. (R5, R7, R9)
14. Given a set of plant conditions, analyze RBC System operation and determine system status and any required actions / corrective actions. (R14, R15)
15. Given a copy of ITS / SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS / SLC LCO's. (R11)
16. Apply all ITS / SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R18)
17. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITS / SLC's. (R19)

4. Manual speed changes should be minimized where possible.
5. During non-emergency operation, the maximum RBCU motor bearing temperature is 220°F. See computer points A0037 through A0042 (RBV CLR FAN IB/OB BRG TEMP).
6. The 1B RBCU may be operated while LPSW is diverted to the Rx Bldg Aux Fan Coolers.
7. Do NOT operate RBCUs in mixed speed combinations. Excessive back pressure is placed on any fan(s) in LOW.
   - Proper damper operation is not required for RBCU operability per ITS. Improper operation could result in high vibration or temperature for the RBCU.
8. When RBC System operability is required (ITS 3.6.5, MODES 1, 2, 3 and 4), LPSW flow to all RBCUs must be ≥ 550 gpm.
   - If LPSW to an RBCU is < 550 gpm, LCO 3.0.3 applies.

   Condensation induced waterhammers will occur in the LPSW piping supplying RBCU's and the RB Aux Fans during a postulated LOCA or MSLB event. Analysis has shown that a minimum of 550 gpm must be maintained to each RBCU to prevent more severe waterhammers (not bounded by current analysis) from occurring. Assuming the minimum flow is maintained, LPSW piping will maintain its integrity during normal and accident conditions.

E. Starting a RBCU

1. Place switch in "HIGH"
   a) Discharge damper opens - this must occur prior to fan actually starting.
   b) Fan starts and goes to low speed.

Refer To OP-PNS-RBC-2

   - Operator monitors RBCU amps on AB-3 above control switches to ensure proper RBCU response.

   c) Low speed windings are deenergized.

5. RBCU Rupture Alarms (1,2,3SA-9)
   a) Two conditions will activate a cooler rupture alarm.
      1) 327 gpm difference in inlet vs. outlet flow
         (a) Uses flow transmitters that supply control room indications.
         (b) A difference in flow indicates water being lost and is representative of a cooler rupture.
      2) 566 gpm LPSW flow, decreasing, indicated at the cooler inlet and the cooler outlet valve is full open
         (a) This is indicative of cooler inlet valve not being open for some reason.
         (b) Since inlet valve is not ES valve, it will not auto open on an ES signal if it is shut.
         (c) If the inlet valve is not open, an ES 5 & 6 actuation should cause the cooler rupture alarm to be activated.
         (d) If received, the inlet valve should be verified to be fully open.
   b) If cooler rupture alarm is received, alarm response manual should be consulted.
      1) Verify valid alarm - check inlet flow and delta flow.
      2) Verify LPSW outlet valve open.
      3) Verify adequate LPSW flow available; check LPSW Pump operation.
      4) Monitor RBNS Level for any unexplained increase.
         - If RBNS Level is increasing, notify Chemistry to sample the RBNS for boron concentration to determine if a cooler rupture has occurred.
      5) IF RBCU Cooler rupture is indicated, then:
         - Isolate LPSW to affected RBCU.
         - Refer to Improved Tech Specs.

QUESTION # 37

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**056000**	**G2.1.28**
Importance Rating	**3.2**	**3.3**

Technical Reference(s): **CF-C**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-C OBJ. #7**

Question Source:
Bank # ______
Modified Bank # **CF-058**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 37

Unit 3 plant conditions:

- MODE 5, Unit startup in progress
- Condensate system startup in progress

Which ONE of the following will the 3C-10 (Hotwell Pump Discharge Controller) interlock prevent?

A. Generator Hydrogen Cooler gaskets rupture.

B. Feedwater pump damage due to windmilling.

C. Hotwell Pump damage during a loss of IA pressure.

D. Powdex resin entering the Condensate system and OTSGs.

1 POINT

QUESTION # 37

056000G1.28 (3.2/3.3) Both PRA 4-6-00

A. Correct - C-10 (HWP Discharge Controller), located in the Control Room on VB1, must be <10% open to be able start the first HWP. Requiring C-10 to be (almost) shut prior to starting the first HWP prevents a water hammer that could rupture downstream components, especially the Generator Hydrogen Cooler gaskets. This interlock was installed after an incident occurred on Unit 1 where the Condensate system was in a started up condition with C-10 fully open. This caused a severe water hammer in the lines due to entrained air. The water hammer ruptured the gaskets in the Generator Hydrogen Coolers.

B. Incorrect - The FDWPT Windmill Protection Interlock is provided to prevent damage to a feedwater pump because of insufficient oil pressure while it is rotating (windmilling) due to condensate flow through its impeller.

C. Incorrect - C-10 fails open on a loss of Instrument Air pressure which could be a problem during Condensate system startup for down stream components but should not damage the HWPs.

D. Incorrect - Originally, it was expected that only 60% of the condensate flow would need to be routed through the Polishing Demineralizers (Powdex), but experience has shown that 100% flow (all five cells) is required to maintain the secondary system chemistry within specifications. Resin discharge from the Powdex has occurred ~~in the~~ recently at Oconee. Resin was discovered in OTSG samples.

TRAINING OBJECTIVES continued

7. Describe the starting interlock between discharge controller C-10 and the HWPs, including the purpose for this interlock. (R7)

8. Explain the automatic features associated with the HWPs. (R8)(R9)

9. Explain the effects on plant operation if the Polishing Demineralizers (Powdex) are not available during power operations. (R10)

10. State who performs routine Powdex operations and the coordination that is required during operations. (R11)

11. Concerning the Powdex Resin Trap: (R12)

    11.1 Explain its purpose.

    11.2 Identify the location of the Resin Trap in the condensate system flow path in relation to the other major components in the condensate system.

    11.3 Describe how the Resin Trap can be bypassed and isolated for cleaning or back washing.

12. Explain the manual and automatic operation of the Powdex control valves, C-14/15. (R13)

13. Describe how to identify that the Powdex has automatically bypassed and explain how to re-establish flow through the Powdex after it has automatically bypassed. (R14)

14. Explain the reason for having condensate coolers. (R15)

15. Explain how condensate temperature is regulated in the Condensate Coolers. (R16)

16. Explain why: (R17)

    16.1 One condensate cooler on each unit is normally isolated.

    16.2 One of the temperature control valves on the Unit 2 and Unit 3 condensate coolers has been failed open.

17. Explain the operation of C-61, Condensate Flow Around Coolers Control, including: (R18)

    17.1 Purpose of controller.

    17.2 Parameter control is based on.

    17.3 Relationship with the temperature control for the Hydrogen Coolers.

5. Hotwell Pump Controls and Indications

   a) The Hotwell Pumps are provided with four position OFF-AUTO-ON-START control switches on AB1 in the Control Room. These switches are spring return from START to ON.

   b) The pumps are also provided with momentary START-STOP local control switches. The local control switches were provided for pre-operational startup testing and are not used to operate the pumps.

   c) The 4160 volt switchgear pump motor circuit breaker is provided with a control switch for test purposes. This switch is functional only when the breaker is racked into the test position.

   d) Each Hotwell pump is provided with a square amber light located above its control switch in the control room that will light when the respective pump's discharge pressure is < 125 psig.

   e) Each hotwell Pump is provided with an ammeter located above its control switch in the control room. Normal operating amperage is approximately 75 to 80 amps, depending on condensate flow.

   f) A hotwell discharge header pressure gauge is provided on AB1 in the main control room.

6. Hotwell Pump Start Interlocks:

   a) C-10 (HWP Discharge Controller), located in the Control Room on VB1, must be <10% open to be able start the <u>first</u> HWP:

      1) Requiring C-10 to be (almost) shut prior to starting the first HWP prevents a water hammer that could rupture downstream components, especially the Generator Hydrogen Cooler gaskets.

      2) This interlock was installed after an incident occurred on Unit 1 where the Condensate System was started up with C-10 fully open. This caused a severe water hammer in the lines due to entrained air. The water hammer ruptured the gaskets in the Generator Hydrogen Coolers.

   b) The FDWPT Windmill Protection Interlock must be clear.

      1) The FDWPT Windmill Protection Interlock is provided to prevent damage to a feedwater pump because of insufficient oil pressure while it is rotating (windmilling) due to condensate flow through its impeller.

      2) The interlock is clear if:

         (a) Feedwater pump discharge pressure of > 770 psig is sensed on both FDWP discharge headers or,

         (b) Either MFDWP's bearing oil pressure is > 4 psig or

         (c) Its respective suction valve is closed.

c) After the first HWP is running, any HWP that is in AUTO will start on a signal of low CBP suction pressure of 55 psig.

7. Automatic Hotwell Pump Trips:

a) The FDWPT Windmill Protection will trip all operating hotwell pumps if there is a simultaneous:

1) Low FDWPT bearing oil pressure of < 4 psig and

2) The respective FDWP suction valve is open and

3) The discharge pressure is < 770 psig on both MFDWP's.

b) If a Hotwell Pump suction or discharge valve is < 50% open the respective pump will trip (operates off of the valve limit switch).

c) Load shed signal.

C. C-10 (HWP Discharge Control)

1. The Hotwell Pump Discharge Control Valve, C-10, is a 24" diameter pneumatically operated butterfly valve. Its purpose is to control hotwell pump flow rate during the fill of the condensate system.

2. C-10 is controlled from VB1 in the control room via a Moore digital controller.

3. C-10 is operated in the "MANUAL" mode only. Except for initial pump start and system fill, the valve is fully open (normal operation).

4. As discussed previously, C-10 can be no more than 10% open in order to start the first HWP:

a) The Operator closes C-10 and the first HWP is started; C-10 is then slowly throttled open to fill the system until the valve is fully open.

b) C-10 then remains fully open throughout subsequent operations; any necessary adjustment of condensate flow is performed via valves further downstream in the system.

5. C-10 fails open on a loss of Instrument Air pressure.

D. Polishing Demineralizers (Powdex)

1. Discharge from the Hotwell Pumps flows through five parallel demineralizer cells.

2. The purpose of the demineralizers is to remove suspended and dissolved metallic cations and anions such as halides, silicates and sulfates, and to act as a high purity filter.

3. Each of the vertically mounted cells contains an array of filter tubes with fine wire openings. These tubes are coated with powdered resins that bridge the wire openings and create a microscopic sieve that allows for the removal of suspended and dissolved solids.

4. The coating is held on the tube by the flow differential pressure. When a demineralizer is off-line, a holding pump provides the necessary differential pressure to retain the coating.

5. Originally, it was expected that only 60% of the condensate flow would need to be routed through the demineralizers, but experience has shown that 100% flow (all five cells) is required to maintain the secondary system chemistry within specifications.

6. The cells are controlled from a local control panel. Operation of the system is under the control of the Chemistry Department and is essentially automatic once its operating mode (filtering, backwash, precoat, or hold) has been selected:

   a) Filtering Mode

      Condensate enters the bottom of the cell, flows up from under a distribution plate, and into the element tubes through the mesh screen. Condensate then flows down the tubes into a chamber formed by the tube sheet and the bottom of the cell vessel. From there, the treated condensate returns to the system through an outlet connection.

   b) Backwash Mode

      When a demineralizer cell exhibits high differential pressure, it must be backwashed and precoated. Backwash water is supplied from the upper surge tank via the demineralizer backwash pump. Backwash water enters the discharge line of the cell, flows through the cell and the elements in a reverse direction, and washes the spent resins and impurities to the demineralizer backwash sump.

   c) Precoat Mode

      After the elements in the cell have been thoroughly cleaned by backwashing, new precoating is applied. The precoat pump transfers resin slurry from the slurry tank to the cell via its inlet line. The resin particles adhere to the external surface of the elements and the water passes through to be returned to the slurry tank.

   d) Holding Mode

      Each cell has its own holding pump. The pump is used to maintain the resin coating in place when the cell is out of service. When the pump is operating, water is recirculated from the outlet of the cell back to the inlet.

      The hold pump automatically starts and stops based on flow rate through the cell.

7. A Resin Trap provided downstream of the polish demineralizer cells traps any resin beads that manage to migrate through the mesh screen on the elements and prevent them from entering the Condensate System.

Exam Question Report

27-Jan-99

Question ID:	CF058	Revision No:	0	Revision Date	10/29/1999
Question Description:	CF058				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	CF-C - Condensate System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 7; LRO = 7; SRO = 7 Reference: OP-OC-CF-C PAGE 21 OF 53 PARAGRAPH 6	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Which ONE of the following requirements must be met to start the first Hotwell Pump?(.25)

A) Oil pressure must be > 10 psig.

B) Hotwell level must be greater than 18".

C) C-10, HWP Discharge Control Valve must be less than 10% open.

D) C-14/15, Polish IX Bypass Control Valves must be < 10% open.

Answer

C
A. incorrect, there is not starting interlock based on oil pressure with the Hotwell pumps. This interlock is with the Condensate Booster Pumps.
B. incorrect, there is no starting interlock associated with Hotwell Level.
C. correct, the HWP Discharge Control Valve has to be < 10% open for the first HWP to start.
D. incorrect, there is not interlock associated with the starting of the HWP's and the position of these valves.

Lessons

ID	Description
CF-C	Condensate System (CF-C) Lesson Plan

Enabling Objectives

ID	Description
CFCR7	Enabling Objective created by conversion

QUESTION # 38

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**059 K4.16**	
Importance Rating	**3.1**	**3.2**

Technical Reference(s): **CF-FDW**

Proposed references to be provided to applicants during examination: **N/A**

Learning Objective: **CF-FDW OBJ. #16**

Question Source:
- Bank # ________
- Modified Bank # ________
- New ___X___

Question History:
- Previous NRC Exam ________
- Previous Quiz / Test ________

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 38

Unit 2 plant conditions:

INITIAL CONDITIONS

- Unit startup in progress
- Reactor power = 18%
- 2A Main FDWPT operating
- Unit 2 TD EFDWP in "RUN" operating in recirc for testing
- 2FDW-35 (2A Startup FDW Control) in MANUAL
- 2FDW-31 and 2FDW-40 (2A and B Main FDW Block) in "OPEN" due to FDW flow swings

CURRENT CONDITIONS:

- 2A MS line pressure rapidly decreases to 400 psig

Which ONE of the following correctly describes the MSLB circuit actuation?

Trips the 2A Main FDWP,...

A. trips the TD EFDWP, closes "A" and "B" FDW Main Block valves, and closes "A" and "B" FDW Startup Control valves.

B. closes "A" and "B" FDW Main Block valves, and closes "B" FDW Startup valve.

C. trips the TD EFDWP, closes "B" FDW Startup Control valve.

D. closes "A" and "B" FDW Startup Control valves.

1 POINT

QUESTION # 38

059 K4.16 common (PRA) 2-1-00 (GTH)

A. Incorrect – If the TD EFDWP were running in the AUTOMATIC mode instead of the "RUN" mode this would be a correct answer with the exception of the Main Block Valves that are in OPEN and will be blocked from the MSLB actuation.

B. Incorrect – The Main Block valves will not close if in OPEN but the Main and Startup Controls valves will close if in manual or automatic. This is testing a misconception of this relationship.

C. Incorrect – The TD will only trip only if in the auto mode and the TD switch is not in "RUN" position. The "B" SU CV will close.

D. Correct – "A" and "B" FDW SU CV will close regardless of auto or manual operation as the solenoid will air dump.

13. State the appropriate values for the following FDW System parameters: (100% power) (R33)

    13.1 FDWP suction temperature

    13.2 FDWP suction pressure

    13.3 FDWP discharge pressure

    13.4 Final Feedwater temperature

    13.5 FDW flow (in both gpm and lbm/hr)

14. State the setpoints and automatic actions that occur based on FDWP discharge pressure and FDWPT hydraulic oil pressure. (R29)

15. Given a set of conditions, determine proper operation of the FDWP Seal Injection System. (R30, R31)

16. Describe the purpose of the MSLB Detection/Isolation system. (R34)

17. Given a set of conditions, verify proper operation of the MSLB Isolation Circuit. (R35)

2.11 Main Steam Line Break Detection/Isolation System

A. Purpose: In the event of a Main Steam line (FDW line) rupture, limit RCS overcooling and prevent the RB design pressure (59 psig) from being exceeded. Recent analysis determined that without operator action, RB design pressure would be exceeded for a MSLB inside containment. Based on this analysis, the MSLB Detection/Isolation System was implemented to reduce operator burden in responding to a MSLB.

B. A recent modification was installed to add a safety related detection and isolation circuitry that upon detecting a Main Steam Line Break (MSLB) will complete the following:

1. Trip <u>both</u> Main FDWP's.
2. Inhibit auto-start or initiate an auto-stop (trip) of the TDEFWP.
3. Close the FDW Main and Startup Block valves.
    a) FDW-31, 33, 40, and 42
    b) The valves must be in AUTO for system to operate.
4. Close the FDW Main and Startup Control valves.
    a) FDW-32, 35, 41, and 44
    b) The valves can be in AUTO or MANUAL for system to operate.
        1) The FDW Control valves are pneumatically controlled valves and will close as long as control air is supplied to the valves.
        2) If Instrument Air is lost the FDW Control valves will fail "as is", this is acceptable for two reasons:
            (a) The FDW Block valves, Main and Startup, can CLOSE to isolate the SG's.
            (b) Unit operation cannot be maintained if the control function is lost to the Main FDW system, this requires an operator manual reactor trip.
            (c) If the FDW Block valves, Main and Startup, are closed the RB design pressure will not be exceeded.

C. The MSLB Detection and Isolation system is designed with two (2) independent trains that use a 2 out of 3 logic (one for each steam header) for system actuation. Either train that is actuated will isolate all Feedwater sources (excluding the MDEFDWP's and FDW-315 and 316) to both SG's.

1. Actuation setpoint is 550 psig on two (2) out of three (3) pressure transmitters on either steam line.
2. The pressure signal requires two (2) seconds to seal-in.
   a) The 2 second time delay prevents instantaneous (erroneous) signal from unnecessarily isolating FDW from the SG's.
3. After the logic is satisfied and is sealed-in (2 second time delay has been satisfied), a signal to complete the following is generated:
   a) Trip both Main FDWP's
      1) This lowers the ΔP across Main and Startup Block valves and ensures the valves close.
   b) Inhibit auto-start or initiate an auto-stop (trip) of the TDEFWP
   c) CLOSE the Main and Startup Block valves (FDW-31, 33, 40, and 42).
      1) This isolates Main Feedwater to BOTH SG's.
4. Five (5) seconds after the actuation logic is satisfied a signal is sent to:
   a) CLOSE the Main and Startup Control valves (FDW-32, 35, 41, and 44).
   b) The 5 second time delay reduces the expected water hammer in the FDW line from the results of tripping the Main FDWP's and closing the FDW Control valves.
5. The seal-in circuit (after the two-second time delay) prevents the valves from cycling during an event where SG pressure would be cycling. The seal-in is manually reset by taking both ENABLE/DISABLE switches to DISABLE. When the switches are taken to DISABLE, the TDEFDWP along with the FDW Block and Control valves will go to the position called for by system conditions (if in AUTO) or to the position selected by the respective switches.

P&I 9/27/99 gts

NOTE: Step may be performed when applicable, no sequence required.

2.40 <u>WHEN</u> 1FDW-35 (1A STARTUP FDW CONTROL) <u>AND</u> 1FDW-44 (1B STARTUP FDW CONTROL) are 90% OPEN,

- Ensure valves open:
  - _____ 1FDW-31 (1A MAIN FDW BLOCK).
  - _____ 1FDW-40 (1B MAIN FDW BLOCK).

NOTE: 1FDW-31 (1A MAIN FDW BLOCK) and 1FDW-40 (1B MAIN FDW BLOCK) control switches must be in AUTO to satisfy requirements for MS Line Break system operability.

2.40.1 <u>IF</u> feedwater control problems occur as result of opening 1FDW-31 (1A MAIN FDW BLOCK) <u>AND</u> 1FDW-40 (1B MAIN FDW BLOCK):

_____ A. Place 1FDW-31 (1A MAIN FDW BLOCK) to "OPEN".

_____ B. Place 1FDW-40 (1B MAIN FDW BLOCK) to "OPEN".

_____ C. Contact Compliance Group for MSLB System operability determination.

2.40.2 <u>WHEN</u> feedwater flow has been stabilized <u>AND</u> 1FDW-32 <u>AND</u> 1FDW-41 (MAIN FDW CONTROL) are ≥ 10% OPEN:

_____ _____ A. Place 1FDW-31 (1A MAIN FDW BLOCK) to "AUTO".

_____ _____ B. Place 1FDW-40 (1B MAIN FDW BLOCK) to "AUTO".

NOTE: OP/0/A/1106/031 (Control of Secondary Contamination) contains guidance for SG tube leak detection during transient Xenon operation.

_____ 2.41 <u>WHEN</u> Reactor Power is 15% to 19%, <u>begin</u> OP/0/A/1106/031 (Control of Secondary Contamination). {18}

MAIN FEEDWAT SYSTEM

AB | RB

Feedwater S rposes:

Increase Condensate System pressure to above OTSG pressure to deliver water from the Condensate System to the SG's.

Increase temperature of Feedwater entering the SG's to increase plant efficiency.

Min Recirc:

Auto: Hyd Oil Press >75psig & FDWP disch Press >770 psig then checks suction flow. If <2300 gpm, ramps open

FDW Heaters:

Vertical, U-Tube, tube and shell heater
FDW on tube side

Main FDWP's:

Purpose: increase the 550 psig discharge of the CBP's to >1000 psig to be sufficient to enter the OTSG's.

Single Stage, double suction, centrifugal pump driven by a steam turbine

Rated Flow: 13,200 gpm (~65% full load)
Discharge Press:
Design 1188 psig, Nominal: 1040 psig
Hp: 7400
Min Flow: 2300 gpm

FDWPT Windmill Protection:

Purpose: trip HWP's, CBP's, ~~'E' HDP's, & 'D' HDP's~~ to prevent wiping oil lubricated FDWPT bearings due to shaft rotation caused by water flow induced rotation of FDWP impeller if lube oil is not being provided

Interlock:
FDWP discharge press ≤ 770 psig on BOTH FDWPs AND
Low FDWPT bearing oil press ≤ 4 psig on either FDWPT with suction valve open

FDWP Trips:

Hi Disch Press: 1275 psig (A), 1240 psig (B)
Low Pump Suction pressure: 235 psig
Hi SG Level: 98% Operating Range
MSLB
Manual

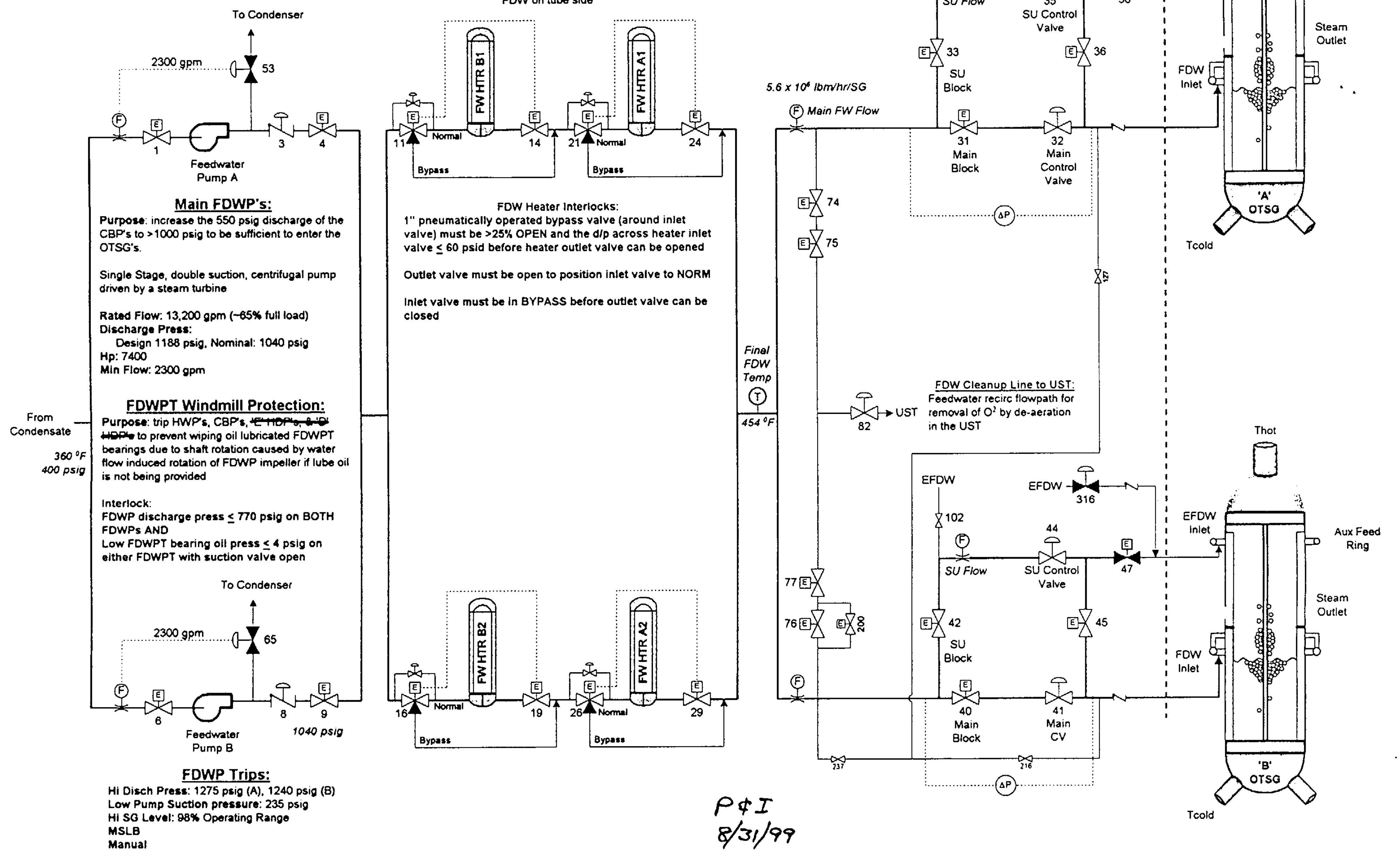
FDW Heater Interlocks:

1" pneumatically operated bypass valve (around inlet valve) must be >25% OPEN and the d/p across heater inlet valve ≤ 60 psid before heater outlet valve can be opened

Outlet valve must be open to position inlet valve to NORM

Inlet valve must be in BYPASS before outlet valve can be closed

FDW Cleanup Line to UST:
Feedwater recirc flowpath for removal of O^2 by de-aeration in the UST



P&I
8/31/99

TRAINING USE ONLY	
FEEDWATER COMPOSITE	
DRAWING #	OP-OC-CF-FDW-1
DRAWN BY: RJL	DATE: 8/19/99
REFERENCE	OFD-121B
APPROVED BY:	Signature on File

QUESTION # 39

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**061000 K4.06**	
Importance Rating	**4.0**	**4.2**

Technical Reference(s): **CF-FDW p.#32**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-FDW OBJ. #17**

Question Source:
Bank # ______
Modified Bank # ______
New ___X___

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 39

Unit 1 plant conditions:

• EFDW in Normal Lineup

- A MS line rupture has occurred
- MSLB circuitry has actuated properly
- operator actions have been completed through . . .

Which ONE of the following is the correct system response when the operator places both trains of the MSLB circuitry to “DISABLE” during the subsequent RCS cooldown?

A. The MD EFDWP will trip.

B. The TD EFDWP will start.

C. 1FDW-315 and 316 (EFDW Control) level control is actuated.

D. 1FDW-35 and 44 (FDW Startup Control) return to auto and control OTSG level.

Need 2 more Bullets to ensure distractors are supported

1 POINT

QUESTION # 39

061000K 4.06 Both PRA 2-9-00

A. Incorrect – The MD EFDWP are started when MSLB circuitry trips the Main FDW Pumps but requires manual action to secure the operating MD EFDWPs.

B. Correct – The TD EFDWP is prevented from automatically starting on a MSLB signal but when cleared by selecting "DISABLE" this action will allow auto start of the TD EFDWP due to the Main FDWP trip signal.

C. Incorrect – FDW-315/316 are on level control when in Auto.

D. Incorrect - The Main and Startup FDW BLOCK valves will return to automatic if selected to automatic. However, they will be selected to "closed" per Rule # 6, Main Steam Line Break Actions.

13. State the appropriate values for the following FDW System parameters: (100% power) (R33)

    13.1 FDWP suction temperature

    13.2 FDWP suction pressure

    13.3 FDWP discharge pressure

    13.4 Final Feedwater temperature

    13.5 FDW flow (in both gpm and lbm/hr)

14. State the setpoints and automatic actions that occur based on FDWP discharge pressure and FDWPT hydraulic oil pressure. (R29)

15. Given a set of conditions, determine proper operation of the FDWP Seal Injection System. (R30, R31)

16. Describe the purpose of the MSLB Detection/Isolation system. (R34)

17. Given a set of conditions, verify proper operation of the MSLB Isolation Circuit. (R35)

C. The MSLB Detection and Isolation system is designed with two (2) independent trains that use a 2 out of 3 logic (one for each steam header) for system actuation. Either train that is actuated will isolate all Feedwater sources (excluding the MDEFDWP's and FDW-315 and 316) to both SG's.

1. Actuation setpoint is 550 psig on two (2) out of three (3) pressure transmitters on either steam line.
2. The pressure signal requires two (2) seconds to seal-in.
   a) The 2 second time delay prevents instantaneous (erroneous) signal from unnecessarily isolating FDW from the SG's.
3. After the logic is satisfied and is sealed-in (2 second time delay has been satisfied), a signal to complete the following is generated:
   a) Trip <u>both</u> Main FDWP's
      1) This lowers the ΔP across Main and Startup Block valves and ensures the valves close.
   b) Inhibit auto-start <u>or</u> initiate an auto-stop (trip) of the TDEFWP
   c) CLOSE the Main and Startup Block valves (FDW-31, 33, 40, and 42).
      1) This isolates Main Feedwater to <u>**BOTH**</u> SG's.
4. Five (5) seconds after the actuation logic is satisfied a signal is sent to:
   a) CLOSE the Main and Startup Control valves (FDW-32, 35, 41, and 44).
   b) The 5 second time delay reduces the expected water hammer in the FDW line from the results of tripping the Main FDWP's and closing the FDW Control valves.
5. The seal-in circuit (after the two-second time delay) prevents the valves from cycling during an event where SG pressure would be cycling. The seal-in is manually reset by taking both ENABLE/DISABLE switches to DISABLE. When the switches are taken to DISABLE, the TDEFDWP along with the FDW Block and Control valves will go to the position called for by system conditions (if in AUTO) or to the position selected by the respective switches. PIP 9/27/99 gcs

QUESTION # 40

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**1**	**1**
	K/A #	**061000 A2.05**	
	Importance Rating	**3.1**	**3.4**

Technical Reference(s): **CF-EFDW**
EL-VPC

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-EFDW #38**

Question Source:
Bank # ______
Modified Bank # CF-73
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 40

Unit 1 plant conditions:

- A loss of Main FDW has occurred
- Motor Driven EFW pumps are maintaining SG levels in AUTOMATIC level control:
  - SG "1A" EMERG LVL CTRL switch selected to "PRIMARY"
  - SG "1B" EMERG LVL CTRL switch selected to "BACKUP" for I&E
- A Loss of 1DIB occurs

Which ONE of the following describes the effect of the loss and restoration of 1DIB?

A. Input to FDW-315 automatic level control circuit will automatically transfer to the backup XSUR level instrumentation train and maintain "1A" S/G level at 30". Upon restoration of 1DIB power the level control circuit will automatically transfer to the primary XSUR level instrumentation train.

B. Input to FDW-316 automatic level control circuit will automatically transfer to the primary XSUR level instrumentation train and maintain "1B" S/G level at 30". Upon restoration of 1DIB power the level control circuit will remain selected to the primary XSUR level instrumentation train.

C. Input to FDW-315 automatic level control circuit will automatically transfer to the backup XSUR level instrumentation train and maintain "1A" S/G level at 30". Upon restoration of 1DIB power the level control circuit must be manually transferred to the primary XSUR level instrumentation train.

D. Input to FDW-316 automatic level control circuit will remain selected to the backup instrumentation train and maintain "1B" S/G level at 30". Upon restoration of 1DIB power the level control circuit will remain selected to the backup XSUR level instrumentation train.

Not sure about explanation for D.

1 POINT

QUESTION # 40

061000A2.05

A. Correct, - DIB is the primary source to FDW-315. Loss of DIB causes the FDW-315 level control circuit to auto swap to DIC backup source and controls at 30". Thirty inches is the setpoint when RCPs are operating. Upon restoration of DIB the "A" level control circuit will auto swap back to the primary source.

B. Incorrect, - DIB is the backup source to FDW-316. When the automatic level control is selected to backup there is no auto swap to primary when power is lost.

C. Incorrect, - See a.

D. Incorrect, - FDW-316 automatic level control circuit will remain selected to the backup instrumentation train. Since the power supply had failed a low level signal will be processed and FDW-316 will open resulting in a level above 30 inches.

transient

30. Explain how to bypass the AUTO START feature of the TDEFDWP. (R28)
31. Describe or make a sketch of the logic/conditions that will AUTO START the TDEFDWP when its control switch is in AUTO, including a description of AMSAC. (R25)
32. Describe the affect a Main Steam OTSG Isolation actuation will have on the EFDW system. (R58)
33. Describe the additional action required to allow emergency feed through the alternate ICS flowpath, following actuation of the MS Line Isolation circuit (R59)
34. List the EFDW SG Level setpoints for the conditions when RCPs are running and when all RCPs are off. (R37)
35. Describe the SG level indicators used in the EFDW System, including whether or not they are temperature compensated, and how to select the PRIMARY/BACKUP indicators. (R30)
36. Detail by sketch the status of the solenoids associated with EFDW Level control (Train A & B) when MFDW is operating and normal power to the solenoids is available. (R31)
37. Explain the operation of the solenoids associated with EFDW Level Control (Train A & B) when MFDW is lost and normal power to the solenoids is available from a provided sketch or by making a sketch. (R32)
38. Explain the operation of the solenoids associated with the EFDW Level Control (Train A & B) when MFDW is lost and normal power to the solenoids is lost, from a provided sketch or by making a sketch (R33)
39. List the locations of FDW-315 & 316. (R27)
40. List which Level Train is the PRIMARY TRAIN for SG A level control and which is the PRIMARY TRAIN for SG B level control. (R35)
41. Explain why the Primary Level Train for SG A is not the same as the Primary Level Train for SG B. (R36)
42. Describe how to manually control FDW-315 & 316, after a loss of MFDW, from the Control Room. (R34)
43. Describe the methods for throttling EFDW flow, available to the operator. (R49)
44. Describe how the TDEFDWP meets "AC Independence" criteria include how each component helps provide this independence. (R38)
45. Explain the purposes of the Nitrogen bottles associated with the EFDW System. (R39)

a) FDW-315 - Primary - Powered from KVIB

b) FDW-315 - Backup - Powered from KVIC

c) FDW-316 - Primary - Powered from KVIC

d) FDW-316 - Backup - Powered from KVIB

	"A" OTSG (FDW-315)	"B" OTSG (FDW-316)
Primary	Train A (KVIB)	Train B (KVIC)
Backup	Train B (KVIC)	Train A (KVIB)

2. Same taps as Startup range except these transmitters are not suppressed.
3. XSUR levels are not temperature compensated.
4. Inputs to FDW-315/316 control circuit and Dryout Protection Logic.
5. Primary/Backup Select (Figure OC-CF-EF-17)
   a) Select level string to control FDW-315/FDW-316
   b) Auto swap on loss of power from Primary to Backup
   c) No auto swap from Backup to Primary if Backup switch selected.
6. Computer Points
   - Power failures on level trains
   - Primary level A & B
   - Backup level A & B

C. FDW-315 Level Control Operation (Figure OC-CF-EF-18)

1. Selected to "AUTO" - Steady State - "A" SG
   a) Solenoid #1 normally energized / Solenoid #2 de-energized.
   b) "Primary" level error is ported through 3-way valve #1 to 3-way valve #2 where it is passed to FDW-315.
   c) Manual loader signal is only applied through 3-way valve #2 to FDW-315 to control the valve when selected to "manual".
   d) Auto level signal from primary train A will pass to FDW-315 to control valve (30" or 240")

e) Loss of 4 RCPs is sensed from RPS pump monitors which is a non - ICS dependent signal

2. Loss of KVIB

a) If DIB is lost, KVIB is lost and solenoid #1 de-energizes.

b) 3-way valve #1 swaps and takes backup level from train B (KVIC) to control FDW-315.

- **This automatic swap from primary to backup level trains requires no operator action.**

c) Solenoid #1 may also be de-energized by the operator selecting backup pushbutton.

- **With backup selected, there is no automatic swap back to primary.**

d) Loss of KVIB - DIB will not inhibit automatic level control, but will prevent manual control.

3. Operator Selects Manual

a) With automatic level control in effect, the operator may desire to throttle FDW-315.

b) Operator should select Manual - this re-energizes solenoid #2 and 3-way valve swaps back to accept manual loader demand.

D. Match valve position demand to manual loader output prior to selecting manual to limit unnecessary swings on system.

E. FDW-316 Level Control Operation

1. Level control for FDW-316 is essentially the same as FDW-315, except as noted below.

- Train B is powered from KVIC and is the primary train for B SG.
- Train A powered from KVIB is the Backup Train.
- Solenoids #1 and #2 are powered from DIC.
- This ensures that upon system actuation and single failure of KVIC or KVIB, only one circuit will auto-swap to its backup level circuit.

4.3 Manual Operation of FDW-315 and FDW-316 (Fig OC-CF-EF-11)

t) **B4** - Alternate Source AC input (100 amp circuit breaker) - Used to connect the AC Line source (KRA) to the Manual Bypass switch. Normally closed during inverter operation.

u) **S1** - Manual Bypass Switch - Make before break switch that allows the operator to manually select either the Normal Source (Inverter Output) position "A" or the Alternate Source (AC Line - KRA) position "B" with no interruption of power to the panel. When the switch is in the Bypass position, the inverter is completely bypassed and the input as well as the output breaker can be opened to perform maintenance on the inverter.

B. **Vital Power System Operation** (Refer to OC-EL-VPC-1&3)

1. During normal system operation, each Vital Instrumentation Power Panelboard is energized via its associated Vital Bus inverter. Each inverter is energized from its associated 125VDC Instrumentation and Control Power Panelboard.

2. Each 125VDC Instrumentation Power Panelboard is energized from one of the unit's 125VDC Control Power buses and battery chargers through the associated set of isolating diodes. Should a problem develop with the battery charger or bus supplying a Vital Bus inverter, whereby this source is lost, an alternate unit's 125VDC supply will be auctioneered to supply the inverter through the isolating diode network without an interruption of power.

3. On the other hand, if a problem develops with a Vital Bus inverter and it fails to provide power, the associated 125VAC vital bus will lose power. There is no automatic transfer to the AC Line supply associated with the Vital Bus inverters.

   a) In order to re-energize the 125VAC vital bus, a manual transfer to AC Line must be made using the Manual Transfer switch.

4. Startup and shutdown:

   a) A simple startup procedure for the vital inverters would be as follows: (Assume that the 125VAC panelboard is being supplied by AC Line)

      1) Press the Precharge switch until the Precharge light is lit.

      2) Close the DC Input circuit breaker B1.

      3) Close the Inverter Output circuit breaker B2.

      4) Verify that the In Sync light is lit.

      5) Transfer the Manual Bypass switch to the Normal Source position.

Exam Question Report

27-Jan-99

Question ID:	CF073	Revision No:	0	Revision Date	10/29/1999
Question Description:	CF073				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	CF-EF - Emergency Feedwater System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	LRO = 33; SRO = 33 Reference: KA:SF4 K4.11 (2.7/2.9) CF-EF NRC #040	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

The following conditions exist on Unit 1:

- A loss of ALL Main FDW has occurred
 Motor Driven EFW pumps are maintaining SG levels at setpoint in AUTOMATIC level control
- A Loss of 1DIB Panel Board occurs

Choose ONE of the following to complete this sentence:

(Back-up level control has NOT been selected.)
The level input to 1FDW-315 (EFDW Line to "A" SG Control) automatic control circuit ____________ to the backup XSUR level instrumentation train, and upon restoration of 1DIB power, the level control circuit ____________ to the primary XSUR level instrumentation train. (.25)

A) will automatically transfer; will automatically transfer.

B) must be manually transferred; must be manually transferred.

C) will automatically transfer; must be manually transferred.

D) must be manually transferred; will automatically transfer.

Answer

A Ref: KA:SF4 K4.11 (2.7/2.9) CF-EF NRC#040
A. CORRECT - DEENERGIZE DIB SWAPS TO BACKUP ENERGIZE WITH PRIMARY SELECTED SELECTS PRIMARY TRAIN
B. INCORRECT - NO MANUAL TRANSFER REQUIRED
C. INCORRECT - NO MANUAL TRANSFER REQUIRED WHEN POWER RETURNED
D. INCORRECT - NO MANUAL TRANSFER REQUIRED WHEN POWER IS LOST

Lessons

ID	Description
CF-EF	Emergency Feedwater

QUESTION # 41

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**1**	**1**
	K/A #	**068000 A3.02**	
	Importance Rating	**3.6**	**3.6**

Technical Reference(s): **RAD-RIA**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **RAD-RIA #5**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 41

Plant conditions:

- 1RIA-54 in alarm and in the NORMAL position
- "A" Turbine Building Sump Pump operating
- "C" Turbine Building Sump Pump operating

Which ONE of the following is correct?

A. "A" Turbine Building Sump Pump will trip and will require manual start after the RIA alarm condition is cleared.

B. "A" Turbine Building Sump Pump will trip and prevent the "B" Turbine Building Sump Pump start.

C. "C" Turbine Building Sump Pump will trip and will require manual start after the RIA alarm condition is cleared.

D. "C" Turbine Building Sump Pump will trip and prevent the "D" Turbine Building Sump Pump start.

1 POINT

QUESTION # 41

068 A302 New 5-25

A. Incorrect –1RIA-54 monitors the Unit 1 and 2 TBS. The "A" pump will trip but will return to automatic operation after the RIA is cleared. The pump does not require a manual restart

B. Correct – If an alarm is received on 1RIA-54 this will trip the operating pump and prevent the start of the alternate pump.

C. Incorrect – 3RIA-54 monitors the Unit 3 TBS (not 1RIA-54). C and D TBSP are for operation of Unit 3. If 3RIA-54 was in alarm the "C" pump will trip but will return to automatic operation after the RIA is cleared. The pump does not require a manual restart

D. Incorrect – If an alarm is received on 3RIA-54 this will trip the Unit 3 (C or D) operating TBSP and prevent the start of the alternate pump.

1 POINT

QUESTION # 41

Plant conditions:

- 1RIA-54 in alarm and in the NORMAL position
- "A" Turbine Building Sump Pump operating
- "C" Turbine Building Sump Pump operating

Which ONE of the following is correct?

A. "A" Turbine Building Sump Pump will trip and "B" will start.

B. "A" Turbine Building Sump Pump will trip and prevent the "B" Turbine Building Sump Pump start.

C. "C" Turbine Building Sump Pump will trip and "D" will start.

D. "C" Turbine Building Sump Pump will trip and prevent the "D" Turbine Building Sump Pump start.

why would a protective function initiate alternate train equipment when it is designed to terminate potential sources of AR.?

1 POINT

QUESTION # 41

068 A302 New 5-25

A. Incorrect – This is normal operation of the TBSPs. When the first pump stops or trips the alternate pump will start. 1RIA-54 monitors the Unit 1 and 2 TBS.

B. Correct – If an alarm is received on 1RIA-54 this will trip the operating pump and prevent the start of the alternate pump.

C. Incorrect – This is normal operation of the TBSPs. When the first pump stops or trips the alternate pump will start. 3RIA-54 monitors the Unit 3 TBS. C and D TBSP are for operation of Unit 3.

D. Incorrect – If an alarm is received on 3RIA-54 this will trip the Unit 3 (C or D) operating TBSP and prevent the start of the alternate pump.

4.1 Scintillation

4.2 Geiger Mueller

4.3 Ionization

5. Describe the basic function of each applicable monitor.
(R2)

5.1 State the purpose of each monitor.

5.2 State where each monitor is located.

5.3 List the interlocks and automatic actions associated with each applicable monitor.

5.4 When given the monitor title, be able to state which system(s) the monitor checks.

6. List seven (7) functions which can be performed at the new RIA Control Room CRT. (R8)

7. Describe the basic procedure to check/set High and Alert alarm setpoints. (R5)

8. Describe the operational relationship between the following components associated with the Sorrento Radiation Monitoring System: (R10)

8.1 RM-80 Microprocessor Unit

8.2 Transient Monitor System Computer

8.3 View Node

9. For the following situations, state whether or not the associated Radiation Monitor is operational and explain why for each case: (R11)

2. If RIA 45/49 is out of service, but RIA 46/49A is in service and the accepted range setpoint (channel item #22 for the high range detector) is left where it currently is, the activity in the Vent/Reactor Building would increase with RIA 46/49A reading zero until the accepted range setpoint (in cpm) is met. At this point the interlock would be actuated because the Accepted range is currently set above the high setpoint. Note that the interlock would NOT have actuated when required by the high setpoint. A PIP has been written to address this problem.

M. (1)(3)RIA-50 - Monitors component cooling to determine any radioactive effluent leakage into system

1. Certain amount of chromate activation by neutron flux expected.
2. Detector sodium iodide.
3. Located component cooling water pump room.

N. (1)(3)RIA-54 - Monitors turbine building sump activity.

1. Sample pump provides continuous flow. (If RIA-54 sample pump is found off, notify Radwaste Chemistry of a possible unmonitored release.)
2. Interlock to trip turbine building sump pumps if:
   a) High Activity in the TBS
   b) Loss of power to RIA
   c) actuates 1SA-18/C5 RM TBS Interlock
   d) the sample pump stops (power failure or clogged strainer)
3. Monitor operation is controlled by a dedicated RM-80 microprocessor mounted on the skid.
4. The RM-80 is programmed to monitor the output of the detector and checksource, and also monitor and control operation of the sample pump, sample inlet valve, purge inlet valve, and other flow instrumentation.
5. A dedicated control/display module on the skid provides an interface where maintenance, calibration and limited diagnostic functions can be performed.
6. The RM-80 automatically performs a periodic checksource check to verify detector response and channel operability and a periodic purge to flush the sample chamber and flow instrumentation to prevent the buildup of contamination and sludge.

QUESTION # 42

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**1**	**1**
K/A #	**072000 K3.02**	
Importance Rating	**3.1**	**3.5**

Technical Reference(s): **FH-FHS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **NONE**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 42

Unit 1 plant conditions:

- MODE 6
- Fuel is being moved in the Spent Fuel Pool (SFP)
- Fuel is being moved in the Reactor Building (RB)
- 1RIA-3 (RB Canal) is out of service

Which ONE of the following is the required action per the SLC 16.12.2 (Refueling Operations) and OP/1/A/1502/007 (Operations Refueling/Defueling Responsibilities)?

A. Suspend all fuel movement. in Both the SFP ~~and~~ RB

B. Continue fuel movement and no other actions are required.

C. Suspend fuel movement in the ~~Reactor Building~~ RB until RIA-4 (RB Entrance) is verified operable.

D. Continue fuel movement and immediately use a portable instrument having the appropriate range and sensitivity to fully protect individuals.

1 POINT

QUESTION # 42

072000 K3.02

A. Incorrect – Fuel movement in the SFP is not required to be suspended.

B. Incorrect – Fuel movement cannot continue unless an alternate radiation monitor is available.

C. Incorrect - RIA-3 is required to be operable for fuel movement in the RB. RIA-4 cannot backup RIA-3.

D. Correct - RIA-3 is required to be operable or a compensatory radiation monitor in place during fuel handling. RIA-3 is not required to be operable for fuel handling in the Spent Fuel Pool.

5. Upenders shall be lowered and carriages driven to the SFP so that SF-1 and SF-2 can be closed.
   a) If the SF pool cooling system is aligned per the SF System Operation During Fuel Loading procedure, the 'B' SF cooling pump should be stopped before closing SF-1 and SF-2.
6. Once the Refueling SRO is notified he should:
   a) Initiate a search to identify and correct (if possible) the cause of the decrease.
      1) If leak is on in-service LPI or SFP cooling train:
         (a) Initiate makeup to the SFP and FTC as necessary.
         (b) Start alternate LPI and SFP cooling systems.
   b) If time permits, direct an operator to deenergize the FTC underwater lights before they become uncovered.
   c) Maintain DHR until leak is stopped or RCS drains below level of leak.
      1) If Loss of DHR is imminent due to loss if inventory, establish RB containment closure.
      2) If FTC leak causes the RCS to drain below 10" on LT-5, align LPI pump suction to the RB emergency sump.

E. Abnormal Procedures
   1. Both the Loss of Power AP (Blackout Section) and the Natural Disaster AP require fully inserting fuel assemblies in the fuel racks and suspending all fuel handling and ISFSI movements if the AP is entered.
   2. The Natural Disaster AP also provides guidance to inspect the ISFSI Horizontal Storage Modules within 24 hours after a tornado event. If debris is found in the air inlets/outlets. If any is found, it must be removed within 40 hours of the tornado event.

2.5 Refueling Procedures

A. Much of the refueling procedure as it used to be has been transferred to Maintenance, and is covered by MP/0/A/1500/09. Operations responsibilities are addressed in OP/1,2,3/A/1502/07, Operations Defueling/Refueling Responsibilities.

B. Limits and Precautions and requirements listed in the refueling system procedures are summarized here.
   1. RIA's 49, 3, 4 to monitor radiation levels in the RB and RIA's 41, and 6 are required to be operable to monitor radiation levels in the SFP..

a) Checked for operability within one week prior to refueling.

b) SLC 16.12.2 states: "Radiation levels in the RB refueling area shall be monitored by RIA-3 and by a portable monitor for each bridge which is being used for fuel handling. Radiation levels in the Spent Fuel Storage Area shall be monitored by RIA-6 and by a portable bridge monitor"..

c) If RIA's 3, or 6 become inoperable, comparable portable survey instrumentation having the app4ropriate ranges and sensitivity to fully protect individuals involved in refueling operations be used immediately.

NOTE: RIA 3 was moved from the Aux. Bridge to the Fuel Transfer Canal Area wall and RIA 6 was moved from the Spent Fuel Bridge to the Spent Fuel Pool Area wall.

d) RIA 3 & 6 both provide signals to the control room and each is equipped with an audible alarm to warn personnel in the area of increased radiation levels.

1) Portable Bridge Monitors have been installed on the Main Fuel Bridge and the Spent Fuel Pool Bridge. Since these monitors are only needed to protect personnel working on or near these bridges, the new monitors do not provide a signal to the control rooms. They do provide a local indication of radiation levels and are equipped with audible alarms. Due to the removal of RIA's 3 and 6 from the fuel bridges, these portable monitors are required to be installed on the bridges and be operable during fuel loading and refueling operations. Contact RP for suitable substitutes.

2. ITS 3.9.2 Nuclear Instrumentation

a) Requires two flux monitors during core alterations or when positive reactivity additions are being made. If these are not operable core alterations and the positive reactivity additions are to be suspended immediately.

b) One monitor is required when in MODE 6 with no evolutions in progress.

c) This provides immediate indication of an unsafe condition.

d) NI-1, 2, 3, and 4 source range instruments are capable of this requirement, but the Defuel/Refuel sequence assumes NI-1 and NI-2 will be used for fuel handling.

e) In March of 1986 during fuel movement an incident occurred because NI-1 failed during actual core geometry change. This went undetected for 8 assembly moves, thus a Tech Spec violation occurred.

O 1 2 K 5 0 2

16.12 REFUELING OPERATIONS

16.12.2 Area Radiation Monitoring for Fuel Loading and Refueling

COMMITMENT Radiation levels in the reactor building refueling area shall be monitored by RIA-3 and by a portable bridge monitor for each bridge which is being used for fuel handling. Radiation levels in the spent fuel storage area shall be monitored by RIA-6 and by a portable bridge monitor.

APPLICABILITY: During fuel handling.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Area Radiation Monitor not monitoring radiation levels.	A.1 Use portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.12.2.1 N/A.	N/A

BASES

The requirement(s) of this SLC section were relocated from CTS 3.8.1 during the conversion to ITS.

Continuous monitoring of radiation levels provides immediate indication of an unsafe condition.

REFERENCES

N/A

QUESTION # 43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**2**	**2**
	K/A #	**006000 A1.13**	
	Importance Rating	**3.5**	**3.7**

Technical Reference(s): **PNS-CF**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-CF OBJ. #6**

Question Source:
- Bank # **PNS-579**
- Modified Bank # ______
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 43

Which ONE of the following is the reason for limiting the pressurization rate of the Core Flood Tanks (CFTs)? (to < 100 psig/15 min)

The pressurization rate is limited to prevent...

A. thermal shock of the CFT.

B. lifting the CFT relief valves.

C. a pressure surge from unseating the check valves .

D. exceeding the CFT pressure Technical Specification limit.

1 POINT

QUESTION # 43

006000A1.13 common /gcw 05/10/00 Bank PNS-579

A. Correct, per engineering studies if the pressurization rate exceeds 100 psig/15 min, thermal shock to the CFTs will occur.

B. Incorrect, The relief valves will lift when setpoint is reached not on a pressurization rate of the CFTs..

C. Incorrect, The RCS is at system pressure and the CFT pressure would have to be higher than CFT design pressure to open the check valve.

D. Incorrect, The TS limit on pressure is not affected by the pressurization rate..

OBJECTIVES

TERMINAL OBJECTIVES

1. Upon completion of this lesson plan the student will be able to demonstrate a knowledge and understanding of the operation, design and limits of the Core Flood System. (T1)

2. Upon completion of this lesson plan the student will be able to apply Improved Technical Specifications Conditions and Required Actions to ensure OPERABILITY of the Core Flood Tanks. (T2)

ENABLING OBJECTIVES

1. Discuss the Design Basis of the Core Flood System. (R1)(R15)

2. Given a set of conditions for the fill sources for the CFT's determine if the source can be used and if not explain the why. (R2) (R6)

3. Discuss the reason for and the procedural requirement for boron concentration in the CFT's (R3)

4. Discuss the procedure for verifying CFT check valve operability. (R4)

   4.1 Method Used

   4.2 Parameters monitored

5. Given a set of conditions on the CFT's evaluate the conditions to determine if the CFT's are aligned for ES. (R5)

6. Discuss the normal pressurization rate of the CFT's and any additional actions required to exceed the normal pressurization rate. (R8)(9)

7. Discuss the depressurization flowpaths for the CFT's and any precautions to be taken while using these flowpaths. (R11)

8. Discuss the drain flowpaths for the CFT's and explain any precautions associated with the different flowpaths. (R12)(R13)

9. Given a set of plant conditions determine if the CFT's can be discharged to the RCS loops and any concerns while performing the evolution. (R14)

10. Describe what would occur if CFT levels and pressures were to be maintained outside their required bands. (R17).

c) The path for CFT makeup from the HPI System exists, but is no longer a procedural option when at normal system pressure. The HPI to CF tank makeup valves (HPI-154, 155, & 156) have experience leakage past their seat, after being used to control makeup flow to the CFTs with HPI pumps running. This has lead to HPI to CFT makeup leakage, which causes level increases and deboration of the CFTs.

1) When this flow path is used to initially fill the CFTs, (HPI-154, 155, & 156) are positioned fully open before starting a HPI pump, and remain open until the HPI pump is stopped.

F. Pressure Makeup to the Core Flood Tanks

1. The source of pressure makeup to the CFTs is Nitrogen from the High Pressure Nitrogen Header.

2. To establishing initial pressurization rate:

a) Open N-137 (Core Flood Tanks Supply).

b) Operator, in constant communication with control Room, stationed at N^2 valve (N-128/130).

c) Open Nitrogen Fill Valve (N-298/299), then throttle open N-128/130 to set the pressurization rate not to exceed 100 psig/15 min (6.6 psig/Min.).

d) When desired pressure is reached, close Nitrogen Fill Valve and N-137.

e) Normal pressurization rate is set at < 100psig/15 min. This was established to prevent thermal shock of the CFT with cold nitrogen, If a faster rate is needed, the Nitrogen Heater must be place in service per Enclosure 4.9 of OP/A/1104/001.

3. Day to day makeup only requires opening N-137. N-298/299 are opened from the Control Room to pressurize the CFT's as necessary.

G. Depressurizing the Core Flood Tanks

1. CFT's are depressurized when no longer needed for ES.RCS pressure is < 800 psig by ITS and ≤ 700 psig by procedure.

a) CF-1 and CF-2 breakers are closed.

b) CF-5 and CF-6 power supplies are energized.

c) CFTs are depressurized.

2. If RB activity levels permit entry for manual valve alignments, the CFTs will generally be depressurized to the RB Containment on Unit 1 or directly to the Waste Gas Filters on unit 2 or 3.

a) If RB activity precludes an entry, the CFTs must be depressurized to the waste gas system via the Quench Tank.

Exam Question Report

27-Jan-99

Question ID:	PNS579	Revision No:	0	Revision Date	10/29/1999
Question Description:	PNS579				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	PNS-CF - Core Flood System
Last Used Date:	02/08/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 9; LRO = 9; SRO = 9 Reference: OP-OC-PNS-CF	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Which ONE of the following is the reason for limiting the pressurization rate of the Core Flood Tanks (CFTs)? (.25)

The pressurization rate is limited to prevent...

A) thermal shock of the CFT

B) lifting the CFT relief valves

C) a pressure surge from unseating the check valves

D) exceeding the CFT pressure Technical Specification limit

Answer

A
A. Correct - per Engineering studies of the presurization rate exceeds 100 psig/ 15 min, thermal shock to the CFTs will occur.
B. Incorrect - The relief valves will lift when setpoint is reached not on a pressurization rate of the CFTs.
C. Incorrect - The pressure of the CFTs is done in a slow controlled manner. For the pressure to increase high enough to lift the check valves the design pressure of the CFTs would have been exceeded.
D. Incorrect - The technical specification limit on pressure is not affected by the pressurization rate.

Lessons

ID	Description
PNS-CF	Core Flood System (PNS-CF)

Enabling Objectives

QUESTION # 44

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**010000 K6.03**	
Importance Rating	**3.2**	**3.6**

Technical Reference(s): **PNS-PZR**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PZR OBJ. #5 & #6**

Question Source:
Bank # ___X___
Modified Bank # _______
New _______

Question History:
Previous NRC Exam ___X___**(1998 Q.# 70)**
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 44

Unit 2 plant conditions:

- Unit shutdown in progress
- 3 RCPs operating (2B2 RCP secured)
- Reactor power = 35% decreasing
- RCS pressure is increasing
- 2RC-1 (PZR Spray) is fully open

Which ONE of the following ALONE could cause the increasing pressurizer pressure?

A. Present combination of RCPs will not achieve adequate spray flow.

B. High concentration of non-condensable gases in the pressurizer.

C. PZR Bank #1 proportional control has failed to maximum.

D. 2RC-2 (PZR Spray Bypass) valve has vibrated closed.

Isn't this a locked throttle valve?

1 POINT

QUESTION # 44

010000 K6.03 common rsi/gcw 05/07/00 BANK 1992 NRC EXAM

A. Incorrect, spray flow is provided by the delta P across the reactor via the "B" loop and RCP 2B1. The pressurizer is a vertical, cylindrical vessel that is connected to the reactor vessel outlet piping by a 10 in. surge line (loop A on Unit 1 and loop B on Units 2 and 3). A 2-1/2 in. spray line originates at the discharge of a reactor coolant pump and terminates in the top of the pressurizer with a spray nozzle. The differential pressure causes flow across the reactor due to pressure losses as the coolant flows through the vessel.

B. Correct, An insurge (pressurizer water level increase) occurs due to an increase in RCS Tc temperature due to the power is decrease. As level increases, system pressure increases due to the compression of the steam bubble in the pressurizer. Some of the steam volume condenses, helping to relieve system pressure. Water is sprayed into the steam space to condense steam and reduce RCS pressure. A condition referred to as a “hard bubble” can occur if non-condensable gases (i.e., N2, H2, air) are not properly vented during pressurizer steam bubble formation. If a significant volume of these non-condensable gases accumulate in the pressurizer, they can inhibit pressurizer response to transients. For example, following an insurge pressurizer spray may not be able to adequately reduce RCS pressure due to the compression of the non-condensable gases.

C. Incorrect, Based on B&W calculations and startup testing experience, a conservative value for the total heat loss from the pressurizer, under normal hot standby conditions, is 107 kilowatts. By design, the first bank (126 kW) of heaters utilizes proportional control and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at the set point.

D. Incorrect, A continuous bypass flow is maintained around the pressurizer spray control valve to prevent the accumulation or dilution of boric acid in the pressurizer and to prevent thermal shock of the spray nozzle. RC-2, (Spray Control Bypass), (a manually operated valve) continuously circulates reactor coolant through the spray loop bypassing RC-1. Since the pressurizer is a remotely located component connected to a hot leg, this bypass flow minimizes temperature differentials in the spray and surge lines, prevents thermal shock of the spray nozzle, and minimizes the boron concentration difference between the pressurizer and RCS.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

Upon completion of this lesson, the student will demonstrate an understanding of the components, indications, controls and operation of the Pressurizer. The student will be able to assess the status of the Pressurizer during normal, abnormal and emergency conditions and determine corrective actions for improper system operation. The student will also be able to apply any ITS/SLC Conditions and Required Actions associated with the Pressurizer.(T1)

ENABLING OBJECTIVES

1. Explain the design basis of the pressurizer. (R21)
2. Describe pressurizer response during load or RCS temperature changes. (R1)(R2)(R3)
3. Given a set of conditions, calculate the change in pressurizer level for a change in RCS temperature. (R33)
4. Explain what is meant by a "subcooled" pressurizer and how to determine if the pressurizer is in a subcooled condition.(R22)(R27)
5. Explain what is meant by a pressurizer "hard bubble" and describe the adverse effects of a "hard bubble" on plant operation, (R23)
6. Identify the source of pressurizer spray for each unit. (R4)
7. Discuss the automatic setpoints and any interlocks associated with pressurizer instrumentation. (R5)
8. Explain the operation of the ICS RC pressure signal median select function as it relates to RC pressure control including: (R28)
   - 8.1 How median select chooses the controlling signal
   - 8.2 Which pressurizer components receive a median selected RC pressure signal.
9. Given a set of conditions, determine which RC pressure signal has been selected for control by the ICS RC pressure signal median select function. (R36)
10. Discuss the reasons for bypass flow around the pressurizer spray valve during normal operation. (R6)
11. Evaluate plant response to a failed open pressurizer spray valve without operator action. (R20)
12. Explain the operation of the Pressurizer Water Space Saturation Recovery Circuit. (R29)

D. Hard Bubble{ TC "Hard Bubble" \f C \l "1" }

1. A condition referred to as a hard bubble can occur if non-condensable gases (i.e., N_2, H_2, air) are not properly vented during pressurizer steam bubble formation.

2. If a significant volume of these non-condensable gases accumulate in the pressurizer, they can inhibit pressurizer response to transients.

3. For example, following an insurge pressurizer spray may not be able to adequately reduce RCS pressure due to the compression of the non-condensable gases, i.e., the hard bubble.

2.3 General Description{ TC "General Description" \f C \l "1" }

A. The pressurizer establishes and maintains RCS pressure and provides a steam chamber and water reserve to accommodate reactor coolant density changes during operation.

B. The pressurizer is a vertical, cylindrical vessel that is connected to the reactor vessel outlet piping by a 10 in. surge line (loop A on Unit 1 and loop B on Units 2 and 3). A 2-1/2 in. spray line originates at the discharge of a reactor coolant pump and terminates in the top of the pressurizer with a spray nozzle.

1. Flow is caused by the differential pressure across the reactor due to pressure losses as the coolant flows through the vessel.

2. A continuous bypass flow is maintained around the pressurizer spray control valve to prevent the accumulation or dilution of boric acid in the pressurizer and to prevent thermal shock of the spray nozzle.

C. The pressurizer contains three removable electric heater assemblies in its lower section. These assemblies are divided into four banks for control purposes.

1. Bank 1 has a variable heater output and is used to compensate for normal heat losses from the pressurizer.

2. Banks 2, 3, and 4 are on-off type heaters and are used for additional heating of the pressurizer contents during plant startup and load changes. Pressurizer water temperature is normally maintained at 648° F, which corresponds to a saturation pressure of 2155 psig.

D. Pressurizer level is controlled automatically by makeup from the High Pressure Injection (HPI) System. Pressurizer level is normally controlled at a level of 220 inches via HP-120, RC Volume Control.

1. The normal operating band is 200" to 260" (low and high statalarm setpoints, respectively)

2. Per Improved Technical Specification 3.4.9, the maximum allowable pressurizer level is $\leq$ 285 in MODES 1,2 and 3 with RCS temperature $\geq$ 325° F. This maximum level is based on a startup accident and a loss of MFW.

*QNUM 33381
*HNUM 33964 (Do NOT change If < 9,000,000)
*ANUM
*QCHANGED FALSE
*ACHANGED FALSE
*QDATE 1992/01/20
*FAC 269 Oconee 1, 2 & 3
*RTYP PWR-B&W177
*EXLEVEL S
*EXMNR
*QVAL
*SEC
*SUBSORT
*KA 010000K603
*QUESTION

Plant conditions on Unit 2 are as follows:

- Reactor coolant pump B2 has tripped.
- Reactor power is stable at 35%.
- Pressurizer pressure is increasing even though the operator has fully opened the spray valve.
- Pressurizer heater indications have been lost.

Which ONE of the following ALONE could cause the increasing pressurizer pressure?

a. Incorrect combination of RCPs are running to achieve spray flow

b. High concentration of noncondensible gases in the pressurizer

c. Proportional heater controller has failed to maximum

d. Pressurizer spray bypass valve has failed closed

*ANSWER
b.
*REFERENCE
OP-OC-PNS-PZR, Revision 05, page 11; LSO 17
1992 Exam
(3.2/3.6) #70

e Lesson Plan OP-OC-PNS-RCS, Fig OP-OC-PNS-RCS-2

010000K603 [3.2/3.6]

QUESTION # 45

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**011000 A4.01**	
Importance Rating	**3.5**	**3.2**

Technical Reference(s): **PNS-PZR**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PZR OBJ. #2 & #3**

Question Source:
Bank # **PNS-307**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 45

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Power = 25%
- Pressurizer level is stable at 210 inches
- 1HP-120 (RC Volume Control) is in MANUAL

CURRENT CONDITIONS:

- Reactor power is slowly increased to 35%

Which ONE of the following is correct?

ASSUME NO OPERATOR ACTIONS and no Unit trip

If RC makeup and letdown is matched the pressurizer level will...

A. decrease and stabilize at a lower value.

B. increase and stabilize at a higher value.

C. initially decrease and then return to the original level.

D. initially increase and then return to the original level.

1 POINT

QUESTION # 45

011000A4.01 common rsi/gcw 05/07/00 1995 NRC #81

A. Correct, With constant makeup flow (not stated in question) from seal injection and constant letdown flow this is balanced. As power increases PZR outsurge occurs as Tc temperature decreases. PZR level will decrease and as long as Makeup and letdown is balanced PZR will stabilize at a lower value since 1HP-120 is in manual.

B. Incorrect, If power was decreasing then this would be correct as power decreases PZR insure occurs as Tc temperature increases. PZR level will increase and as long as Makeup and letdown is balanced PZR will stabilize at a higher value.

C. Incorrect, With constant makeup flow (not stated in question) from seal injection and constant letdown flow this is balanced. As power increases PZR outsurge occurs as Tc temperature decreases. PZR level will decrease and as long as Makeup and letdown is balanced PZR will stabilize at a lower value. Level would return to setpoint if 1HP-120 was in automatic and not in manual.

D. Incorrect, If power was decreasing then this would be correct as power decreases PZR insurge occurs as Tc temperature increases. PZR level will increase and as long as Makeup and letdown is balanced PZR will stabilize at a higher value. Level would return to setpoint if 1HP-120 was in automatic and not in manual.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

Upon completion of this lesson, the student will demonstrate an understanding of the components, indications, controls and operation of the Pressurizer. The student will be able to assess the status of the Pressurizer during normal, abnormal and emergency conditions and determine corrective actions for improper system operation. The student will also be able to apply any ITS/SLC Conditions and Required Actions associated with the Pressurizer.(T1)

ENABLING OBJECTIVES

1. Explain the design basis of the pressurizer. (R21)
2. Describe pressurizer response during load or RCS temperature changes. (R1)(R2)(R3)
3. Given a set of conditions, calculate the change in pressurizer level for a change in RCS temperature. (R33)
4. Explain what is meant by a "subcooled" pressurizer and how to determine if the pressurizer is in a subcooled condition.(R22)(R27)
5. Explain what is meant by a pressurizer "hard bubble" and describe the adverse effects of a "hard bubble" on plant operation, (R23)
6. Identify the source of pressurizer spray for each unit. (R4)
7. Discuss the automatic setpoints and any interlocks associated with pressurizer instrumentation. (R5)
8. Explain the operation of the ICS RC pressure signal median select function as it relates to RC pressure control including: (R28)
   - 8.1 How median select chooses the controlling signal
   - 8.2 Which pressurizer components receive a median selected RC pressure signal.
9. Given a set of conditions, determine which RC pressure signal has been selected for control by the ICS RC pressure signal median select function. (R36)
10. Discuss the reasons for bypass flow around the pressurizer spray valve during normal operation. (R6)
11. Evaluate plant response to a failed open pressurizer spray valve without operator action. (R20)

2. Presentation

2.1 Purpose

The pressurizer provides reactor coolant system pressure control during operation and limits pressure transients by accommodating RCS volume changes.

2.2 Principles of Pressurizer Operation

A. Insurge

1. An insurge (pressurizer water level increase) occurs due to an increase in RCS volume. As level increases, system pressure increases due to the compression of the steam bubble in the pressurizer. Some of the steam volume condenses, helping to relieve system pressure. Water is sprayed into the steam space to condense steam and reduce the pressure.

   a) Pressurizer level changes approximately 7 inches for each 1 degree of RCS temperature change. (Assumes HP-120 in manual).

   b) Pressurizer (and RCS) volume changes approximately 24 gallons per inch of pressurizer level.

2. An insurge is usually the result of coolant expansion due to an RCS heatup (i.e.... MT trip, loss of MFW).

3. An insurge also occurs during a power reduction when greater ≥15% power. RCS hot leg temperature will decrease, but cold leg temperature will increase. Since there is more total cold leg volume than hot leg volume in the RCS, the overall coolant volume will increase resulting in an insurge. Relatively slow insurges will be masked by the action of the pressurizer level controller to decrease RC makeup to the RCS.

4. More rapid insurges can occur due to an upset in the primary/secondary system heat balance, (i.e., loss of feedwater) that can result in the rapid expansion of RCS volume.

B. Outsurge

1. An outsurge (pressurizer water level decrease) occurs when RCS volume is reduced. As the pressurizer level decreases, the steam bubble expands and RCS pressure decreases. As RCS pressure decreases, some of the pressurizer water flashes to steam, which assists in maintaining the existing pressure. Heaters are then actuated to restore the normal operating pressure.

2. An outsurge occurs during an RCS cooldown due to coolant contraction (i.e. load increase, overcooling).

Exam Question Report

27-Jan-99

Question

KA: 011000A4.01

The following conditons exist:

- Reactor power at 25%
- Pressurizer level is stable at 210 inches
- HP-120 (RC Volume Control) is in MANUAL

Which ONE of the following describes the response of pressurizer level for slowly raising Reactor power to 50%? (ASSUME NO OPERATOR ACTIONS and no Unit trip) (.25)

Pressurizer level will...

A) decrease and stabilize at a lower value.

B) increase and stabilize at a higher value.

C) initially decrease and then return to the original level.

D) initially increase and then return to the original level.

Answer

A

A. Correct, with constant makeup flow (not stated in question) from seal injection and constant letdown flow this is balanced. As power increases PZR outsurge occurs as Tc gets colder and contracts; PZR level will decrease and as long as Makeup and letdown is balanced PZR will stablize at a lower value.

Lessons

ID	Description
PNS-HPI	High Pressure Injection System PNS-HPI

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag
NRC #81	Reference created by conversion		

KA'S

Exam Question Report

27-Jan-99

Question ID:	PNS307	Revision No:	0	Revision Date	10/29/1999
Question Description:	PNS307				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	PNS-HPI - High Pressure Injection System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 4; SRO = 4 Reference: NRC #81 1995	Max. Point Value:	0.25
		Passing Point Value:	0.25

QUESTION # 46

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**2**	**2**
	K/A #	**012000 K6.03**	
	Importance Rating	**3.1**	**3.5**

Technical Reference(s): **IC-RPS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-RPS #6**

Question Source:
Bank # **IC-141**
Modified Bank # _______
New _______

Question History:
Previous NRC Exam _______
Previous Quiz / Test _______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 46

Unit 1 plant conditions:

- Reactor power = 100%
- 1NI-5 has failed low
- Operators have completed all required actions per PT/600/01, Periodic Surveillance Requirements, to place the "A" RPS Channel in Manual Bypass

Which ONE of the following is the RPS **trip logic** for Low RCS Pressure?

A. two (2) out of three (3)

B. one (1) out of three (3)

C. two (2) out of four (4)

D. one (1) out of four (4)

1 POINT

QUESTION # 46

012000K6.03 common rsi/gcw 05/07/00 Bank IC-141
(1)

A. Correct - The RPS will initiate a reactor trip if two of the four RPS channels trip; this constitutes a two-out-of four logic. If the automatic trip functions of one channel are bypassed, two RPS channels are still required to actuate a reactor trip, but only three channels are left available to do this. So, the logic with one channel in Manual Bypass becomes two-out-of three.

B. Incorrect - With 1 Channel in Manual Bypass then 3 channels are remaining in the trip logic. If the Channel were placed in trip instead of bypass then this would be correct.

C. Incorrect – This is the normal operating mode of RPS without any channels in bypass or tripped. If all channels were operable then this would be correct.

D. Incorrect – If the channel was in the trip state instead of bypass this would be correct if all 4 channels were operable.

1 POINT

QUESTION # 46

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Power = 100%
- ~~Four (4) operable RPS channels~~, NI (5) Failed downscale

CURRENT CONDITIONS: operators and I&C have completed all compensatory measures IAW . . . .

- ~~RPS Channel "A" is placed in Manual Bypass~~ (and placed)

Which ONE of the following is correct?

The RPS trip logic ~~is~~ for Low S/G Level is . . .

A. two (2) out of three (3).

B. one (1) out of three (3).

C. two (2) out of four (4).

D. one (1) out of four (4).

1 POINT

QUESTION # 46

012000K6.03 common rsi/gcw 05/07/00 Bank IC-141
(1)

A. Correct - The RPS will initiate a reactor trip if two of the four RPS channels trip; this constitutes a two-out-of four logic. If the automatic trip functions of one channel are bypassed, two RPS channels are still required to actuate a reactor trip, but only three channels are left available to do this. So, the logic with one channel in Manual Bypass becomes two-out-of three.

B. Incorrect - With 1 channel in Manual Bypass then 3 channels are remaining in the trip logic. If the Channel were placed in trip instead of bypass then this would be correct.

C. Incorrect – This is the normal operating mode of RPS without any channels in bypass or tripped. If all channels were operable then this would be correct.

D. Incorrect – If the channel was in the trip state instead of bypass this would be correct if all 4 channels were operable.

5.4 What administratively controlled action is required when SD Bypass is selected, and the basis for that action.

5.5 The basic operation required to place each RPS channel in SD Bypass, and the indications that alert the operator that a channel is in SD Bypass.

5.6 The consequence of selecting SD Bypass during full power operation.

6. Explain the following concerning the Manual Bypass (channel trip bypass) function in RPS: (R6)

6.1 The effect on an RPS channel of placing that channel in Manual Bypass.

6.2 The meaning of two-out-of four and two-out-of three logic in RPS.

6.3 When Manual Bypass is used.

6.4 The basic operation required to place an RPS channel in Manual Bypass, and the indications that alert the operator that a channel is in Manual Bypass.

6.5 What administrative limit is imposed on the use of Manual Bypass, and what safeguards are used to insure compliance with this limit.

7. Explain the following relative to a bistable: (R7)

7.1 basic electronic operation

7.2 two basic functions bistables serve in RPS

7.3 function and/or operation of each operator-related indication and control on a bistable module

8. Explain the following relative to a STAR Module (R29)

8.1 Inputs

8.2 Outputs

8.3 Normal operation

8.4 Trip conditions; indications when tripped and methods used to manually trip.

9. Explain the following relative to a dummy bistable: (R8)

9.1 purpose

9.2 restrictions placed upon use of dummy bistables and how those restrictions are enforced

9.3 how a dummy bistable is distinguished from a normal bistable module, and how the operator is alerted to the use of a dummy bistable in an RPS channel.

f) The setpoint of ≤1720 psig is selected for the new high pressure trip so that the plant must first be shutdown, using normal procedures, before S/D Bypass can be initiated; 1720 psig is below the normal low pressure trip of 1800 psig, so that the plant must first be maneuvered past the normal low pressure trip point before going to S/D bypass. 1710 psig is the actual setpoint used for conservatism.

g) Selecting S/D Bypass at full power will result in a trip of the associated RPS channel on high RCS pressure.

h) By administrative procedure, the high flux trip set points are manually reset to approximately 4% when in S/D bypass. 4% is below the Improved Tech Spec requirement of less than or equal to 5% when shutdown.

   1) While the normal high flux trip of ≤105.5% power is not electrically bypassed it is basically nonfunctional because RPS will trip before the setpoint can be reached.

i) Resetting the high flux trip to this value prevents any significant power from being produced when performing zero power physics testing. Sufficient natural circulation flow would be available to remove up to 5% of rated power if no RCPs were operating.

2. Manual Bypass (Channel Trip Bypass)

   a) A Manual Bypass key switch located in each RPS channel Cabinet (A2, B2, C2, and D2) on the Reactor Trip Module, bypasses all automatic trip functions associated with that channel. **(OC-IC-RPS-9)**

   b) As will be discussed in a later section of this lesson plan, the RPS will initiate a reactor trip if two of the four RPS channels trip; this constitutes a two-out-of four logic. If the automatic trip functions of one channel are bypassed, two RPS channels are still required to actuate a reactor trip, but only three channels are left available to do this. So, the logic with one channel in Manual Bypass becomes two-out-of three.

   c) Manual Bypass is used, normally, for testing individual RPS channels while the plant is operating (so that the likelihood of inadvertent reactor trip is reduced); but it can also be used to bypass an inoperable channel due to a component failure in that channel.

Exam Question Report

27-Jan-99

Question

KA: 012000K6.03

Unit #1 is operating at 100% power with four (4) operable RPS channels. RPS Channel "A" is placed in Manual Bypass. Which ONE of the following describes the trip logic on Unit #1, after RPS Channel "A" is placed in Manual Bypass? (.25)

A) Two (2) out of three (3).

B) One (1) out of three (3).

C) Two (2) out of four (4).

D) One (1) out of four (4).

Answer

A

Lessons

ID	Description
IC-RPS	Reactor Protective System IC-RPS

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

Exam Question Report

27-Jan-99

Question ID:	IC141	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC141				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-RPS - Reactor Protective System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 6; SRO = 6	Max. Point Value:	0.25
		Passing Point Value:	0.25

QUESTION # 47

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**2**	**2**
	K/A #	**016000 K1.01**	
	Importance Rating	**3.4**	**3.4**

Technical Reference(s): **IC-RCI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-RCI OBJ. #7**

Question Source:
- Bank # **IC-315**
- Modified Bank # ______
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test __X__

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis __X__

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 47

Unit 1 plant conditions:

- Power = 100% power.
- Loop "A" RC flow is 99.8% of normal flow.
- Loop "B" RC flow is 99.5% of normal flow.

A significant leak develops on the LOW PRESSURE side of the loop "B" RC flow instrument header resulting in a loop flow change of 5.5%.

Which ONE of the following Tave signals is being used by the ICS after this event?

A. Average of selected Th and Tc from Loop "A".

B. Average of selected Th and Tc from Loop "B".

C. Average of Loops "A" and "B" Tave.

D. Highest Tave from either LOOP "A" or "B".

1 POINT

QUESTION # 47

016000 K1.01 Both PRA 04-11-00 (bank question Exam #2, IC-315)

A. Incorrect – Tave does not swap to “A” loop because indicated “B” Loop flow has not decreased below the controlling Tave transfer value.

B. Incorrect – Tave does not swap to “B” loop because indicated “A” Loop flow has not decreased below the controlling Tave transfer value.

C. Correct – ICS continues to use the Unit Tave because the “B” loop flow has failed high due to a leak on the low pressure side of the instrument.

D. Incorrect – The Tave swap doesn’t select based on high or low loop Tave (the swap is based on FLOW)

4. During all modes of operation, analyze proper operation of "Dixon" meters and differentiate between a loss of power, overranged, and underranged instrument (R21)

5. Given a set of conditions describe the required operator actions when selecting an alternate controlling signal. (R20)

6. Applying the knowledge of simplified instrumentation drawings be able to determine how various indications and control functions are processed for RCS temperature, pressure, level and flow including: (R2, 3, 62)

    6.1 Range of the indicator

    6.2 Source of the signal

7. Given a set of conditions analyze proper operation of RCS TEMPERATURE indications that the operator uses to monitor and control unit operation including the following: (R3, 4, 5, 6, 10)

    7.1 RCS T-hot

    7.2 RCS T-cold

    7.3 Core exit temperature (CETCs)

    7.4 Pressurizer temperature

8. Given a set of conditions analyze proper operation of RCS PRESSURE indications that the operator uses to monitor and control unit operation including the following: (R6, 7, 9, 63, 10)

    8.1 RCS Loops

    8.2 ICCM WR Pressure

    8.3 Low Range Cooldown

9. Given a set of conditions analyze proper operation of RCS LEVEL indications that the operator uses to monitor and control unit operation including the following: (R13, 15, 16, 17, 18)

    9.1 Pressurizer level and pressure

    9.2 Reactor Vessel (LT-5)

    9.3 Ultrasonic Level Indication (ULI)

    9.4 Tygon tubing

c) Ultrasonic Level Measurement (ULI)

1) Consists of:

(a) Ultrasonic transducer mounted on the bottom of pipe.

(b) Signal processing circuits

D. Flow Detectors

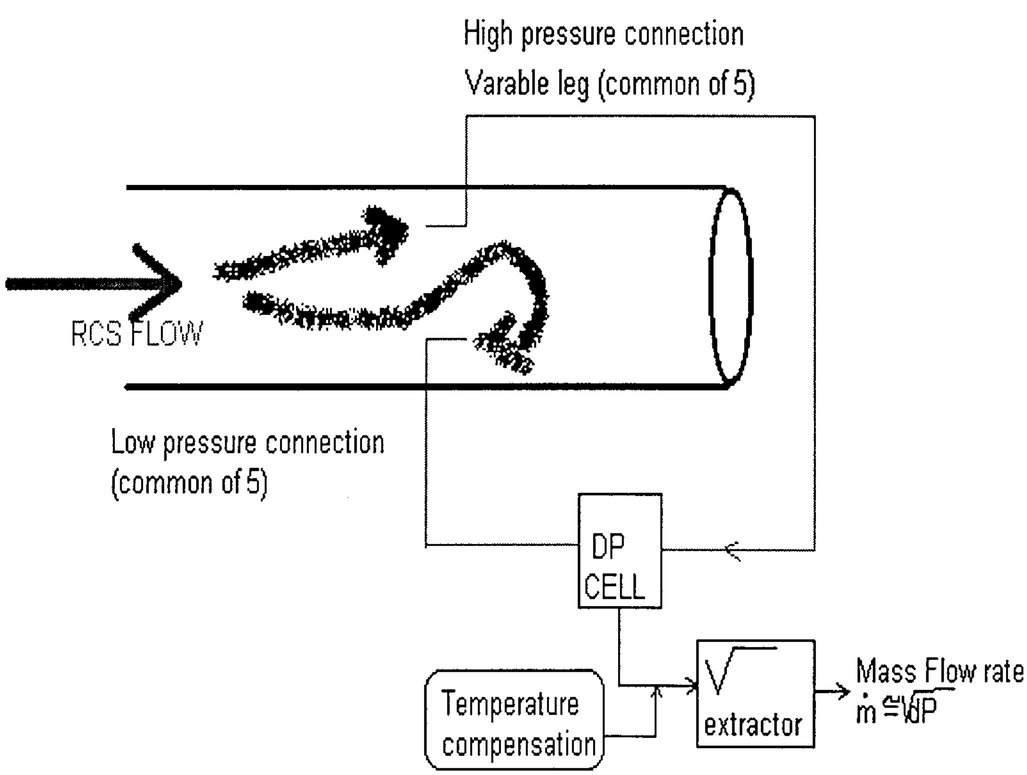
1. Type

a) Differential Pressure (ΔP) Transmitter (ROSEMOUNT)

b) Operating Principle

1) Similar to that for the ΔP transmitter used for level indication.

2) The variable leg is connected to the high pressure header of a Gentilli Tube (facing into the flow path) and the reference leg is connected to the low pressure header of the Gentilli Tube (facing the same direction of the flow path).



c) Gentilli Tube consists of 5 pairs of tubes, (one for each RPS Channel A-E) projecting into the RCS piping. Each pair consists of one tube pointing in the direction of the flow and one tube pointing against the flow. The five tubes pointing in the direction against the flow are joined outside the RCS piping to form one common high pressure header and the five tubes pointing in the direction of the flow are joined together outside the RCS piping to form one common low pressure header. Five (5) separate ΔP transmitters are used to provide RC flow parameters for unit operation.

d) With no flow, the high pressure header equals the low pressure header.

1) Zero ΔP developed

2) Zero flow indicated

e) When flow is developed the pressure in the high pressure header will increase.

1) A ΔP will develop.

2) The ΔP developed is proportional to the square of the volumetric flow rate (gpm).

3) An increase in flow will be shown.

4) The opposite is true for a decrease in flow.

f) Failure modes: If the low side input fails then ΔP will increase. If high side fails then ΔP will decrerase.

2. Volumetric flow rate verse Mass flow rate:

a) Volume flow rate is indicated in gpm (gallons = volume)

b) Gallons of water is not affected by a change in temperature. A gallon of water at 70°F is the same measured gallon of water at 600°F. Volumetric flow rate does not change significantly as temperature of the fluid changes. Volume flow rate cannot be used in a heat balance equation for this reason.

c) In a heat balance equation or heat balanced system such as the ICS, mass flow rate has to be used because the RCS properties changes as temperature is changed during different modes of unit operation. The function of ICS is to balance heat generation (RCS) and heat removal (Steam) via heat balances thus the need to use mass flow rate as an indication/input for ICS operation is necessary.

5) ICS "CONTROLLING" Tave digital meter - green LED

(a) RC Tave is calculated for loop A and loop B using the respective loop Thot and Tcold temperatures. The loop A and loop B Tave signals are averaged to calculate the Unit Tave. Unit Tave provides an overall indication of both operating loops and provides an accurate input for ICS Tave and control rod control.

(b) ICS Controlling Tave function:

(1) If RC flow is normal (70E6 lbm/hr) in both loops, then the UNIT Tave is output for ICS Tave control. If RC Loop flow is <u>low</u> (<62E6 lbm/hr*) in one loop, the opposite loop Tave is output for ICS Controlling Tave. If BOTH Loop RCS Flow is < 62E6 lbm/hr then Unit Tave is selected. This system provides an accurate Tave signal to ICS during 3 RCP operation and any other RC flow mismatch conditions.

(2) The Loop RC Flow LOW statalarm setpoint is 62.5E6 lbm/hr. This is a good indication or tool for the operator to use. If the LOW flow alarm has actuated then the Tave control for ICS has probably swapped to the loop with the higher RC flow.

(3) If either OTSG is on LLL then the highest Tave is selected.

(c) NR recorder and Controlling Tave LED. (520 - 620°F)

4. Wide Range T-cold (50 - 650°F)

a) WR Supplies Tc indication to the RCP starting interlock for the 4th (LAST) RCP started during a startup.

1) The lowest A1 and A2 WR Tc indication supplies starting permissive for the A1 and A2 RCPs

2) The lowest B1 and B2 WR Tc indication supplies starting permissive for the B1 and B2 RCPs

5. Pressurizer (RTDs) Temperature Dixon meter - (0-700°F)

a) PZR temperature "A" and "B" (ICCM). RTDs used to temperature compensate Pressurizer level signals. RTD "A" provides temperature for PZR saturation temperature calculation and indication (ICCM but not PAM).

Exact Question Report

27-Jan-99

Question ID:	IC315	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC315				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-RCI - RCS Instrumentation
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 75; SRO = 75 Reference: IC RCI OBJ 1, 32	Max. Point Value:	0.25
		Passing Point Value:	0.25

27-Jan-99

Question

The following initial conditions exist:

- Unit #1 is operating at 100% power.
- Loop "A" RC flow is 99.8% of normal flow.
- Loop "B" RC flow is 99.5% of normal flow.

A significant leak develops on the LOW PRESSURE side of the loop "B" RC flow instrument header resulting in a loop flow change of 5.5%

Which ONE of the following Tave signals is being used by the ICS after this event? (.25)

A) Average of selected Th and Tc from Loop "A".

B) Average of selected Th and Tc from Loop "B".

C) Average of Loops "A" and "B" Tave.

D) HIGHEST Tave from either Loop "A" or "B".

Answer

C
A: incorrect - Tave does not swap to "A" loop
B: incorrect - Tave does not swap to "B" loop
C: correct - ICS continues to use the Unit Tave because the "B" loop flow has failed high due to a leak in the low pressure side of the instrument.
D: incorrect - the Tave swap doesn't select based on high or low loop Taves (the swap is based on FLOW).

Lessons

ID	Description
IC-RCI	Reactor Coolant System Instrumentation

QUESTION # 48

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**029000 A3.01**	
Importance Rating	**2.9**	**3.1**

Technical Reference(s): **RAD-RIA**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **RAD-RIA OBJ. #5**

Question Source:
Bank # **RAD-57**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 48

Plant conditions:

INITIAL CONDITIONS:

- RB Purge in progress

CURRENT CONDITIONS:

- 1RIA-45, Unit Vent Gas Monitor is **operable** and reads 0 cpm
- 1RIA-46, Unit Vent High Gas Monitor is in ALERT

Which ONE of the following describes the status of the RB Purge System?

A. RB Purge System has been automatically isolated.

B. Manual isolation of the RB Purge System is required.

C. Normal purge in progress and no operator action required.

D. RB Purge exhaust fan is tripped but 1PR-2 through 1PR-5 will not close until 1RIA-46 reaches the high alarm.

1 POINT

QUESTION # 48

029000A3.01 common rsi/gcw 05/07/00 Bank RAD-57

QUESTION SETUP: RIA-46 provides backup for RIA-45 and has the same interlock features when the HIGH alarm is reached. RIA-45 indicating 0 cps indicates that it has increased to the switchover acceptance range and has also exceeded its HIGH setpoint and isolated the in progress purge.

A. Correct - Both RIAs perform the same interlock functions. 1RIA-46 operates at the Switchover Acceptance Range Setpoint, which also provides the interlock feature.

B. Incorrect – Manual isolation would be required if both RIAs did not share the same interlock.

C. Incorrect - The interlock functions will isolate the purge. Normal purge should not remain in progress.

D. Incorrect – The purge fan will trip and isolation valves PR-2 through 5 will close upon actuation of the interlock signal from either RIA.

1 POINT

QUESTION # 48

Which ONE of the following statements is correct regarding the operation of 1RIA-45, Unit Vent Normal Gas Monitor and 1RIA-46, Unit Vent High Gas Monitor?

1RIA-45 and 1RIA-46...

A. ~~both~~ are always on scale. ~~at the same time~~

B. alert alarms will isolate the RB Purge System. T/F

C. cause the same interlock functions to occur.

D. utilize switchover acceptance range setpoint.

initial conditions • RB purge in progress

current conditions . 1RIA-45 Reads 0
1RIA-46 is in Alert

which 1 of the following describes the status of the RB purge system.

a. Normal purge in progress and no operator action required

b. Purge system is isolated

c. Purge exhaust fans tripped but PR 2.5 Remains open until 1RIA-46 High Alarm received

d. manual isolation of purge system is required

1 POINT

QUESTION # 48

029000A3.01 common rsi/gcw 05/07/00 Bank RAD-57

A. Incorrect - 1RIA-45 and 46 are not both on scale at the same time.

B. Incorrect - The interlock functions actuate on the High not the Alert setpoint.

C. Correct - Both RIAs perform the same interlock functions.

D. Incorrect - Only 1RIA-46 operates at the Switchover Acceptance Range Setpoint.

4.1 Scintillation

4.2 Geiger Mueller

4.3 Ionization

5. Describe the basic function of each applicable monitor. (R2)

   5.1 State the purpose of each monitor.

   5.2 State where each monitor is located.

   5.3 List the interlocks and automatic actions associated with each applicable monitor.

   5.4 When given the monitor title, be able to state which system(s) the monitor checks.

6. List seven (7) functions which can be performed at the new RIA Control Room CRT. (R8)

7. Describe the basic procedure to check/set High and Alert alarm setpoints. (R5)

8. Describe the operational relationship between the following components associated with the Sorrento Radiation Monitoring System: (R10)

   8.1 RM-80 Microprocessor Unit

   8.2 Transient Monitor System Computer

   8.3 View Node

9. For the following situations, state whether or not the associated Radiation Monitor is operational and explain why for each case: (R11)

5. Detects potential leaks in SF coolers, primary sample coolers and seal return coolers.

I. (1)(3)RIA-42 - Monitors RCW return from auxiliary building.

1. Detector is sodium iodide.
2. Located in turbine building basement behind backwash pumps.
3. A pump on the skid ensures sufficient sample flow.
4. RIA-42 is basically the same as RIAs-31, 35 and 50 except that they have slightly different pumps.

J. (1)(2)(3)RIA-43, 44, 45, 46 - Unit Vent Monitors

1. Unit vent monitors

1. Particulate (RIA-43), Iodine (RIA-44), Normal gas RIA-45), High Gas (RIA-46) "PIGG"
2. RIAs-43 & 45 are plastic beta scintillation detectors.
3. RIA-44 is a NaI scintillation detector.
4. RIA-46 is a Cadmium Telluride solid state detector
5. Located on 6th floor Auxiliary Building in the Purge Equipment room close to the Unit Vent Stack.
6. Interlocks associated with the Unit vent monitors
7. On HIGH alarm, RIA-45 will do the following:
   a) close PR-2 through PR-5
   b) trip the main and mini purges
   c) actutes statalarm "RM Reactor BLDG Purge Disch RAD Inhibit"
8. When RIA-46 reaches the "switchover acceptance range setpoint", the following occurs:
   a) RIA-45 will read zero
9. RIA-46 will now perform the same interlock functions that RIA-45 performed
10. This provides a backup function so that in case of a failure of RIA-45 HIGH alarm, then RIA-46 HIGH alarm will actuate the required interlock functions. Normally RIA-45 HIGH alarm setpoint will be reached prior to RIA-46 reaching the "switchover accceptance range setpoint".

K. Reactor Building Monitors: (1)(2)(3)RIA-47, 48, 49, 49A

1. Particulate (RIA-47), Iodine (RIA-48), Normal gas (RIA-49), and High gas (RIA-49A)

Exact Question Report

27-Jan-99

Question

KA: 029000A3.01

Which ONE of the following statements is correct regarding the operation of 1RIA-45, Unit Vent Normal Gas Monitor and 1RIA-46, Unit Vent High Gas Monitor? (.25)

1RIA-45 and 1RIA-46...

A) actuate at different alert and high setpoints.

B) alert alarm will isolate the RB Purge System.

C) cause the same interlock functions to occur.

D) utilize switchover acceptance range setpoint.

Answer

C

A. Incorrect. 1RIA-45 and 46 actuate at the same Alert and High Setpoints.
B. Incorrect. The interlock functions actuate on the High not the Alert setpoint.
C. Correct. Both RIAs perform the same interlock functions.
D. Incorrect. Only 1RIA-46 operates at the Switchover Acceptance Range Setpoint.

Lessons

ID	Description
RAD-RIA	Radiation Indicating Alarms RAD-RIA

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

Exam Question Report

27-Jan-99

Question ID:	RAD057	Revision No:	0	Revision Date	10/29/1999
Question Description:	RAD057				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	RAD-RIA - Radiation Indicator Alarms
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 2; SRO = 2 Reference: PT/0/A/230/01	Max. Point Value:	0.25
		Passing Point Value:	0.25

QUESTION # 49

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**033000**	**G2.1.25**
Importance Rating	**2.8**	**3.1**

Technical Reference(s): **OP/3/1104/06E**
Encl. #4.1 & #4.2

Proposed references to be provided to applicants during examination: **OP/3/1104/06E**
Encl. #4.2

Learning Objective: **FH-SFC OBJ. #16**

Question Source:
Bank # ______
Modified Bank # **ADM-759**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ______
55.43 __X__

Comments:

1 POINT

QUESTION # 49

Unit 3 plant conditions:

- SFP temperature = 118 degrees F
- Time after shutdown is 73 days
- CETCs = 98 degrees F
- RV Head has been installed following a Refueling outage
- A loss of Spent Fuel Pool cooling has just occurred

SEE ATTACHMENT

Which ONE of the following is the estimated time to boiling in the SFP?

A. 128 hours

B. 35 hours – OK Based on using 98 F°

C. 30 hours

D. 28 hours

1 POINT

QUESTION # 49

033000G2.1.25 common rsi/gcw 05/07/00 Bank ADM759 (modified)

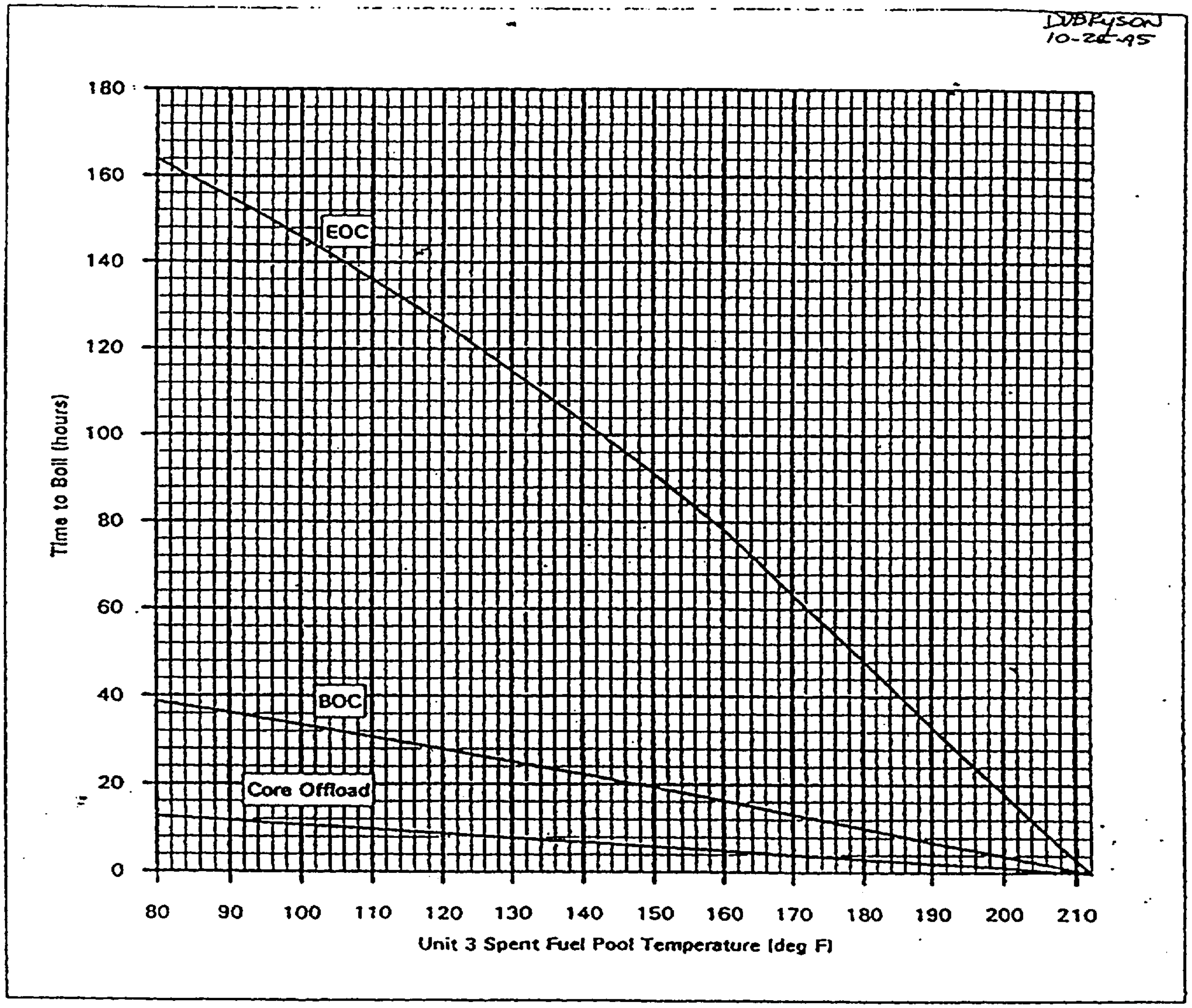
A. Incorrect, This distracter is based on EOC curve.

B. ~~Incorrect, This distracter is based on BOC curve for a similar question in the exam bank~~ 98 F°

C. Incorrect, This distracter is based on misinterpretation of BOC curve. (ie misreading that the horizontal lines are in 5 hour increments).

D. Correct, The vertical SFP temperature (118 degrees F) intersects the BOC curve at approximately the 28-hour horizontal line. (The horizontal lines are in 4-hour increments).

Information Use



Note: Graph assumes SFP water level at -2.0' when SF Cooling is lost and SSF is required.

Curve	Condition
Core Offload	Complete core offloaded to SFP
BOC	After core loading (≈ 1/3 core added to SFP inventory) and between refuelings
EOC	After unit has shutdown for refueling, but prior to core offload.

Continuous Use

1. Initial Conditions

_____ 1.1 SFP temperature increasing.

_____ 1.2 Review Limits and Precautions.

2. Procedure

_____ 2.1 Estimate time to SFP boiling per Enclosure "U3 SFP Time To Boil After Loss Of SF Cooling".

_____ 2.2 Notify RP to check for SFP dose rates **AND** air-borne radioactivity.

_____ 2.3 Verify Fuel Receiving Area roll-up door closed. (continue with procedure)

NOTE:	SF Pumps interlocked to trip at -2.5'.

_____ 2.4 Verify SFP level between 0.7' and -0.7'.

_____ 2.5 **IF** SFP level low, begin SFP makeup to minimize boiling in SFP. {2}

_____ 2.6 **IF** SFP in purification, secure purification.

2.7 **IF** a leak degraded SF system, isolate leak **AND** restore cooling.

- _____ **IF** required, refer to AP/3/A/1700/009 (Spent Fuel Damage).
- _____ **IF** required, initiate makeup to SFP. {2}

2.8 Verify SF System flow:

_____ 2.8.1 **IF** 3A **AND/OR** 3B SF Cooling Pump(s) operating, throttle 3SF-21 (Pool Coolant Supply Hdr Blk) as required. (A-2-SF Clr Rm, SW Corner)

- Verify 800-1000 gpm for single pump operation.
- Verify 1600-2000 gpm for two pump operation.
- Refer to CP 03A0783 (SF CLNG H20 FLOW) **OR** 3SF FT0017P.

_____ 2.8.2 **IF** 3C SF Cooling Pump operating, throttle 3SF-90 (SF Pump 'C' Disch) to obtain 900-1000 gpm. (A-2-SF Clr Rm)

- Refer to CP O3A2058 (SF CLR 3C FLOW) **OR** 3SF FT0175.

_____ 2.9 <u>**IF**</u> excessive vibration <u>**OR**</u> motor temperature noted, secure SF pump(s).

_____ 2.10 <u>**IF**</u> required, verify valve alignment per Enclosure "SF System Valve Checklist" in OP/3/A/1104/006 (SF Cooling).

_____ 2.11 <u>**IF**</u> SF pump cavitation occurs, vent SF pumps <u>**AND**</u> coolers.

_____ 2.11.1 Restart SF System.

_____ 2.12 <u>**IF**</u> SF pump breaker tripped, have I&E check out SF pump motor <u>**AND**</u> breaker.

_____ 2.13 <u>**IF**</u> required, start additional SF pumps.

_____ 2.14 Verify RCW flow adequate to in service SF coolers.

_____ 2.15 Verify SF Cooler Temperature Control Valve(s) on idle SF coolers adjusted to ≈ 120°F. {3}

ENABLING OBJECTIVES (continued)

10. Discuss the following related to the Limits and Precautions of OP/1&2/A/1104/06, Spent Fuel Cooling System : (R10)

    10.1 The precaution related to preventing BWST overflow.

    10.2 The reason why SF-4 must be shut if the SF Pool Level is less than zero inches.

    10.3 The reason for the special precaution related to valving the "cold" and "hot" fluids in and out very slowly for the "C" Spent Fuel Coolers.

    10.4 The reason for maintaining specified spent fuel pool level, boron concentration and temperature.

11. Briefly describe the Spent Fuel Cooling System flowpath including all major valves and components for the following modes of operation: (R11)

    11.1 Normal operation

    11.2 Refueling operation

    11.3 Testing of SFC Pumps

12. Given a copy of PT/1&2,3/A/0251/002, Spent Fuel Cooling Pump Test, and a set of data, determine if acceptance criteria is being met. (R21)

13. Explain why SF-90 should be throttled if the "C" SF Cooling System Pump is to be started. (R12)

14. Describe the effect of changes in Reactor Building pressure on Spent Fuel Pool Level. (R13)

15. State the purpose for the skimmer associated with the SF Pool. (R14)

16. Given a set of conditions and the appropriate enclosure(s), calculate the time required for boiling to occur in the SFP. (R16)

17. Describe the relationship between Spent Fuel Pool Level and Temperature, and SSF RCMU System operability. (R17)

18. Given a set of conditions and the appropriate enclosure(s), determine the required Spent Fuel Pool level. (R18)

19. Given a set of conditions and the appropriate enclosure(s), determine SSF RCMU System operability. (R19)

H. Recovery of Spent Fuel Cooling

1. This enclosure provides Operators with guidance on how to trouble shoot SF Cooling problems and return the system to normal operation. Recovery steps include:

   a) Estimating time to SF Pool boiling following loss of SF Cooling.

   b) Verifying SF Pool level between +0.7 and -2.0 ft.

   c) Isolating SF Cooling System leaks and initiating makeup to the SF Pool.

   d) Notifying RP of occurrence and verifying that FRA roll up door is closed

   e) Verifying adequate SF Cooling System flow.

2. An *Estimated Time To Units 1&2 SFP Boiling Following A Loss Of Spent Fuel Cooling* is provided as an aid to operators in determining time to SFP boiling following a loss of SF Cooling. This enclosure contains the following information:

   a) A graph of <u>initial</u> SFP temperature vs. Time to Boil (in hours) with curves for the three most probable scenarios, core off load, BOC and EOC.

   b) The core off load curve is the most restrictive curve (i.e., shortest time to pool boil for a given initial temperature. This curve is based on the condition where one unit's complete core has been off loaded to the SFP. Of the three scenarios addressed, this one introduces the highest heat load to the SFP and, therefore, results in the shortest time to SFP boil.

   c) The BOC curve is the next most restrictive and is based on the discharge of approximately 1/3 of a core to the SFP <u>following refueling</u>. *This curve is also used if a loss of SF Cooling occurs between refueling outages.*

   d) The EOC curve is the least restrictive and is based on the scenario where a unit is shutdown for refueling, but has yet to unload fuel. Additional decay heat from the core's spent fuel has not yet been added to the SF Pool and the spent fuel in the pool has decayed for a considerable length of time.

   e) The graph is based on an initial SF Pool level of -2.0 ft. which is less than that allowed by procedures. Therefore, any estimated time to SF Pool boil will be conservative.

f) The graph is used as follows to calculate the estimated time to SF Pool boiling:

1) The current SF Pool temperature is determined.

2) The current plant condition is determined, i.e., core off load, BOC or EOC.

3) The *minimum* estimated time to SF Pool boil is determined from the intersection of the current SF Pool temperature line and the appropriate curve for the current plant condition.

g) This graph is also based on SSF RCMU System operability. The relationship between SF Pool level and temperature, and RCMU System operability is described in section 2.6 below.

I. Other Operations Involving the SF Cooling System (Refer to OP/1&2/1104/06):

1. Purification of SFP and BWST

2. Use of the SF Pool Skimmer to remove debris that may accumulate on the surface of the pool water.

a) SF Pool level ≥ 0 inches.

Note: If the level decreases to approximately the 0 inch level, air could enter the system causing cavitation of the SF Cooling Pumps.

b) The basic procedure is to position SF-4 ~ 50% open while monitoring SF Cooling Flow. Close SF-4 when skimming is no longer needed.

J. Abnormal SF Cooling System Operations

1. During an SSF event, the spent fuel pool supplies borated water to the SSF RC Make Up System for injection into the RCS via the RC Pump seal injection lines.

2. Following a loss of 4160 volt power and the BWST (i.e., tornado), the HPI System can be aligned to take a suction on the SFP.

a) Initially, the HPI System is aligned to take a suction on the common "A" and "B" SF pump suction line from the SFP.

b) When the level in the SFP has decreased to less than -2.5 feet, HPI pump suction is swapped to the SFP fill line.

1) The SFP fill line extends down farther into the SFP extending the available inventory.

2) A SFP fill line priming pump is provided to prime (i.e., setup siphon flow in) the SFP fill line prior to making this alignment.

Exam Question Report

27-Jan-99

Question

KA: 033000G1.33

Given the following Unit 3 conditions:

- SFP temperature is 112□
- SFP level is -1.9 ft
- Time of shutdown is 73 days
- CETCs = 98□F
- RV Head has been installed following a Refueling outage.
- A loss of Spent Fuel Pool cooling has just occurred.

Which ONE of the following is the amount of time before boiling is expected in the SFP? (.25)

A) 148 hours

B) 135 hours

C) 35 hours

D) 30 hours

Answer

D

Utilize SFP offload curve in OP/3/1104/06 encl. 3.23

A. Wrong. Based on EOC curve and CETC temp.

B. Wrong. Based on EOC curve

C. Wrong. Based on BOC curve and CETC temp.

D. Correct. Based on BOC curve and SFP temp.

Lessons

ID	Description
H2O HAMMER	Lesson Plan created by conversion

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

Exam Question Report

27-Jan-99

Question ID:	ADM759	Revision No:	0	Revision Date	10/29/1999
Question Description:	ADM759				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	WATER HAMM
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = R14; SRO = R14	Max. Point Value:	0.25
		Passing Point Value:	0.25

QUESTION # 50

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**2**	**2**
	K/A #	**035000 K1.09**	
	Importance Rating	**3.8**	**4.0**

Technical Reference(s): **PNS-RCS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **RCS OBJ. #14**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 50

Unit 2 plant conditions:

- 4 RCPs are operating
- Reactor power level = 27%
- Controlling Tave = 579°F

Which ONE of the following is correct?

SEE ATTACHMENT

RCS Th = __________ / Tc = __________.

A. 602 / 555

B. 582 / 575

C. 592 / 566

D. 585 / 573

1 POINT

QUESTION # 50

035000K 1.09 Both PRA 2-10-00

Question setup:
0-15% Tc, Th, and Tave ramped from 532° to Tave = 579° with a 7° ΔT
15 – 100% Tave is maintained constant at 579°F, Tc decreases 21°F (85% @ .25°/%), Th increases 21°F (85% @ .25°/%)

A. Incorrect - This is the 100% power values.

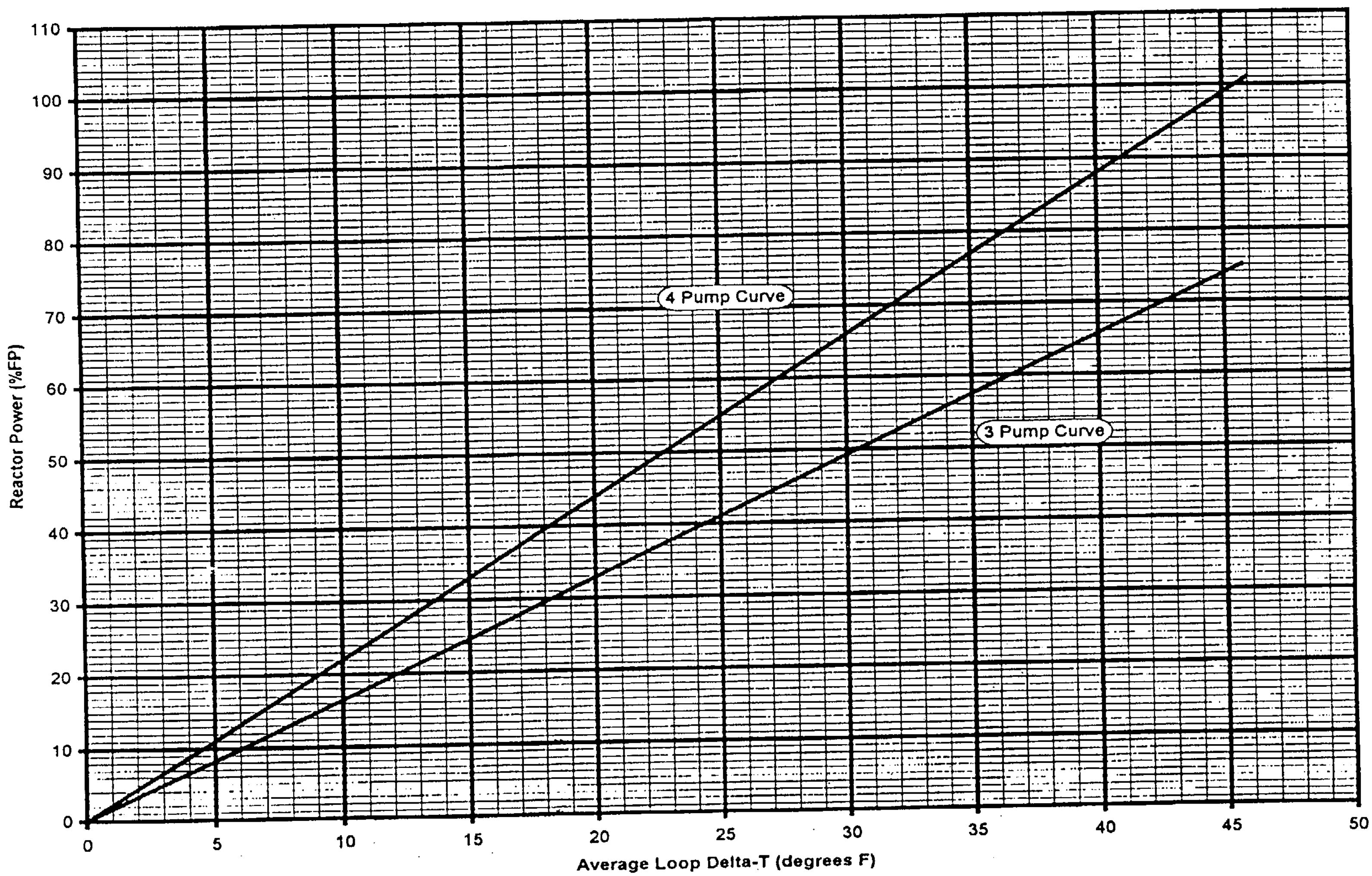
B. Incorrect - This is the 15% value.

C. Incorrect – Since reactor power = 27% and assuming a linear 1°F per 1% power then 579°F – 13.5°F = 566°F. 579°F + 13.5°F = 592°F

D. Correct - Th = 583°F+ (.25°/%x12%=3) = 586°F / Tc = 576°F - (.25°/%x12%=3) = 573°F.

Enclosure 13.12

Unit 2 Cycle 18
Loop ΔT Vs. Reactor Power



9. List the general location, on each Oconee unit RCS, for the following piping connections during normal and ECCS operation. (R9)

<u>NORMAL</u>	<u>ECCS</u>
A. Letdown Line	A. HPI
B. Pressurizer Spray Line	B. LPI
C. Decay Heat Drop Line	C. CFT
D. Pressurizer Surge Line	
E. Vent and Drain Lines	

10. List the three major process parameters monitored by RCS instrumentation. (R10)

11. State how RCS temperature is increased from cold shutdown to hot shutdown and describe how RCS system volume is controlled during the heatup. (R11)

12. State the normal RCS pressure and average temperature during full power operation. (R12)

13. State how RCS temperature is increased above hot shutdown. (R13)

14. By using a graph of RCS temperature (x-axis) versus reactor power (y-axis) describe how RCS hot leg, cold leg, and average temperature varies from 0% power to 100% power, including approximate values for each temperature at 0%, 15%, and 100% power. (R14)

15. State the location for the power supplies of the RCS high point vents. (R15)

16. Draw the Reactor Coolant System. Include the five major components that comprise the RCS and the nine (9) major groups of piping or penetrations that connect to the RCS. (R16)

17. Given a copy of ITS/SLCs and associated Bases, analyze a given set of conditions for applicable ITS/SLC LCOs. (R17)

18. Apply all ITS/SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R18)

19. Compute the maximum Completion Time allowed for all applicable Required Actions to ensure compliance with ITS/SLCs. (R19)

2.6 System Operations

A. Startup

1. Pressurization - Bubble formation and pressurization will be discussed in the Pressurizer Lesson Plan.

2. Heatup - Reactor Coolant System heatup from ambient temperature to 532°F is accomplished by using the heat generated by decay heat and the operation of the reactor coolant pumps. Above 532°F (Hot Shutdown), heatup is accomplished by the fission process.

   a) Service Water (LPSW) to the LPI coolers is throttled to decrease cooling water flow, this allows the RCS to heatup from decay heat energy. Pressurizer heater operation heat the inventory in the pressurizer, increasing pressure until RCS pressure is above NPSH for starting RCPs.

   b) After the RCPs are started the forced flow condition causes temperature to increase from friction heating called Pump Heat.

   c) Control Rods are withdrawn at 532°F to allow heat from the fission process to increase T_{ave} to 579°F and RC Pressure is approx. 2155 psi.

3. RCS volume is controlled by allowing the Pressurizer level to increase to the desired level, then maintaining the desired level via letdown section of the HPI system to the BHUT.

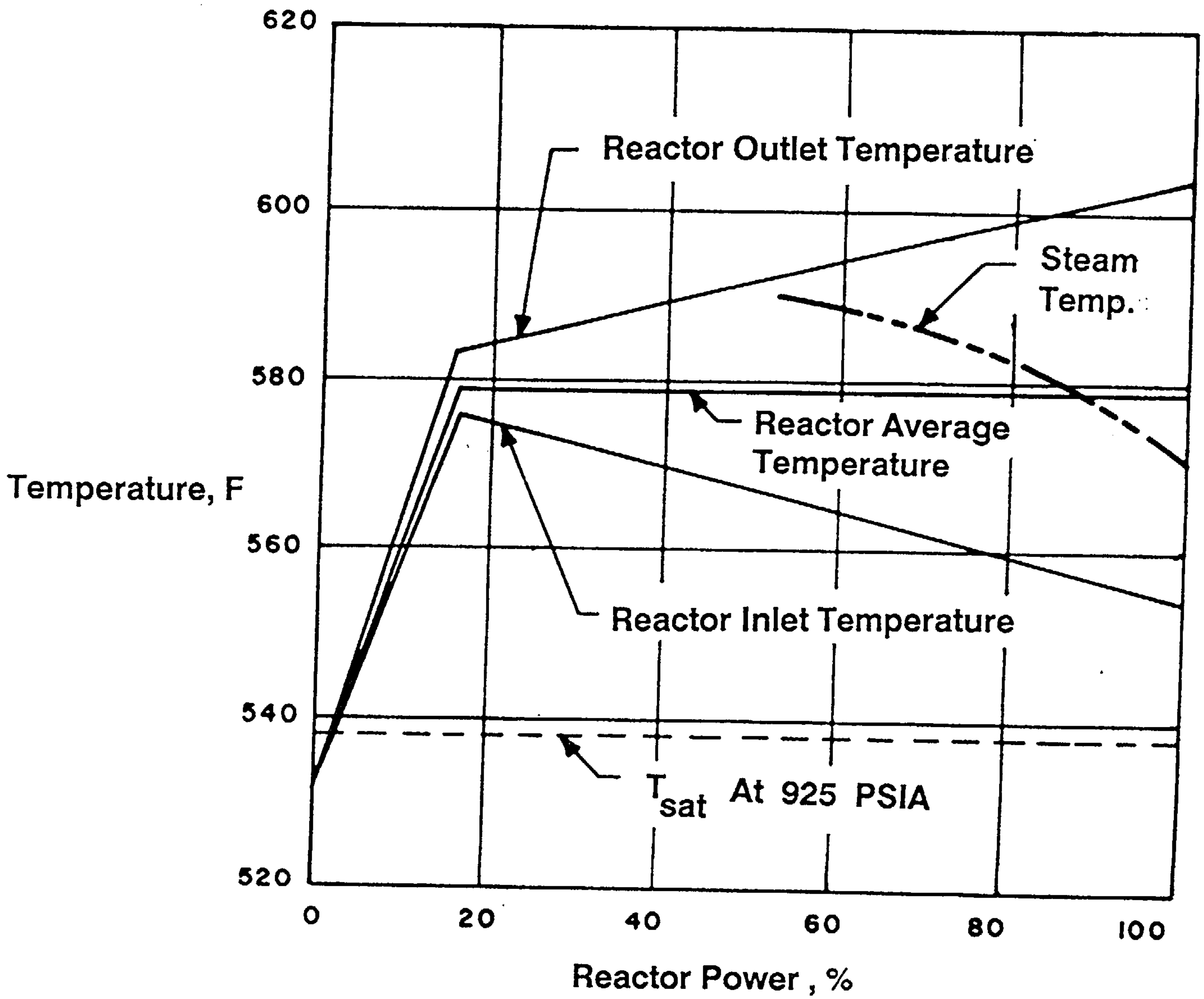
B. Power Operations - Refer to OC-PNS-RCS-5

1. Normal Parameters - 15% Power

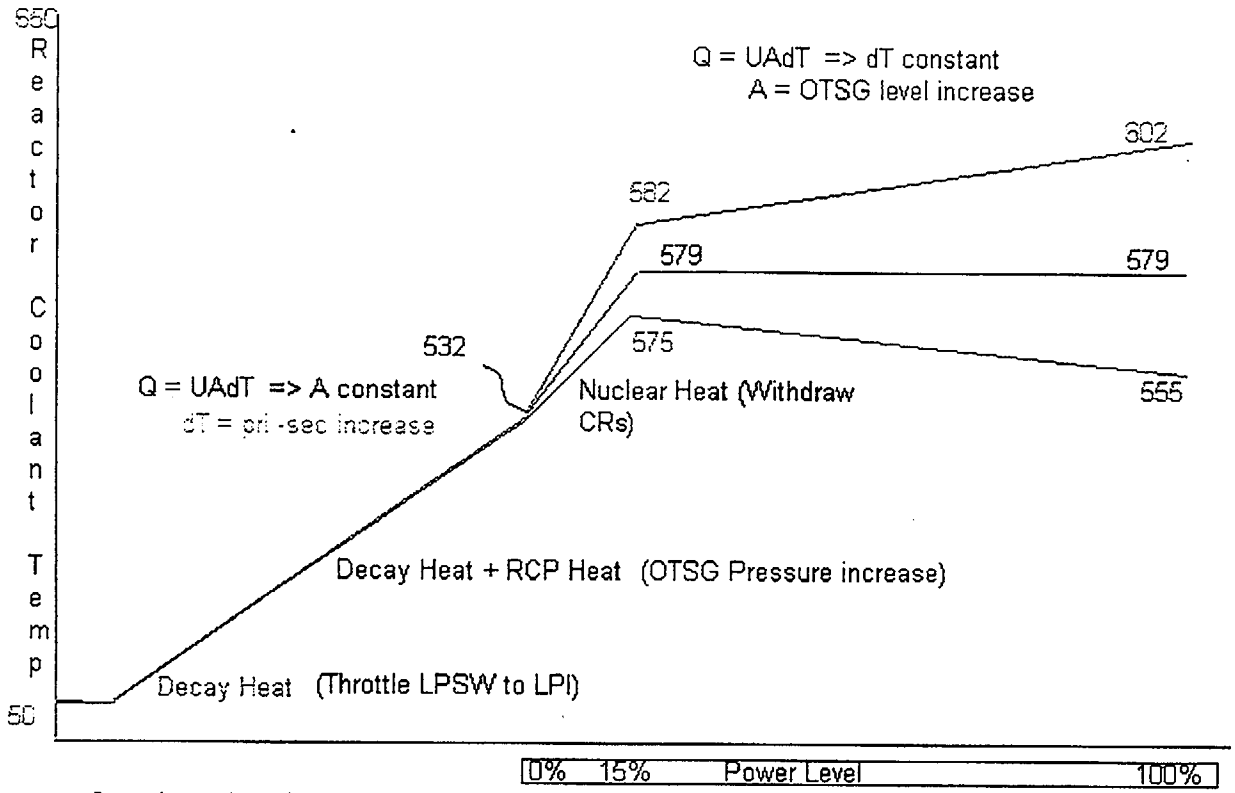
a)	Average Temperature	579°F
b)	Hot Leg Temperature	582.5°F
c)	Cold Leg Temperature	575.5°F
d)	Temperature Difference	7°F
e)	RCS Pressure	2155 psig

2. Normal Parameters - 100% Power

a)	Average Temperature	579°F
b)	Hot Leg Temperature	602.5°F
c)	Cold Leg Temperature	555°F
d)	Temperature Difference	47.5°F
e)	RCS Pressure	2155 psig



TITLE: REACTOR COOLANT SYSTEM	NOTES: Reactor and Steam Temperatures Versus Reactor Power	ID. NO: OC-PNS-RCS-5	DATE: 3-13-90
		REF. FSAR	
		DRN. BY: WC/ARB	APR. BY:
		MEDIA NO. OPNSRCS5	TRAINING USE ONLY



Core decay heat is controlled via the LPI coolers with LPSW used as cooling. Decay heat is rejected to the final heat sink, Lake Keowee.

RCP heat and decay heat is controlled via the OTSGs. As RCS temperature changes the OTSG pressure is controlled to control heat transfer rate from primary to secondary.

Nuclear heat is generated from the fission event and controlled by control rods and OTSG level control as OTSG steam pressure is held constant.

QUESTION # 51

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**035000 K6.01**	
Importance Rating	**3.2**	**3.6**

Technical Reference(s): **STG-MT**
STG-EHC

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **STG-MT T #2 & #3**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 51

Unit 2 plant conditions:

- Reactor power = 100%
- A Main turbine trip has occurred
- All four (4) Main Steam Stop Valves (MSSVs) CLOSE by actuating the TSV Closure signal in **<u>19 seconds</u>**

Which ONE of the following is correct?

The MSSVs are ______ because…

MSSVs

A. operable / all four TSV's are in the closed position.

within TS limits

B. operable / both of the 2 TSV's Closure Channels have actuated ~~as required.~~

C. inoperable / one of the two TSV's Closure Channels ~~has~~ closed all the MSSVs ~~as required.~~ ~~outside required time limits.~~
within TS limits

ID D. inoperable / neither of the two TSV's Closure Channels have closed all the MSSVs ~~as required.~~
~~Designed~~ within TS limits.

you stated they were closed in stem

1 POINT

QUESTION # 51

035000K6.01 COMMON PRA 05/09/00 NEW

A. Correct – TSV Closure Channel A is required to actuate the MSSV in ≤ 1 second(s) to close and Channel B is required to actuate MSSV in ≤ 15 second(s) to close. If the MSSVs were closed in 19 second neither of the Channels are operable. Per ITS 3.3.15 if the MSSVs are closed then the MSSVs are operable.

B. Incorrect – The MSSVs are operable if all the MSSV are closed. TSV Closure Channel A and B are required to be operable. Channel A and B TSV Closure due to the 19 second closure time.

C. Incorrect – Neither of the TSV Closure Channels actually closed the MSSVs within the required time but the MSSVs are closed and are operable.

D. Incorrect – The MSSVs are operable because the valves are all closed. The time limit for either of the TSV Closure Channels was not met.

TRAINING OBJECTIVES

TERMINAL OBJECTIVES

At the conclusion of this lecture, the student will be able to:

1. Describe the purpose and operation of the main turbine and its related components.
2. Explain the function of and purpose for the various protective actions/devices associated with the main turbine.

ENABLING OBJECTIVES

1. Describe the steam flow path from entry into the high pressure turbine to the exit of the low pressure turbine. (R1)
2. Explain what is meant by "Double Flow" as it relates to the main turbines at Oconee Nuclear Station. (R2)
3. Explain the two purposes of the Main Steam Stop Valves. (R3)
4. Explain why the # 2 Main Steam Stop Valve has an internal bypass valve. (R4)
5. Explain the purpose of the Control Valve above seat drains. (R5)
6. Explain the purpose of the Reheat Stop Valves. (R8)
7. Explain the purpose of the Intercept Valves. (R9)
8. Explain the purpose of the exhaust hood spray on the Low Pressure Turbines. (R10)
9. Describe the conditions that could result in high exhaust hood temperatures. (R11)
10. Discuss the consequences of high exhaust hood temperatures. (R12)
11. Identify the trip setpoint for "Low Condenser Vacuum". (R13)
12. Explain why the turbine should be placed on turning gear when it is shutdown. (R14)
13. Describe three methods of placing the turbine on turning gear. (R15)
14. Identify the causes of an auto trip of the turning gear motor. (R16)
15. Discuss the purpose of the extraction check valves. (R17)
16. Concerning Main Turbine vibration indication, identify that from: 0-600 RPM indication is invalid, 600-780 RPM indication is approximate, and from 780-1800 RPM indication is accurate. (R18)

2.2 Component Description

A. Main Steam Stop Valves (OC-STG-MT-3)

1. Four Main Steam Stop Valves (MS-102, MS-103, MS-104 & MS-105) are located in the four Main Steam inlet lines to the High Pressure Turbine.

2. Each MSSV outlet is welded directly into the inlet of a Main Control Valve casing.

3. The four MSSV's are physically located on the Mezzanine Floor (3rd floor) of the Turbine Building at the front end of each unit's High Pressure Turbine.

4. The four MSSV's are also welded together to form a common casing below the stop valve seats. This common connection forms the Main Steam Chest.

5. The primary purpose of the MSSV's is to quickly shut off steam flow to the turbine under emergency conditions. MSSV closure is initiated by a reactor trip signal. To keep from rapidly cooling off the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. From the CRDI system, a Channel "A" trip circuit will close all the MSSVs in ≤ 1 second. A Channel "B" trip circuit will close all MSSVs in ≤ 15 seconds.

6. The MSSV's also serve as the isolation valves between the OTSG's in the RB and the Main Turbine. All four MSSVs are needed. If for example a MSLB occurred. Both S/Gs would discharge through the break. Once the MSSVs closed, then steam would be isolated to just the single affected steam generator.

   a) ONS ITS 3.7.2 – Turbine Stop Valves states that both MSSVs in each main steam line shall be operable in Modes 1, 2, and 3. ITS Bases states: an operable MSSV is one that closes within its required actuation signal time limits. The MSSVs must be Operable or closed. A Closed MSSV is performing its safety function.

7. MSSV Construction (Valves 1, 3 & 4)

   a) Basically consists of a casing that holds a spherical-shaped hardened valve seat that is ~ 24" in diameter.

   b) A valve stem passes through the bottom of the casing and attaches to a conical-shaped valve disc that mates with the valve seat when the stop valve is closed.

c) Any steam leakage down the stem is routed to the Steam Seal Header.

d) A steam drain line and control valve are located above the stop valve seat to drain condensation from the casing that collects when the stop valve is closed.

e) An EHC Control Pack pushes the valve stem upwards and carries the valve disc with it to open the stop valve. Full stroke of the stem is ~ 8.5 inches.

8. #2 MSSV Construction (OC-STG-MT-4)

a) Basic construction is the same as the other MSSV's except that the #2 MSSV has an additional internal valve disc to regulate turbine warming prior to placing the turbine in operation.

b) The normal valve disc is modified:

1) The top portion of the disc is fabricated into an integral valve seat.

2) Orifices have been milled from the integral valve seat through the bottom portion of the main valve disc.

c) A smaller valve disc is attached to the normal valve stem of the #2 MSSV to mate with the integral valve seat formed into the main valve disc.

1) This smaller valve disc is called the #2 Main Stop Valve Bypass Valve.

2) When the proper logic (Shell Warming or Chest Warming) is selected, the operator can position the Bypass Valve disc from a fully shut to a fully open position from the EHC Control Panel in the Control Room.

3) With the Bypass Valve open, steam will flow through the main valve disc and through the orifices in the disc.

4) The Bypass Valve is used to:

(a) Slowly warm the HP Turbine Shell prior to placing the MT in service.

(b) Slowly warm the Main Control Valve components and the below seat metal of the MSSV's before admitting full steam through these valves.

(c) Reduce the ΔP across the MSSV's to allow the valves to open. For this reason, MSSV #2 is referred to as a "balanced" valve.

f) Loss of 24V DC

1) Loss of 24V DC will render the EHC electrical trip system inoperable.

2) A loss of 24V DC will trip the turbine by deenergizing the pilot solenoids and repositioning the master trip solenoid.

g) Loss of 125V DC

1) Loss of 125V DC will render this EHC electrical trip system inoperable.

2) A loss 125V DC will trip the turbine by energizing the 24V DC trip bus.

h) Manual Trip Button

i) Manual Trip Handle

6. Summary of Turbine Trips

a) Mechanical Overspeed - ≈ 1980 RPM / 110% of rated speed

b) Backup Overspeed - ≈ 2003 RPM / 111.25% of rated speed

c) Loss of Both Speed Feedback signals to the turbine speed control circuitry

d) Low Condenser Vacuum - ≈ 21.75 inches Hg.

e) Loss of Stator Coolant - If runback does not reduce load below 740 MWe within 2 minutes and below 256 MWe within an additional 1½ minutes if condition has not cleared when first plateau is reached.

f) Low Bearing Oil Pressure - Originates from old thrust bearing wear detector pressure switches at <8 psig

1) This trip was changed to incorporate 3 pressure switches and a 2 out of 3 trip logic.

g) High Steam Generator Level Trips -

1) 98% Level on Operating Range on either SG

2) SG Overfill Protection - An additional auxiliary relay was added to the OTSG level control system circuitry. Existing hi level contacts feed this relay. When both level signal monitors for A or B SGs sense a high level, a redundant trip signal is sent to both MFDWPS and the Main Turbine.

h) High Moisture Separator Level Trip – level at bottom of MSRH
   1) This trip incorporates 3 new level switches on each MSRH to provide a 2 out of 3 Turbine Trip logic.
i) Reactor Trip
j) Low Feedwater Pump Discharge Header Pressure - ≈ 800 psig
k) Low EHC Discharge Header Pressure - 1100 psig decreasing
l) Loss of 24V DC
m) Loss of 125V DC - 2 out of 3 relays which monitor 125V DC at three locations in the EHC cabinet
n) Generator Lockout - 86 GA
o) Manually initiated from:
   1) Turbine Oil Fire Trip
   2) Manual Trip Handle on Front Standard
   3) Master Trip Button in Control Room

P&I Oem 1-12-00

p) Emergency Low Hydraulic Oil Pressure – < 800 psig

E. Control systems

1. Speed Control

a) Six different speed sets can be selected by the operator:
   1) Close Valves
   2) 100 RPM - Used to sound out turbine bearings
   3) 500 RPM - Not used at Oconee
   4) 1500 RPM - Not used at Oconee
   5) 1800 RPM - Used to bring turbine to rated speed after 100 RPM checks are complete.
   6) Overspeed Test - This pushbutton can be used to test either the mechanical overspeed device or the back-up overspeed trip circuit.

- The MASTER IVs will not have opened far enough for the SLAVE IVs to open.

- Both the MASTER/SLAVE IVs will be open.

b) Three different acceleration rates can be selected by the operator.
   1) Slow - 60 RPM/MIN
   2) Medium - 90 RPM/MIN
   3) Fast - 180 RPM/MIN

QUESTION # 52

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**2**	**2**
	K/A #	**055000 G2.1.2**	
	Importance Rating	**3.0**	**4.0**

Technical Reference(s): **STG-CVS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **STG-CVS**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 52

Unit 3 plant conditions:

- Reactor power = 80%
- SGTL = 1.0 gpm has just developed

Which ONE of the following will be the first means of detecting the SGTL?

A. CSAE off gas

B. FDW mismatch

C. MS Line monitors

D. Chemistry sampling

1 POINT

QUESTION # 52

055000 G2.1.2 common rsi/gcw 05/08/00 Bank TA-097 (modified)

A. Correct - Non-condensable gaseous activity and will be transported to the main condenser where the air ejectors would extract the non-condensable from the condenser and release to the unit vent. This would be the first operator indication of the SGTL

B. Incorrect - FDW mismatch is a method of SGTL detection but not during power operation

C. Incorrect – MS line monitors will not be able to detect a small SGTL

D. Incorrect – Chemistry sampling requires collection and analysis time to sample the SG inventory.

TERMINAL OBJECTIVE

Upon completion of this lesson, the student will be able to describe the purpose, operation, and response of the Condenser vacuum System during normal and abnormal plant conditions as it relates to his or her job responsibilities.

ENABLING OBJECTIVES

1. State the purposes of the Condenser Vacuum System. (R1)
2. Concerning the Main Vacuum Pumps: (R2)
   - 2.1 State the maximum achievable condenser vacuum using the Main Vacuum Pumps.
   - 2.2 Recognize that the three (3) Main Vacuum Pumps serve all three units.
3. Concerning the Main Condenser: (R3)
   - 3.1 Briefly describe the operation of the Main Condenser
   - 3.2 Briefly describe the method by which the high condenser vacuum is created and maintained.
   - 3.3 State the component used to remove air and non-condensible gases from the Main Condenser.
   - 3.4 Briefly describe the operation of the condenser boot seals.
4. Concerning the Air Ejectors: (R4)
   - 4.1 Briefly discuss the principle of operation of the Air Ejectors including the reason for using two stages.
   - 4.2 State the cooling medium of the Air Ejectors.
   - 4.3 State the purpose for the inter condenser drain loop seal.
   - 4.4 State where the inter condenser and after condenser drains discharge to.
   - 4.5 State the purpose of the Air Ejector Off Gas blower.
   - 4.6 Briefly describe how the Off Gas volume is determined at Oconee.
   - 4.7 State the reason for monitoring Off Gas flow for radioactivity.

TRAINING OBJECTIVES continued

3. Main Condenser Controls and Indications

   a) There is condenser vacuum gauge located in the control room that indicates vacuum in the 'C' condenser section.

   b) There are computer points for vacuum and absolute pressure for each section of the condenser on the OAC.

   c) There is a statalarm and computer alarm to alert the operators to decreasing condenser vacuum (24" Hg decreasing).

C. Boot Seals (CVS-4)

1. The low pressure turbines are connected to their respective condenser section through a boot seal assembly.

2. The purpose of the boot seal is to allow for thermal expansion and contraction of the low pressure turbine shell and the neck of the condenser while preventing any air in-leakage around the seal area.

3. The boot seal assembly is a rubber collar bolted to the LP turbine exhaust and to the condenser neck on the inside.

4. The outside at the joint is surrounded by a trough of water kept full from the condensate system that acts as a seal against air in-leakage by covering up the joint area.

5. Proper boot seal water flow is established when the pinwheel on the boot seal trough drain spins or water can be seen through the sightglass, vacuum is holding and boot seal water is not overflowing down the outside of the condenser.

D. Condenser Steam Air Ejectors

1. Air In-Leakage

   a) If the condenser were perfectly airtight, and if no air or other non-condensible gases were present in the exhaust steam entering the condenser, it would be necessary only to condense the steam and to remove the condensate in order to create and maintain a vacuum.

   b) The sudden reduction in the volume of the steam as it changed to water would create a vacuum, and pumping the water from the bottom of the condenser as fast as it is formed would maintain the vacuum so created.

   c) Unfortunately, valve seat leaks and packing leaks, pipe flange leaks, and non-condensible gases entrained in the exhausted turbine steam allows for air and non-condensible gas leakage into the condenser.

d) If air is allowed to occupy space in a condenser, it interferes with the transfer of heat by tending to insulate the tube surfaces from the steam.

e) Thus the air inhibits the condensing of the steam resulting in a decreased condenser vacuum.

f) Vacuum is reduced further by the partial pressure exerted by the air itself, which is added to the pressure of the steam in the condenser.

2. Since it is practically impossible to prevent air and other non-condensible gases from entering the condenser, it is necessary to use an air ejector to maintain the vacuum in the condenser.
3. The primary functions of air ejectors are to remove the non-condensible gases (mainly air) that are discharged into condensers operating under a vacuum, and to compress those gases to the pressure necessary for discharge from the condensing system.
4. Principles Of Operations (CVS-5)

a) The air ejector is a jet pump. A jet of high pressure motive steam, expanding through a nozzle, acquires a high velocity (supersonic). It entrains the air/vapor mixture surrounding the nozzle exit and accelerates it to the steam velocity.

b) Friction between the steam jet and the low pressure air causes the air to move with the steam into the converging section of the diffuser tube where the steam and air mix.

c) The divergent section at the downstream end of the diffuser tube serves to decrease the velocity of the moving gas and to increase its pressure, thus converting energy in the form of motion (kinetic energy) into energy in the form of pressure, so that the pressure at the diffuser outlet is higher than at the air vapor entrance.

d) In this way the air and other non-condensible gases removed from the vacuum system are compressed, thus accomplishing the purpose of the air ejector element.

e) The air ejector has a poor efficiency when the pressure of its outlet is more than 8 or 9 times its suction pressure. Yet the purpose of the air ejector is to raise the pressure of the air from condenser pressure to atmospheric pressure, a ratio of 15 or 20 to 1. The problem is solved by placing two ejectors in series.

3. Summary

3.1 System Overview

A. The purpose of the main vacuum system is to establish the initial vacuum on the condenser, upper surge tank, and feedwater pump turbines during a unit startup and to remove air and other non-condensible gases from the steam space of the condenser shells during normal and abnormal (i.e., steam generator tube rupture) plant

B. Three main vacuum pumps that are shared between the three Oconee units are used to establish the initial system vacuum during a unit

C. During normal operation, condenser vacuum is maintained by the reduction in volume of the steam exhaust from the low pressure turbines as it is condensed into water.

D. The three parallel condenser steam air ejectors assist in maintaining system vacuum by continuously removing air and other non-condensible gases from the condensers.

E. Air and non-condensible gases that is removed by the CSAEs is transferred to the vent stack via the CSAE blower.

F. As the gases are being released, they are monitored by process radiation monitor RIA-40 in order to detect and alert the operators to the presence of a steam generator tube leak.

3.2 Review Objectives

QUESTION # 53

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**062000 A3.04**	
Importance Rating	**2.7**	**2.9**

Technical Reference(s): **EL-VPC**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EL-VPC OBJ. #4.4**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 53

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Power level = 100%
- Static Inverter 1DID is connected to regulated power source MCC 1X0
- MCC 1XP is de-energized for maintenance

CURRENT CONDITIONS:

- A loss of power to MCC 1XO occurs

REGULATED POWER RESTORATION SEQUENCE:

- Power is **first** restored to 1XP then power is restored to 1XO

Which ONE of the following describes the operation of the ASCO Transfer Switch (ABT)?

The ASCO Transfer Switch...

A. is manually transferred to 1XP then re-transferred to 1XO by the NLO as directed by the control room operator.

no sense

B. must be is manually transferred to 1XP when power is restored then automatically re-transfers to 1XO when power is restored to 1XO.

ok C. automatically transfers to 1XP when power is restored to 1XP and automatically re-transfers to 1XO when power is restored to 1XO.

D. automatically transfers to 1XP when power is restored to 1XP and remains positioned to 1XP when power is restored to 1XO.

1 POINT

QUESTION # 53

062000A3.04 (2.7/3.0) BOTH PRA 4-6-00

A. Incorrect - The ASCO Transfer Switch operation is automatic not a manual transfer function. If Source 1 is interrupted (say XO de-energizes) the ABT will automatically transfer over to Source 2 (XP) and the LOAD CONNECTED TO EMERGENCY light will come on.

B. Incorrect - The ASCO Transfer Switch operation is automatic not manual. The retransfer must be manual

C. Incorrect - If Source 1 is interrupted (XO de-energizes) the ASCO Transfer Switch will automatically transfer to Source 2 (XP) and the LOAD CONNECTED TO EMERGENCY light will come on. Once a transfer to Source 2 has occurred, the ABT can only be returned to a normal alignment (Source 1) by manually positioning the RESET toggle switch spring returns to the Reset position.

D. Correct – Source 2 is initially deenergized. No transfer occurs upon the loss of Source 1. Power being regained to Source 2 first causes the ABT to select XP. Following restoration of Source 1, a manual transfer is required to return the ABT to Source 1 (XO).

A. List three conditions that will cause a statalarm and identify the location of the alarms.

2.4 Explain the operation of the inverter fans and how their operation affects the inverter operation.

3. Describe, or draw, the power path from the DC power bus to the KU, KOAC, KI, and KX inverters, including the backup or AC Line source. (R3)

4. Discuss the Essential inverters. (R4)

4.1 Explain the difference between the vital and essential inverters.

4.2 Explain the difference between the KI&KX inverters and the KU inverter.

4.3 Demonstrate the ability to locate and explain all panel meters, lights, switches and breakers.

4.4 Discuss the operation of the essential inverters:

A. Explain the operation of the Manual Bypass switch.

B. Explain the operation of the Static Transfer switch.

C. Explain the function and operation of the ASCO Transfer switch.

D. Explain the operation and location of the Inverter Bypass switches.

1. Explain what would happen if both SW #2 and SW #3 Bypass switches were opened.

E. Explain the operation of the precharge switch.

1. Explain why the Precharge Light should be lit before closing the DC INPUT circuit breaker on an Essential inverter.

F. Describe the basic startup and shutdown of the inverters.

G. Describe the inverter status during normal operation.

4.5 Explain the statalarms associated with the essential inverters:

A. Identify three conditions that will cause a statalarm and identify the location of the alarms.

4.6 Explain the operation of the inverter fans and how their operation affects the inverter operation.

5. Explain the difference in the switching arrangements for Inverter and AC Line between the Vital Power panelboards and the Essential Power panelboards. (R5)

6. Explain how it is possible to interrupt power to the AC power panelboard if the three switches of the Static Inverter Bypass Switch are operated incorrectly. (R6)

C. The voltage regulators receive power from either 600V MCC XO or XP, through the respective 600/240/120VAC transformer.

1. **(2)(3)XO** from (2)(3)X5; (1)(2)(3)XP from (1)(2)(3)X6.

a) X5 and X6 will load shed if the SL Breakers are closed to CT-5 and a loss of AC electrical power causes a "LOAD SHED" signal to be generated -provided NO ES signal is present.

1) If X5 and X6 are load shed, they will automatically re-energize thirty (30) seconds after power is restored to the MFB's. Therefore, AC panelboards previously powered by AC line will lose power until X5 and X6 automatically regain power.

2. **1XO** receives its normal power from 1X7.

a) 1X7 always load sheds.

b) The alternate power supply for 1XO is 1X5. 1X5 will respond as described above.

D. The ASCO Transfer Switch, transformers, regulators, and 600V MCCs are all located in the equipment room of the unit. The KRA/KRB Panelboards are in the cable room.

E. The ASCO Transfer Switch panel includes the following controls and indications: (OC-EL-VPC-10 & 10A)

1. **LOAD CONNECTED TO EMERGENCY** light:

a) Normally this light will be off, indicating that Voltage Regulator and Transformer Circuit A (Source 1) is energizing KRA/KRB.

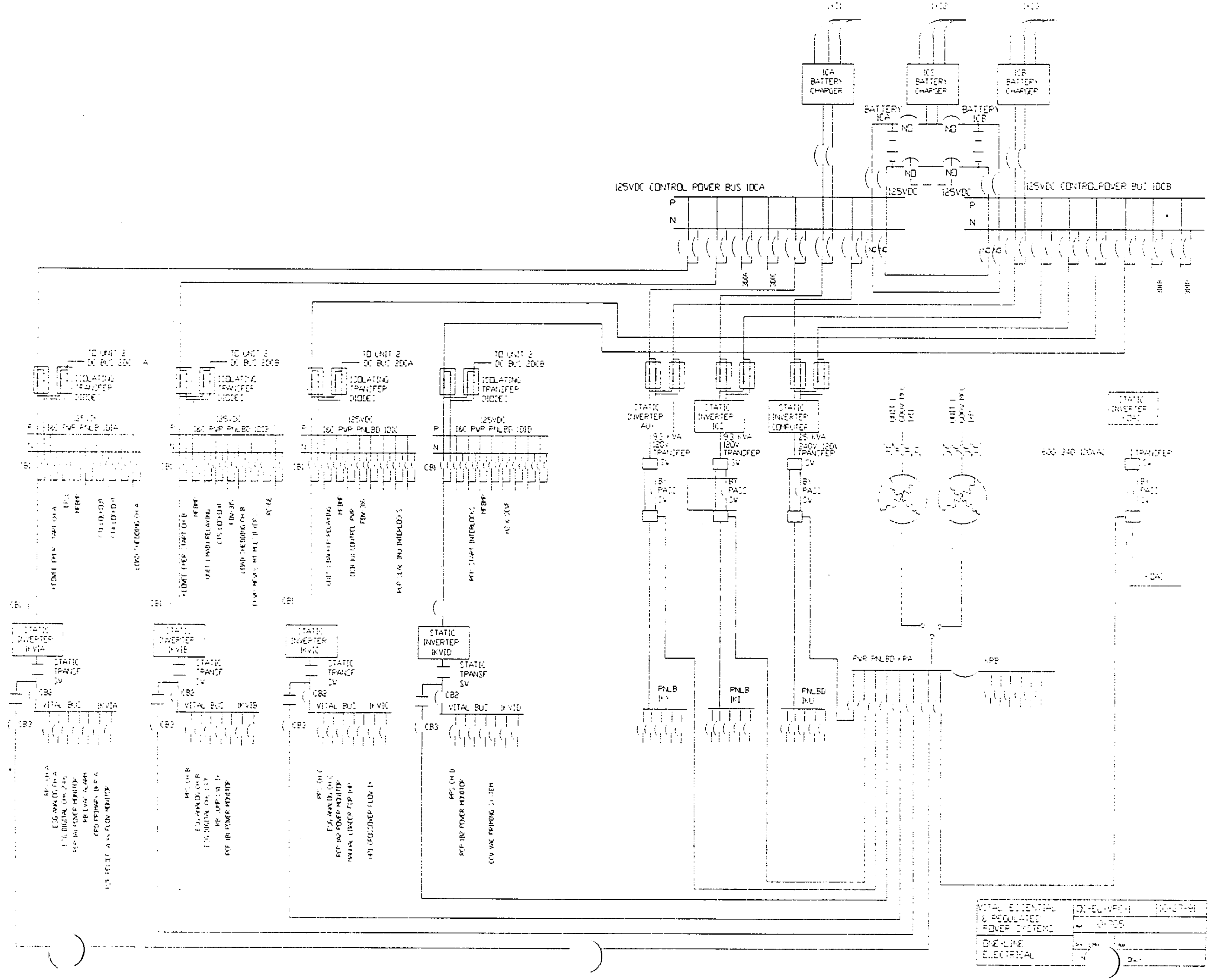
b) If Source 1 is interrupted (say XO de-energizes) the ABT will automatically transfer over to Source 2 (XP) and the LOAD CONNECTED TO EMERGENCY light will come on.

2. A **TRANSFER TEST** switch is provided on the front of the ABT panel for checking ABT action or for manually transferring over to Source 2. Pushing the Transfer Test switch to TEST simulates a low voltage on Source 1.

3. **RESET** - Once a transfer to Source 2 has occurred, the ABT can only be returned to a normal alignment (Source 1) by manually positioning the RESET toggle switch momentarily to the Reset position.

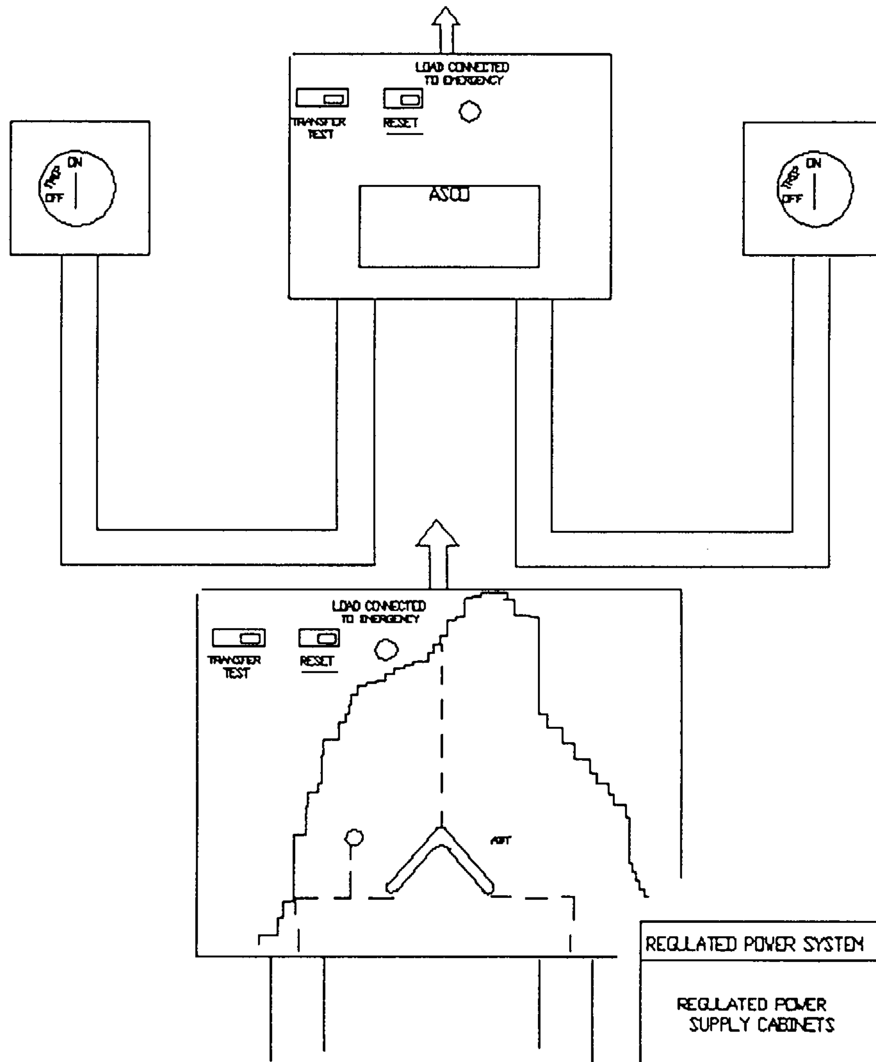
Note: The ASCO Transfer Switch is power seeking. So, if source 2 is supplying KRA/KRB and source 1 is available, a loss of source 2 will result in a transfer to source 1.

F. Indication as to which source is supplying Regulated Power is provided on the vertical boards (VB1) in the control room for the unit.



VITAL ESSENTIAL & REGULATED POWER SYSTEMS

ONE-LINE ELECTRICAL



REGULATED POWER SYSTEM	DC-EL-VPC-14	10-23-91
REGULATED POWER SUPPLY CABINETS	DRAWN FROM PANEL	
	LMH	App
	TRAINING USE ONLY	

QUESTION # 54

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**064000**	**A3.09**
Importance Rating	**4.0**	**4.0**

Technical Reference(s): **EL-KHG EL-EPD**
EL-PSL

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EL-KHG-12 EL-EPD-16 EL-PSL-7**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

QUESTION ______

Unit 3 plant conditions:

INITIAL CONDITIONS: 10-7-00/0430

- Reactor power = 68%
- ACB-3 Closed
- 230kv Switchyard Yellow buss voltage
  - X Phase = 226.3 kv
  - Y Phase = 225.7 kv
  - Z Phase = 226.5 kv

CURRENT CONDITIONS: 10-7-00/0432

- 230kv Switchyard Yellow buss voltage
  - X Phase = 226.5 kv
  - Y Phase = 225.9 kv
  - Z Phase = 226.7 kv
- RCS pressure rapidly decreases to 1375 psig
- At 10-7-00/0450 KHU #1 experiences a generator differential.

nutes
ter

ASSUME NO operator action

Which ONE of the following is correct?

Keowee Hydro Unit #_____ will energize the Unit 3 Main Feeder Buses via ______.

A. 1 / CT-3

B. 1 / CT-4

C. 2 / CT-4

D. 2 / CT-3

A & B : KHU #1 differential lockout => #1 KHU inoperable
simplistic conclusion to eliminate A & B.

C 2

1 POINT

QUESTION # 54

Unit 3 plant conditions:

INITIAL CONDITIONS: 10-7-00/0430

- Reactor power = 68%
- ACB-3 Closed

CURRENT CONDITIONS: 10-7-00/0432

- SWITCHYARD ISOLATION has occurred
- ES 1 and 2 automatically actuate on low RCS pressure

Which ONE of the following is correct if KHU #1 experiences ~~a~~ high generator differential ~~lockout~~ current at 10-7-00/0450?

ASSUME NO operator action

Keowee Hydro Unit #_____ will energize the Unit 3 Main Feeder Buses via ______.

A. 1 / CT-3

B. 1 / CT-4

C. 2 / CT-4

D. 2 / CT-3

Q-29

1 POINT

QUESTION # 54

064000A3.09 BOTH GCW 02/10/2000 (GTH/PMS)

A. Incorrect, KHU # 1 will emergency lock out and cannot provide power. Other relays and events could cause a normal or Alarm lockout, which would not prevent the unit from supply power after an emergency start.

B. Incorrect, KHU # 1 will emergency lock out and cannot provide power. Other relays and events could cause a normal or Alarm lockout, which would not prevent the unit from supply power after an emergency start.

C. Incorrect, The conditions are not met to cause the underground feeders to swap.

D. Correct, The KHU #1 will emergency lock out and the Oconee unit will lose power a retransfer to startup will occur causing the “E” breakers to close thus supplying power from KHU #2 via CT-3.

4. State the following about the Main Feeder Bus Monitor Panel Logic:
   4.1 Purpose (R11)
   4.2 Location of panel (R12)
   4.3 The conditions that will initiate a MFBMP signal. (R13)
   4.4 The events which will occur following a MFBMP actuation. (R14)

5. Concerning Emergency Power Switching Logic, state the following:
   5.1 Purpose (R15)
   5.2 Location of panel (R16)

6. For the Startup Breaker Anti-Recycle Relay, recall the following:
   6.1 Purpose (R17)
   6.2 The conditions that will generate a STAR relay signal. (R18)
   6.3 The events that will occur following a STAR relay actuation. (R19)

7. For the Transfer to Standby and Retransfer to Startup Logic, state the following: (R20)
   7.1 The conditions which will initiate a transfer to Standby operation.
   7.2 The conditions which will initiate a retransfer to startup operation.

8. Discuss the operation of the SK breakers as they relate to power switching logic. (R21)

9. Discuss the operation of the SL breakers as they relate to power switching logic including the following. (R22)
   9.1 The under-voltage and degraded voltage protection when the Standby Bus is receiving power from CT-5.
   9.2 The purpose and operation of the SL-1 and SL-2 TRIP INTERLOCK DEFEAT selector switches.

2.8 Retransfer to Startup

A. Function

1. The Retransfer to Startup logic provides the emergency power switching logic the capability to retransfer essential loads from the Standby Bus to the startup source, if available, should power to the Standby Bus be lost for more than 5 seconds.

B. Retransfer to Startup Logic **(OP-EL-PSL-15)**

1. The Retransfer to Startup (STD) logic senses that the emergency power switching logic has or has attempted to transfer power for the essential loads to the Standby Bus during an accident situation. The STD logic provides the capability to retransfer the essential loads back to the startup power source if power on the Standby Bus is lost, or if the startup source becomes available before power is applied to the Standby Bus. This logic is accomplished by the coincident occurrence of each of the following input signals:

    a) A standby breaker close initiation signal (SBCX/1A and SBCX/1B), as described above, provides the retransfer logic with the knowledge that a transfer to the Standby Bus has been initiated. Once the retransfer logic is satisfied, a seal-in (SDS/1A) is provided to maintain the retransfer to startup (STD) signal in the event of a failure of the SBC input signal.

    b) An input signal is provided from relay RX/1 to indicate the need for an alternate power source. The signal is initiated by either an ESG signal (ESG-1) or indication of under-voltage on both Main Feeder Buses ($TX1_{1A}$ and $TX2_{1A}$).

    c) Under-voltage on two out of three phases of Standby Bus 1 (27SY) and Standby Bus 2 (27SY). These signals provide the retransfer logic with the knowledge that the standby buses are not available to supply essential loads.

    d) Under-voltage is not sensed on two out of three phases of the startup source (27EY). This signal indicates to the STD logic that power has been regained at the startup source since the transfer to the Standby Bus was initiated.

2. Five seconds after the logic is satisfied, a retransfer to startup signal is generated which provides a trip signal to breakers S1 and S2, and un-blocks the closure of the startup breakers (E1 and E2).

13. Explain why restoration to normal conditions is more critical following a 230 KV Yellow Bus Differential Lockout than following a 230 KV Red Bus Differential Lockout. (R-15)

14. Explain briefly the purpose for the "230 KV Grid Protection Relay" (SY Isolation) signal at Oconee. (R-16)

15. Identify the preferred path for unit auxiliaries following a SY Isolation event, if the unit main generator has also tripped. (R 17)

16. Describe the following about the SY Isolation circuits at Oconee: (R-18)

    16.1 Initiation logic.

    16.2 State the major items performed by the Switchyard Isolation Circuit.

    16.3 The basic actions required restoring normal conditions following initiation.

17. Evaluate the affect on the plant should the applicable PCBs fail to operate properly following a Switchyard Isolate signal. (R-51)

18. List in order of preference the possible means of supplying auxiliary electrical power to an Oconee unit. (R-20)

19. State the purpose of the Kirk Key Interlock associated with the Unit 1&2 "B" LPSW Pump 4160V breakers on 1TD and 2TD. (R-21)

20. Describe the basic procedure for transferring motor control centers (MCC's) between NORMAL and EMERGENCY POWER load centers (LC's) including: (R-22)

    20.1 The proper position for the Auto-Manual selector switches on the Normal and Emergency LC's prior to manually transferring power.

    20.2 How power is manually transferred between sources.

    20.3 The proper position for the Auto-Manual selector switches on the Normal and Emergency LC's after power has been manually transferred.

21. Explain why it is necessary to place 600V load center Auto-Manual selector switches in Manual prior to performing any dead-bus transfers. (R-23)

22. Describe the basic procedure for supplying alternate power to safety related 600V/208V MCC's. (R-24)

23. State the purpose of the Kirk Key Interlock associated with Load Center 2X11 and Distribution Center 2X11A. (R-25)

12. For an emergency lockout (ELO) or normal lockout (NLO) of a KHU: (R10)
    12.1 describe automatic actions that occur.
    ~~12.2~~ determine events that that will cause an ELO or NLO.
    12.3 determine actions required following an ELO or NLO. (R21)
13. Given a copy of OP/0/A/1106/19, Keowee Hydro at Oconee, verify the proper sequence of actions have occurred for: (R4)
    13.1 an automatic start of a Keowee Hydro unit.
    13.2 a manual start of a Keowee Hydro unit.
    13.3 a manual shutdown for a Keowee Hydro unit.
    13.4 an Emergency Start of a Keowee Hydro unit.
14. State the locations of the manual emergency start controls. (R6)
15. List all signals that will initiate an emergency start of a Keowee Hydro unit. (R5)
16. Given a set of conditions, verify proper sequence of actions have occurred for an Emergency Start of the Keowee Hydro Units. (R18)
17. Given a copy of AP/O/A/2000/002, KHS Emergency Start, discuss the reason for the performance of specific steps. (R14)
18. Explain the basis for the critical action steps of the following NLO JPMs associated with the KHG: (R25)
    18.1 NLO-045, Restore power to the 600 volt switchgear 1X
19. Describe how various degraded conditions of this component could affect continued safe plant operation and the impact on accident mitigation, if any. (R16)
20. Draw and explain the Electrical Distribution System of the Keowee Hydro Station down to the 600V load centers. (R13)
21. Given a set of conditions, diagnose the status of the KHS 600V power supply system. (R22)
22. Given a copy of ITS/SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's. (R26)
23. Apply all ITS /SLC rules to determine applicable Conditions and Required Actions for a given set of plant conditions. (R27)

JAN 1 1 1997
Approval Date

2SA-17/A-01
Statalarm Number
Rev. #3 (10/96)

GEN. #1 EMERG. LOCKOUT

1.0 Alarm Setpoint

86E-1 - Hand reset lockout relay activated by following which will prevent an Emergency Start:

- S3SUIX - GATE SAFETY (Key Inhibit) switch at Keowee,
- 12XTD/1 - Overspeed Switch (23 second time delay ≥ 180 RPMs),
- 63FX - Generator Fire, CO_2 Release Relay,
- 62-1/TD - ACB-1 and ACB-3 Backup Trip Timer,
- 59GN/1 - Generator Neutral Ground Relay,
- 87T-1E - Transformer 1E Differential Relays,
- 87G-1 - Generator Differential Relays,
- 87GB-1 - Generator Bus Differential Relays,
- 40G1 - Generator Loss of Field Relay,
- 76D - Generator Field Overcurrent Detector Relay,
- 76T2 - Maximum Excitation Timer Relay,
- V/HZ-TD - Over Volt/Cycle Relay,

2.0 Automatic Actions

2.1 Operates supervisory system points 2SA-17/5 (2SA-17/A-05) and 2SA-17/1 (2SA-17/A-01), and Keowee statalarm 1SA2/03 (1SA2/A-03).

2.2 Trips and blocks closure of ACB-1 and ACB-3 at Keowee.

2.3 Trips and blocks closure of field flashing, supply, and field breakers.

2.4 Blocks startup of unit under all conditions.

2.5 Operates 86T (Main Step-up Transformer Lockout Relay, see 2SA-17/B-04) if ACB-1 fails to trip (62-1/TD).

2.6 Operates 86EF (Emergency Feeder Lockout Relay, see 2SA-17/A-04) if ACB-3 fails to trip 62-1/TD).

3.0 Manual Actions

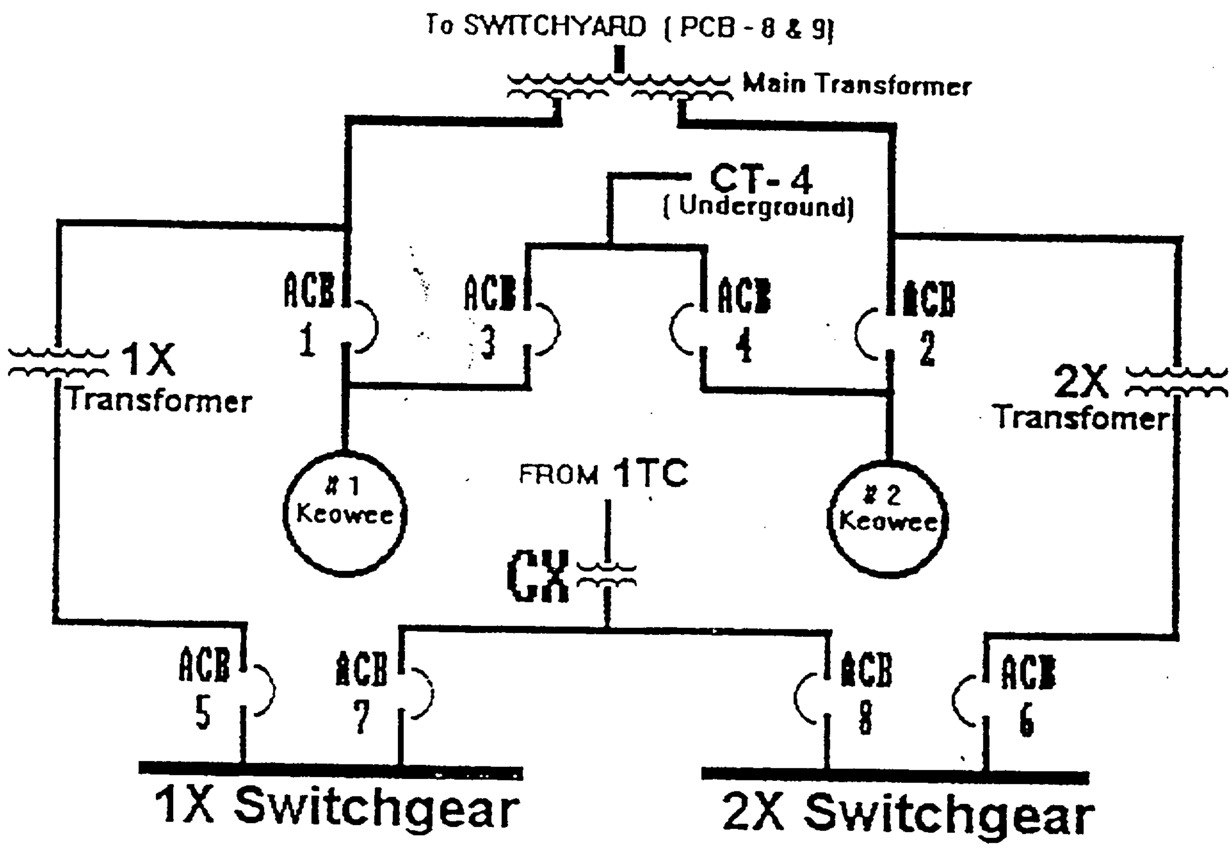
3.1 Check and record statalarms before resetting.

3.2 Contact Keowee Operating personnel at ext. 3326 or 3327 for information concerning this alarm and unit availability.

4.0 Alarm Sources and References

4.1 SOURCES and REFERENCES: 86E-1; Keowee Alarm Response Guideline 1SA2/03 (1SA2/A-03), KEE-106-1, KEE-106-2, KEE-111, KEE-112-3, KEE-114-3, KEE-114-4, K-711-D, K-712-C, K-734, Keowee Station Relay Instructions, OEE-218-33, OEE-218-34.

TITLE:	NOTES:	ID#: OC-EL-KHG-13	DATE: 02/14/97
KEOWEE	KEOWEE	REF: ARG 2SA17/A-01	
HYDRO	HYDRO	DRN BY: JAW	APR BY: PMS
GENERATOR	LOAD LIMITS	TRAINING USE ONLY	


To SWITCHYARD (PCB - 8 & 9)
Main Transformer
CT-4
(Underground)
ACB 1
ACB 3
ACB 4
ACB 2
1X Transformer
2X Transfomer
#1 Keowee
#2 Keowee
FROM 1TC
CX
ACB 5
ACB 7
ACB 8
ACB 6
1X Switchgear
2X Switchgear

QUESTION # 55

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**064000 K4.04**	
Importance Rating	**3.1**	**3.7**

Technical Reference(s): **EL-PSL**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EL-PSL OBJ. #3 & #7**

Question Source:
Bank # ______
Modified Bank # **EL-008**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 55

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Power = 100%
- ACB-4 closed

CURRENT CONDITIONS:

provide condition which would cause this

- Switchyard Isolation has occurred
- Keowee Unit 1 Emergency Lockout

Which ONE of the following is correct?

Load Shed _________ occur ________.

A. will / to prevent overloading CT-4.

B. will / to prevent overloading the Standby Buss.

C. will not / and power is restored via CT-2 and a Keowee Unit.

D. will not / and power is restored via CT-2 and the 230 KV Switchyard.

1 POINT

QUESTION # 55

064000K4.04 common GCW 05/08/00 new

Question setup:
The Unit will trip as a result of the Switchyard Isolation and the subsequent load rejection. Keowee Unit 1 would be supplying the yellow buss and the Unit 2-startup transformer but it locks out. This will cause a loss of power and a Load Shed and the Unit will regain power from Keowee Unit 2 via CT-4.

A. Correct, Load Shed will occur to protect CT-4.

B. Incorrect, first part is correct however the load shed is to protect CT-4.

C. Incorrect, load shed will occur. If the other Keowee were tied to the underground this answer would be correct.

D. Incorrect, load shed will occur. If the Unit had tripped without a Switchyard Isolation this answer would be correct.

OBJECTIVES

Terminal Objective

1. Discuss the EPSL, including the various power supplies and how each power source can be aligned to supply power to ONS Units MFB during Design Bases Events. (T1)

2. For a given set of plant conditions evaluate the status of the MFB power sources including automatic system actions, time frames for re-energizing the MFB, and what contingency actions are required if automatic actions do not occur. (T2)

Enabling Objectives

1. Concerning the Design Bases for the 4KV Essential Auxiliary Power System, describe the following: (R1)

   1.1 The System Functional Design Bases

   1.2 The Design Bases Events

2. Concerning a Keowee Emergency Start, describe the following: (R2)

   2.1 Purpose

   2.2 Panel location

   2.3 Emergency Start signals

3. Describe the following for the Load Shed Logic:

   3.1 Purpose (R3)

   3.2 Panel location (R4)

   3.3 The conditions, which will initiate a load-shed signal and the logic, involved. (R5)

   3.4 Loads which will be load shed (R6)

   3.5 How to reset a load shed signal (in the Cable Room) (R7)

   3.6 How to reset a load shed signal (R8)

   3.7 The location of the fuses for load shed in the 4160V switchgear (R9)

   3.8 How to verify power is available to the load shed trip relays (R10)

4. State the following about the Main Feeder Bus Monitor Panel Logic:
    4.1 Purpose (R11)
    4.2 Location of panel (R12)
    4.3 The conditions that will initiate a MFBMP signal. (R13)
    4.4 The events which will occur following a MFBMP actuation. (R14)

5. Concerning Emergency Power Switching Logic, state the following:
    5.1 Purpose (R15)
    5.2 Location of panel (R16)

6. For the Startup Breaker Anti-Recycle Relay, recall the following:
    6.1 Purpose (R17)
    6.2 The conditions that will generate a STAR relay signal. (R18)
    6.3 The events that will occur following a STAR relay actuation. (R19)

7. For the Transfer to Standby and Retransfer to Startup Logic, state the following: (R20)
    7.1 The conditions which will initiate a transfer to Standby operation.
    7.2 The conditions which will initiate a retransfer to startup operation.

8. Discuss the operation of the SK breakers as they relate to power switching logic. (R21)

9. Discuss the operation of the SL breakers as they relate to power switching logic including the following. (R22)
    9.1 The under-voltage and degraded voltage protection when the Standby Bus is receiving power from CT-5.
    9.2 The purpose and operation of the SL-1 and SL-2 TRIP INTERLOCK DEFEAT selector switches.

d) Channel A (B) Keowee Start Oconee 1(2)(3) Instrument Coil Circuit Good.

1) monitors Keowee emergency start relay at Oconee (K relay)

2) normally on

e) Keowee Emergency Start Local Initiate Channel A (B)

1) Key switch

2) Initiates emergency start signal to both Keowee units

f) Channel A(B) Keowee Start Relays at Keowee Coil Circuit Good

1) monitors emergency starting relays at Keowee

2) normally on

2.3 Load Shed

A. Purpose

1. Trip the feeder breakers to non-essential loads on the 4KV switchgear. Shedding the non-essential loads reduces the load on the Main Feeder Bus to a level within the capacity of the standby transformers (CT4 and CT5) in the event a standby source is required.

B. Source capabilities and load considerations

1. The total required AC power for a single unit at hot shutdown is approximately 6.2MVA (after Load Shed) for the units 4KV auxiliaries.

2. A single unit at hot shut down with a LOCA (ES actuation) would require 7.8 MVA (after Load Shed).

3. Two units at hot shutdown and the third with a LOCA would require a total AC capacity of 20.2 MVA.

4. The continuous AC power capacity available from the on-site power systems (Keowee Hydro Units) is 22.4 MVA (limited by CT-4) if furnished by the underground circuit or 30MVA (limited by CT-1, CT-2, or CT-3) if furnished by the 230 KV transmission lines. Capacity available from the backup 100 KV off-site transmission line (Lee Gas Turbine Generator) is 22.4 MVA (limited by CT-5).

5. Thus, the minimum available capacity from any of the multiple sources of AC power, 22.4 MVA, is adequate but only if non-essential loads are shed.

Exact Question Report

27-Jan-99

Question

KA: 064000K4.04

The reason for load shedding electrical loads is to prevent overloading which ONE of the following? (.25)

A) The Standby Buses.

B) The CT4 transformer.

C) The Main Feeder Buses.

D) The Startup transformer.

Answer

B

Lessons

ID	Description
EL-PSL	Power Switching Logic EL-PSL

Enabling Objectives

ID	Description
ELPSLR3	3. Describe the following for the Load Shed Logic: 3.1 Pu

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

Exam Question Report

27-Jan-99

Question ID:	EL008	Revision No:	0	Revision Date	10/29/1999
Question Description:	EL008				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EL-PSL - Power Switching Logic
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 3; LRO = 3; SRO = 3	Max. Point Value:	0.25
		Passing Point Value:	0.25

QUESTION # 56

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**073000 K3.01**	
Importance Rating	**3.6**	**4.2**

Technical Reference(s): **SLC 16.11.3**

Proposed references to be provided to applicants during examination: **SLC 16.11.3**

Learning Objective: **WE-T50 OBJ. #1**

Question Source:
Bank # **ADM-001**
Modified Bank # ________
New ________

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 56

Unit 1 plant conditions:

- MODE 5
- Reactor Building purge has been in progress for the past twelve (12) hours.
- I&E investigation of a sudden drop in purge flow indicates that the RB Purge flow monitor is inoperable.
- Replacement monitor has been ordered and will arrive within the next twenty-four (24) hours.

Per SLC 16.11-3, Radioactive Effluent Monitoring Instrumentation, which ONE of the following is a required action(s) concerning the purge release?

SEE ATTACHMENT

The release...

A. may continue if the flow rate is estimated immediately and once every four (4) hours.

B. may continue if the position of 1PR-3 is unchanged for the duration of the release.

C. must be stopped until a redundant containment sample can be taken.

D. must be stopped until two independent samples can be analyzed.

RO STA?

1 POINT

QUESTION # 56

073000 K3.01 COMMON rsi/gcw 05/08/00 Bank ADM-001

A. Correct, - The conditions provided indicate that a failure has occurred which meets the requirements of 16.11-3 Condition " J". It is NOT a "short controlled outage" so the required ACTIONs of J.1 or J.2 must be met. J.1 suspends the release or J.2 allows continued release if the flow rate is estimated immediately and once every four (4) hours.

B. Incorrect, - PR-3 is in the RB purge flow path which will be open for the given conditions, and if closed then the release would be terminated.

C. Incorrect, - There is no procedural guidance or requirement to take a redundant containment sample.

D. Incorrect, - This action would be required for one or more liquid effluent monitoring instrument channels.

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more required liquid effluent monitoring instrument channels inoperable.	B.1	Enter the Condition referenced in Table 16.11.3-1 for the function.	Immediately
	AND		
	B.2	Restore the instrument(s) to OPERABLE status.	30 days
C. One or more required gaseous effluent monitoring instrument channels inoperable.	C.1	Enter the Condition referenced in Table 16.11.3-2 for the function.	Immediately
	AND		
	C.2	Restore the instrument(s) to OPERABLE status.	30 days
D. Required Action and associated Completion Time of Required Action B.2 or C.2 not met.	D.1	Explain in next Annual Radiological Effluent Release Report why inoperability was not corrected in a timely manner.	April 30 of following calendar year

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. As required by Required Action B.1 and referenced in Table 16.11.3-1. (RIA-33)	E.1.1	Analyze two independent samples in accordance with SLC 16.11.4.	Prior to initiating subsequent release
	AND		
	E.1.2	Conduct two independent data entry checks for release rate calculations	Prior to initiating subsequent release
	AND		
	E.1.3	Conduct two independent valve lineups of the effluent pathway.	Prior to initiating subsequent release
	OR		
	E.2	Suspend release of radioactive effluents by this pathway.	Immediately
F. As required by Required Action B.1 and referenced in Table 16.11.3-1. (RIA-54)	F.1	Suspend release of radioactive effluents by this pathway.	Immediately
	OR		
	F.2	Collect and analyze grab samples for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml.	Prior to each discrete release of the sump

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action B.1 and referenced in Table 16.11.3-1. (Liquid Radwaste Effluent Line Flow Rate Monitor)	-----------------NOTE------------------ Not required during short, controlled outages of liquid effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments, service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. ---	
	G.1 Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>	
	G.2 Estimate flow rate during actual releases.	Immediately <u>AND</u> Once per 4 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action B.1 and referenced in Table 16.11.3-1. (RIA-35, #3 Chemical Treatment Pond Composite Sampler and Sampler Flow Monitor (Turbine Building Sumps Effluent))	---------------NOTE--------------- Not required during short, controlled outages of liquid effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments, service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. --	
	H.1 Suspend release of radioactive effluents by this pathway.	Immediately
	OR	
	H.2 Collect and analyze grab samples for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} µCi/ml.	Immediately AND Once per 12 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action C.1 and referenced in Table 16.11.3-2 for effluent releases from waste gas tanks (RIA-37, RIA-38) or containment purges (RIA-45).	---------------NOTE--------------- Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments, service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. --	
	I.1.1 Analyze two independent samples.	Prior to initiating subsequent release
	AND	
	I.1.2 Conduct two independent data entry checks for release rate calculations	Prior to initiating subsequent release
	AND	
	I.1.3 Conduct two independent valve lineups of the effluent pathway.	Prior to initiating subsequent release
	OR	
	I.2 Suspend release of radioactive effluents by this pathway.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. As required by Required Action C.1 and referenced in Table 16.11.3-2. (Effluent Flow Rate Monitor (Unit Vent, Containment Purge, Interim Radwaste Exhaust, Hot Machine Shop Exhaust, Radwaste Facility Exhaust, Waste Gas Discharge))	-------------NOTE--------------- Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments, service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. --	
	J.1 Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>	
	J.2 Estimate flow rate	Immediately <u>AND</u> Once per 4 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. As required by Required Action C.1 and referenced in Table 16.11.3-2. (4RIA-45, RIA-53)	---------------NOTE--------------- Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments, service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. --	
	K.1 Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>	
	K.2.1 Collect grab sample.	Immediately <u>AND</u> Once per 8 hours
	<u>AND</u>	
	K.2.2 Analyze grab samples for gross activity (beta and/or gamma).	24 hours from collection of sample

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. As required by Required Action C.1 and referenced in Table 16.11.3-2. (Unit Vent Monitoring Iodine Sampler, Unit Vent Monitoring Particulate Sampler, Interim Radwate Building Ventilation Monitoring Iodine Sampler, Interim Radwaste Building Ventilation Monitoring Particulate Sampler, Hot Machine Shop Iodine Sampler, Hot Machine Shop Particulate Sampler, Radwaste Facility Iodine Sampler, Radwaste Facility Particulate Sampler)	----------------NOTE---------------- Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments, service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. --	
	L.1 Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>	
	L.2.1 ----------NOTE---------- The collection time of each sample shall not exceed 7 days. ------------------------------ Collect samples continuously using auxiliary sampling equipment.	Immediately
	<u>AND</u>	
	L.2.2 Analyze each sample.	48 hours from end of each sample collection

CONDITION	REQUIRED ACTION	COMPLETION TIME
M. As required by Required Action C.1 and referenced in Table 16.11.3-2 for effluent releases from ventilation system or condenser air ejectors. (RIA-40)	-----NOTE----- Not required during short, controlled outages of gaseous effluent monitoring instrumentation. Short controlled outages are defined as planned removals from service for durations not to exceed 1 hour, for purposes of sample filter changeouts, setpoint adjustments, service checks, and/or routine maintenance procedures. This guidance may be applied successively, provided that time between successive short, controlled outages is always at least equal to duration of immediately preceding outage. ----------	
	M.1 Continuously monitor release through the unit vent.	Immediately
	<u>OR</u>	
	M.2 Suspend release of radioactive effluents by this pathway.	Immediately
	<u>OR</u>	
	M.3.1 Collect grab sample.	Immediately <u>AND</u> Once per 8 hours
	<u>AND</u>	
	M.3.2 Analyze grab sample for gross activity (beta and/or gamma).	24 hours from collection of grab sample

SURVEILLANCE		FREQUENCY
SR 16.11.3.6	--------------------NOTE-------------------- The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room annunciation occurs if any of the following conditions exist: 1. Instrument indicates measured levels above the alarm/trip setpoint. 2. Circuit failure (downscale only). -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 16.11.3.7	--------------------NOTE-------------------- The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room annunciation occurs if any of the following conditions exist: 1. Instrument indicates measured levels above the alarm/trip setpoint. 2. Circuit failure (downscale only). -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 16.11.3.8	Perform CHANNEL FUNCTIONAL TEST.	92 days

Radioactive Effluent Monitoring Instrumentation
16.11.3

SURVEILLANCE		FREQUENCY
SR 16.11.3.9	-----NOTE----- The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with the National Institute of Standards and Technology (NIST). The standards shall permit calibrating the system over its intended range of energy and measurement. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for these requirements.) ---------- Perform CHANNEL CALIBRATION.	12 months
SR 16.11.3.10	Perform CHANNEL CALIBRATION.	12 months
SR 16.11.3.11	Perform leak test.	When cylinder gates or wicket gates are reworked
SR 16.11.3.12	Perform Source Check.	Within 24 hours prior to each release via associated pathway

Table 16.11.3-1
LIQUID EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS AND SURVEILLANCE REQUIREMENTS

	INSTRUMENT	MINIMUM OPERABLE CHANNELS	APPLICABILITY	SURVEILLANCE REQUIREMENTS	CONDITION REFERENCED FROM REQUIRED ACTION B.1
e.	Keowee Hydroelectric Tailrace Discharge [a]	NA	NA	SR 16.11.3.11	NA
4.	Continuous Composite Sampler				
	#3 Chemical Treatment Pond Composite Sampler and Sampler Flow Monitor (Turbine Building Sumps Effluent)	1	At all times	SR 16.11.3.2 SR 16.11.3.10	H

(a) Flow is determined from the number of hydro units operating. If no hydro units are operating, leakage flow will be assumed to be 38 cfs based on historical data.

Table 16.11.3-2
GASEOUS EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS AND SURVEILLANCE REQUIREMENTS

INSTRUMENT	MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	SURVEILLANCE REQUIREMENTS	CONDITION REFERENCED FROM REQUIRED ACTION C.1
1. Unit Vent Monitoring System				
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Containment Purge Release (RIA-45 - Purge Isolation Function)	1	At All Times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	I
b. Noble Gas Activity Monitor Providing Alarm. (RIA-45 - Vent Stack Monitor Function)	1	At all times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	K
c. Iodine Sampler	1	At All Times	SR 16.11.3.2	L
d. Particulate Sampler	1	At All Times	SR 16.11.3.2	L
e. Effluent Flow Rate Monitor (Unit Vent Flow) (GWD CR0037)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	J
f. Sampler Flow Rate Monitor (a) (Annunciator)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	NA
g. Effluent Flow Rate Monitor (Containment Purge) (PR CR0082)	1	During Containment Purge Operation	SR 16.11.3.2 SR 16.11.3.10	J
h. CSAE Off Gas Monitor (RIA-40)	1	During Operation of CSAE	SR 16.11.3.2 SR 16.11.3.5 SR 16.11.3.8 SR 16.11.3.9	M
2. Interim Radwaste Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor (RIA - 53)	1	At All Times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	K
b. Iodine Sampler	1	At All Times	SR 16.11.3.2	L
c. Particulate Sampler	1	At All Times	SR 16.11.3.2	L
d. Effluent Flow Rate Monitor (Interim Radwaste Exhaust) (GWD FT0082)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	J
e. Sampler Flow Rate Monitor(a) (Annunciator)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	NA

Table 16.11.3-2
GASEOUS EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS AND SURVEILLANCE REQUIREMENTS

INSTRUMENT	MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	SURVEILLANCE REQUIREMENTS	CONDITION REFERENCED FROM REQUIRED ACTION C.1
3. Hot Machine Shop Ventilation Sampling System				
a. Iodine Sampler	1	At All Times	SR 16.11.3.2	L
b. Particulate Sampler	1	At All Times	SR 16.11.3.2	L
c. Effluent Flow Rate Monitor (Hot Machine Shop Exhaust) (Totalizer)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	J
d. Sampler Flow Rate Monitor [(a)] (Annunciator)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	NA
4. Radwaste Facility Ventilation Monitoring System				
a. Noble Gas Activity Monitor (4-RIA-45)	1	At All Times	SR 16.11.3.2 SR 16.11.3.4 SR 16.11.3.7 SR 16.11.3.9	K
b. Iodine Sampler	1	At All Times	SR 16.11.3.2	L
c. Particulate Sampler	1	At All Times	SR 16.11.3.2	L
d. Effluent Flow Rate Monitor (Radwaste Facility Exhaust) (0VS CR2060)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	J
e. Sampler Flow Rate Monitor [(a)] (Annunciator)	1	At All Times	SR 16.11.3.2 SR 16.11.3.10	NA
5. Waste Gas Holdup Tanks				
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RIA-37,-38)[b]	1	During Waste Gas Holdup Tank Releases	SR 16.11.3.1 SR 16.11.3.6 SR 16.11.3.9 SR 16.11.3.12	I
b. Effluent Flow Rate Monitor (Waste Gas Discharge Flow) (GWD CR033)	1	During Waste Gas Holdup Tank Releases	SR 16.11.3.1 SR 16.11.3.10	J

(a)Alarms indicating low flow may be substituted for flow measuring devices.
(b)Either Normal or High Range monitor is required dependent upon activity in tank being released.

1 POINT

QUESTION # 56

073000K3.01 COMMON rsi/gcw 05/08/00 Bank ADM-001

A. Correct, - The conditions provided indicate that a failure has occurred which does NOT meet the requirements of 16.11-3 Condition " J" in the it is NOT a "short controlled outage", however J.2 allows continued release if the flow rate is estimated immediately and once every four (4) hours.

B. Incorrect, - PR-3 is in the RB purge flow path which will be open for the given conditions, and if closed then the release would be terminated.

C. Incorrect, - There is no procedural guidance or requirement to take a redundant containment sample.

D. Incorrect, - This action is not required if the flow rate is estimated immediately and once every four (4) hours.

OBJECTIVES

TERMINAL OBJECTIVE

1. Given a set of conditions, review Selected Licensee Commitment 16.11-3, Radioactive Effluent Monitoring Instrumentation, to determine if any section of the commitment has been violated and make appropriate recommendations for corrective action. (T1)

2. Review Selected Licensee Commitment 16.11-3, Radioactive Effluent Monitoring Instrumentation, and state the basis for each section of the commitment. (T2)

Enabling Objectives

1. Given a copy of the Selected Licensee Commitments, or applicable sections thereof:

   1.1 Given a situation involving inoperable liquid effluent monitoring equipment, recognize those instruments that are required to be operable and which require compensatory actions until repaired. (R1)

   1.2 Determine required compensatory actions for an inoperable liquid effluent instrument. (R2)

   1.3 Recognize that inoperable effluent monitors should be returned to service within 30 days, and that a reporting requirement applies if this is not achieved. (R3)

   1.4 Given a situation involving inoperable gaseous effluent monitoring equipment, recognize those instruments that are required to be operable and which require compensatory actions until repaired. (R4)

   1.5 Determine required compensatory actions for an inoperable gaseous effluent instrument. (R5)

2. Describe the required actions to take if an effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required. (R6)

2.4 Short term inoperability of monitoring instrumentation

A. For certain applicable cases, grab samples or flow estimates are required at frequencies between every 4 hours and every 12 hours upon RIA removal from service. SLC 16.11-3 does not explicitly require operator compensatory action (grab samples or flow estimates) to be initiated immediately upon RIA removal from service, when removal is for the purposes of sample filter changeouts, setpoint adjustments, service checks, or routine maintenance. Therefore, during the defined short, controlled outages, operator action is not required.

B. For the cases in which operator compensatory action is defined as continuous sampling by auxiliary equipment (Table 16.11-2, note (d)), initiation of continuous sampling by auxiliary sampling equipment requires approximately 1 hour. One hour is an accepted reasonable time to initiate, collect and change samples. Therefore, for the defined short, controlled outages (not to exceed l hour), operator action is not required.

Exam Question Report

27-Jan-99

Question

KA: 073000K3.01

Unit 1 is in cold shutdown with the following conditions:

- Reactor Building purge has been in progress for the past twelve (12) hours.
- Investigation by I&E, of a sudden drop in purge flow, indicates that the flow monitor is inoperable.
- Replacement monitor has been ordered and will arrive within the next twenty-four (24) hours.

Per SLC 16.11-3, Radioactive Effluent Monitoring Instrumentation, which ONE of the following actions should be taken concerning the purge release? The release...(.25)

A) may continue if the flow rate is estimated once every four (4) hours.

B) may continue if the position of 1PR-3 is unchanged for the duration of the release.

C) must be stopped until the flow monitor is fully operational.

D) must be stopped until a redundant containment sample can be taken.

Answer

A

Lessons

ID	Description

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

Exam Question Report

27-Jan-99

Question ID:	LP-ADM001	Revision No:	0	Revision Date	10/29/1999
Question Description:	LP-ADM001				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	SLC 16.11-
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:		Max. Point Value:	0.25
		Passing Point Value:	0.25

QUESTION # 57

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**086000 K4.01**	
Importance Rating	**3.1**	**3.7**

Technical Reference(s): **SSS-HPW, AGR SA-3/B7**

Proposed references to be provided to applicants during examination: NONE

Learning Objective: **SSS-HPW OBJ. #4**

Question Source:
Bank # ______
Modified Bank # ______
New __X__

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 57

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Plant fire has occurred in the TB Basement
- TB Basement area sprinklers have actuated
- EWST level = 70,000 gallons decreasing
- HPSW pumps:
  - "A" in BASE
  - "B" in STANDBY

CURRENT CONDITIONS:

- Fire is extinguished
- EWST level = 55,000 gallons increasing

Which ONE of the following is correct?

When EWST level indicates 76,000 gallons and increasing _____ HPSW pump(s) should be secured.

A. "A"

B. "B"

C. "A" and "B"

D. No

1 POINT

QUESTION # 57

086000 K4.01 Both NEW PRA 2-14-00

A. Incorrect – This would be correct if the BASE HPSW pump was stopped at its auto start setpoint but, both HPSW pumps would remain operation until the EWST was full (90,0000 gallons)

B. Incorrect - Level is higher than the STANDBY HPSW pump auto start setpoint but, both HPSW pumps would remain in operation until the EWST was full (normal level) 90,000 gallons.

C. Incorrect - Normal level (90,000 gallons) has not been achieved.

D. Correct –The lowest EWST level during the event was low enough that BASE and STANDBY HPSW pumps would have started. Both HPSW pumps should operate until normal level (90,000 gallons) psi is established.

TERMINAL OBJECTIVES

1. Upon completion of this lesson plan the student will have an understanding of the functions and operation of the HPSW) System during normal and off normal operations. They will also have an understanding of their assigned responsibilities as NLOs during system operations. (T1)

2. Upon completion of this lesson plan the student will be able to explain the operation of the HPSW System during normal and abnormal operations, their assigned responsibilities as ROs/SROs and apply any ITS/SLC Conditions and Required Actions associated with the system. (T2)

ENABLING OBJECTIVES

1. State the purpose(s) of the HPSW system. (R1)

2. Given a copy of the Oconee Flow Diagram, trace the basic flowpath(s) of the HPSW system and locate the following major components: (R2)

   2.1 HPSW pumps

   2.2 the common suction source

   2.3 the fire main loop and major isolation valves

   2.4 Elevated Water Storage Tank and Altitude Valve

3. Describe the major components of the HPSW System listed below: (R4)

   3.1 HPSW pumps

   3.2 the common suction source

   3.3 the fire main loop and major isolation valves

   3.4 Elevated Water Storage Tank and Altitude Valve

4. Describe the control function(s) of the HPSW Auto-Initiation Circuit based on changes in the EWST Level. (R5)

5. Identify (9) nine non-fire related functions performed by the HPSW system. (R6)

6. Describe each of the four-(4) sprinkler systems, including the type of sensor, spray header, and method of actuation. (R7)

7. Explain the basis for the Critical Action Steps of the following NLO JPMs associated with the HPSW System: (R21)

   7.1 NLO-038, Align Sprinkler System to the Equipment Room During a Fire

G. Auto Initiation Logic

1. Provides auto start/stop capabilities for the safety-related HPSW Pumps based on EWST level.

2. The relationship of EWST level Vs Auto function:

LIGHT	INDICATION	VOLUME	AUTO FUNCTION
"RED"	Overflow	100,000 Gals.	
"YELLOW"	Full	90,000 Gals.	Auto Stop HPSW Pump(s)
"WHITE"	Low	70,000 Gals.	Auto Start "Base" Pump
"RED"	EWST Emer Low	60,000	Auto Start "Stby" Pump

3. Loss of Auto Initiation Logic

   a) A loss of the Site Assembly Air Horn air supply will cause loss of HPSW control room EWST level indications, Auto Initiation Logic and associated alarms. (Refer to the Alarm Response Guide for 1SA3/B-7 / Site Assembly Horn Air Failure)

   b) The Auto Initiation Logic is a Fire Suppression function. The HPSW Pumps can still be Manually started from the Control Room. The altitude valve is totally hydraulic with no electrical controls.

   c) Backup indication of the EWST Level is located outside Unit #1 control room under Unit 1 UST Mezzanine.

      1) Heise Gage is read in psi. Conversion to gallons located on the gage is as follows:

         71.63 psi = 60,000 gals REFERENCE: O-422H-14

         72.5 psi = 70,000 gals

         74.52 psi = 90,000 gals

         76.03 psi = 100,000 gals

   d) If AUTO start/stop capability is lost, Operations should open relief valves 3HPSW-557 and 558 to help prevent potential deadheading of an HPSW pump.

      NOTE: The altitude valve will close upon EWST refill, thus preventing the EWST overflow line from providing minimum flow protection.

SITE ASSEMBLY ALARM

1. Alarm Setpoint

1.1 80 psig decreasing.

2. Automatic Action

None

3. Manual Action

3.1 AUTO start of the HPSW pumps is defeated. (Air is lost to EWST Level Transmitter).

3.2 Monitor Heise Gauge outside Unit 1&2 Control Room to determine EWST level and operate HPSW pumps as necessary.

3.3 Refer to SLC 16.9.1 (Fire Suppression Water System).

4. Alarm Sources and References

4.1 Pressure Switch PS-147.

4.2 OEE-23.

QUESTION # 58

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**2**	**2**
K/A #	**086000 A4.03**	
Importance Rating	**3.5**	**3.4**

Technical Reference(s): **OP/1/A/6101/003**
1SA-03 / B-6

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-FDS OBJ. #5**

Question Source:
Bank # **IC-170**
Modified Bank # ______
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 58 ?

~~Plant~~ unit 2 plant conditions:

- Power = 100%
- A fire alarm on the Honeywell system is actuated

Which ONE of the following is utilized to determine where an NLO should be dispatched to investigate the problem?

A. System display indicates the plant location code

B. Information from zone indicating unit ? is this utilized at all?

C. Audible alarm at the detector

D. Fire alarm response guide

1 POINT

QUESTION # 58

086000 A4.03 common rsi/gcw Bank IC-170

A. Correct, The Operator should use the PREVIOUS/NEXT switch to scroll through all alarms to determine the affected areas or detectors.

B. Incorrect, Zones apply only to general areas and do not specify the specific location.

C. Incorrect, There are no audible alarms at the detectors -- only at the fire alarm cabinet.

D. Incorrect, The Fire alarm response guide (2SA03 / B-6) does NOT provide possible fire locations.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

1. Describe the basic design and operation of the fire detection systems used at Oconee Nuclear Station.
2. Describe proper response of a non-licensed or a licensed operator following the receipt of a fire alarm.

ENABLING OBJECTIVES

1. State the purpose of the plant, SSF and KHS fire detection systems. (R1)
2. Briefly describe the two types of detectors utilized in the ONS fire detection system, including: (R2)
   - 2.1 The advantages and disadvantages of the ionization type fire detectors
   - 2.2 The operation of the neon bulbs located in the ionization type fire detectors.
3. Briefly describe the design and operation of the plant fire detection system. (R3)
4. List the general types of areas in the plant that are covered by the plant fire detector system. (R4)
5. Briefly discuss the proper actions of the operator sent to investigate a fire alarm on the plant fire detector system. (R5)
6. Discuss briefly the proper actions of the control room operator should the fire alarm annunciator in the control room sound. (R6)
7. State the location of the SSF Fire Alarm Control Unit. (R7)
8. State the eight zone locations monitored by the SSF fire detection system. (R8)
9. Briefly describe the design and operation of the SSF fire detection system. (R9)
10. Describe how to determine the location of a fire from the SSF Fire Alarm Control Unit. (R10)
11. Briefly describe the design and operation of the Keowee Hydro Station fire detection system. (R11)
12. Evaluate the Fire Suppression/Detection System SLC operability requirements and demonstrate proper compliance with the SLC. (R12)

1) Reactor Building (RCP oil flash point is below RCS temperature)
2) Operator Kitchens
3) Laundry
4) MTOT

c) Is Normally Unoccupied

1) Chemical storage
2) Storage areas.

d) Areas specifically required to be monitored by NRC regulations.

E. Operator Alarm Response

1. Fire Alarm is received in the control room via 1(3) SA-03 / B-6, FIRE ALARM.

   a) No automatic actions associated with the alarm actuation itself.

2. Monitor the alarm cabinet to determine the cause of the alarm.
3. Push the ACKNOWLEDGE switch at the fire alarm cabinet to silence the audible alarm.
4. If more than one alarm or trouble condition exists, use the PREVIOUS/NEXT switch to scroll through all alarms to determine the affected areas or detectors.
5. If one or more detectors are in alarm, dispatch an operator to survey the immediate area to determine the cause of the alarm.

NOTE: The operator should be aware that since the system uses particle of combustion detectors, there may not be "visible" smoke if a fire exists (in the incipient phase). Therefore, if no cause for the alarm is readily apparent, someone should remain in the area for sufficient time to assure that no fire exists.

6. If a fire does exist, the operator will notify the Control Room and/or perform actions to mitigate the situation that can be safely accomplished.
7. Control Room Operator Action after local status report of a fire / valid alarm:

   a) Notify the Control Room Supervisor.
   b) Refer to RP/0/A/1000/01 (Emergency Classification Procedure).
   c) Refer to the Fire Plan for specific information concerning the fire location.

d) The Control Room Supervisor should dispatch the fire brigade if the situation dictates. Situations such as multiple reports from individuals, multiple alarms, or other system indications (affected area equipment problems) are some examples where activation may be appropriate.

1) activate the fire brigade using the plant paging system.

e) Establish contact with the fire brigade by phone or radio, as required, to relay information.

f) Conduct unit operations per instructions from the Control Room Supervisor.

8. If a fire exists in the Reactor Building

a) Shutdown of the reactor may be necessary to gain access to an area that is off limits during operation.

b) For oil fires involving stainless steel, use only CO_2 extinguishers.

c) Oil fires generated by RCS surface temperatures may require the RCS to be cooled below the flash point (425°F) if the oil leak and/or fire cannot be secured.

9. If the alarm is not valid, push the RESET switch to clear the alarm.

10. If one or more trouble condition exists, or an invalid alarm cannot be reset:

a) Notify I&E to repair the affected detector(s).

11. If I&E cannot immediately repair the cause of the alarm:

a) Declare the affected detector(s) inoperable and record in OP/0/A/1102/006, R&R of Station Equipment

b) Verify the requirements of Selected Licensee Commitment 16.9 are met. (Consult the system engineer for Appendix R equipment.)

c) Have I&E lockout the affected detectors.

B-6

FIRE ALARM

1. Alarm Setpoint

None

2. Automatic Action

None

3. Manual Action

3.1 Monitor fire alarm cabinet in rear of control room to determine cause of the alarm.

3.2 Push the ACKNOWLEDGE Switch at the fire alarm cabinet to silence the audible alarm.

3.3 **IF** more than one alarm or trouble condition exists,

THEN press the PREVIOUS/NEXT Switch to scroll through all of the alarms on the display unit to determine the affected areas or detectors.

3.4 **IF** one or more detectors are in the alarm condition,

THEN dispatch operator(s) to the affected area(s) to determine validity of the alarm(s).

3.4.1 **IF** the alarm(s) is valid,

THEN perform the following:

- Notify the Shift Supervisor
- Refer to the Fire Plan
- Refer to RP/0/A/1000/01 (Emergency Classification Procedure)

3.4.2 If a fire exists inside the Reactor Building:

- Shutdown the reactor if it is necessary to gain access to an area that is off limits during reactor operation.
- For oil fires involving stainless steel, use only CO_2 Fire Extinguishers.

B-6

- For oil fires generated by RCS surface temperatures, the RCS should be cooled down below the flash point (425°F) if the oil leak and/or fire cannot be secured.

3.4.3 **IF** the alarm(s) is not valid,

THEN push the RESET Switch to clear the alarm.

3.5 **IF** one or more detectors are in the trouble condition,

OR an invalid alarm condition cannot be reset,

THEN notify I&E to investigate and repair cause of the alarm(s).

3.6 **IF** I&E cannot immediately repair cause of the alarm(s),

THEN perform the following actions:

3.6.1 Declare the affected detector(s) inoperable and record in OP/0/A/1102/006 (Removal and Restoration of Station Equipment).

3.6.2 Verify the requirements of Selected Licensee Commitment 16.9 are met.

3.6.3 Have I&E lockout the affected detectors.

4. Alarm Sources and References

4.1 Any detector alarm condition.

4.2 Any detector trouble condition.

4.3 Reflash Module in Fire Alarm Cabinet.

4.4 IP/1&2/A/250/5A (Fire Detection System Accessible Detector Functional Test).

4.5 IP/0/A/250/5B (Fire Detection System Inaccessible Detector Functional Test).

Exam Question Report

27-Jan-99

Question

KA: 086000A4.03

Assume you have received a fire alarm on the Honeywell system (located in either the Unit 1 & 2 or the Unit 3 control rooms). Which ONE of the following allows the operator to know the location to send an operator to investigate the problem? (.25)

A) System printout indicates the plant location code.

B) Information from zone indicating unit.

C) Audible alarm at the detector.

D) Fire alarm response guide.

Answer

A

Lessons

ID	Description
IC-FDS	Fire Detection System IC-FDS

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

Exam Question Report

27-Jan-99

Question ID:	IC170	Revision No:	0	Revision Date	10/29/1999
Question Description:	IC170				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	IC-FDS - Fire Detection System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 4; SRO = 4	Max. Point Value:	0.25
		Passing Point Value:	0.25

QUESTION # 59

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**3**	**3**
	K/A #	**00500 K2.01**	
	Importance Rating	**3.0**	**3.2**

Technical Reference(s): **PNS-LPI**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-LPI 2**

Question Source:
- Bank # ________
- Modified Bank # ________
- New ___X___

Question History:
- Previous NRC Exam ________
- Previous Quiz / Test ________

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X___
- 55.43 _____

Comments:

1 POINT

QUESTION # 59

Unit 1 plant conditions:

INITIAL CONDITIONS:

- MODE 5
- "1A" LPIP is in service

CURRENT CONDITIONS:

- E1 MFB1 STARTUP FDR breaker opens due to an internal fault

Which ONE of the following describes the LPI Pumps available for core cooling?

A. A and B

B. A and C

C. B and C

D. A, B and C

1 POINT

QUESTION # 59

005000 K2.01 BOTH NEW RSI/GCW 04/28/00

A. Incorrect – If 1TE were de-energized instead of the E1 MFB1 STARTUP FDR breaker this would be correct, as C LPIP would not be available.

B. Incorrect – If 1TD were de-energized instead of the E1 MFB1 STARTUP FDR breaker this would be correct, as B LPIP would not be available.

C. Incorrect – If 1TC were de-energized instead of the E1 MFB1 STARTUP FDR breaker this would be correct, as A LPIP would not be available.

D. Correct - A, B and C LPIP are fed from 1TC, 1TD and 1TE respectively. When Breaker E1 opens all three buses are still available because Breaker E2 is closed and providing power to 1TC, 1TD and 1TE.

TRAINING OBJECTIVES

Terminal Objective:

1. Describe the purpose, location of the major equipment and the various modes of operation, and precautions related to the LPI System. Recognize the major power supplies associated with this system. (T01)
2. Define the purpose, the various modes of operation, and precautions related to the LPI System. (T02)
3. Recognize and respond to various degraded LPI Operating Conditions.(T03)
4. Properly operate the LPI System in both the normal and emergency injection modes by remaining familiar with and following the requirements and information contained in the Low Pressure Injection System Operation Procedure. (T04)

Enabling Objectives:

1. State the six major functions of the LPI System. (R1)
2. Recognize that the LPIPs are powered from the 4160V Switchgear (TC, TD, and TE) on the associated Unit. (R2)
3. State the building and the room in which the LPI pumps are located. (R3)
4. State the cooling medium used to remove heat from the LPI System via the LPI Coolers. (R4)
5. Describe the function of the following: (R5)
   - 5.1 LPI Coolers
   - 5.2 BWST
   - 5.3 RBES
6. State the major design differences between the LPI systems at Oconee. (R6)
7. Compare the major design differences between the LPI systems at Oconee, to include the reasons for the differences. (R7)
8. Recognize that with the LPI System in operation in the Normal Decay Heat Removal Mode, and with the HPI System secured, the normal purification demineralizers can be placed in service through the LPI System for RCS chemistry control. (R19)
9. Identify the method used to provide RCS chemistry control when the LPI System is in the Normal Decay Heat Removal Mode and the HPI System is secured. (R20)

10) Statalarm SA3 LP Decay Heat Loop B Flow Low

11) Statalarm SA3 LP Injection Loop A Flow High/Low

12) Statalarm SA3 LP Injection Loop B Flow High/Low

d) Power for the non-safety grade outputs comes from KC.

2. The two non-safety grade transmitters are Rosemount electric transmitters.

a) LPIFT004A feeds B LPI Flow to computer point.

b) LPIFT005A feeds A LPI Flow to computer point.

3. Since these flow instruments are electric, they will not be affected by a loss of instrument air.

4. Individual trains of the Low Pressure Injection System will be considered inoperable when either train's safety grade LPI flow instrument is out of service. If the safety grade LPI flow instrument is lost, the required action statement of ITS 3.5.3 must be entered.

2.4 Power Supplies:

A.	A LPI Pump	1TC	2TC	3TC
B.	B LPI Pump	1TD	2TD	3TD
C.	C LPI Pump	1TE	2TE	3TE

2.5 System Operations

A. Normal Operations

1. LPI Design Differences at Oconee

a) Units 1 and 2 have the High Pressure Mode available through the "A" Decay Heat Removal Cooler to allow use of either cooler above 125 psig Reactor Coolant pressure.

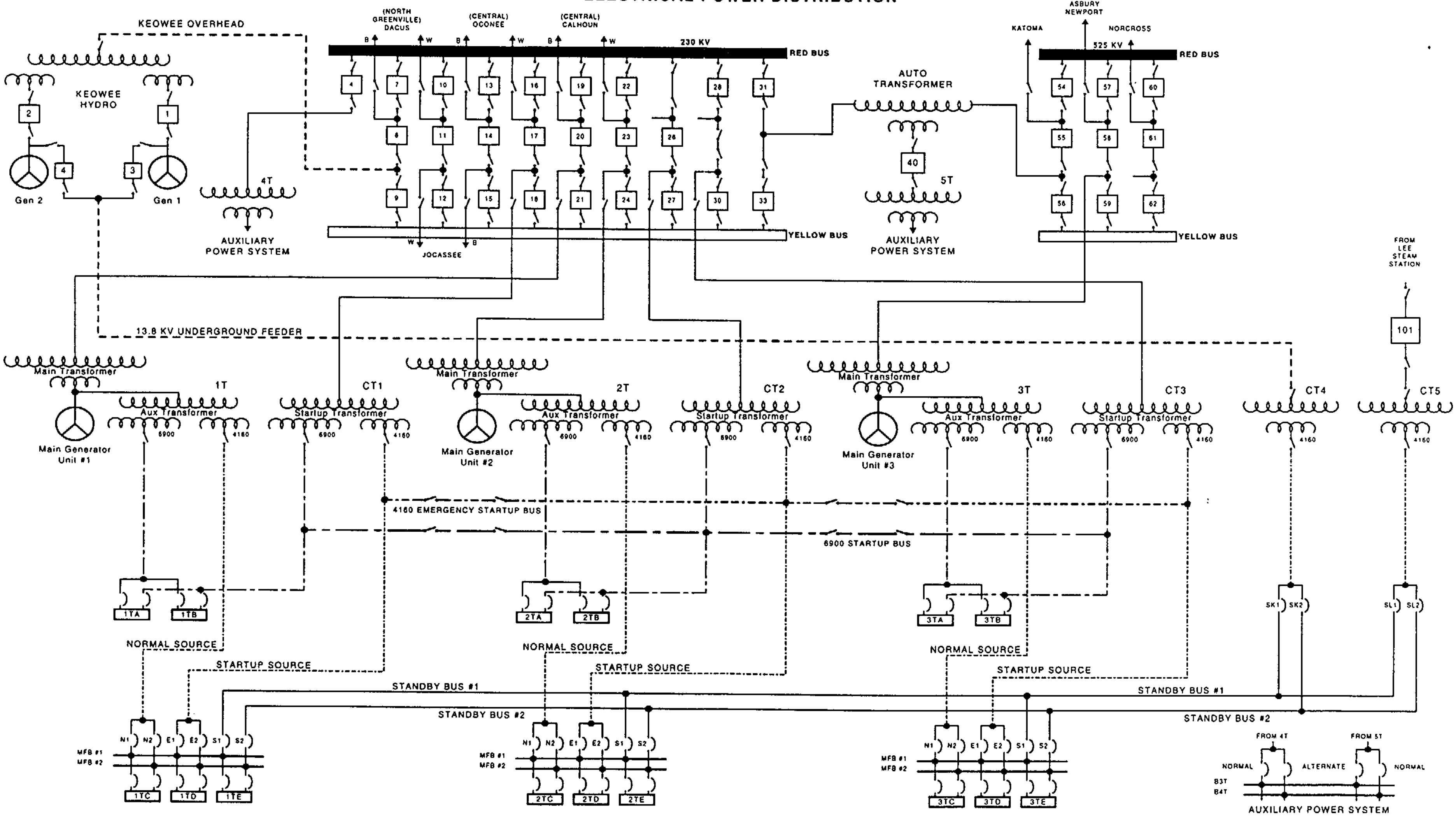
b) Units 1 and 2 have a Switchover (S.O.) Mode of operation. Unit 3 does not require switchover, due to the increased design operating pressure of the Decay Heat Removal Coolers.

c) Units 1 and 2 use valve (1)(2)LP-4 for switchover and high pressure mode operations. Unit 3 does not require valve LP-4 in the Decay Heat Drop Line.

d) Unit 1 & 2 do not have cooler bypass valves LP-92 & LP-93.

e) Unit 1 and 2 have electric cooler inlet valves (1)(2)LP-11 and 12. Unit 3 LPI Cooler Inlet Valves 3LP-11 & 3LP-13 are manual.

ELECTRICAL POWER DISTRIBUTION



OP-OC-EL-EPD	TRAINING USE ONLY	APPROVED BY-
REF. DRAWING BY RJL		

QUESTION # 60

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**3**	**3**
K/A #	**007000 G2.2.3**	
Importance Rating	**3.1**	**3.3**

Technical Reference(s): **PNS-CS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-CS OBJ. #7**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 60

All three Oconee Unit's Quench Tank levels are above the high alarm setpoint and will be pumped to the BHUTs using the Component Drain Pump during the shift.

Which ONE of the following describes the expected radiation total dose received by the ~~operators~~ NLOs involved in each unit's pumping evolution?

The total dose for...

A. all three units will be the same.

B. Unit 1 will be higher.

C. Unit 2 will be higher.

D. Unit 3 will be higher.

1 POINT

QUESTION # 60

007000 G2.2.3 COMMON rsi/gcw 05/09/00 NEW

A. Incorrect, If all three Units Quench tanks were pumped from the Control Room this distracter would be correct.

B. Correct, The flow path for pumping the Quench tank to the BHUT requires opening CS-20. On Unit 1, CS-10 and CS-20 are manual valves, located in the HPI room. On Units 2 and 3, they are remotely operated from the control room. Manually opening CS-20 will result in some dose that will increase the total dose for the Unit 1's NLO.

C. Incorrect, If Unit Two's was not pumped from Control Room this distracter would be correct.

D. Incorrect, If Unit three's was not pumped from Control Room this distracter would be correct.

7.2 Limits on QT pressure and level, the reasons for these limits, and how QT pressure and level are regulated

7.3 The primary purpose for the QT Cooler and how it is placed in operation

7.4 The general list of components that discharge to the QT

8. List the six major uses for the Component Drain Header and Component Drain Pump. (R8) (PNSCS008)

9. Briefly describe the test method of the: (R9) (PNSCS009)

   9.1 PT/0/A/150/53, Coolant Storage System Leakage Test

   9.2 PT/0/A/251/08, CS-73 Functional Test

10. Describe the basic operation of the Deborating Demineralizers to include: (R10) (PNSCS010)

    10.1 Primary purpose.

    10.2 When the Deborating Demineralizers are used.

11. Briefly describe the general purpose of the Component Drain Header of the Coolant Storage System.(R11) (PNSCS011)

12. Describe the extra precautions relating to CBAST operation due to the highly concentrated boric acid solutions contained in the tank. (R12) (PNSCS012)

13. Evaluate the overall affect on plant operations from the loss of the: (R13)

    13.1 Bleed transfer pumps

    13.2 Bleed holdup tanks

    13.3 CBAST

14. For PT/0251/003, CBAST Pump Test, describe: (R14)

    14.1 The purpose

    14.2 How the test is performed

1. The QT Drain Pump is used to lower QT level by pumping it directly back to the LDST. The Component Drain Pump lacks sufficient discharge head to be able to do this.

2. located in HPI room

G. Component Drain Pump (and Header)

1. Pump (located in the HPI room), performs the following functions;

a) Provides for draining the RC Loops and the primary side of the Steam Generators for refueling operations.

b) Provides for draining the Core Flood Tanks to allow for maintenance on the tanks.

c) Can be used to pump out the Quench Tank to the BHUT if the level increases or for maintenance of the tank. On Unit 1 CS-10 and CS-20 are manually operated valves, while on Unit 2 & 3 CS-10 and 20 are remotely operated from the control room.

d) Allows draining the Secondary side of the Steam Generators in the event of SG Tube leak resulting in high activity water that must be disposed of.

e) Allows recirculating the Quench Tank through the QT cooler to cool the Quench Tank contents after PZR Relief Valves have lifted.

f) Used during Pressurizer cooldown to pump water from Surge Line drains and for Pressurizer heater maintenance.

NOTE: The effluents drained may be pumped to the RC Bleed Holdup tanks or the Miscellaneous Waste Holdup Tank.

H. Deborating Demineralizers (See OC-PNS-CS-3 & 8)

QUESTION # 61

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**2**	**2**
Group #	**3**	**3**
K/A #	**041000 K5.01**	
Importance Rating	**2.9**	**3.2**

Technical Reference(s): **EAP-E24**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E24 OBJ. #5**

Question Source:
Bank # ______
Modified Bank # **EAP-263**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test __X__

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 61

Unit 1 plant conditions:

INITIAL CONDITIONS:

- Power = 100%

CURRENT CONDITIONS:

- 1A OTSG leak rate = 450 gpm
- Cooldown to 532°F in progress

Which ONE of the following is correct?

The primary reason the RCS is initially cooled down to 532°F per EOP Section 504, "SG SG Tube Leak" is...

A. to conserve BWST inventory.

B. to prevent the MSRVs from lifting.

C. because it will minimize the flow rate through the tube rupture.

D. because 532°F is the saturation pressure for the lowest MSRV setpoint.

1 POINT

QUESTION # 61

041000K5.01 common GCW 05/08/00 Bank EAP-263 (modified)

A. Incorrect, the cooldown later in section 504 is to conserve BWST inventory but not at the beginning of the section.

B. Correct, saturation pressure for 532°F is below the pressure for the lowest MSRV setpoint.

C. Incorrect, SCM is reduced to minimize the flow rate through the tube rupture.

D. Incorrect, 532°F is BELOW the saturation pressure for the lowest MSRV setpoint.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

Describe the use of Section 504 (SG Tube Leak) of the Emergency Operating Procedure in order to perform the required actions during an event involving a primary to secondary leak greater than Tech. Spec. limits. Be able to discuss Section 504 procedure steps and their bases in an oral or written format. Discuss in an overview format how Section 504 mitigates a SGTR event and places the plant into MODE 5 with the affected SG(s) isolated and heat removal via LPI.

ENABLING OBJECTIVES

1. Using an overview format describe the intent of this procedure. (R1)
2. Given a set of conditions, be able to identify a OTSG with a tube leak. (R2)
3. Explain why it is important to place the TBVs in hand just prior to manually tripping the reactor. (R3)
4. During a SGTL shutdown explain the importance of maintaining PZR levels $\geq$ 220, $\geq$ 200, and > 80 inches at different times during the shutdown and cooldown to 532°F. (R4)
5. Explain how the affected SG is isolated and why it is not done until an RCS temperature of 532°F is reached. (R5)
6. Explain the reason for maintaining the subcooled margin as close as possible to 0°F during the cooldown. (R6)
7. Given a set of conditions determine the proper Cooldown Plateaus and state the bases behind the specified plateaus. (R13)
8. Given a situation understand that throttling FDW flow may be the only method for controlling the cooldown rate when feeding a SG with a steam leak. (R7)
9. Describe the reason for the increased concern over available BWST inventory during a SGTR event if the cooldown rate is not properly maintained. (R8)
10. Given a set of plant conditions calculate and explain the basis for the limit(s) for SG tube-to-shell ΔT. (R9)
11. Explain how an uncontrolled release to the environment can possibly result if the MS lines are allowed to fill with primary water from the affected SG. (R10)

B. MS to SSRH – MS-79 or MS-76

C. MS Supply to TDEFWP – MS-82 or MS-84

D. FDWPT MS Supply – MS-35 or MS-36

23.4 Ensure Air Ejectors are on Aux Steam Header by opening AS-40 and closing MS-47

23.5 Ensure the following are closed: SSH-1, SSH-3, and SSH-9

NOTE 24:	• *Cool down using Tc with RCPs or CETCs without RCPs.*

24. •Initiate RCS cool down to 525°F - 532°F

24.1 Limit cooldown rate only by the ability to maintain PZR level > 80 inches.

- **REFER TO** IT #4 "SGTL PZR Level Control".

INSTRUCTOR NOTE: Cooldown to 532°F must be commenced using ADVs within 40 minutes of recognizing a SGTR for the Design Bases Event.

A. The saturation pressure at 532°F is well below the lowest Main Steam Relief Valve setpoint of 1050 psig (550°F Tsat). Therefore, the affected steam generator can be completely isolated when the RCS is cooled to 532° F.

B. The cooldown rate to 532° F will only be limited by the ability to maintain PZR level > 80 inches (via IT #4). The pressurizer level should be kept above 80 inches by HPI injection and by limiting the cooldown rate. This will ensure that PZR heaters are still operable if needed and no unexpected loss of subcooled margin occurs so that the RCPs can be kept in operation.

C. This cooldown plateau provides an adequate SDM and that can be attained without additional RCS boron concentration analysis being determined.

D. This cooldown plateau (532°F) is the RCS temperature that the operator can cool the RCS down to without first having a shutdown margin calculation performed by reactor engineering.

E. When a subsequent RCS boron concentration analysis and shutdown margin calculation allows, the cooldown can continue below the 532° F.

NOTE 25:	Disabling the MSLB Isolation circuit will allow the TDEFWP to start if selected to "AUTO" and an automatic start signal exists.

25. Disable both trains of the MSLB Isolation circuit:

Exam Question Report

27-Jan-99

Question ID:	EAP263	Revision No:	0	Revision Date	10/29/1999
Question Description:	EAP263				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-E24 - SG Tube leak
Last Used Date:	01/28/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	0
Inactive Comment:	LRO = 1; SRO = 1	Max. Point Value:	0.25
		Passing Point Value:	0.25

Question

WHICH ONE (1) of the following is the PRIMARY reason the RCS is initially cooled down to 532 degrees F. following diagnosis of a Steam Generator Tube Rupture per EOP 504, "Tube Rupture"?

A) To prevent MSRVs from opening.

B) To prevent Pressurizer PORV from opening.

C) To minimize the flow rate through the tube rupture.

D) To minimize loss of Subcooing Margin to maintain 30 degrees or more.

Answer

A - correct, Psat for 532F is < MSRV lift setpoint.
B - incorrect, no connection
C - incorrect, SGTR flow rate dependent on Pri- Sec dP
D - incorrect, no connection

Lessons

ID	Description
EAP-E24	SG Tube Leak EAP-E24

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

QUESTION # 62

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**2**	**2**
	Group #	**3**	**2**
	K/A #	**103000 A4.01**	
	Importance Rating	**3.2**	**3.3**

Technical Reference(s): **PNS-PRV**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PRV #13**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 62 WINDOW DRESSING

Unit 2 plant conditions:

- A LBLOCA has occurred
- ES Channel 1 - 8 actuation
- IA and AIA pressure = 0 psig

Which ONE of the following is correct concerning

The purpose of the travel stops installed on 2PR-13 and 17 (A&B Penetration Room Filter Outlet)...

A. ensures adequate PRV flow during ES actuation. when ~~valves fail close~~ instrument air is lost.

B. prevents excessive PRV flow during ES actuation.

C. maintains proper PRV flow to prevent PRV system PAC filter channeling.

D. ~~regulates~~ maintains adequate PRV flow to eliminate charcoal filter loading and ignition ~~during low flow operations.~~

stops don't regulate!

1 POINT

QUESTION # 62

103000 A4.01 Both PRA 05-14-00

A. Correct – During a Loss of IA event the PRV filter outlet valves fail closed (Loss of IA). Travel stops are installed on the PRV filter outlet valves to keep them properly positioned (100 cfm/train) during a loss of IA and ES.

B. Incorrect – most normal application of travel stops prevents valve movement in the open direction therefore preventing excessive flow. i.e. HP-120 during LTOP operation.

C. Incorrect – Filter channeling is not a problem with the PRV filters.

D. Incorrect – Due to engineering evaluations charcoal ignition/burn due to filter loading is no longer a concern with the PRV filters at Oconee. With no flow through the filter natural circulation is adequate to prevent charcoal filter burn.

TRAINING OBJECTIVES

TERMINAL OBJECTIVE

T1 Describe the operation and purpose of the Penetration Room Ventilation System.

ENABLING OBJECTIVES:

1. Draw a basic flow diagram of the Penetration Room Ventilation System. (R1)
2. State the purpose of the Penetration Room Ventilation System. (R2)
3. List the types of filters used in the Penetration Room Ventilation System. (R3)
4. Explain the advantages of the external carbon sampling canisters. (R4)
5. State the purpose of the charcoal filter. (R5)
6. Explain the reason for the maximum flow limit through the Penetration Room Ventilation System. (R6)
7. State the purpose of PR-13 and PR-17, A & B filter outlet valves. (R7)
8. Explain the two purposes of PR-20, PRV fan suction cross-connect valve. (R8)
9. Describe the operation of the Penetration Room Ventilation System following Engineered Safeguards actuation. (R9)
10. State the reason for the high humidity limit in the penetration rooms and any associated actions. (R10)
11. Briefly describe the requirements for opening a Penetration Room floor drain while containment integrity is required. (R11)
12. Briefly describe the requirements for draining a system to the Penetration Room floor drains. (R12)
13. Describe the purpose of the travel stops installed on PR-13 and PR-17, including when they should be adjusted. (R13)
14. Describe two circumstances when it may be necessary to periodically adjust Penetration Room Ventilation flow. (R14)
15. Explain the various means available to the operator in the control room and locally to identify degraded Penetration Room Ventilation flow. (R15)

2) The purpose of having these sample canisters is to satisfy the Tech. Spec. sample frequency for the carbon filters without having to disassemble the entire filter container. The Tech. Spec. sample frequency is:

(a) on a refueling frequency

(b) every 720 hours of operation

(c) after painting, fire, or chemical release in any ventilation zone communicating with the system.

Thus, these sample canisters prevent taking the main filter out of service for any extended periods and they also minimize having to leak test the filter housing, which must be done each time the housing is disassembled.

B. Penetration Room Ventilation Fans

C. Each Unit's penetration room is provided with two fans. Each PRV train contains one fan. Each fan is a direct drive, constant speed, single inlet design. Each fan is designed for 900 to 1000 CFM of air flow and a pressure of -11" H_2O.

D. Penetration Room Ventilation Valves

1. PR-12 and PR-16 - Filter A & B manual inlet valves

   Normal position is locked open.

PKI GRW 7-12-99

2. PR-13 and PR-17 - Filter A & B pneumatic outlet valves

   Normal position is throttled per PT/0/A/170/05 to 1000 CFM. This positioning assures flow through each filter train during a loss of IA since PR-13 and PR-17 fail "as is" on a loss of IA due to installation of mechanical travel stops on the valve stems. These valves are remote manual valves and must be periodically adjusted, during operation, to offset the effects of increased leakage and filter loading. Travel stops **should be reset** anytime flows are increased.

PR-13/17 fail closed on loss of IA, but the travel stops prevent full closure.

3. PR-14 and PR-18 - Fan A & B manual inlet valves

   Normal position is locked open.

4. PR-34 and PR-35 - Fan A & B discharge check valves

   These check valves prevent recirculation flow in the event of a single fan failure.

5. PR-15 and PR-19 - Fan A & B motor operated discharge valves

   These valves are electrically operated butterfly valves which open when its respected fan starts. This valve will close to prevent recirculation if one fan fails.

6. PR-20 - PRV fan pneumatic suction tie

   This remotely operated valve is normally closed but can be opened from its remote manual station to:

   a) maintain adequate cooling air through an idle filter train

      Analysis completed per PIR 4-090-0057 indicates that natural circulation around the filter will provide adequate cooling to prevent carbon ignition.

   b) utilize an idle filter train.

E. Penetration Room Ventilation Dampers

1. These normally closed self-actuating dampers are located in the inlet of each PRV train to prevent moisture from being carried into the PRV system filters by natural circulation.

2. A minor mod (#OE-6565) assigned numbers to these valves and put them on flow diagrams. They are # 115 (A Train) and 116 (B Train).

2.2 Penetration Room Description

A. The penetration rooms are formed adjacent to the outside surface of each reactor building to enclose the area around the majority of the penetrations into the containment building. The Penetration Room floor drains are equipped with loop seals, to allow the maximum negative pressure in the penetration room to be obtained, while allowing the floor drains to remain open.

B. The penetrations that do not pass through the penetration rooms are:

1. Main Steam Lines:

   The Main Steam Lines are not considered a source of significant leakage because they are welded to the liner plate.

2. Personnel and Emergency Personnel Hatches:

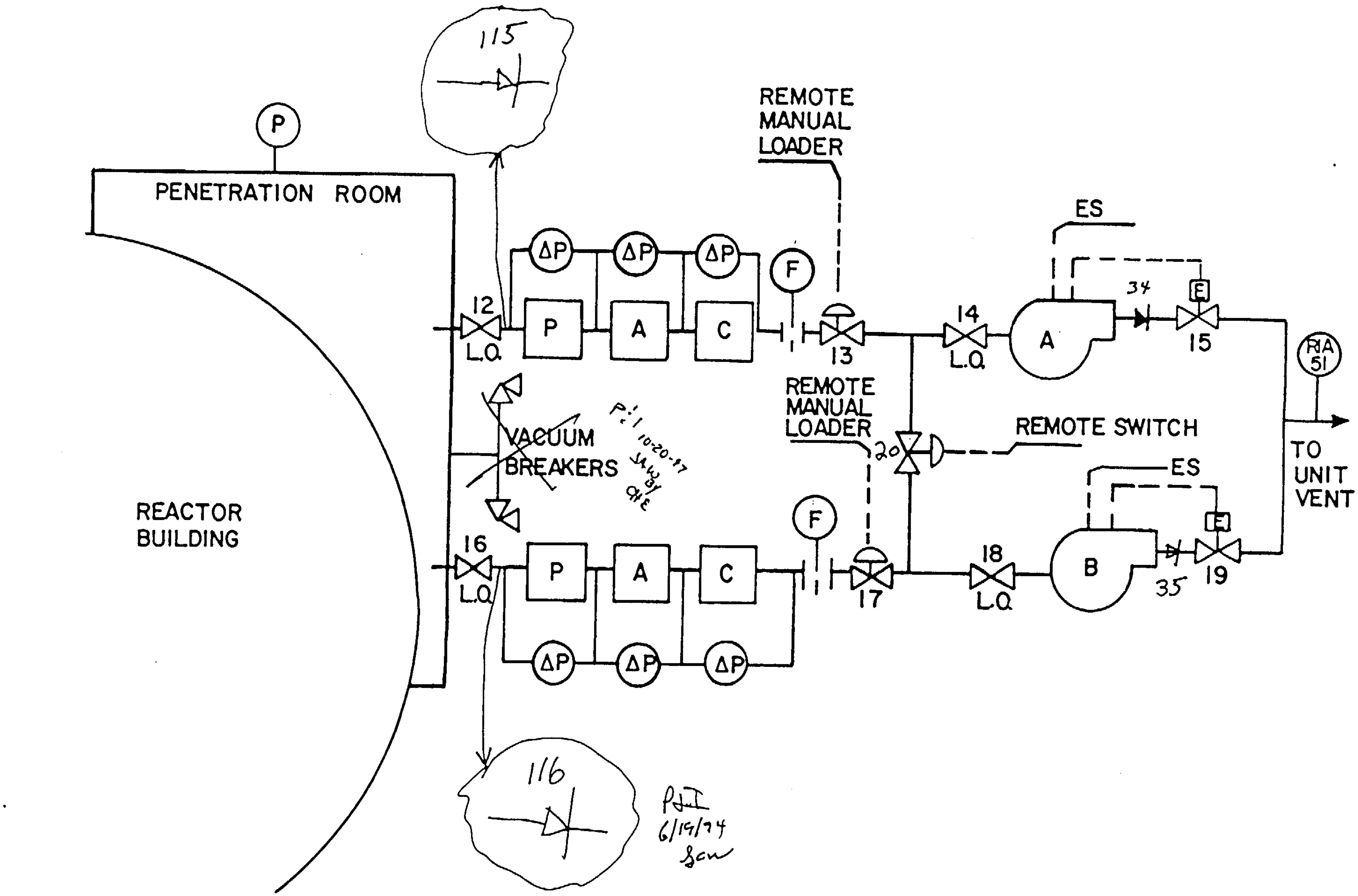
   These access openings can be tested during normal operation and are not considered significant sources of leakage. There are double seals at each of these openings, and the space between these double seals is connected to the penetration room.

3. Permanent Equipment Hatch:

   This hatch is closed and sealed by a double gasket closure.

4. Embedded Lines:

   These lines include the normal sump drain, the emergency sump drain, and the decay heat lines (Units 2 and 3 only).



TITLE:	NOTES:		
PENETRATION ROOM VENTILATION SYSTEM	PENETRATION ROOM VENTILATION SYSTEM FLOWPATH	ID NO OC-PNS-PRV-1	DATE: 5-27-88
		REF. F.S.A.R	
		DRN. BY: DMC/ms	APR. BY:
		TRAINING USE ONLY	

QUESTION # 63

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**T3**	**T3**
	Group #	**C1**	**C1**
	K/A #	**G2.1.17**	
	Importance Rating	**3.5**	**3.6**

Technical Reference(s): **SF-164**
NSD-509

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **SF-164 #5**

Question Source:
- Bank # ______
- Modified Bank # ______
- New ___X___

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 63

Which ONE of the following alarm conditions will the RO verbally identify as an EXPECTED alarm per NSD 509?

A. Channel "A" RCS High Temperature alarm during Channel "A" RPS On-Line testing.

B. NI Calibration Error alarm received for the first time during power decrease.

C. CFT Pressure Low alarm that occurs twice each shift due to small Nitrogen leak.

D. RPS Channel Manual Bypass alarm received while an HLP student is performing OJT and crew is aware of training in progress.

1 POINT

QUESTION # 63

G2.1.17 PRA/5-8-00

A. Correct – This is an operational situation that the operating crews use EXPECTED ALARM. RPS On-Line test alarms are part of the crew brief prior to the test and at this time the alarms that will be received is discussed by the crew and I&E.

B. Incorrect - An NI Calibration Error alarm is an expected as it is a normal alarm condition during a power change. This alarm pertaining to reactivity management and should be investigated as it is the first time it has alarmed and not handled as an EXPECTED ALARM per NSD 509.

C. Incorrect – This alarm is expected for the situation (N2 leak) and should be pre-briefed during shift turnover but this component is important to safe operation of the plant and has no Lo-Lo alarm feature to warn the operator if the pressure continues to decrease. Each time this alarm is received it should be evaluated for changing conditions.

D. Incorrect – per NSD 509 this defines an UNEXPECTED alarm.

1 POINT

QUESTION # 63

Which ONE of the following alarm conditions will the RO verbally identify as an EXPECTED alarm per NSD 509?

An alarm that has...

A. been announced each time it alarms, requires no action, and occurs so frequently that announcing it detracts from the attention of the control room crew.

B. occurred the and the crew has discussed it prior to occurrence including any planned required actions.

C. been received from a control panel external to the main control room i.e. Heater Panel.

D. alarmed for the first time and the reason for the alarm has not been discussed among the crew.

Better to give 4 situations AND ID the Expected alarm.

A. Accumulator pressure alarm that occurs ~~routinely~~ twice each shift due to small N2 Leak.

B. LD 45% pressure alarm received for 1st time during letdown restoration lineup.

C. condensate demin alarm received during planned downpower maneuver.

D. NI bypass alarm received while students were performing OJT on NI cabinets AND crew was aware of training.

1 POINT

QUESTION # 63

G2.1.17 PRA/5-8-00

A. Incorrect – per NSD 509 this defines a NUISANCE ALARM.

B. Correct - per NSD 509 this properly defines the EXPECTED ALARM.

C. Incorrect – per NSD 509 this defines an UNEXPECTED alarm from another source other than the control room.

D. Incorrect – per NSD 509 this defines an UNEXPECTED alarm.

TRAINING OBJECTIVES

Terminal Objectives

The operator will have an understanding of the layout and information contained in the ARM for all unit alarms. The operator will demonstrate proper use of the ARM when responding to statalarms in accordance with Operation Management Procedures. Use correct communication techniques to inform all crew members or alarm status. (T1)

Enabling Objectives

1. Describe the layout and information contained in the ARM. (R1)
2. Determine when the ARM for a particular alarm should be consulted. (R2)
3. Discuss the requirements and priority for using the ARM. (R3)
4. Explain the proper procedure to follow if information in the ARM conflicts with other procedure response. (R4)
5. Demonstrate proper communications when receiving alarms. (R5)
6. Comply to all the OMPs that address response to alarms. (R6)

E. After an event has occurred, and the unit is stable and in a safe condition, all alarms should be checked to ensure the reason for them being in alarm is known. If the reason is not known, then refer to the ARM.

3.6 Discuss the different OMPs concerning alarms and the use of the ARM.

A. OMP 1-18, Communications and EOP Implementation Standards:

1. Control Room Annunciators, including computer alarms, must be clearly communicated to all Control Room Operators standing watch on a unit. The operator acknowledging an alarm is responsible for initiating this communication and obtaining appropriate repeatbacks.

2. Alarms, that are directly related to actions being taken by the operator, can be verbalized by "expected alarm" and do not require repeatbacks.

3. Expected alarms, such as those associated with ES On-line Testing, can be verbalized by "expected alarm" and do not require a repeatback. During the pre-job briefing, all operators standing watch on that unit shall define what alarms are expected.

4. During a reactor trip or unexpected transient, alarm annuniciators must become an integral part of the control board sweep and operation communication. The verbalization of alarms should help rather than hinder the overall command and control process. The relative importance of an alarm should be evaluated by the individual acknowledging the alarm. If a problem that needs immediate attention is identified, then the individual acknowledging the alarm shall report this to the Control SRO.

B. OMP 2-1, Enclosure 4.5, Responsibilities of the Reactor Operator.

1. A Reactor Operator shall acknowledge all alarms. When an alarm is received, he/she shall take appropriate actions in response to the alarm. This action may include a comparison/check of relative supporting parameters to validate the alarm, taking such actions as designated in the ARM, EOP, or APs. When an alarm is received that is unexpected for the existing plant conditions or without apparent cause, he/she shall notify the Control Room SRO immediately.

2. SRO (Unit Supervisor) may acknowledge alarms during times when the crew is busy, but the alarms must be communicated to the crew

C. OMP 2-2, Unit Log.

1. All statalarms pertaining to reactor core conditions and other important alarms (even if printed on the Alarm Typer), on the RCS, ES, RPS, RIAs, TG, and others appropriate will be listed in the log with an explanation. This applies only to alarms that would not be in the alarm state for the existing plant conditions. Repeating or nuisance alarms may be recorded once with a notation.

9. **Control Room Annunciators and Instrumentation:** The Operator at the controls is responsible for ensuring that the appropriate follow-up action is taken for all alarms. Alarms shall be considered valid until determined otherwise.

   Defective Control Room Annunciators and Instrumentation shall be identified and repaired promptly. If an annunciator is determined to be defective, the Control Operator shall ensure that alternate monitoring means are available to monitor parameters of importance.

10. **Control Room Equipment:** Cabinets, chairs, desks, tables, etc. shall be professional in appearance and limited to what is absolutely necessary and authorized by designated Operations Supervision. Work stations shall be arranged in the Control Room and Control Room area to prevent unnecessary distraction of Control Room Personnel.

11. **Control Room and Control Room Area Equipment & Component Storage:** Storage of I&E test equipment and other components shall be strictly limited. Only I&E test equipment and components that are absolutely necessary to be in the Control Room or Control Room Area on a continuous basis shall be authorized by designated Operations Supervision

12. **Control Room and Control Room Area Informational Notes:** Informational notes shall only be used on control panels in the Control Room and Control Room area when authorized by and signed by designated Operations Supervision. Informational notes shall be for information only and shall never be used to circumvent or supplement approved procedures. However, place keeping aids may be placed on instrumentation and procedures in accordance with OMP guidance

13. **Eating Meals in the Control Room:** Only personnel required to remain in the Control Room will be allowed to eat meals there. However, food and drink shall be kept away from the control panels.

14. **Control Room Professionalism Criteria Evaluation:** Evaluation of Control Room professional standards shall be part of the Management/Supervisory Observation Program.

509.3 NUCLEAR SITE COMMUNICATION STANDARDS

509.3.1 COMMUNICATION STANDARDS

Attachment A defines the standards at Duke's Nuclear Sites for 3 way communications and use of the phonetic alphabet.

509.3.1.1 Applicability to plant equipment during maintenance.

The standards described in Attachment A apply to installed plant equipment. They apply even if that equipment has been tagged out for maintenance or otherwise removed from normal service. They do not apply when working on equipment that has been removed to shop areas.

Three part communications and the phonetic alphabet should be used in training settings, (whether they be classroom, lab, or in-plant) when the training circumstances parallel in-plant conditions under which these standards would apply.

509.4 CONTROL ROOM ALARM MANAGEMENT STANDARD

NOTE: This section should be followed beginning on the effective date of the Rev 1 issue of this standard. To allow time for training and communication of this standard, compliance will be required after 4/1/2000.

509.4.1 ALARM MANAGEMENT STANDARD

This section defines the standard at Duke's Nuclear Sites for the management of alarms in the main control room(s) and the standard for management of alarms on local panel alarms for which Operations is the Owner Control Group.

509.4.1.1 Applicability

This standard applies to all alarms received in the Operations main Control Rooms (CR), including annunciators/statalarms, computer alarms, loose parts monitor alarms.

509.4.1.2 Standard

Notify the Control Room of Alarms: All plant personnel identify expected alarms to the CR ahead of time, and notify the CR upon causing or recognizing unexpected alarm conditions.

Announce Alarms: When acknowledging an alarm, the reactor operator (RO) announces it to the control room crew and/or "expected" as appropriate.

Use Alarm Response Guidance: ROs review and use alarm response guidance, with CR SRO oversight.

509.4.1.3 Definitions

Expected Alarm: A specific alarm which the crew knows will occur, why it will occur, and has planned any required actions.

Nuisance Alarm: A repetitive alarm which has been announced, requires no action, and occurs so frequently that announcing it detracts from the control room environment (determined by the SRO).

Priority OAC Alarm: Those alarms which appear in the lower "priority" portion of the OAC alarm screen. The alarms that appear in this section include HI-HI, LO-LO, and Priority 1 alarms.

509.4.1.4 Guidance and Examples

1. Notify the Control Room of Alarms

   Expected Alarms: Prior to the alarm, notify the control room of the specific alarms to be caused by planned activities.

   Unexpected Alarms: Upon causing or recognizing potential alarm conditions, notify the control room immediately.

EXAMPLES

- Maintenance provides a list of alarms prior to starting on-line engineered safeguards testing.
- NLO calls the control room immediately prior to generating alarms during battery testing on rounds.
- CR SRO identifies potential Loose Parts Monitor alarms in the pre-job briefing for an RCP start.
- Painter calls the control room after accidentally striking a level float in a sump.

2. Guidance and Examples on Announcing Alarms (Applicable to all annunciator/statalarms, all priority OAC alarms and any other OAC alarms deemed important by the RO.)

Note: IF the CR SRO is not available in the immediate area of the control boards to acknowledge the receipt of unexpected alarms, he will be provided a summary of those alarms received in his absence upon his return.

Unexpected Alarm: Acknowledging RO announces the noun name of the alarm. The other RO and the CR SRO repeats or paraphrases the noun name of the alarm.

Individual Expected Alarm: Acknowledging RO announces the noun name of the alarm and "previously reviewed" or "expected" along with the reason for the alarm. If the alarm response has been reviewed during the shift, acknowledging RO states "previously reviewed" or "expected". The other RO and CR SRO repeat or paraphrase the announcement of the alarm.

Multiple Expected Alarms due to preplanned (briefed) evolutions: Acknowledging RO announces the noun names of the alarms and "expected alarms", or announces the reason for the alarms and "expected alarms". The other RO and CR SRO repeat or paraphrase the alarms.

Nuisance Alarm: The CRSRO may direct the RO to acknowledge the alarm without announcing the alarm or announce "expected" as appropriate.

Exceptions:

1. The SRO may choose to require only the acknowledging RO and the SRO to verbalize alarms if critical plant evolutions are in progress which require increased focus on plant indications/controls. For this exception to apply:

   1.1. The SRO will communicate to the RO that his activity is considered 'critical'.

   1.2. The RO will not be involved in communications about alarms not directly related to his task while performing the 'critical' task.

   1.3. The SRO shall determine which alarm(s) to communicate to the RO involved in the critical evolution.

   1.4. The RO shall inform the SRO when his critical task is completed and will participate in 3 way communication on all subsequent CR alarms.

EXAMPLES

- Unexpected alarm "Turbine Building Temperature High".

  Communication: 1st RO: "Bill and Pete, 'Turbine Building Temperature High' ".

  CR SRO: "Turbine building high temp".

  2nd RO "Turbine building high temperature".

  1st RO: "That's correct". (BOP RO reviews alarm response.)

- Expected alarm "Turbine Building Sump In Purge" received for the first time on shift

  Communication: BOP RO: "Bill and Pete 'Turbine Building Sump In Purge', expected alarm".

  2nd RO "Turbine Building Sump In Purge, expected alarm".

CR SRO: "Expected alarm on Turbine Building Sump In Purge".

BOP RO: "That's correct." (BOP RO reviews alarm response).

- Expected alarms "NIS Channel 1 in Test" (crew has previously discussed applicable test alarms during a pre-job brief and reviewed alarm responses if desired).

  Communication: 1st RO: "Bill and Pete, NIS Channel 1 test, previously reviewed".

  2nd RO "NIS Channel 1 test, previously reviewed".

  CR SRO: "I understand, NIS Channel 1 test, previously reviewed."

  1st RO: "That's correct."

- Announcement of previously reviewed nuisance alarm hotwell level that is causing distraction in the Control Room.

  Communication: CR SRO: "Fred, acknowledge further 'Hotwell Low' alarms without announcing."

  RO (Fred): "Acknowledge 'Hotwell Low' alarm without announcing."

  CR SRO: "That's correct." (RO acknowledges subsequent alarm without announcing.)

3. <u>Guidance and Examples on Use of Alarm Response Guides</u>

Current Alarms at Beginning of Shift: Review available alarm response guidance for other current alarms which are unusual or unexpected for current plant conditions.

Unexpected Alarm: Upon receipt of an unexpected alarm, review and implement alarm response guidance as applicable.

Expected Alarm: During review of expected alarms (prior to alarms), crew will determine if alarm response guidance should be reviewed and plan it's use as applicable with the CR SRO.

BOP RO preferred: The BOP RO is the first choice crew member to review alarm response for new alarms.

Written Alarm Response Not Available: Determine response to the alarm. CR SRO determines whether assistance is needed.

EXAMPLES

- Unexpected annunciator alarm / statalarm "Upper Surge Tank Level Low".

  CNS/MNS: BOP RO reviews associated Alarm Response Procedure (ARP); discusses ARP with CR SRO. The BOP RO conducts ARP actions with CR SRO oversight.

  ONS: BOP RO reviews the associated Alarm Response Guide (ARG); discusses ARG with CRSRO. The ARG actions are taken per appropriate plant procedure with CR SRO oversite.

- Unexpected priority computer alarm received: "Generator Exciter cooling water low D/P".

  CNS/MNS: BOP RO reviews OAC ARG, and communicates required action to CRSRO.

  ONS: BOP RO and CR SRO discuss response to the alarm. CR SRO decides that additional support is needed. BOP RO directs the Turbine Building NLO to check system locally and to write a WR to check the instrument if needed.

  CRSRO directs the Shift Work Manager to obtain engineering support regarding response to alarm.

- A list of expected alarms is reviewed prior to the alarm occurring. Pre-job brief will determine if alarm response guidance needs to be reviewed for any of the listed alarms. ROs reviews alarm

QUESTION # 64

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	**Gen**	**Gen**
Group #	**1**	**1**
K/A #	**Gen. 2.1.3**	
Importance Rating	**3.0**	**3.4**

Technical Reference(s): **OMP 2-16**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **None Found**

Question Source:
- Bank # **1998 NRC EXAM**
- Modified Bank # ______
- New ______

Question History:
- Previous NRC Exam __X__
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ______

Comments:

1 POINT

QUESTION # 64

Unit 3 plant conditions:

- Time = 1800
- Shift turnover is in progress

As the on-coming Unit 3 Reactor Operator, which ONE of the following describes a responsibility that you must perform?

A. Complete the plant status checklist within one hour after assuming the shift.

B. Initiate shift turnover sheet and plant status checklist within one hour after assuming the shift.

C. Make a complete tour of the control room with the aid of the plant status checklist before assuming the shift.

D. Review unit turnover sheet for any equipment out of service that places the unit in an LCO ACTION statement before assuming the shift.

1 POINT

QUESTION # 64

Gen 2.1.3 3.0/3.4 Both PRA 4-5-00

A. Incorrect - Identifies the on-coming Reactor Operator as responsible for completing the shift turnover checklist. The on-coming Reactor Operator is responsible for reviewing and signing the shift turnover sheet.

B. Incorrect - Identifies the on-coming Reactor Operator as responsible for maintaining the shift turnover sheet. The on-shift Reactor Operator, during the shift, maintains the shift turnover sheet and completes shift turnover checklist within the last hour of the shift.

C. Incorrect - The on-coming Reactor Operator should make a complete Control Room round within the first hour after shift turnover. A tour and signing of shift turnover sheet are duties of on-coming operator.

D. Correct - Required action for shift turnover is to review turnover sheet. In addition OP/1/A/1102/20 notes that equipment taken out of service that places the Unit in an Action Statement of an LCO will be documented on shift turnover sheet.

3.3 Shift personnel shall conduct the shift turnover in a professional manner at all times.

3.4 Each individual shall be responsible for reviewing and understanding the Turnover Sheet applicable to the shift position prior to assuming that position.

3.5 Each shift position shall be relieved by a person qualified to that position.

3.6 The Operations Shift Manager shall be responsible for ensuring only fully qualified and fit-for-duty Operators assume License, NLO, and Fire Brigade positions.

3.7 Keowee Hydro Operators Turnover sheets shall be controlled by OP/0/A/2000/043 (Keowee Shift Turnover And Rounds).

3.8 The Shift Technical Advisor, Work Control Center SRO(s), and Plant SRO(s) shall utilize copies of the Operations Shift Manager and Unit Turnover Sheets to ensure adequate turnover of plant status is performed.

4. Turnover Guidelines

4.1 Each individual ensures that all significant conditions are appropriately recorded on the Turnover Sheets, Checklists, Round Sheets, and Logs for which they are responsible.

4.2 Evolutions in progress shall be evaluated by the Operations Shift Manager for turnover purposes. Turnovers may be delayed or take place where the evolution/task is being performed, if required.

A 4.3 The off-going Operator shall complete each section of the Turnover Sheet required for turnover prior to the on-coming Operator assuming the watchstation.

B 4.4 During the turnover process, the off-going Operator shall be able to explain each entry on the Turnover Sheet. Each line item shall be addressed during turnover; providing additional information, if any, to the entry.

4.5 The on-coming Operator shall ask questions as necessary to gain a full understanding of all turnover items.

4.6 The off-going operator shall not be relieved of the watchstation until all questions are resolved to the satisfaction of the on-coming Operator.

4.7 Monitoring of the ONS Control Boards shall be maintained during the Control Room turnover process.

5. Turnover and Turnover Sheet Instructions

5.1 All Turnover Sheets

5.1.1 On night shift, each watch station Operator shall prepare the applicable Turnover Sheet to the detail necessary to ensure adequate turnover.

5.1.2 During turnover, each operator will review the watchstation status with the on-coming Operator, emphasizing off-normal conditions, trends, tests in progress, and planned evolutions for the shift.

5.1.3 The number of entry lines under each section of the Turnover Sheets can be adjusted (added or deleted) as necessary to accommodate the necessary information under each heading. (i.e., the sections within each Turnover Sheet shall be as defined in this OMP. The number of entry lines under each section, however, can differ from the number illustrated within this OMP).

5.2 Operations Shift Manager Turnover Sheet

5.2.1 Safety items/issues designated by the OSM shall be recorded in the Safety section. The corresponding corrective process (work request, work order, PIP, etc.) and accountable individual for update/resolution shall be recorded.

5.2.2 Operability of the SSF, Keowee, and number of Lee Combustion Turbines operable will be recorded.

5.2.3 Technical Specification and Selected Licensee Commitment Action conditions in effect will be recorded, including the out-of-service date/time, date/time the action statement is applicable, and the applicable Technical Specification or Selected Licensee number.

5.2.4 Operability Evaluations in progress per NSD 203 (Operability) will be recorded in the available section of the Turnover Sheet. The corresponding PIP number and accountable individual for the evaluation shall also be recorded.

5.3 Unit Turnover Sheet

5.3.1 The Unit Turnover Sheet shall be completed by the Control Room Operators

5.3.2 Completion of Control Room Rounds shall be documented in the Control Room Rounds section.

5.3.3 Significant Safety items/issues shall be recorded in the Safety section. The comments field can include information such as the corresponding corrective process (work request, work order, PIP, etc.), accountable individual for resolution, expected resolution date, etc.

5.3.4 Unit Status information will be recorded on the turnover sheet for the approximate times denoted. The following information will be recorded:

- Operating mode and reactor power
- Electrical megawatts
- Reactor Coolant System Temperature and Pressure
- If in mode 4 or higher, the last available reactor coolant system leakage results. Otherwise N/A.
- When CSAEs are in operation, RIA-40 (CSAE off-gas monitor) reading. Otherwise N/A.
- When OP/0/A/1106/31 (Control of Secondary Contamination) is in effect, the last available OTSG tube leak calculation. Otherwise N/A.
- Two boration sources available for RCS makeup
- Demineralizer currently in use for delithiation
- Status of the GWD vent header (split or cross-connected)
- Unit supplying the Auxiliary Steam header
- Chemical Treatment Pond alignment for the turbine building sump
- Units available to supply, and be supplied, alternate source of Emergency Feedwater per Selected Licensee Commitment 16.10.7 (Alternate Source of Emergency Feedwater (EFW))

5.3.5 Technical Specification and Selected Licensee Commitment Action conditions in effect will be recorded, including the out-of-service date/time, date/time the action statement is applicable, and the applicable Technical Specification or Selected Licensee number.

5.3.6 Plant Concerns and Action Register Items designated by the OSM per Site Directive 2.1.7 (Top Equipment Problem Resolution Process) will be recorded in the applicable sections of the turnover sheet. The comments field can include information such as the corresponding corrective process (work request, work order, PIP, etc.), accountable individual for resolution, expected resolution date, etc.

5.3.7 Required compensatory actions in effect such as fire watches, equipment monitoring, testing, etc. will be recorded. The comments field can include information such as the corresponding corrective process (work request, work order, PIP, etc.), accountable individual for resolution, expected resolution date, etc. Compensatory Actions in effect should be submitted to the Operations Work Process Manager's Group as candidates for Operator Workarounds or Operator Burdens per Site Directive 2.1.7 (Top Equipment Problem Resolution Process).

5.3.8 Any controls in manual which are outside of normal procedure position will be designated.

5.3.9 The Turnover Items section will be utilized to record equipment status or configuration information, activities in progress, etc. at the Control Room operator's discretion.

5.3.10 Major equipment deficiencies associated with the Unit will be recorded, including the component, problem description, associated tracking/corrective action process, and scheduled resolution timeframe. The work request/work order/PIP number and resolution timeframe information shall be updated weekly each Sunday day shift.

5.3.11 Radiation monitors which are out-of-service will be recorded. The associated tracking/corrective action process will be included. The comments field can include information such as the accountable individual for resolution, expected resolution date, etc.

5.3.12 The following items will be reviewed/performed and signed prior to accepting turnover: {PIP 99-2179}

- Unit log
- Technical Specification Log
- Control Room in-progress procedures
- Control Board Walkdown

5.3.13 Out of normal OAC and/or Statalarms shall be documented and reviewed. Documentation of alarms will include Alarm Description, Date of Alarm, Reason For Alarm (if known), and Corrective Action Taken.

5.3.14 The following items shall also be performed/reviewed by the Control Room operators (Control Room SRO and Reactor Operators) at the earliest opportunity upon completion of turnover:

- Removal and Restorations
- Unit work schedule
- Operator Workarounds
- Operations Guidelines
- Computer Alarm Summary
- Computer Temporary Alarm Summary (Saturday night only)
- Computer Point Processing Log
- Unit Events Recorder Alarm Summary
- Switchyard Events Recorder Alarm Summary
- Statalarm Test

5.3.15 Upon completion of turnover to day shift, the day shift will complete and sign the designated reviews. Upon completion of turnover to the night shift, the night shift will complete and sign the night shift reviews sections on the same turnover sheet. A new turnover sheet will then be prepared for day shift with the Reviews section sign-offs blank.

5.3.16 Prior to concluding turnover, the Control Room SRO shall verify and designate that the Work Management System (WMS) has been properly updated for the shift's activities.

5.3.17 Upon completion of turnover, the on-coming and off-going Control Room SROs will sign the turnover sheet, designating relief of the watchstation.

5.3.18 Upon completion of turnover, the on-coming and off-going Reactor Operators will sign the turnover sheet, designating relief of the watchstation.

QVALUE 1.0

QUESTION 66 B60

Which ONE of the following describes one of the responsibilities that must be performed by the oncoming Unit 1 reactor operator?

A. Complete the shift turnover checklist within one hour after assuming the shift.

B. Initiate shift turnover sheet and shift turnover checklist within one hour after assuming the shift.

C. Make a complete tour of the control room with the aid of the turnover checklist before assuming the shift.

D. Review turnover sheet for any equipment out of service that places the unit in an LCO action statement before assuming the shift.

B60
ANSWER D
COGNITIVE: Knowledge
REFSPECIFIC OP/1/A/1102/20, SHIFT TURNOVER, PAGE 3 OF 5 (4.4.2.5)
MODULE Lesson Plan: Operations Management Procedures (ADM-OMP)
OBJECTIVE: ELO-4.3.B

ABASISIncorrect, identifies the on-coming Reactor Operator as responsible for completing the Shift Turnover Checklist. The on-coming Reactor Operator is responsible for reviewing and signing the Shift Turnover Checklist.

BBASISIncorrect, identifies the on-coming Reactor Operator as responsible for maintaining the shift turnover sheet. The on-shift Reactor Operator, during the shift, maintains the shift turnover sheet and completes shift turnover checklist within the last hour of the shift.

CBASIS Incorrect, A complete CR round should be made by the on-coming RO within the first hour after shift turnover. A tour and signing of shift turnover checklist are duties of on-coming operator.

DBASIS Correct, Required action for shift turnover is to review R&R book to attain status of equipment. In addition 1102/20 notes that equipment taken out of service that places the Unit in an Action Statement of an LCO will be documented on Shift turnover sheet.

QNUM	60
HNUM	
QCHANGE	NEW
ACHOICE	NEW
BCHOICE	NEW
CCHOICE	NEW
DCHOICE	NEW
ANSCHANGE	
DAREA	
EXAM TYPE	NRC
QDATE	9/21/98
FAC	269 Oconee 1, 2, & 3
RTYP	B&W 177
EXLEVEL	B
AUTHOR	Sonalysts, Inc.
REFKEY	
KA1	G2.1.3
KA1RO	3.0
KA1SRO	3.4
KA2	
KA2RO	
KA2SRO	

QUESTION # 65

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**T3**	**T3**
Group #	**C1**	**C1**
K/A #	**G 2.1.7**	
Importance Rating	**3.7**	**4.4**

Technical Reference(s): **THF-CTP**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **THF-CTP #3**

Question Source:
Bank # **THF-219**
Modified Bank # ________
New ________

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 ______
55.43 __X__

Comments:

1 POINT

QUESTION # 65

Unit 1 plant condition

- Steady State power operations for previous 2 days
- P0889, Core Thermal Power Best = 84.95
- E2082, ICS Core Thermal Power Best = 84.02
- E2085, ICS Core Thermal Power Demand Setpoint = 85.00

Statalarm 1SA-2/C11, Loss of OAC CTP Signal and 1SA-4/E7, OAC Trouble is received.

Which ONE of the following is the expected plant response?

SEE ATTACHMENT

The Unit will _______ power approximately 1% / ________.

A. increase / over the next hour

B. decrease / over the next hour

C. increase / immediately

D. decrease / immediately

1 POINT

QUESTION # 65

G 2.1.7 Common

A. Correct – If the ICS CTP reading is lower that the OAC reading when the signal is lost the ICS will increase actual power. OAC calibration is removed over a 60 minute time constant.

B. Incorrect – If the ICS CTP reading is higher that the OAC reading, then this would be correct.

C. Incorrect –First part is true however the change is made by a slow integral and occurs over about 1 hour.

D. Incorrect – If the ICS CTP reading is higher that the OAC reading, then this would be correct but would occur over an hour.

C-11

ICS

LOSS OF OAC CTP SIGNAL

1. Alarm Setpoint

1.1 OAC value differs from the ICS average value by more than 2%, or the OAC value fails to update in a timely manner.

2. Automatic Action

2.1 Depending on calibration of feedwater fouling coefficient:

2.1.1 If the ICS CTP reading is lower than the OAC reading when the signal is rejected, the plant may operate above 100% licensed power as calculated by the OAC.

2.1.2 If the ICS reading is greater than the OAC reading when the data loss occurs, the ICS will decrease actual power and avoid operation above 100% licensed power.

3. Manual Action

3.1 Refer to PT/1/A/0600/001 for CTP calculation and adjust power downward as necessary to ensure that power is within required limits.

3.2 Contact Systems Engineering to verify the correct value for the feedwater fouling coefficient.

4. Alarm Sources and References

4.1 OM 201.H-0179 001

4.2 STAR Module 1ICSCOIM01

OBJECTIVES

TERMINAL

1. During operation at all power levels utilize redundant control room indications to ensure reactor thermal power is maintained within procedural and licensing limits.

ENABLING

1. Given ΔT power level, determine which one of the following OAC calculations will be utilized as Thermal Power Best estimate and why: (R1)
   1.1 Secondary Power
   1.2 Primary Power
   1.3 Weighted Average TPB
2. Explain why the ICS thermal power calculation is calibrated by the OAC thermal power calculation. (R2)
3. Compare OAC and ICS calculated thermal power values and predict plant response if the OAC calibration to the ICS were removed. (R3)
4. Describe how the Thermal Output Program can be used to verify the status of input and outputs to the thermal power calculation. (R4)
5. Explain the thermal power response that occurs during the following events at Oconee: (R5)
   5.1 Letdown Flow instrument isolation
   5.2 FDW Heater extraction isolation
6. Given a copy of PT/600/01, Periodic Instrument Surveillance, determine the most accurate indication for Core Thermal Power at given power levels. (R6)
7. Given plant conditions, determine the requirements to maintain compliance with Improved Technical Specifications instrumentation surveillance. (R7)

2. The correction term is applied through a long time period filter to prevent demand upsets from data loss.

D. Using the corrected thermal power best value for feedback in the ICS ensures precise control of the plant to the ASME standard calculation.

E. Because the ICS may reject the OAC reading at any time, the potential exists that the plant may operate above 100% licensed power as calculated by the OAC, if the ICS reads lower than the OAC when data rejection occurs.

1. To prevent this undesirable response, the ICS feedwater fouling coefficient may be adjusted so that the ICS thermal power best reading is equal to or greater than the OAC reading at all times.

2. If the ICS reading is greater than the OAC reading, when data loss occurs, the ICS will decrease actual power, and thus avoid operation above 100% licensed power.

F. Loss of this data link is annunciated to the operator via 1SA-2 / C-11, Loss of OAC CTP Signal

1. Alarm Response Guide alerts the operator that the power change following the loss of the OAC signal will be dependent on the setting of the ICS FDW fouling coefficient.

   a) If the ICS CTP reading is lower than the OAC reading when the signal is rejected, the ICS will increase actual power and the plant may operate above 100% as calculated by the OAC.

   b) If the ICS CTP reading is greater than the OAC reading when the signal is rejected, the ICS will decrease actual power and avoid operation above 100%.

   c) The alarm has a five minute timer that will prevent recurring alarms during plant transients.

2. Loss of OAC Procedure

   a) Loss of ICS Gateway

      1) SA-02/C-11, Loss of OAC CTP Signal

      2) An expected decrease in Unit power level, about 1 to 2%, will occur over a 1-hour period. This is initiated by the ICS system in order to match the ICS Core TPB with the value in the CTPD set window.

      3) For planned loss, estimate power change, OAC (P0889) – ICS (E2082)

      4) Upon loss of OAC Core Thermal Power Calculation, the Reactor Engineering Group must calculate the Heat Balance per PT/0/A/0205/002 (Thermal Power Calculation) once every 12 hours.

f) Feedwater temperature is applied to a function generator to obtain feedwater enthalpy at reference conditions of 1000 psia.

1) Feedwater enthalpy is subtracted from OTSG A steam enthalpy, and the resulting difference term is multiplied by selected loop A feedwater flow to obtain OTSG A secondary power.

2) Feedwater enthalpy is subtracted from OTSG B steam enthalpy, and the resulting difference term is multiplied by selected loop B feedwater flow to obtain OTSG B secondary power.

g) The loop A power and loop B power are summed to obtain total secondary side thermal power.

h) The total secondary power is output as Thermal Power Secondary, and includes the RCP power. Indicated on Dixon on front board.

i) The RCP power will be removed later in the Thermal Power Best calculation to obtain reactor power.

- In the ICS thermal power calculation, reactor coolant pump power is derived from total RC flow.

j) An ICS feedwater fouling coefficient correction term is available which allows the thermal power secondary reading to be scaled to agree with the results from the precision heat balance measurements, or to force the ICS to operate conservatively in conjunction with normal thermal power correction from the OAC.

  - This coefficient is NOT to be confused with the OAC fouling coefficient that is used during fuel cycle to increase plant efficiency.

2.4 ICS/ OAC Power Calculation Interactions

A. The ICS thermal power best calculation is less sophisticated than the ASME standard calculation performed in the OAC, because several inputs are missing.

1. Also, the ICS calculation is based on selected signals, rather than the (potentially) more accurate results from statistical averaging.

B. To improve the accuracy of the ICS control, a correction term is applied to the thermal power best integral feedback signal.

1. The correction term is based on the difference between the OAC periodic average thermal power best value and an ICS average thermal power best value taken over the same period.

C. Before the correction is allowed, the OAC term is validated by the ICS.

1. If the OAC value differs from the ICS average value by more than 2%, or if the OAC value fails to update in a timely manner, the ICS will ignore the OAC and set the correction term to zero.

Exam Question Report

27-Jan-99

Question ID:	THF219	Revision No:	0	Revision Date	10/29/1999
Question Description:	THF219				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	CTP
Last Used Date:	02/25/2000	Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 3; SRO = 3	Max. Point Value:	0.25
		Passing Point Value:	0.25

27-Jan-99

Question

Unit 1 plant conditions:

- Steady State power operations for previous 2 days

- P0889, Core Thermal Power Best..................84.95
- E2082, ICS Core Thermal Power Best.............84.02
- E2085, ICS Core Thermal Power Demand Setpoint...85.00

Statalarm 1SA-2/C-11, Loss of OAC CTP Signal and 1SA-4/E7, OAC Trouble is received. Which ONE of the following describes the expected plant response? (.25)

The Unit will _____________ power approximately 1% ________.

A) increase / over the next hour

B) decrease / over the next hour

C) increase / immediately

D) decrease / immediately

Answer

A
Per LP (THF-CTP) Page 21, Section F on Loss of data Link states that "if the ICS CTP reading is lower than than OAC reading when the signal is rejected (LOST), the ICS will increase actual power and the plant may operate above 100% power as calculated by the OAC".
ARG for Statalarm SA-02/C-11 (Loss of OAC CTP Signal) states that "an expected decrease in Unit power level... will occur over a 1-hour period". This assumes that the ICS CTP reading is > the OAC CTP reading, which in this question is not the case.
A. Correct- See Above
B. Incorrect-See Above.
C. Incorrect-See Above.
D. Incorrect-See Above.

Lessons

QUESTION # 66

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**T3**	**T3**
	Group #	**C1**	**C1**
	K/A #	**G 2.1.30**	
	Importance Rating	**3.9**	**3.4**

Technical Reference(s): **CF-EFD**
AP/1700/19-502 p.#8

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **CF-EFW #43**

Question Source:
- Bank # ____
- Modified Bank # **CF-18**
- New ____

Question History:
- Previous NRC Exam ____
- Previous Quiz / Test ____

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ____

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 ____

Comments:

1 POINT

QUESTION # 66

Unit 2 plant conditions:

- A Loss of Main FDW has occurred
- Emergency Feedwater (EFW) system is in operation
  - ➢ FDW-315 (SG 2A EFDW Control Valve) has failed at 50% OPEN

Which ONE of the following describes the local control of the EFW flow control valves using the manual handwheel?

Assume the valve disk is free to move.

The Handwheel can be used to...

A. position the valve in the open direction ONLY.

B. position the valve in the closed direction ONLY.

C. open, close, or throttle the valve upon a loss of the control room auto positioning signal.

D. open, close, or throttle the valve if instrument air and N2 is unavailable to stroke the valve.

1 POINT

QUESTION # 66

G2.1.30

Question Setup:

.

A. Incorrect, Some manual operators on be used to open and/or close failed valves. (CC-8, HP-5). On this valve the handwheel can only throttle a failed open valve.

B. Correct, Manual Operation of FDW-315 and FDW-316 (Fig OC-CF-EF-11) Handwheel on top of each valve. Normally "backed-off" so the controller can operate the valve. If the handwheel is not fully backed off the valve travel will be limited. This has caused EFDW train inoperability at ONS. Can only be used to close the valve or throttle the valve if it fails open, i.e. loss of IA. There is a locking nut on the handwheel stem that has to be "backed- off" of the bottom of the stem in order to allow free operation of the handwheel. (This nut is normally backed off) The handwheel can only throttle a failed open valve

C. Incorrect, Some manual operators on air operated valves can be used to open and/or close failed valves. (CC-8, HP-5). On this valve the handwheel can only throttle a failed open valve.

D. Incorrect, Some manual operators on air operated valves can be used to open and/or close failed valves. (CC-8, HP-5). On this valve the handwheel can only throttle a failed open valve.

31. Describe or make a sketch of the logic/conditions that will AUTO START the TDEFDWP when its control switch is in AUTO, including a description of AMSAC. (R25)

32. Describe the affect a Main Steam OTSG Isolation actuation will have on the EFDW system. (R58)

33. Describe the additional action required to allow emergency feed through the alternate ICS flowpath, following actuation of the MS Line Isolation circuit (R59)

34. List the EFDW SG Level setpoints for the conditions when RCPs are running and when all RCPs are off. (R37)

35. Describe the SG level indicators used in the EFDW System, including whether or not they are temperature compensated, and how to select the PRIMARY/BACKUP indicators. (R30)

36. Detail by sketch the status of the solenoids associated with EFDW Level control (Train A & B) when MFDW is operating and normal power to the solenoids is available. (R31)

37. Explain the operation of the solenoids associated with EFDW Level Control (Train A & B) when MFDW is lost and normal power to the solenoids is available from a provided sketch or by making a sketch. (R32)

38. Explain the operation of the solenoids associated with the EFDW Level Control (Train A & B) when MFDW is lost and normal power to the solenoids is lost, from a provided sketch or by making a sketch (R33)

39. List the locations of FDW-315 & 316. (R27)

40. List which Level Train is the PRIMARY TRAIN for SG A level control and which is the PRIMARY TRAIN for SG B level control. (R35)

41. Explain why the Primary Level Train for SG A is not the same as the Primary Level Train for SG B. (R36)

42. Describe how to manually control FDW-315 & 316, after a loss of MFDW, from the Control Room. (R34)

43. Describe the methods for throttling EFDW flow, available to the operator. (R49)

44. Describe how the TDEFDWP meets "AC Independence" criteria include how each component helps provide this independence. (R38)

45. Explain the purposes of the Nitrogen bottles associated with the EFDW System. (R39)

a) With automatic level control in effect, the operator may desire to throttle FDW-315.

b) Operator should select Manual - this re-energizes solenoid #2 and 3-way valve swaps back to accept manual loader demand.

D. Match valve position demand to manual loader output prior to selecting manual to limit unnecessary swings on system.

E. FDW-316 Level Control Operation

1. Level control for FDW-316 is essentially the same as FDW-315, except as noted below.

- Train B is powered from KVIC and is the primary train for B SG.
- Train A powered from KVIB is the Backup Train.
- Solenoids #1 and #2 are powered from DIC.
- This ensures that upon system actuation and single failure of KVIC or KVIB, only one circuit will auto-swap to its backup level circuit.

4.3 Manual Operation of FDW-315 and FDW-316 (Fig OC-CF-EF-11)

A. Valves located in East (FDW-315) and West (FDW-316) Penetration Rooms

1. Handwheel on top of each valve.
2. Normally "backed-off" so the controller can operate the valve.
3. If the handwheel is not fully backed off the valve travel will be limited. This has caused EFDW train inoperability at ONS.
4. Can <u>only be used</u> to close valve or throttle valve if it fails open, i.e. loss of IA.
5. There is a locking nut on the handwheel stem that has to be "backed-off" of the bottom of the stem in order to allow free operation of the handwheel.(This nut is normally backed off)

5. A.C. Independence (Figure OC-CF-EF-19)

5.1 Loss of all power - steam driven TDEFDWP available

A. Cooling Water

1. LPSW-138 (TDEFDWP Cooling Bypass)

a) LPSW-138 is an air-operated valve, normally shut.

Alternate Methods For Controlling EFDW Flow

3.4 Ensure open any of the following valves that were previously closed:

_____ 1FDW-372 (1A MD EFDWP DISCHARGE BLOCK)

_____ 1FDW-368 (TD EFDWP DISCH TO 1A SG BLOCK)

_____ 1FDW-382 (1B MD EFDWP DISCHARGE BLOCK)

_____ 1FDW-369 (TD EFDWP DISCH TO 1B SG BLOCK).

NOTE 3.5:	The handwheels for 1FDW-315 and 1FDW-316 are locked at full open position. The handwheel can be used to close an open valve. The handwheel will **NOT** open a failed closed valve.

3.5 Dispatch an operator(s) to the Penetration Room to close the affected Emergency FDW Control Valve(s).

_____ 1FDW-315 (SG 1A EFDW CONTROL VALVE)

_____ 1FDW-316 (SG 1B EFDW CONTROL VALVE),

_____ 3.5.1 Establish communications with the operator.

Exam Question Report

27-Jan-99

Question ID:	CF018	Revision No:	0	Revision Date	10/29/1999
Question Description:	CF018				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	CF-EF - Emergency Feedwater System
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 50; LRO = 50; SRO = 50	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

KA: 2.1.30

A transient has occurred requiring the use of Emergency Feedwater (EFW). The EFW flow control valves (FDW-315 and FDW-316) fail to operate from the Control Room and have failed open. Which ONE of the following describes the local control of the EFW flow control valves using the manual handwheel? (Assume the valve disk is free to move.)
The Handwheel can be used to... (.25)

A) fully close or throttle closed on the valve.

B) fully open or throttle open the valve.

C) open, close, or throttle the valve upon a loss of the remote control positioning signal.

D) lock the valve in its fully open or closed position, if instrument air is available to stroke the valve.

Answer

A
A is correct handwheel can only throttle a failed open valve.

Lessons

ID	Description
CF-EF	Emergency Feedwater

Enabling Objectives

ID	Description
CFEFR50	Enabling Objective created by conversion

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

QUESTION # 67

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**T3**	**T3**
Group #	**C2**	**C2**
K/A #	**G 2.2.2**	
Importance Rating	**4.0**	**3.5**

Technical Reference(s): **EAP-E32**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-E32 #10**

Question Source:
Bank # ______
Modified Bank # **EAP-04**
New ______

Question History:
Previous NRC Exam ______
Previous Quiz / Test ______

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 67

Unit 3 plant conditions:

- A LOCA has occurred
- RCS subcooling = 0°F
- BWST level = 19 feet

Which ONE of the following best describes the reason for aligning LPI suction to the RBES at this time?

A. BWST level instrument errors can cause suction vortexing before indicated level reaches 10 feet.

B. Waiting longer would result in radiation fields in LPI Pump Rooms that prohibit local operation of the required valves.

C. Sufficient level still remains to ensure adequate suction if manual alignment of the necessary valves is required.

D. To ensure adequate LPI and HPI pump suction pressure during piggyback operation when suction is being provided from the BWST.

1 POINT

QUESTION # 67

G2.2.2

A. Incorrect, - Level instrument errors are considered in the engineering calculations that prescribe the BWST level at which LPI pump suctions should be shifted to the RBES. The value of 10 feet was previously used as a transfer point.

B. Incorrect, - Radiation levels in the LPI pump rooms should not be affected by BWST level changes. Rad levels will increase when suction is aligned to the RBES.

C. Correct, - The 19 foot level for the LPI pump suction shift was selected based on allowing sufficient time for manual alignment of the necessary valves should the remote operating capabilities fail prior to BWST level being to low.

D. Incorrect, - Adequate NPSH to the LPI pumps is available down to the 6 foot level in the BWST. As long as the LPIP has proper suction pressure the HPIP will be delivered adequate suction pressure.

TERMINAL OBJECTIVE

1. Explain the purpose(s) for specified steps in CP-602, SG Cooldown with Saturated RCS, and, where applicable, the basis behind each step or decision path. (T1)
2. Become familiar with the sequence of operation and reasons for actions performed in CP-602, to be able to better perform these steps as a crew while using the EOP directions and guidance, both on the Simulator, and should the need arise, in the plant. (T2)

ENABLING OBJECTIVES

1. State the common link among all transients that may result in the use of CP-602. (R1)
2. Explain why HPI piggyback operation may be required during SBLOCAs; describe the equipment failures that could require the use of the HPI piggyback alignment even before BWST inventory is depleted during a LOCA. (R2)
3. Given a copy of the "Total HPI flow vs. RCS Pressure" curve from CP-602, be able to determine, for a given set of conditions, whether HPI flow is "adequate" or "inadequate". (R3)
4. Describe the two (2) actions taken to cool and depressurize the RCS if total HPI flow is not adequate or if it cannot be established to each header. (R4)
5. Discuss the reasons for not attempting to depressurize the RCS to LPI conditions by latching the PORV open. (R5)
6. State under what two conditions primary to secondary heat transfer will exist when the primary is saturated, and explain why heat transfer will or will not exist for these conditions. (R8)
7. Given hypothetical system parameters, be able to determine whether secondary heat transfer does or does not exist. (R9)
8. Explain why controlling the tube to shell ΔT of an isolated SG during plant C/D is important; describe the proper procedure for maintaining the proper positive ΔT for an isolated SG per the Loss of Main Feedwater AP(R10)
9. Explain why it is important to maintain the level of a SG that has been isolated because of a SG Tube Leak to < 285" XSUR. (R11)
10. Describe the basis for swapping LPI and RBS suction to the RBES when the BWST level decreases to 19 feet. (R27)
11. Discuss briefly, the basis for verifying $\geq$ 1000 gpm LPI flow per header before transferring from CP-602 to the Large Break LOCA section of the EOP. (R7)

2. If for some reason SG level exceeds this level, Encl. 7.3B should be used to determine if the possibility of water in the steam lines exists. If Encl. 7.3B indicates water in the steam lines, discontinue steaming the SGs and initiate HPI Forced Cooling by referring to Rule #6.

2.7 If at any time BWST level reaches 19 feet and RB level increasing, then transfer LPI and RBS suction to the RBES. (step 10.0)

A. When BWST level is ~ 19 feet, alignment must be started to supply the HPIPs from LPI, since a swap to the RBES must soon be made.

B. **REFER TO** Enclosure 7.12, "ECCS Suction Swap To RBES With Either LPI Header Flow < 1000 gpm."

1. This 19-foot BWST level is chosen to provide sufficient time in case LP-15 or LP-16 does not operate electrically and has to be operated manually.
2. **At 19 feet in the BWST, at least 30 minutes would be available for manually opening LP-15 and for LP-16 before BWST inventory required swapping to the RBES. This is a Time Critical/Risk Significant Operator Action. "Swap for the injection mode to the Recirculation mode prior to depleting the BWST (when the BWST is approximately 94% empty).**
3. While LP-15 and LP-16 are environmentally qualified, and safety-grade valves, the power supplies for the motors (XL and XN) are NOT qualified as safety-grade (even though XL and XN are non-load shed panels). So conservative measures are required to ensure LP-15 and LP-16 availability for HPI piggyback operation.
4. The 19-foot level also allows sufficient time for manual operation of LP-9 and LP-19 should these valves fail to operate electrically when alignment for HPI piggyback is made.

C. (Refer to Enclosure 7.12, "ECCS Suction Swap To RBES With Either LPI Header Flow <1000 gpm."

Note: (Refer to PIP 98-00825) The Reactor Building Wide Range Level Instruments, LT-90 and LT-91 have large uncertainties which could potentially create a conflict in the Emergency Operating Procedure guidance provided to the reactor operators when mitigating LOCA's. Specifically, the guidance for initiating swap over of suction source for the BS and LPI pumps from the BWST to the RBES states that the reactor building water level must be greater than 4 ft. AND the BWST level must be greater than 6 ft. Due to the large uncertainty in the indication on the RB level instruments, these conditions may not be indicated simultaneously. While the operator is waiting for the RB level to reach 4 ft., the BWST could fall well below 6 ft., possibly to the point that air might be ingested into the BS and LPI pumps. An operability evaluation must be performed to determine if the guidance would lead

Exam Question Report

27-Jan-99

Question ID:	EAP004	Revision No:	0	Revision Date	10/29/1999
Question Description:	EAP004				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	EAP-E32 - SG Cooldown with a Saturated RCS
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 19; SRO = 19	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

KA: 2.2.2

While operating in CP-602, SG Cooldown With Saturated RCS, HPI piggyback alignment is started when BWST level indicates 10 feet. Which ONE of the following best describes the reason for commencing the alignment at this time? (.25)

A) BWST level instrument errors can cause suction vortexing well before indicated level reaches 6 feet.

B) Waiting longer would result in radiation fields in LPI Pump Rooms that prohibit local operation of the required valves.

C) Sufficient level still remains to ensure adequate suction if manual alignment of the necessary valves is required.

D) Adequate NPSH for piggyback operation CANNOT be assured below 10 feet in the BWST.

Answer

C

Lessons

ID	Description
EAP-E32	SG Cooldown With Saturated RCS EAP-E32

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

QUESTION # 68

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**T3**	**T3**
Group #	**C2**	**C2**
K/A #	**G 2.2.13**	
Importance Rating	**3.6**	**3.8**

Technical Reference(s): **NSD-500.7**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **SD 2.11**

Question Source:
Bank # **TAGS 035**
Modified Bank # ________
New ________

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 68

Unit 1 plant conditions:

- 1A MD EFDW pump is OOS for motor bearing repair
- Maintenance has completed repairs
- The breaker Red tag has been lifted for initial testing to check motor rotation. AND THAT (pump is uncoupled) portion is completely filled
- Following rotation checks, pump coupling will be performed followed by additional testing.

Which ONE of the following is correct?

Prior to pump coupling the tag…

A. ~~must~~ SHALL be cleared. AND reissued.

B. and stub should be marked "VOID", ~~and~~ filed, and a new tag hung.

C. may remain lifted during the completion of the pump coupling.

D. should be replaced on the breaker and stub returned to work supervisor.

NON-DISCRIMINATORY

1 POINT

QUESTION # 68

G2.2.13

A. Incorrect - Section 500.7.2.10 describes using Temporary Lifts for simple tasks only. This is a simple task and does not require the tag to be cleared. The tag is replaced on the component to allow further work to be performed.

B. Incorrect - Cannot void the tag in this case.

C. Incorrect - The temporary lifted tag must be re-hung on the breaker to continue work.

D. Correct - Per Section 500.7.2.10, Temporary Lifts can be used for simple tasks. When this is done the tag is re-hung for the work to be completed.

2. For the following Nuclear System Directives (NSD's): (R2)

- Discuss the purpose, Operations / individual responsibilities, and management expectations (where applicable).
- Apply the guidance within the NSD to determine or interpret required responses for a given situation.

2.1 NSD 104 (Housekeeping, Material Condition, and Foreign Material Exclusion)

2.2 NSD 109 (Confined Space Entry)

2.3 NSD 112 (Fire Brigade Organization, Training, and Responsibility)

NSD 114 (Site Assembly & Evacuation)

2.4 NSD 117 (Emergency Response Organization, Training, and Responsibilities)

NSD 213 (Conduct of Infrequently Performed Test or Evolution)

2.5 NSD 224 (Severe Accident Management Guidance)

2.6 NSD 301 (Nuclear Station Modifications)

2.7 NSD 304 (Reactivity Management)

2.8 NSD 311 (Safety-Related DC Systems Ground Response)

2.9 NSD 316 (Fire Protection Impairment and Surveillance)

2.10 NSD 317 (Freeze Protection Program)

NSD 403 (Shutdown Risk Management)

2.11 NSD 500 (Safety Tags / Equipment Protection Tags)

2.12 NSD 502 (Corporate Conduct of Operations in the Switchyard)

2.13 NSD 506 (Operator Workarounds)

2.14 NSD 507 (Radiation Protection)

2.15 NSD 604 (Stop Work)

2.16 NSD 700 (Independent Verification)

2.17 NSD 704 (Technical Procedure Use and Adherence)

P&I
2/9/00
gs

2. Red Tag Stub Control Process

   - Red Tag stubs are controlled by the OPERATIONAL CONTROL GROUP and WORK GROUP with the following provisions:
   - Red and White Tag stubs shall not be removed from the site
   - When Red Tag stubs are transferred, the responsibility for verifying safe work conditions is also transferred with the stubs.

     CAUTION:
     A lock or blocking device shall not be removed unless the individual is in possession of the final signed stub for the only tag on the isolation device
   - After work is complete, the WORK GROUP SUPERVISOR or designee, is responsible for removing Red Tag locks before returning Red Tag stubs to the OPERATIONAL CONTROL GROUP.
   - Red Tag stubs may be attached to station work orders or work packages
   - The WORK GROUP shall maintain a designated area to store Red Tag stubs during work interruptions or upon leaving the site. Red Tag stubs shall be reclaimed prior to resuming work
   - When all work is completed the WORK GROUP SUPERVISOR or designee shall sign, date and time Red Tag stubs and return them to the OPERATIONAL CONTROL GROUP.

3. Lost Red Tag Stub Process

   - For lost Red Tag stub, determine work requiring the Red Tag is complete. This shall require a WORK GROUP SUPERVISOR inspection at the work location. Document WORK GROUP SUPERVISOR approval to remove Red Tag on the Tag Occurrence Report Form.
   - For situations that the Red Tag stub and WORK GROUP SUPERVISOR cannot be located perform the following:
     - WORK GROUP and OPERATIONAL CONTROL GROUP supervisors shall inspect the work area to determine if Red Tag(s) can be safely removed.
     - The OPERATIONAL CONTROL GROUP shall complete a TAG OCCURRENCE form. The Red Tag Occurrence Report Form will be substituted for the Red Tag stub. (Appendix D)
     - The WORK GROUP superintendent/manager shall document authorization to remove the Red Tag(s) on the Tag Occurrence Report Form. The associated Red Tag(s) shall not be removed prior to WORK GROUP Superintendent approval.
     - When a WORK GROUP SUPERVISOR cannot be located and Red Tag stub is not lost the same procedure is used, except the WORK GROUP Superintendent/designee shall sign the Red Tag stub.

   Contaminated Red Tag or stub may be photocopied in poly. The photocopy may then replace the contaminated tag or stub.

500.7.2.10 Tags Lifted For Testing (TLFT)

Red Tags may be temporarily removed for testing. TLFT should be used for simple task only. For more involved tasks, the original tags should be cleared and a new set issued. The following is applicable for TLFT:

- The OPERATIONAL CONTROL GROUP SUPERVISOR/designee shall evaluate the need to TLFT versus perform an additional tagout.

- The Red Tag and locking devices shall be removed only if WORK GROUP SUPERVISOR/designee approval is documented on back of Red Tag stub.
- Person lifting Red Tag stub shall initial and date back of Red Tag after removal.
- Person replacing Red Tag stub shall verify correct component position and document with date/time/initials on the TLFT portion of the tag.
- Lockout/blocking devices shall be reinstalled as appropriate and reverification of Safe Working Conditions shall be performed.
- If the TLFT portion of Red Tag is completely filled, a new Red Tag should be issued.

Note: The OPERATIONAL CONTROL GROUP is responsible for correct component alignment.

500.7.2.11 Red Tag Audits

1. Each OPERATIONAL CONTROL GROUP shall administratively review tagouts periodically. The periodic review of tagouts (R&R's) by Operations should have a licensed or previously licensed operator involved because of the level of knowledge systems and equipment is necessary to accurately evaluate the continuing need for a tagout. This review shall be documented.
2. An audit of randomly selected tags and tagouts shall be performed at least quarterly. All active tags should be subject to an audit during the course of a year. Audits include the following checks:
   - Proper placement of each safety tag
   - Proper position of tagged equipment
   - Accuracy, completeness, and legibility of information on tags and tagout sheets
   - Condition of tags
   - Continued need for tags issued for more than 6 months.
3. At least annually, also include interviews with a representative number of authorized and affected employees for the purpose of determining employee knowledge of the program.
4. During the course of performing audits, a tour of plant spaces should be made to check for the presence of unauthorized tags, superseded tags, and tags for which no record exists.
5. Deficiencies noted in the audits and tours shall be documented and corrective action taken. If a number of deficiencies were found, the extent of the audit should be increased to provide statistical assurance that the remaining tagouts are in order.

500.7.2.12 Red Tag Process Exceptions

1. Red Tag Logs / Tagout Sheets

   OPERATIONAL CONTROL GROUPS outside of Operations shall control removal and restoration of equipment with processes approved by the Operations Superintendent.
2. Exceptions to OSM or designated SRO involvement in the Red Tag issue and removal authorization process is the responsibility of the Operations Superintendent and is part of OPERATIONAL CONTROL GROUP determination.
3. Exceptions to the use of the Red Tag Computer for the TAGOUT process shall be the responsibility of the Operations Superintendent.
4. The use of any check valve for a tagout isolation boundary, shall be approved by the Operations Shift Manager.

Exam Question Report

27-Jan-99

Question ID:	TAGS035	Revision No:	0	Revision Date	10/29/1999
Question Description:	TAGS035				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	NSD 500
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	NLO = 12 Reference: NSD-500	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

Given the following conditons:

• 1A MD EFDW Pump has been removed from service for motor bearing repair.

• Maintenance has completed repairs.

• The breaker tag has been lifted for testing to check motor rotation. (Pump is Uncoupled)

• While the 1A MD EFDW pump motor was running for rotation checks, the orginial motor bearing problem is identified as NOT having been properly repaired.

Which ONE of the following statements describes the correct handling of the safety tag lifted for testing and the associated stub?(.25)

The tag...

A) must be cleared

B) and stub should be marked "VOID" and filed.

C) may remain lifted following the testing in order to complete the maintenance.

D) should be replaced on the breaker and stub returned to work supervisor.

Answer

D
D-Correct
A-Incorrect Section 500.7.2.10 describes using Temporary Lifts for simple tasks only. This is a simple task and does not require the tag to be cleared. The tag is replaced on the component to allow further work to be performed.
B-Incorrect (See answer Explanation for A) Can not Void a tag in this instance.
C-Incorrect (See answer Explanation for A) The temporary lift tag must be rehung on the breaker to continue work.

QUESTION # 69

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	**Gen**	**Gen**
	Group #	____	____
	K/A #	**Gen-Equip Cont 2.2.30**	
	Importance Rating	**3.5**	**3.5**

Technical Reference(s): **IC-NI OP/1502/07 L/P #2.5**
ITS 9.2.2

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-NI OBJ. #27**

Question Source:
Bank # ____
Modified Bank # __FH-01__
New ____

Question History:
Previous NRC Exam ____
Previous Quiz / Test ____

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ____

10 CFR Part 55 Content:
55.41 ____
55.43 ____

Comments:

1 POINT

QUESTION # 69

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Unit 2 in MODE 6
- Fuel handling operations in progress in the SFP
  - new fuel receiving in progress
  - spent fuel storage in progress
- Fuel handling operations in progress in the RB
  - core alteration in progress

CURRENT CONDITIONS:

- Only **<u>ONE</u>** Source Range NI is operable

Which ONE of the following is correct?

A. Continue fuel handling operations in the SFP and RB.

B. Suspend fuel handling within the core until an additional NI is operable.

C. Suspend <u>ALL</u> fuel handling operations until an additional NI is operable.

D. Continue fuel handling operations in the core after verifying boron concentration within required limits of the COLR.

similar to loss of RIA question 42
But ok - need to compare
K/As to ensure no overlap

1 POINT

QUESTION # 69

Gen.-Equipment Control 2.2.30 Both PRA 2-8-00

A. Incorrect - 1 operable NI is required during refueling operations and 1 additional NI is required when moving fuel within the core (core alterations)

B. Correct – Core alterations must be stopped immediately but fuel-handling operations within the SFP can continue. No NIs are required for SFP fuel handling. 2 NI's are required to alter the core arrangement.

C. Incorrect – SFP fuel handling operations can continue.

D. Incorrect – Core alterations must be stopped immediately. Boron concentration is required if no NIs are operable.

25. When given a copy of Improved Tech. Specs / SLCs and associated Bases., analyze a given set of plant conditions for applicable: (R27)

    -LCOs

    -Conditions and Required Actions

    -maximum Completion Time(s) allowed for required actions

26. Discuss the effects of a loss of power on:

    26.1 Nuclear Instruments (R28)

    26.2 Chessel Recorder (R29)

27. Given various plant conditions, explain actions to take upon loss of any or all NIs. (R30)

30. Describe the individuals ' responsibilities as directed by NSD 104, Housekeeping, Material Condition, and Foreign Material Exclusion. (R30)

31. Evaluate the effect on fuel movement/refueling from the loss or degradation of various components. (R31)

2. Mode 6: Refueling S/D—**One** SR monitor shall be operable and **One additional** SR shall be operable during Core Alterations and during positive reactivity additions. (**ITS 3.9.2**)
   a) **If** one required SR becomes inoperable during core alterations or positive reactivity additions, **then**

      -Immediately suspend core alterations and positive additions

   b) **If** No required SRs are operable, **then** immediately take actions to fix one.

   c) The Defueling / Refueling procedure requires that the two SRs to be used are "designated". If something should happen to one of the two designated SRs; then stop and have Engrg evaluate.

B. Wide Range NIs

1. Mode: 2, or

   Modes: 3, 4, and 5 with any CRD trip breaker in the closed position.

   Two Wide Range NIs are required to be operable. (**ITS 3.3.10**)

   a) **If** one required channel becomes inoperable **then**,

      -Reduce thermal power to <4E-4% in two hours

   b) **If** two required channels become inoperable **then**,

      -suspend all positive reactivity addition operations immediately and in 1-hour trip open all CRD breakers

2. During startup overlap between the Wide Range and the Source Range instruments shall not be less than one decade. (ITS SR 3.3.10.3)

C. Power Range NI (**ITS 3.3.1**) Modes: as applicable per Table 3.3.1.-

1. **Three** Power Range instruments are required to be operable.

   a) Guidance for PRNIs operability is covered in the ITS for RPS including **Table 3.3.1-1.** Guidance is also provided for use of dummy bistables and for manual bypass of RPS channels fed by the Power Range instruments in the ITS for RPS.

   b) **If**, one required PR becomes inoperable **then**, trip RPS Channel w/in 1 hour.

   c) **If** , two or more PRs become inoperable **then**, be in Mode 3 and open all CRD trip breakers w/in 12 hrs.

2.5 When in MODE 6, ITS requires no less than two flux monitors shall be in operation whenever core geometry is being changed **OR** when positive reactivity additions are being made to the core. When core geometry is **NOT** being changed, at least one Neutron Flux Monitor shall be in service.

2.6 Two trains of Spent Fuel Pool Ventilation shall be operable with the following exceptions:

2.6.1 With one train of Spent Fuel Pool Ventilation inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the operable Spent Fuel Pool Ventilation train is in operation and discharging through the Reactor Building Purge Filters.

2.6.2 With no Spent Fuel Pool Ventilation Filter operable, suspend all operations involving movement of fuel within the storage pool or crane operations with loads over the storage pool until ITS 3.7.17 is satisfied.

2.7 At least one Low Pressure Injection Pump and Cooler shall be operable. However, a pump may be stopped temporarily at the discretion of the SRO in charge of refueling per ITS 3.9.4.

CAUTION: **IF** a direct flow path from the Reactor Building atmosphere to the outside atmosphere is discovered, immediately suspend all operations involving core alterations or movement of irradiated fuel in the Reactor Building.

2.8 The following Reactor Building Containment Closure guidelines shall be followed:

2.8.1 Provide containment penetration closure per OP/1/A/1502/009 (Containment Closure Control).

2.8.2 To verify operability of RB Purge isolation, ITS SR 3.3.16.2 **AND** SR 3.9.3.2 must be performed prior to fuel movement. RB Purge isolation must remain operable during fuel movement or the flow path must be isolated per ITS 3.3.16.

2.8.3 **IF** the secondary side of a Steam Generator is isolated inside the RB, then the secondary side of that Steam Generator must be intact (i.e. secondary manways, handholes, nozzles, etc. installed).

2.8.4 During the handling of fuel in the RB, at least one door on the personnel hatch and one door on the emergency hatch shall be closed. The equipment hatch cover will be in place with a minimum of four bolts (approximately 90° from each other) securing the cover to the surfaces. This must be inspected within 100 hours of refueling.

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2

a. One source range neutron flux monitor shall be OPERABLE, and

b. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS and during positive reactivity additions.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required source range neutron flux monitor inoperable during CORE ALTERATIONS or positive reactivity additions.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
B. No OPERABLE source range neutron flux monitors.	B.1	Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND		
	B.2	Perform SR 3.9.1.1.	4 hours AND Once per 12 hours thereafter

Exam Question Report

27-Jan-99

Question ID:	FH001	Revision No:	0	Revision Date	10/29/1999
Question Description:	FH001				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	FH-T53 - LOC - Fuel Loading and Refueling
Last Used Date:		Question Type:	Fill in the Blank
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 2; SRO = 2	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

According to Tech. Spec. 3.8, Fuel Loading and Refueling, during refueling operations, when core geometry is being changed, core subcritical neutron flux shall be continuously monitored by ____________________. (.25)

Answer

at least two neutron flux monitors.

Lessons

ID	Description
FH-T53	Fuel Loading and Refueling Limiting Condition For Operation

Enabling Objectives

ID	Description

Referenced Documents

ID	Description	Review Date	Ref Flag

KA'S

ID	Description

QUESTION # 70

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**T3**	**T3**
	Group #	**C2**	**C2**
	K/A #	**G 2.2.22**	
	Importance Rating	**3.4**	**4.1**

Technical Reference(s): **ADM OMP 2-12**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **ADM OMP-1**

Question Source:
- Bank # ______
- Modified Bank # ______
- New __X__

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 __X__

Comments:

1 POINT

QUESTION # 70

Which ONE of the following conditions **REQUIRES** prior SRO approval/oversite?

A. Withdrawal of Group 1 CRDs to 50%.

B. Placing "A" HPI pump to manual following an ES actuation.

C. Manually tripping the reactor when CRD temperatures exceed 180°F.

D. Performance of RULE # 2, Loss of SCM Actions, when core SCM is 0°F.

K/A does not match

1 POINT

QUESTION # 70

Gen. Equipment Control 2.2.22

A. Correct – Reactivity Management requires SRO approval to add positive reactivity per OMP 2-12, Reactivity Management.

B. Incorrect – Placing an ES component to manual does not reposition the component and does not require SRO approval unless the component operational state will be changed.

C. Incorrect - The RO has the authority to manually trip the reactor without SRO approval

D. Incorrect – OMP 1-18 requires the operator to perform EOP RULES without SRO guidance.

OBJECTIVES

TERMINAL OBJECTIVES

1. When given a copy of applicable OMP's or sections, be able to demonstrate an understanding of the guidance or rules within the OMP. (T1)
2. Become familiar with and be able to use and follow the requirements and information contained in the Operations Manual. (T2)

ENABLING OBJECTIVES

1. Be able to answer questions specific to the duties and responsibilities of an operator by referencing applicable copies or portions of the Operations Manual for the following OMP's:

   1.1 OMP 1-2, Rules of Practice (R6)

   1.2 OMP 1-7, Emergency Response Organization (R7)

   1.3 OMP 1-9, Use of Procedures (R8)

   1.4 OMP 1-17, Key and Locks (R9)

   1.5 OMP 1-18, Implementation Standard during Abnormal and Emergency Events(R10)

   1.6 OMP 1-20, Control of Test and Measuring Equipment (R30)

   1.7 OMP 1-22, Job Assignments (R11)

   1.8 OMP 2-1; Duties and Responsibilities of On-Shift Operations Personnel (R12)

   1.9 OMP 2-4, Recorder Charts (R13)

   1.10OMP 2-5, Relay Trips of Station Breakers (R14)

   1.11OMP 2-7, SSF-LCO Required Actions (R15)

   1.12OMP 2-9, Control Room Indication Checklist (R17)

   1.13OMP 2-10, Promotion Criteria for Non-Exempt Shift Personnel (R18)

   1.14OMP 2-12, Reactivity Management (R19)

   1.15OMP 2-13, Weekly Check of Emergency Communications Equipment (R20)

   1.16OMP 2-14, Operations Test Group Use of Blue Tags (R29)

1.17 OMP 2-16, Shift Turnover (R33)

1.18 OMP 3-1, Operations Training (R21)

1.19 OMP 3-9, New RO/SRO Mentoring Guides (R34)

2. Be able to recite, from memory, any required procedure or administrative items as detailed in OMP 1-18, Licensed Operator Memory Items Attachment: (R1)

   2.1 The student is not required to be able to list each item in the attachment from memory.

   2.2 The student is expected to be able to recall from memory those actions or statements listed in the attachment as they relate to the specific task or evolution being performed.

3. When given a copy of the Operations Manual, or portions thereof, be able to demonstrate an understanding of the guidance or rules within specific OMP's by locating the answer to or interpreting required responses for a given situation. (R2)

4. The operator will become well versed in the requirements set forth in the following OMP's, in order to meet the expectations of Operations Management and conduct safe reliable operations of all Oconee units at all times. The operator will comprehend and exercise the OMP as it relates to the following conditions:

   4.1 OMP 1-2, Rules of Practice (R3)

      A. Acceptable means of operator conduct and operational practices.

      B. Limits for acceptable work schedules.

      C. Minimum shift staffing requirements.

   4.2 OMP 1-9, Use of Procedures (R4)

      A. Provide guidance to the operator in the following areas concerning procedures:

         - establish consistent methods for using procedures
         - control of approved procedures
         - use of approved procedures
         - completion of procedures
         - control of procedure changes
         - deviation from approved procedures

4.3 OMP 1-10, Usage and Testing of the NRC Emergency Notification System (ENS) (R26)

A. Use and testing of the ENS.

B. Determine reportability of an event.

C. Making a report, including follow-up as needed.

4.4 OMP 1-12, NRC License Maintenance (R27)

A. Requirements for maintaining an active NRC license.

B. Methods for restoring an inactive license to active status.

4.5 OMP 1-24, Operations Communications Standard (R32)

A. Responsibilities

B. Principles of Effective Communications

C. Crew Update

D. Crew Briefing

4.6 OMP 2-1, Duties and Responsibilities of On-Shift Operations Personnel (R5)

A. Roles and responsibilities of the OSM

B. Responsibilities of the Control Room SRO

C. Responsibilities of the Plant SRO

D. Responsibilities of the Reactor Operators

E. Responsibilities of non-licensed operators

F. Responsibilities of Refueling SRO and RB SRO

G. Normal lines of communication and shift organization during plant operation

H. Required boundaries within the control room to ensure the controls are adequately monitored by the operator

I. Shift Staffing Requirements

4.7 OMP 2-2, Unit Log (R24)

A. Maintain the Unit Log in a manner that documents the shift activities, accomplishments and problems.

4.8 OMP 2-7, SSF LCO Required Actions (R25)

A. Define actions required when SSF equipment is declared inoperable.

B. Identify criteria for determining SSF operability.

4.9 OMP 2-12, Reactivity Management (R23)

A. Controlling reactor power during system transients.

B. Management expectations in the mitigation and control of system transients during power operation.

C. Management expectations during manual rod pulls.

4.10 OMP 4-7, Reactor Coolant Leakage Safety Evaluations (R28)

A. Prepare and document a Reactor Coolant Leakage Safety Evaluation as required by ITS.

5. Given a set of conditions, use guidance given in OMP 1-25, (Safety Function Determination Program (SFDP) to: (R31)

5.1 Ensure a loss of safety function is identified and appropriate actions are taken when LCO 3.0.6 are in place.

5.2 Ensure that multiple occurrences of Support Feature inoperability cannot cause the Supported Feature completion time to be inappropriately extended.

OCONEE NUCLEAR STATION
OPERATIONS MANAGEMENT PROCEDURE 2-12
REACTIVITY MANAGEMENT

4.4 **All** manual rod pulls shall be performed under SRO oversight (similar to control rod pulls for criticality). This includes manual rod pulls at power and a sub-critical reactor (e.g. Group 1 to 50%). Communication for these rod pulls shall meet the guidance found in OMP 1-24 (specifically, three way communications shall be used). The RO still has the authority and responsibility to insert control rods or manually trip the reactor during any system transient.

4.5 All reactivity changes shall be communicated with the CR SRO prior to the change (following the communication requirements found in OMP 1-24). The expectation is that the RO and CR SRO fully understand the need for the reactivity change and the expected reactor response. Some examples of activities that require prior notification to the CR SRO are demineralizer activities, water additions to the RCS, and ICS manual operations.

Exception:

The RO still has the authority and responsibility to insert control rods or manually trip the reactor. Communication with the CR SRO is desired during these types of transients, but conditions may prevent this communication.

4.6 During Shift Turnover, there should be NO manual positive reactivity changes on-going. This would include no manual control rod pulls, no power increases and no deboration. However, the intent of this policy is not to prevent the operator from taking conservative action when reacting to plant conditions. For example, if the unit has experienced a transient and Xenon is building in, the operator would be allowed to add demin water to offset the Xenon change during the turnover. The turnover process would then include the actions taken and the expected response of the action.

There may be times when decisions occur that go outside the above statement. When this occurs there needs to be boundaries in place to protect both the off-going and oncoming crews. Therefore, decisions to allow operation outside the above statement should:

- be rarely used
- be prearranged
- require dedicated individuals
- require approval of the Operations Shift Manager

Example: CRD Patch Verification

QUESTION # 71

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**3**	**3**
	Group #	**C3**	**C3**
	K/A #	**G 2.3.9**	
	Importance Rating	**2.5**	**3.4**

Technical Reference(s): **OP/1104/35 L/P PNS-RBP**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-RBP #6**

Question Source:
- Bank # ______
- Modified Bank # ______
- New ___X___

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 __X__

Comments:

1 POINT

QUESTION # 71

Unit 2 Plant conditions:

- Core is defueled
- Unit 2 RB Purge is in progress
- RP has requested that the SFP Filtered Exhaust System be used to ventilate the SFP to reduce airborne contamination

Which ONE of the following is required to place the Spent Fuel Pool Filtered Exhaust System in service?

A. Start either F1 or F2 filter fan.

B. Start both F1 and F2 filter fans.

C. Secure the RB Purge fan then start either F1 or F2 filter fan.

D. Secure the RB Purge fan then start both F1 and F2 filter fans.

Objective Does not match
K/A knowledge of process for performing
a containment purge.

1 POINT

QUESTION # 71

G2.3.9

A. Incorrect – The RB Purge must be secured prior to starting the SFP Filtered Exhaust fan.

B. Incorrect – Both SFP Filtered Exhaust fans cannot be operated at the same time.

C. Correct – The Spent Fuel Pool Filtered Exhaust System is interlocked with the RB Purge system. The Spent Fuel Pool Filtered Exhaust System cannot be started unless the RB Purge is secured.

D. Incorrect - Both SFP Filtered Exhaust fans cannot be operated at the same time

4. Explain the purpose for each RBP system interlock and when given plant conditions: (R4)

   - Predict system/component/indication response to RBP system interlock actuation.
   - Describe necessary actions and/or plant status required to return system/component/indication to normal operating status.

5. Describe the response of the RBP system to a "High" alarm on RIA-45 or RIA-46. (R5)

6. Given the enclosure for removal/restoration of the Equipment Hatch: (R6)

   - Explain the personnel safety considerations that result in stopping the RB Purge fan prior to Equipment hatch removal/replacement
   - Describe the purpose of the desired minimum flow limit on RB Purge flow with the Equipment hatch removed
   - Describe the major difference in the enclosures for Equipment hatch removal/replacement with and without the RB Purge in operation

7. Describe the response of the RBP system to an actuation of Engineered Safeguards channels 1&2. (R7)

8. Given a specific RBP Limit and Precaution describe the purpose for the Limit or Precaution. (R8)

9. Given a specific set of conditions, determine if "favorable" or "unfavorable" conditions exist for a release. (R9)

10. Discuss the "required level of approval" for the various releases which may be in progress at the Station. (R10)

11. Given a copy of ITS/SLC's and associated Bases, analyze a given set of plant conditions for applicable ITS/SLC LCO's.

Spent Fuel Pool Filtered Exhaust System

1. Purpose

Provides procedural guidance for operation of the SFP Filtered Exhaust System in filtered mode utilizing Unit 2 or Unit 3 RB Purge filters.

2. Limits and Precautions

2.1 The RB Purge Fan should NOT be operated at the same time the associated SFP Filtered Exhaust System is on.

NOTE:	Tagging may be waived for short term opening of Fuel Receiving Area roll-up door provided no fuel handling is in progress.

2.2 When the SFP Filtered Exhaust System is NOT operable, the following equipment associated with specific unit's filtered exhaust system should be white tagged:

- Unit 1&2 spent fuel bridge/hoist PIE crane (1XP F4AB).
- Unit 1&2 spent fuel pool crane (100 ton) (1XUA F1F).
- Unit 3 spent fuel bridge/hoist crane (3XR 4CB).
- Unit 3 spent fuel pool crane (100 ton) (3XR 1B).

2.2.1 Tags should be removed when the SFP Filtered Exhaust System is returned to service.

NOTE:	This is an operating decision to minimize release of radioactivity should a fuel handling accident occur and NOT an ITS requirement.

2.3 A dedicated lock is placed on the roll up door breaker during fuel movement operations or when load is suspended over the SFP.

3. Procedure

3.1 Perform Enclosure "Operating Unit 1&2 SFP Filtered Exhaust System" to operate Unit 1&2 SFP Filtered Exhaust System.

3.2 Perform Enclosure "Operating Unit 3 SFP Filtered Exhaust System" to operate Unit 3 SFP Filtered Exhaust System.

RB Purge System

1. Purpose

1.1 To provide procedural guidance for operating RB Purge System.

2. Limits and Precautions

2.1 RB Purge filters should be changed:

- When radiation level exceeds 20 mR/hr at 1 ft.

  or

- When filter ΔP reaches alarm setpoint.

2.2 RB Purge filter alarm setpoints are:

- Absolute and pre-filter ΔP of 1.75 inches H_20.
- Carbon filter ΔP of 1.5 inches H_20.

2.3 Operation of RB purge valves only allowed in Mode 5 or 6.

2.4 When equipment hatch is open:

- RB Purge flow should be > 10,000 cfm.
- RB Mini Purge Fan should <u>NOT</u> be operated.

NOTE:	Proper venting through vent stack reduces potential for an unmonitored release with equipment hatch removed.

2.5 If equipment hatch removal required while RB Purge is <u>NOT</u> available, RB should be vented to vent stack prior/during removal.

2.6 During refueling operations with 1SF-1&2 open and equipment hatch installed, RB Purge provides vent for compressed air used inside RB.

- If RB Purge stopped, RB pressure may increase resulting in changes to FTC level and SFP level.

2.7 Meteorological conditions should be considered when performing RB purges.

2.8 If unit vent radiation monitors inoperable during RB Purge, grab samples must be taken daily.

h) Close the following valves:

- PR-1 (RB Purge Outlet (RB)).
- PR-2 (RB Purge Outlet (PR)).
- PR-3 (RB Purge Control).
- PR-4 (RB Purge Inlet).
- PR-5 (RB Purge Inlet (PR)).
- PR-6 (RB Purge Inlet (RB)).

i) Complete R. B. Gaseous Waste Release Form and route to RP.

REFER to OC-PNS-RBP-3

D. Equipment Hatch Removal and Replacement:

- Purge and Mini-Purge Fan operation is suspended for Hatch Removal to prevent air from being drawn in around the hatch for safety reasons during removal. If the fan was running when the last bolts of the Equipment Hatch were loosened, the hatch could swing inward.

1. Permission is required from RP and the Shift Manager prior to hatch removal.
2. Equipment Hatch Removal

a) Stop the running Purge Fan and close PR-3.

b) Red Tag "open" the following breakers:

- Main Purge Fan breaker red tagged.
- Mini Purge Fan breaker red tagged.

c) Grant permission to remove Equipment Hatch provided no work is in progress that would prevent running of the Purge Fan.

d) When the Equipment Hatch has been removed, resume purge of the Reactor Building using the Main Purge Fan.

1) This purge should continue as long as the Equipment Hatch is open.

2) Place Purge back on under existing GWR, but using the Main Purge Fan only at > 10,000 cfm, but < than GWR limit.

- This ensures a negative pressure at the hatch area, so air is drawn into the Reactor Building.

e) Note hatch open on SRO turnover sheet.

3. Equipment Hatch Replacement

a) Secure the running Main Purge Fan.

b) Red tag open the following breakers:

- Main Purge Fan breaker
- Mini Purge Fan breaker

c) Grant permission to replace the Equipment Hatch.

d) When the Equipment Hatch is in place, remove red tags on the Main and Mini Purge Fan breakers.

e) Rack in the breaker on the Reactor Building Main Purge Fan.

f) If desired, continue purge of the Reactor Building under the existing GWR and notify RP.

2.3 Abnormal Operations

A. Engineered Safeguards Operation

1. The R.B. Purge Isolation Valves must be operable under the most severe design-basis-accident (DBA) flow-condition loading and these valves must close within their design time limit under these conditions.

- By meeting their operability requirements under these conditions adverse amounts of radiation will not escape the containment building, following an accident.

2. Non-essential Containment Isolation Valves must actuate immediately following any size LOCA.

a) To achieve this actuation, low reactor coolant system pressure of 1600 psig is used as the initiating signal to actuate E.S. digital channels 1 and 2.

b) A diverse signal will also actuate these channels. This signal is a High Reactor Building Pressure of 3 psig.

3. The R.B. Purge Isolation Valves are non-essential containment isolation valves and receive signals to isolate as follows:

Valve	Channel

- This RTD is located downstream of the inlet heat exchanger and controls the steam flow to the heating coil when steam is valved into the coils.

2.6 Limits and Precautions

A. R.B. Purge System

1. Filters should be changed when the radiation level exceeds 20 mR/hr @ 1 ft. or the filter DP reaches the alarm point.
2. If low flow is experienced through the filters, the filters should be performance tested.
3. If equipment hatch removal is required while RB Purge is NOT available, the RB should be vented (via PR- 1, 2, &3) to the stack prior/during removal.
4. If Unit Vent RIA's inoperable during RB Purge operations, alternate sampling should be established per SLC 16.11.3-2.
5. Purge valve operation is allowed if RCS is in MODE 5, 6, or NO MODE. (Instructor Note: To clarify when Purge operation is allowed).
6. When the equipment hatch is open with the RB Purge system available, the RB Purge flow should be > 10,000 cfm to prevent release to the environment. The mini-purge should not be operated.
   - **Instructor Note: Tests have shown that a low flow rate does not ensure positive air flow into the RB through the hatch.**
7. There is also an enclosure in OP/1/A/1102/014 that allows opening the Equipment Hatch with the Purge Fan not available. This enclosure vents all positive RB pressure out of the stack prior to removal of the Equipment Hatch.
8. During a refueling outage when the Equipment Hatch is installed, the RB Purge System provides a vent flowpath for compressed air used inside the Reactor Building.
   - If the Purge Fan is stopped, Reactor Building pressure will gradually increase, resulting in a gradual decrease in Fuel Transfer Canal level and a subsequent increase in SFP level.
9. Reactor Building purges should be coordinated, as far as practicable, with favorable meteorological conditions.

QUESTION # 72

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**T3**	**T3**
	Group #	**C3**	**C3**
	K/A #	**G 2.3.4**	
	Importance Rating	**2.5**	**3.1**

Technical Reference(s): **RAD-RPP**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **RAD-RPP 8**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 ______

Comments:

1 POINT

QUESTION # 72

Which ONE of the following plant areas is posted **INCORRECTLY** based upon recent sample-survey results?

A. Turbine Building 5th floor / Contaminated Area – 125 dpm/100 cm^2 β-γ (loose)

B. Unit 1 LDST Hatch Area / High Radiation Area – 210 mrem/hr @ 30 cm

C. Unit 2 Powdex Filter / Hot Spot – 273 mrem/hr on contact

D. Unit 3 CBAST Room / Radiation Area – 7 mrem/hr @ 30 cm

1 POINT

QUESTION # 72

G2.3.4 PRA Bank RAD061 modified

A. Correct – A Contaminated Area is > 1000 dpm/100 cm^2 β-γ .

B. Incorrect – A High Radiation Area is > 100 mrem/hr @ 30 cm.

C. Incorrect – A Hot Spot is 100 mrem/hr on contact.

D. Incorrect – A Radiation Area is > 5 mrem/hr @ 30 cm.

ENABLING OBJECTIVES (continued)

4. State the approval requirements for an individual at Duke Power Company to exceed the **basic** permissible exposure limit of 2.0 rem. (R4)

5. State the special dose limits established for the general public. (R5)

6. Describe the special dose control measures used to protect the fetus of a "declared" pregnant radiation worker. (R6)

7. Recognize that in "exceptional situations", it is possible to allow an adult radiation worker to receive additional exposure, apart from normal occupational exposure. (R7)

8. Define and describe the specific site area for each of the following terms relating to the control of station areas: (R8)

   8.1 Unrestricted Area

   8.2 Restricted Area

   8.3 Controlled Area

   8.4 Radiation Control Area (RCA)

   8.5 Radiation Control Zone (RCZ)

   8.6 Radiation Area (RA)

   8.7 High Radiation Area (HRA)

   8.8 Extra High Radiation Area (EHRA)

   8.9 Very High Radiation Area (VHRA)

   8.10 Airborne Radioactivity Area

   8.11 Hot Spot

   8.12 Significant Dose Contributor

   8.13 Low Exposure Waiting Area

   8.14 Contaminated Area

G. Radiation Area

Any area in which there exists radiation levels such that a major portion of the body could be exposed to > **5 mrem/hour** (.005 rem/hour) at 30cm is, by 10CFR20, a Radiation Area.

H. High Radiation Area (HRA)

Any accessible area to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent in excess of **100 mrem in one hour, but $\leq$ 1000 mrem in one hour**, at 30 cm from the radiation source.

I. Extra High Radiation Area (EHRA)

1. An area where major portions of the body may be exposed to > **1000 mrem/hr** (@ 30 cm) **but$\leq$ 500 rads in an hour** at 1 meter.
2. Some EHRA's are located within large open areas, such as the RB's, where locked access or enclosures cannot be reasonably constructed. These EHRA's will be barricaded, posted as EHRA, and a yellow flashing light will be activated in lieu of being locked or guarded.

J. Very High Radiation Area (VHRA)

1. An area accessible to individuals where they could receive a dose > **500 rads in one hour at one meter**. (RADs, which are the units of absorbed dose, are used rather than rems, which are units of equivalent dose, when doses at very high dose rates are involved).

   NOTE: At very high doses received at high dose rates units or absorbed dose (rads) are appropriate, rather than units of dose equivalent (rems).

2. At Oconee, some probable VHRA's, depending on operating conditions, are:

   ·the reactor annulus area

   ·incore wire cage area

   ·fuel transfer tube areas

K. Entry Requirements for HRA's, EHRA's, and VHRA's

1. At one time, all HRA's were maintained locked, with special entry requirements and key controls. With the new EDC System and alarming MG's, HRA's are no longer required to be maintained locked. Entry into HRA's are controlled per the RWP/SRWP.

2. Personnel requiring access to EHRA's and VHRA's **must** contact RP for permission, and so that RP can provide **continuous** coverage for the entry.

3. In addition to general RP permission, entries into VHRA's requires the approval of the RP Manager.

L. Key Issue and Control

While HRA's are no longer typically required to be maintained locked, some HRA's may remain that way for control purposes. Where EHRA's or VHRA's are able to be controlled with locking devices, the following guidelines also apply.

1. Permanent HRA and EHRA master keys and individual HRA room keys may be issued to station management if required. **Key issue for the LDST room hatch area is controlled by the Shift Supervisor.** Entry into the LDST room is still controlled by RP whenever EHRA conditions exist.

   EHRA master keys may only be issued to the Operations Shift Manager, and these keys must be maintained in a sealed key box.

2. Keys to VHRA's shall only be issued to RP. RP Manager must approve each entry.

3. Before keys may be issued for entrance into a reactor annulus area, the incore wires must be fully inserted into the reactor core such that there is no possibility that they will be withdrawn while personnel are in the annulus area.

4. For access to the incore wire caged areas, the incore wires shall be fully inserted into the core, or "parked" such that there is no possibility of moving them while people are in the area.

5. To access ladders to the shielded fuel transfer tubes, no fuel assemblies or irradiated components shall be moved through a tube while personnel have access.

6. Entries into the reactor annulus area, incore wire caged area, and fuel transfer tubes area, are controlled by RP. An RP supervisor is responsible for ensuring that entry conditions are appropriate, and that nothing will be done to change the radiological conditions while people are in the area. Continuous RP coverage is required for entries into these areas, and a minimum of one pre-job briefing among all of the individuals involved shall be conducted.

7. Access into HRA's, EHRA's, and VHRA's shall be limited to Level II radiation workers.
8. If the entrance into a HRA, EHRA, or VHRA is controlled by a guard, the guard must be stationed in a location so that positive control over the area is maintained (normally within 15 feet and within sight of the entrance).
9. The use of padlock to secure doors is permitted **only on a temporary basis** (for 24 hours or less, unless special circumstances require a longer period).
10. Radiation Protection (RP) must approve and install the use of a padlock for securing an entrance.
11. Free exit requirements shall exist at all times when personnel are inside a normally locked area.
12. Anyone who enters a locked area is responsible for ensuring that the entrance into the area is properly secured upon leaving the area (i.e. make certain that the door closes and locks or that the area is guarded). Each person leaving the area is responsible for performing a door check upon exit (close the door, turn the knob, and pull the door to ensure that it is secured).
13. Anyone finding a radiation door open or unlocked, and not guarded, shall perform a door check. If the door cannot be secured, notify RP and your supervisor from the nearest phone, and then stand guard over the entrance until relieved.

    NOTE: The RP Shift Supervisor notified of a malfunctioning door is responsible for ensuring that a Priority Work Request is written to repair the door.

M. Contaminated Area and Personnel Contamination Monitoring

1. A **Contaminated Area** is designated whenever <u>removable</u> contamination levels are:
    a) **$\geq$ 1000 dpm/100 cm²** β-γ
    b) $\geq$ 20 dpm/100 cm² α
2. Individuals and personal clothing are considered to be contaminated if contamination levels on them are:
    a) **> 5000 dpm/100 cm²** β-γ**(> 100 cpm above bkg)**
    b) **any net observable** α **(> 20 dpm/100 cm²).**

NOTE: Monitoring of personnel for alpha contamination is only performed when it is expected that alpha contamination is present; i.e., personnel monitoring for alpha contamination is not routinely performed.

N. Airborne Radioactivity Area

The 10CFR20 definition of an Airborne Radioactivity Area is a room, enclosure, or area in which airborne radioactive material exists or has the potential to exist, in concentrations in excess of DAC's specified in App. B; or if a person would exceed 0.6 % of ALI (12 DAC-hours) in one week (40 hours) without a respirator.

At Duke Power it is a room where airborne radioactivity exists in concentrations:

$\geq$ 25% of the weighted Derived Air Concentration (DAC) listed in App. B of 10CFR20 for the specific radionuclide.

$\geq$ 25% of the DAC for alpha.

where an individual, within a week's time, without respiratory protection, could receive an intake of 0.6% of the Annual Limit on Intake (ALI) or 12 DAC hours.

O. Neutron Exposure Area

Any area where personnel are likely to receive a neutron dose at any measurable dose rate is designated as a neutron exposure area. Neutron exposure is not normally a problem at the plant because neutron radiation is confined to the reactor core area.

P. Hot Spot

A hot spot is any location where the contact radiation dose rate (½") exceeds the general area dose rate (30 cm) by a factor of five (5), is > 100 mrem/hr on contact, and where the potential exists for significant exposures.

Q. Significant Exposure Contributor

Significant Exposure Contributors are posted items/components that do not meet hot spot criteria.

R. Low Exposure Waiting Area

A Low Exposure Waiting Area is the lowest exposure area in a room or work area. The Low Exposure Waiting Area will be posted in all rooms with a General Area dose > 10 mr/hr.

QUESTION # 73

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	**3**	**3**
Group #	**C4**	**C4**
K/A #	**G 2.4.12**	
Importance Rating	**3.4**	**3.9**

Technical Reference(s): **EAP-SEP**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-SEP #17**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __X__
55.43 _____

Comments:

1 POINT

QUESTION # 73

Which ONE of the following is correct concerning the notification to offsite agencies during E-Plan implementation?

~~The~~ ________ notifications shall be made within ________ minutes of declaration of the EAL.

A. Initial / 15

B. Initial / 30

C. Upgrade / 30

D. Upgrade / 60

3 Notifications must be made

~~THE~~

1 POINT

QUESTION # 73

G2.4.12

A. Correct – The initial notification must be made within 15 minutes.

B. Incorrect – Initial notification must be made within 15 minutes.

C. Incorrect – Upgrade is 15 minutes.

D. Incorrect – Upgrade is 15 minutes.

15. Given a reading from one of the Control Room Wind Direction Meters be able to determine the direction the wind is coming from and the direction the wind is going. (R-18)

16. Briefly describe the Alert and Notification System for Oconee Nuclear Site and the procedure for its use. (R-13)

17. Describe the duties and responsibilities of the Control Room Offsite Communicators, (RP/0/B/1000/15A) (R-14)

    17.1 List the various communication equipment available at Oconee for making required notifications, (RP/0/B/1000/15A).

    17.2 Given an Emergency Notification Form, be able to complete the appropriate sections of the form for initial notifications, follow-up messages, and change of classifications.

    17.3 Be able to discuss items that should be reviewed or given to the TSC Offsite Communicator during communicator turnover.

B. Control Room Offsite Communicator Duties and Responsibilities

1. The Operations Offsite Communication must use the time reference that the Emergency Coordinator (OSM) is using.

   a) We could miss the 15 minute time limit to notify the Offsite Agencies if the time reference used to log when the Emergency declaration was made and the time reference used to log when the notification to the Offsite Agencies occurred were different!

2. Refer to RP/0/B/1000/15A, Offsite Communications from the Control Room.

3. Notification of Offsite Agencies (State/County)

   a) Complete Emergency Notification Form as outlined in Enclosure 4.1.

      1) Obtain Emergency Description/ Remarks wording from Operations Shift Manager and the Emergency Action Level Guideline Manual.

      2) Obtain Plant Condition/Reactor Status information from Operations Shift Manager.

      NOTE: For an initial Emergency Notification, lines 11-14 are not required.

   b) Provide completed form to Emergency Coordinator (Operations Shift Manager) for his review and approval on line 16.

      1) Initial Emergency Notifications and classification upgrades must be provided to Oconee County, Pickens County, and SC State within <u>15 minutes</u> of event declaration or upgrade.

         (a) The timeframe for making the required notifications ***shall start at the time that the Emergency Coordinator/ EOF Director determines the correct classification for the event.*** (Recorded on Line 6)

         (b) The time at which the State and Counties are considered to have been "notified" of an ***event will be the time recorded on Line 3***. Procedure Step 2.5.4 states that the <u>time</u> at which the number has been dialed and phone begins to rings is recorded on Line 3.

QUESTION # 74

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	**3**	**3**
Group #	**C4**	**C4**
K/A #	**G 2.4.32**	
Importance Rating	**3.3**	**3.4**

Technical Reference(s): **E-Plan RP/1000/01**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-SEP T #2**

Question Source:
Bank # ________
Modified Bank # ________
New ___X___

Question History:
Previous NRC Exam ________
Previous Quiz / Test ________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __X__
55.43 __X__

Comments:

1 POINT

QUESTION # 74

Which ONE of the following events warrants entrance into the associated document?

A. Safety related fire detector out of service / Fire Plan

B. Loss of all Control Room statalarms / Emergency Plan

C. Dropped Control Rod / Emergency Operating Procedure

D. SGTL of 30 gpm / Excessive Leakage Abnormal Procedure

2.4.32 Knowledge of operator response to Loss of all annunciators

2.4.4 Ability to recognize abnormal indications which are entry-level conditions for . . .

1 POINT

QUESTION # 74

G2.4.32

A. Incorrect – An OOS fire detector will place the crew in the SLC's

B. Correct – 52% of all statalarm panels in the control room would exceed the criteria for E-Plan implementation. E-Plan implementation is required if 50% of the following statalarm panels are lost: 1SA-1-9, 14-16, and 18 Incorrect

C. Incorrect – A Dropped Control Rod places the crew into the Dropped Rod AP not the Emergency Operating Procedure

D. Incorrect – A SGTL of 30 gpm will place the crew into the EOP not and in the Excessive Leakage Abnormal Procedure.

OBJECTIVES

Terminal Objective

1. Describe the duties and responsibilities of a non-licensed operator associated with the Site Emergency Plan.(T-1)
2. Describe the duties and responsibilities of a reactor operator associated with the Site Emergency Plan.(T-2)
3. Describe the duties and responsibilities of a Senior Reactor Operator associated with the Site Emergency Plan.(T-3)

Enabling Objectives

1. Define <u>and</u> discuss the following terms associated with the Oconee Nuclear Site Emergency Plan: (R-1)
   - 1.1 Site Boundary
   - 1.2 Protected Area
   - 1.3 Vital Area
   - 1.4 Emergency Planning Zone (EPZ)
   - 1.5 Plume Exposure Pathway
   - 1.6 Ingestion Exposure Pathway
   - 1.7 Protective Actions
   - 1.8 Protective Action Guides (PAG)
   - 1.9 Emergency Action Levels
   - 1.10 Technical Support Center (TSC)
   - 1.11 Operational Support Center (OSC)
   - 1.12 Emergency Operations Facility (EOF)
   - 1.13 Emergency Coordinator
2. Briefly describe the Emergency Classification System used at Oconee. (R-2)
3. Be able to state at what level of the Oconee Emergency Classification System that the TSO, OSC, and the EOF are required to be activated. (R-3)
4. Be able to state at what level of the Oconee Emergency Classification System that Evacuation of Nonessential personnel is required. (R-4)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. RCS LEAKAGE (BD 14) OPERATING MODE: 1, 2, 3, 4 • Unidentified leakage ≥ 10 gpm • Pressure boundary leakage ≥ 10 gpm • Identified leakage ≥ 25 gpm 2. UNPLANNED LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/INDICATION IN CONTROL ROOM FOR > 15 MINUTES (BD 15) OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND Loss of anunciators or indicators requires additional personnel (beyond normal shift complement) to safely operate the unit 3. INABILITY TO REACH REQUIRED SHUTDOWN WITHIN LIMITS (BD 16) OPERATING MODE: 1, 2, 3, 4 • Required operating mode not reached within TS LCO action statement time (CONTINUED)	1. UNPLANNED LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/INDICATION IN CONTROL ROOM (BD 19) OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND Loss of anunciators/indicators requires additional personnel (beyond normal shift complement) to safely operate the unit AND EITHER OF THE FOLLOWING: • Significant plant transient in progress OR • Loss of the OAC and ALL PAM indications (END)	1. INABILITY TO MONITOR A SIGNIFICANT TRANSIENT IN PROGRESS (BD 21) OPERATING MODE: 1, 2, 3, 4 • Unplanned loss of > 50% of the following anunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1-9, 14-16, and 18 3 SA1-9, 14-16, and 18 Unit 2 2 SA1-9, 14-16 AND A significant transient is in progress AND Loss of the OAC and ALL PAM indications AND Inability to directly monitor any one of the following functions: 1. Subcriticality 2. Core Cooling 3. Heat Sink 4. RCS Integrity 5. Containment Integrity 6. RCS Inventory (END)	
INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4	INITIAL NOTIFICATION REQUIREMENTS: SEE EMERGENCY TELEPHONE DIRECTORY NOTIFY 1, 2, 3, 4

QUESTION # 75

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	**T3**	**T3**
	Group #	**C4**	**C4**
	K/A #	**G 2.4.16**	
	Importance Rating	**3.0**	**4.0**

Technical Reference(s): **TA-AT**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **TA-AT #1.1**

Question Source:
- Bank # ______
- Modified Bank # **TA-26**
- New ______

Question History:
- Previous NRC Exam ______
- Previous Quiz / Test ______

Question Cognitive Level:
- Memory or Fundamental Knowledge __X__
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 __X__
- 55.43 __X__

Comments:

1 POINT

QUESTION # 75

Which ONE of the following describes one of the reasons for tripping RCP's on a loss of subcooling margin?

A. Allows RC pumps to be “bumped” to mitigate interruption of two-phase natural circulation.

B. Allows greater RCS inventory and a higher Reactor Vessel level to be established during a SBLOCA.

C. Prevents excessive current (amperage) in the 6.9kV startup transformer due to high RCS void concentration.

D. Prevents high vibration and possible damage of the RC Pumps due to a high RCS void fraction during a SBLOCA.

Distractors analysis not lined up correctly

1 POINT

QUESTION # 75

G2.4.16

C A. Incorrect - The CT transformer is the power source for the RCPs during post trip conditions. The CT transformer is protected by over current relays that would prevent transformer damage. If the power source were CT-4 or 5 then the RCP would not be allowed to operate. CT-4 and 5 can not power RCPs because over current conditions on the transformers.

B. Correct - The RCS pumps will have an effect in the response to a SBLOCA. If RCPs are operating they will pump the RC inventory out of the break therefore reducing the total RCS inventory and RV level.

A C. Incorrect - RC pump bumps are an option in the later stages of two-phase NC when gas binding interrupts the boiler-condenser mode (BCM) of NC but this is not the reason we secure RCPs on loss of SCM.

D. Incorrect - The RCPs must be tripped within 2 minutes of the loss of subcooling margin. If RCPs are not tripped within the two minutes then the RCP will remain in operation and will pump a two phase mixture caused by a high void fraction. The higher the void fraction the lower the RCP amperage as pumping steam requires much less work than pumping water.

Terminal Objectives:

1. At the completion of this training, the non-licensed operator should have a better understanding of the consequences of a turbine building flood on plant integrity and what actions have been taken to mitigate the severity of those consequences. In doing so, the non-licensed operator will better understand his role in ensuring those actions are capable of being carried out should a turbine building flood occur. (T1)

2. At the completion of this training, the licensed operator will be better able to assess and mitigate the consequences of several significant abnormal transients, including SBLOCAs, OTSG Overfills and Turbine Building Flooding. (T2)

Enabling Objectives:

1. Be able to correctly predict the plant response to SBLOCAs and indicate how that response will be affected by the following conditions: (R1)

   1.1 RCP operation following a LOCA

   1.2 Small breaks with feedwater:

   A. LOCAs large enough to depressurize the Reactor Coolant System.

   B. LOCAs which stabilize at approximately secondary side pressure.

   C. LOCAs which may repressurize the RCS in a saturated condition.

   D. LOCAs which stabilize at a primary system pressure greater than secondary system pressure.

   E. Breaks in the Pressurizer.

   1.3 Small breaks without feedwater:

   A. LOCAs capable of relieving all decay heat via the break.

   B. LOCAs that relieve decay heat with both HPI and via the break.

   C. LOCAs which do not automatically actuate the HPI and result in system repressurization.

2. Be able to correctly predict the plant response (primary as well as secondary) to a OTSG overfill casualty by considering the following effects to assess plant status: (R2)

   2.1 Hydraulic forces

   2.2 Excessive dead weight forces

   2.3 Failure of relief valves to reseat

   2.4 Loss of EFDWP turbine

   2.5 OTSG tube rupture

C. Large LOCAs (> .5 ft^2 to 14 ft^2)

1. Rapid, violent events that go through clearly defined phases of blowdown, refilling and reflooding.
2. Operators can do little to control a large LOCA until its later stages.

D. Small LOCAs (> .25" to < .5 ft^2)

1. More probable than large LOCAs.
2. Progression can be significantly influenced by the operator.
3. Improper or incorrect operator action can increase the severity of the consequence.
4. The response of the primary system to a small break will greatly depend on break size, its location in the system, operation of the reactor coolant pumps, the number of ECCS trains functioning, and the availability of secondary side cooling.

   a) Impact of RCP operation on a small LOCA:

      1) With RCPs on, the steam and water remain mixed.
      2) Liquid is therefore discharge out the break continuously.
      3) Liquid in the RCS can evolve to a high void fraction. (OC-TA-AT-01)
      4) Maximum void fraction depends upon break size and location.

   b) Continued RCP operation will provide sufficient core flow to keep cladding temperature within a few degrees of the saturated fluid temperature.

      1) Cools even with high void fraction
      2) Loss of the RCP, after evolving to a high void fraction, can lead to inadequate core cooling.
      3) If the pumps are lost after the RCS has evolved to a void fraction of > 70%, there will not be enough water left to cover the core after phase separation.
      4) The pumps must be tripped immediately upon loss of any one of the three subcooling margins.

         (a) The RCPs <u>will</u> be tripped within 2 minutes of the loss of any subcooling margin.

         (b) The two minutes ensures that the 70% void fraction won't be reached.

Exam Question Report

27-Jan-99

Question ID:	TA026	Revision No:	0	Revision Date	10/29/1999
Question Description:	TA026				

Exam Question Status		Exam Question Criteria	
Reference Flag:		Topic Area:	TA-AT - Abnormal Transients
Last Used Date:		Question Type:	Multiple Choice
Inactive:	N	Response Time:	
Inactive Comment:	LRO = 1; SRO = 1	Max. Point Value:	0.25
		Passing Point Value:	0.25

Exam Question Report

27-Jan-99

Question

KA: 2.4.16

Which ONE of the following is the primary reason that the EOP instructs the operator to trip RCP's on a loss of subcooled margin? (.25)

A) prevent pump damage due to cavitation which occurs when pumping a two phase mixture.

B) minimize inventory loss out the break, in the event of a LBLOCA.

C) prevent excessive current flows through the "CT" transformer.

D) prevent core uncovery that would occur if RCP's were subsequently lost and the RCS had a high void fraction.

Answer

D

Lessons

ID	Description
TA-AT	Abnormal Transients TA-AT

Enabling Objectives

ID	Description

Referenced Documents

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