FINAL SUBMITTAL

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

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

FINAL RO/SRO WRITTEN EXAM

ANSWER KEY

OCONEE 2000
ANSWER KEY

COMMON						RO ONLY		SRO ONLY	
1.	B	26.	D	51.	A	76.	D	76.	B
2.	A	27.	C	52.	A	77.	D	77.	C
3.	D	28.	A	53.	D	78.	A	78.	D
4.	A	29.	C	54.	D	79.	C	79.	D
5.	C	30.	D	55.	A	80.	D	80.	A
6.	A	31.	A	56.	A	81.	B	81.	A
7.	A	32.	A	57.	D	82.	C	82.	C
8.	D	33.	B	58.	A	83.	B	83.	A
9.	A	34.	A	59.	D	84.	B	84.	C
10.	B	35.	B	60.	B	85.	B	85.	B
11.	C	36.	C	61.	B	86.	C	86.	B
12.	B	37.	A	62.	A	87.	A	87.	D
13.	B	38.	D	63.	A	88.	A	88.	D
14.	D	39.	B	64.	D	89.	C	89.	D
15.	B	40.	A	65.	A	90.	B	90.	A
16.	D	41.	B	66.	B	91.	C	91.	B
17.	B	42.	D	67.	C	92.	A	92.	D
18.	B	43.	A	68.	D	93.	C	93.	B
19.	B	44.	B	69.	B	94.	A	94.	A
20.	A	45.	A	70.	A	95.	C	95.	C
21.	B	46.	A	71.	C	96.	A	96.	D
22.	A	47.	C	72.	A	97.	C	97.	B
23.	A	48.	A	73.	A	98.	C	98.	B
24.	D	49.	D	74.	B	99.	B	99.	D
25.	C	50.	D	75.	B	100.	C	100.	B

FINAL SUBMITTAL

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FINAL AS ADMINISTERED

RO WRITTEN EXAMINATION

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I / II / III / IV
Date:	Facility/Unit: OCONEE 1, 2, 3
License Level: RO / SRO	Reactor Type: W / CE / BW / GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	________ Points
Applicant's Score	________ Points
Applicant's Grade	________ Percent

1 POINT

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  - Group 1 Rods 2 through 9 withdrawn to 10%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 # 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 # 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.

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.

chg #9

chg #9

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.

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 # 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 positive 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

Unit 1 core reactivity will **initially** ____ and the reactor will immediately ____.

A. increase / trip

B. decrease / trip

C. increase / NOT trip

D. decrease / NOT trip

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 # 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 **5910 gpm**
- 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.

D. was positioned to the "fail-open" position and LPSW-4 (LPI Cooler Outlet) will be manually throttled to achieve < 5200 gpm.

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 # 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 ≥ 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 # 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 # 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 # 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 slowly 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 # 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 switch / "2"

B. RESET button / "2"

C. Test toggle switch / "3"

D. RESET button / "3"

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?

1HP-5 (Letdown Isolation) will automatically close as letdown temperature increases to ______ °F which was **designed** to prevent a release of ________ from the purification demineralizer into the RCS.

A. 130 / Boron

B. 135 / Boron

C. 130 / Sulfur

D. 135 / Sulfur

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 # 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 # 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

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 ______ be throttled / _______.

A. can / when PZR level is ≥ 100 inches

B. can / when core SCM increases to ≥ 5°F

C. cannot / until the core and both Loops SCM are ≥ 5°F

D. cannot / until RCS pressure is above the ES 1 and 2 actuation setpoint

1 POINT

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?

A. Shifting RBCUs to high speed.

B. Penetration Room ventilation filters.

C. Reactor Building Purge inlet pre-filter.

D. Triple isolation valves in both intake/exhaust ductwork.

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 shutdown 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. 6.9

B. 8.7

C. 9.9

D. 12.3

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

<u>Curves</u>

Figure 1: Time To Core Boiling Prior To Refueling

Figure 2: Time To Core Uncovery Prior To Refueling

Figure 3: Time To Core Damage Prior To Refueling

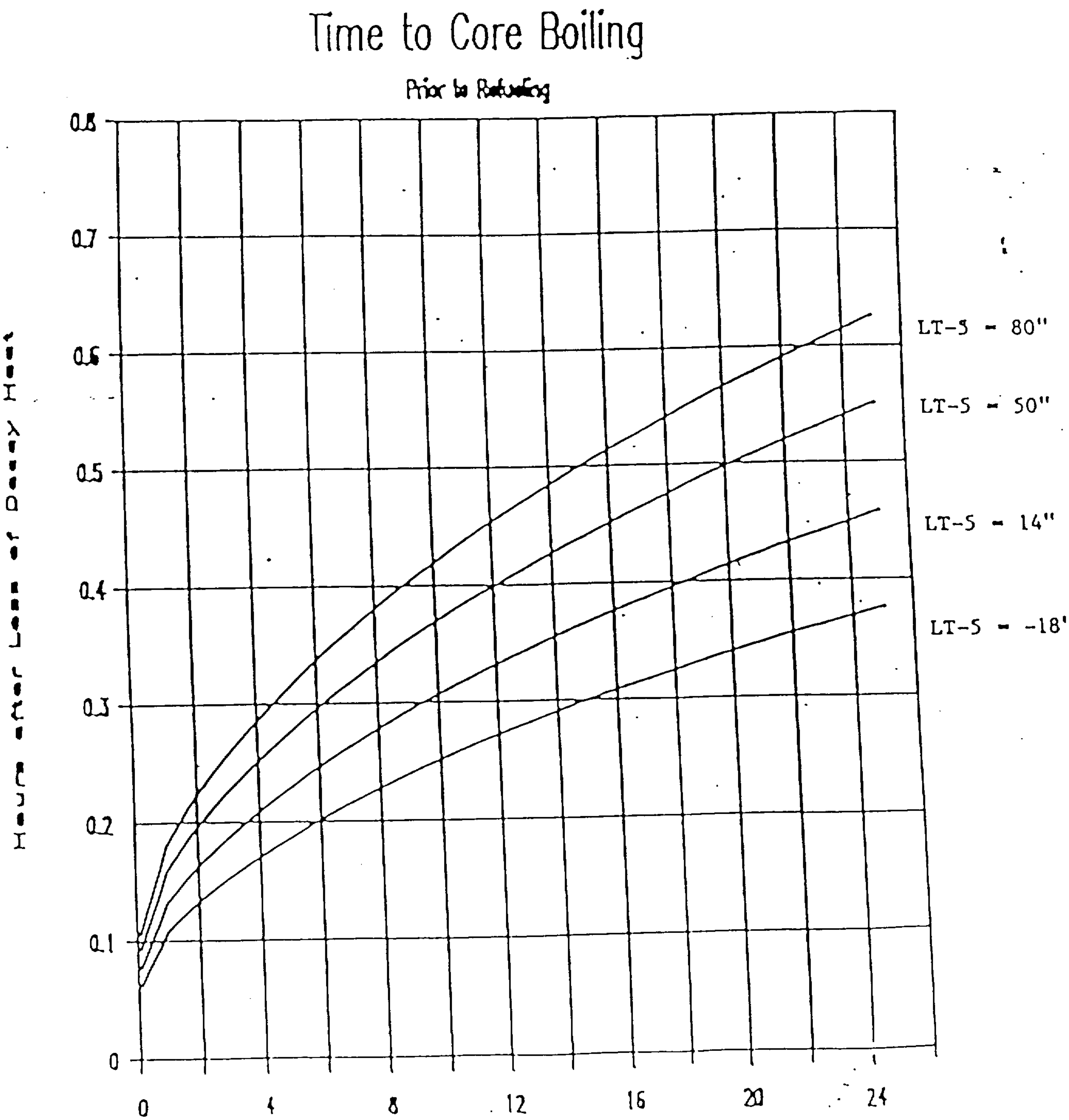
Figure 4: Time To Core Boiling After Refueling

Figure 5: Time To Core Uncovery After Refueling

Figure 6: Time To Core Damage After Refueling

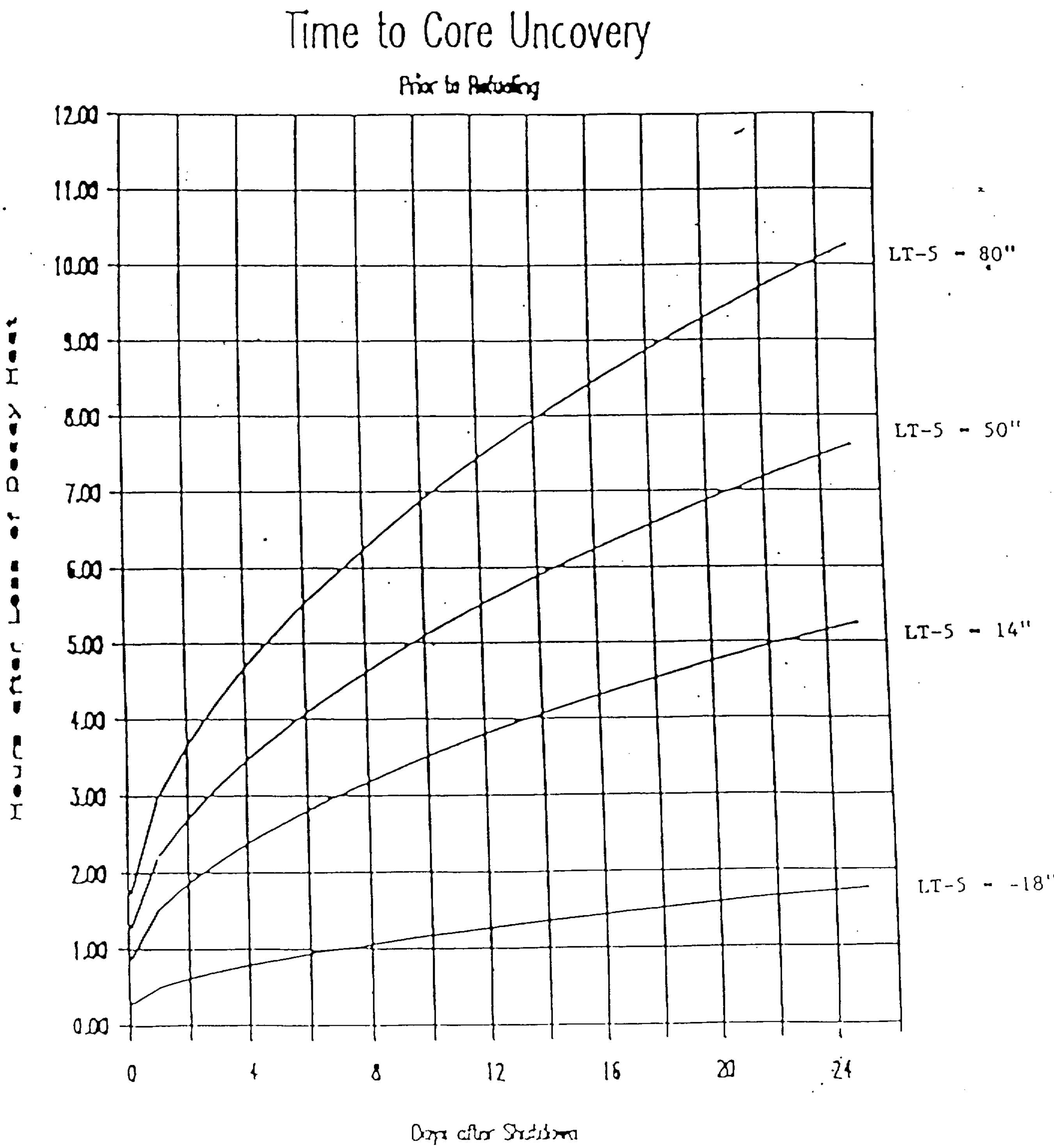
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 1



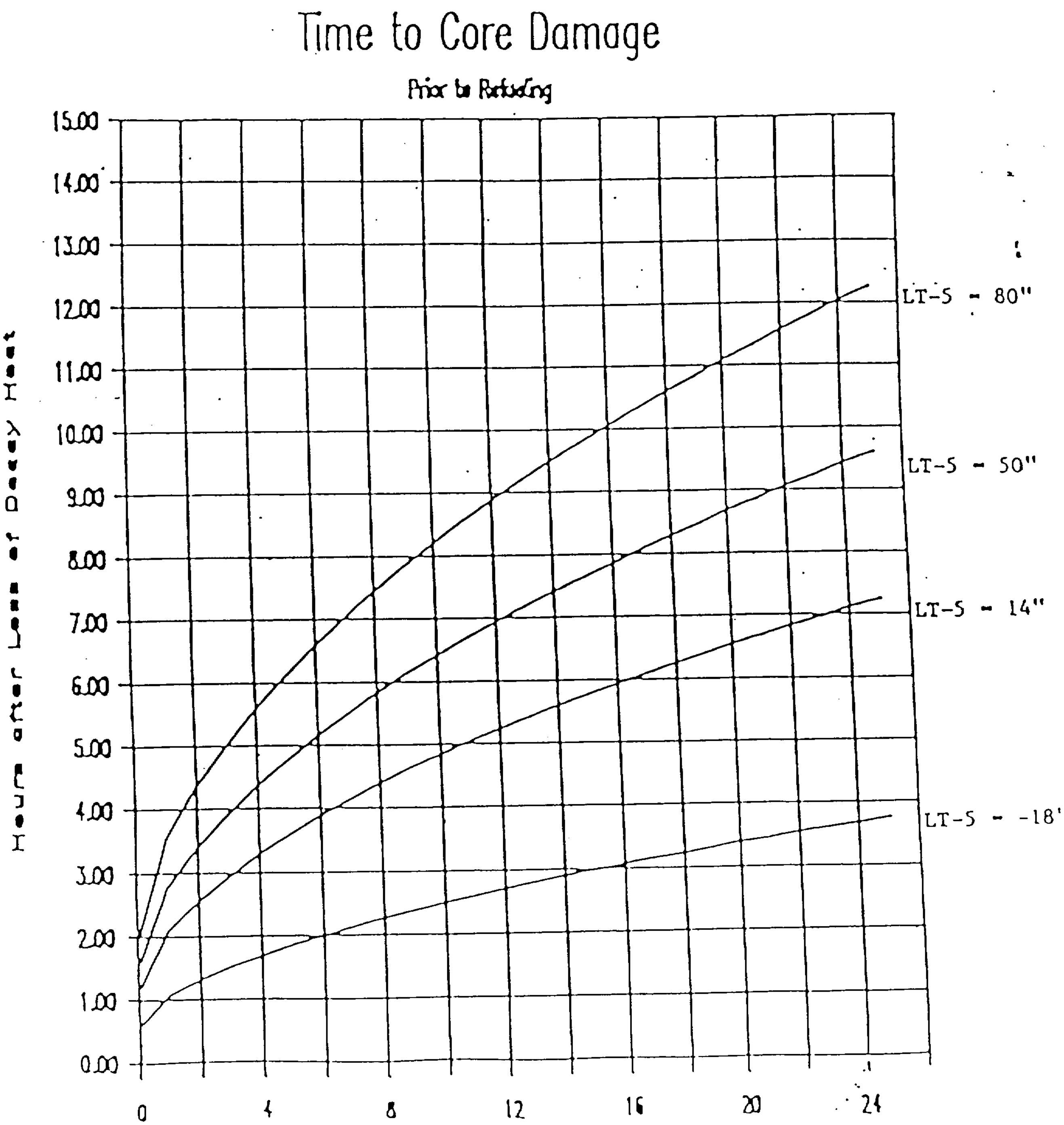
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 2



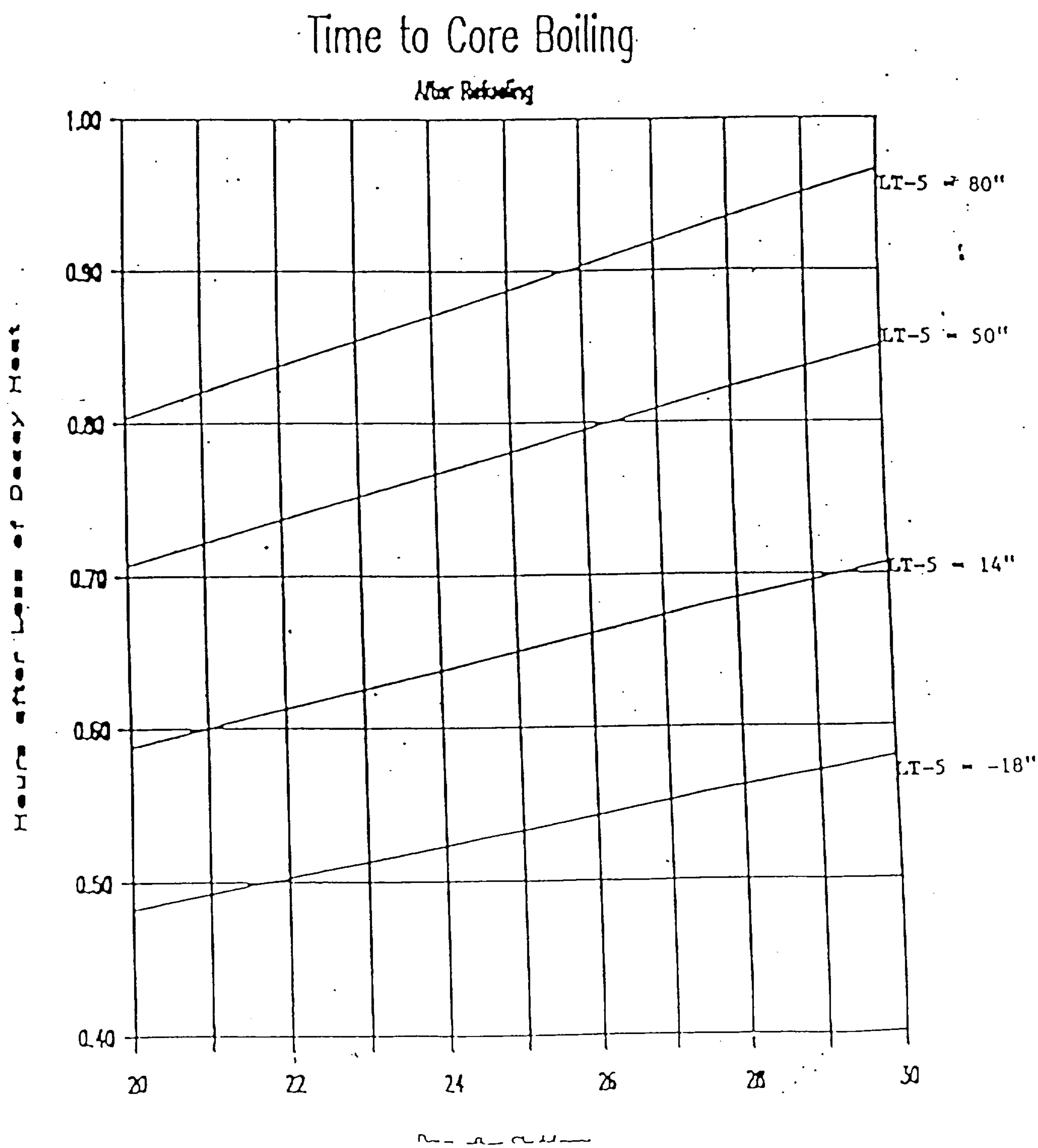
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 3



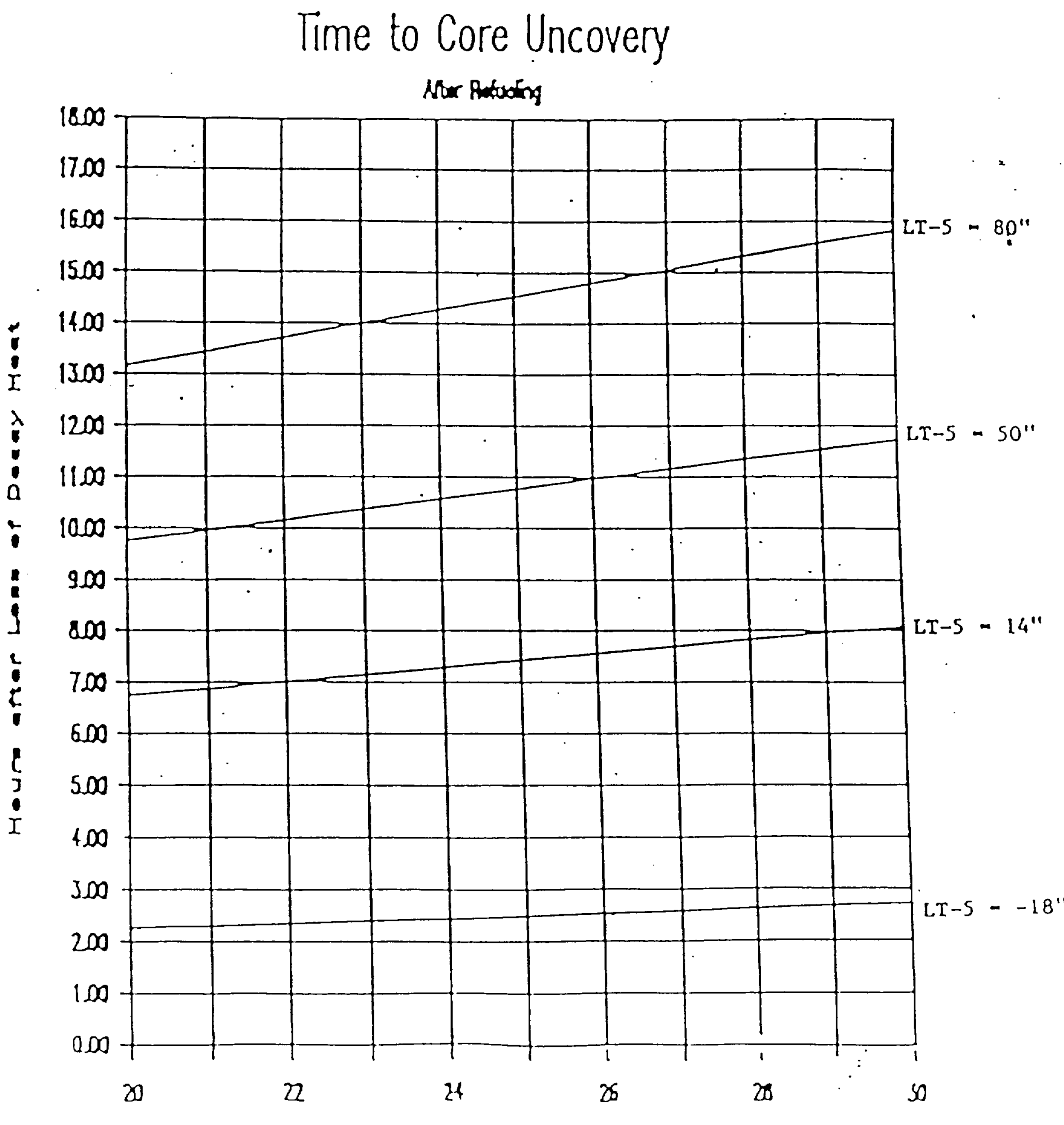
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 4



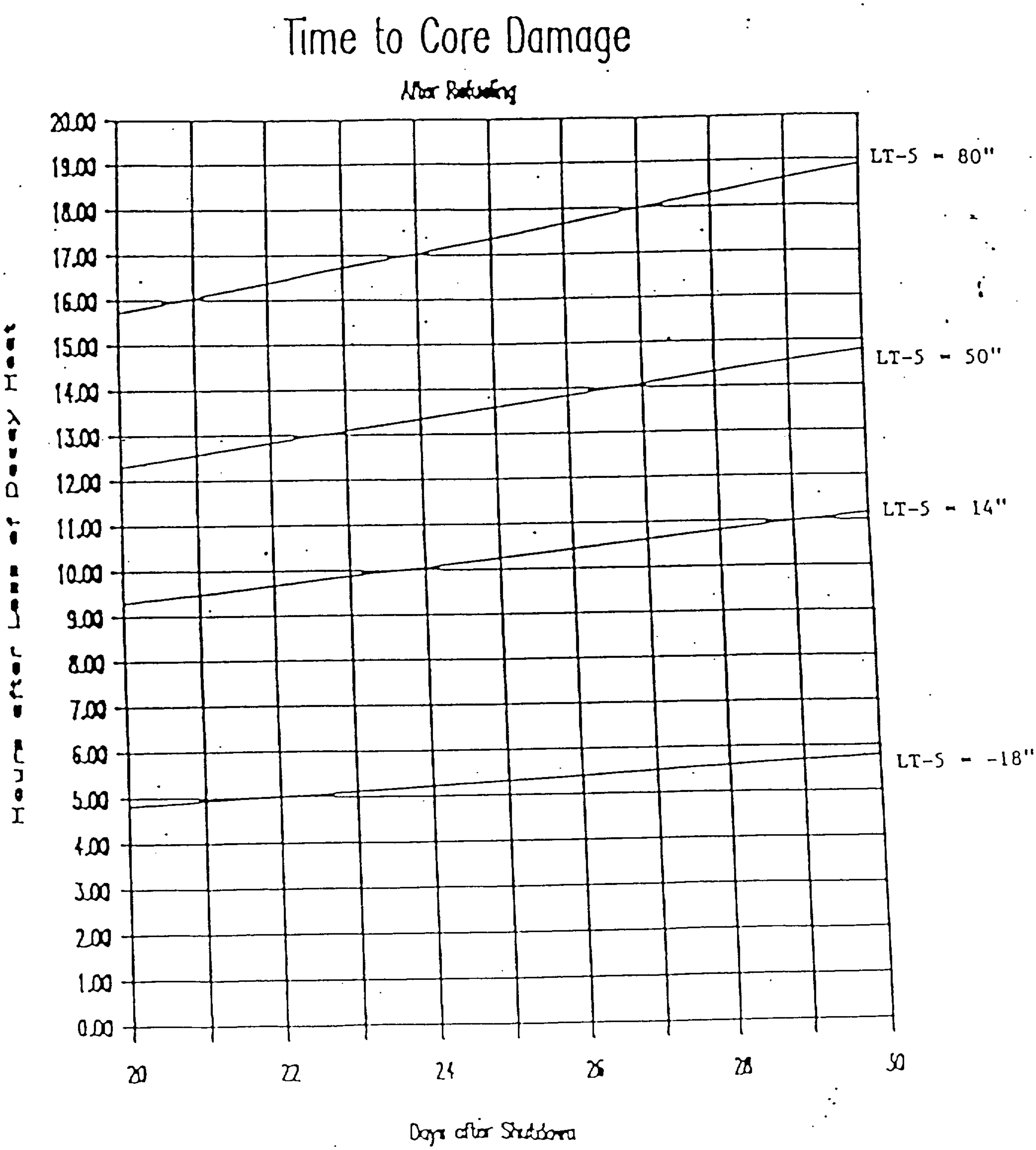
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 5



ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 6



1 POINT

QUESTION # 21

Which ONE of the following correctly completes the below statement?

The LPI system design pressure interlock 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 # 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. automatically stop.

B. not stop because the Diamond is in MANUAL.

C. not stop as NI-1 has ONLY failed to mid-scale.

D. stop ONLY if the operator positions the Diamond Control ("joy stick") to "neutral".

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 # 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 # 25

Unit 1 plant conditions:

- Recovery from HPI Cooling is in progress
- EFDW has been restored
- 1TA and 1TB 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 # 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 # 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

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

The RCS will...

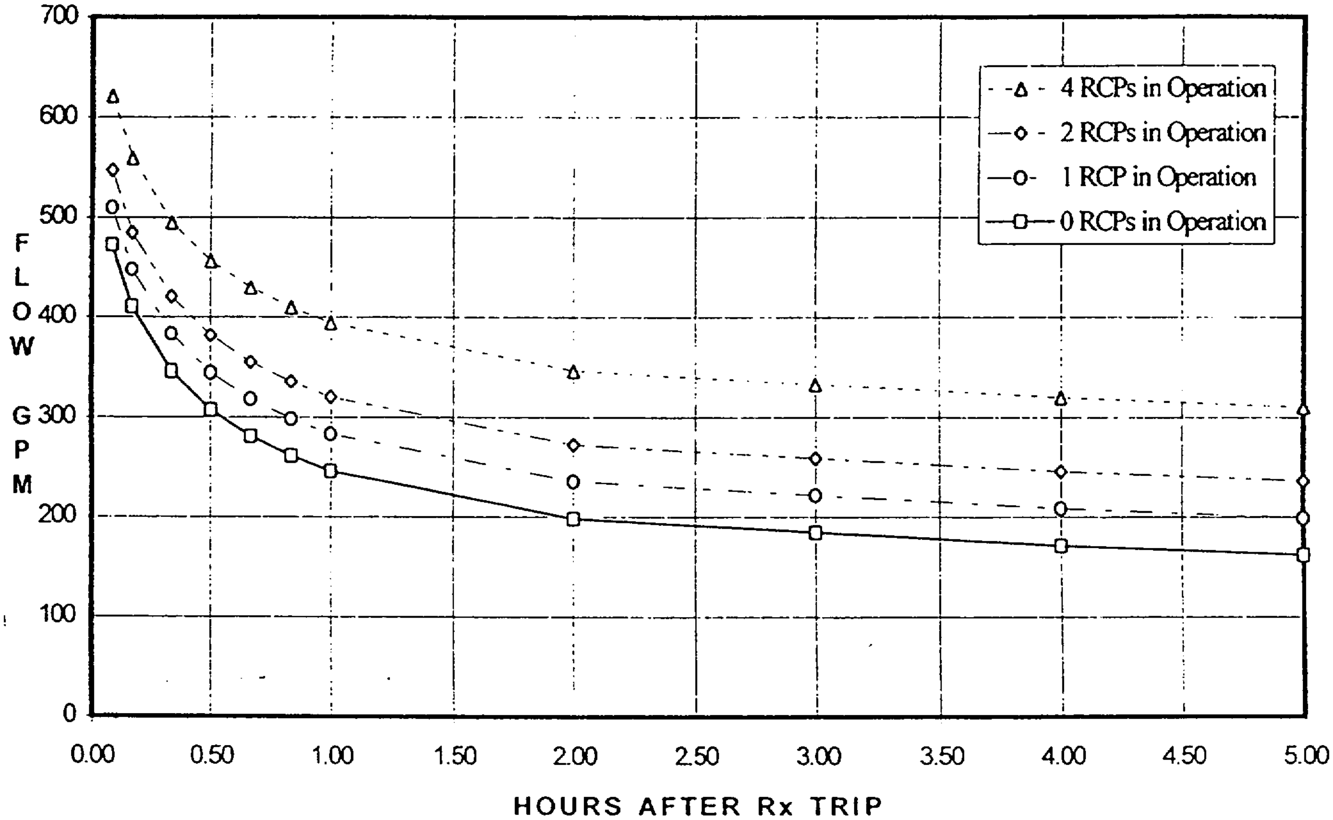
A. cooldown resulting in a decrease in SCM.

B. gradually heat up with a reduction in SCM.

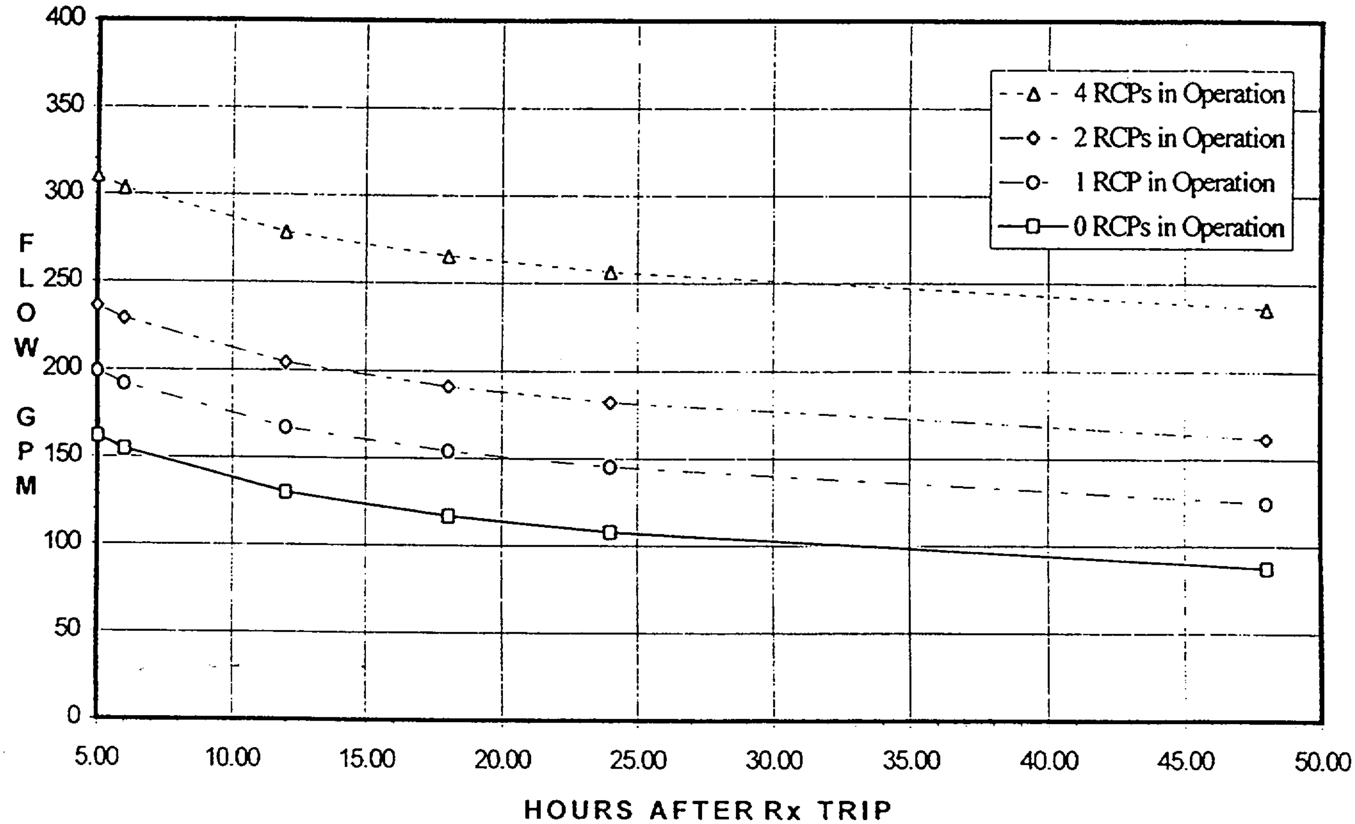
C. continue to gradually cool with an increase in SCM.

D. remain at the same temperature, pressure, and SCM.

Total Feedwater Flow Required To Match NSSS Heat



Total Feedwater Flow Required To Match NSSS Heat



○○○END○○○

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

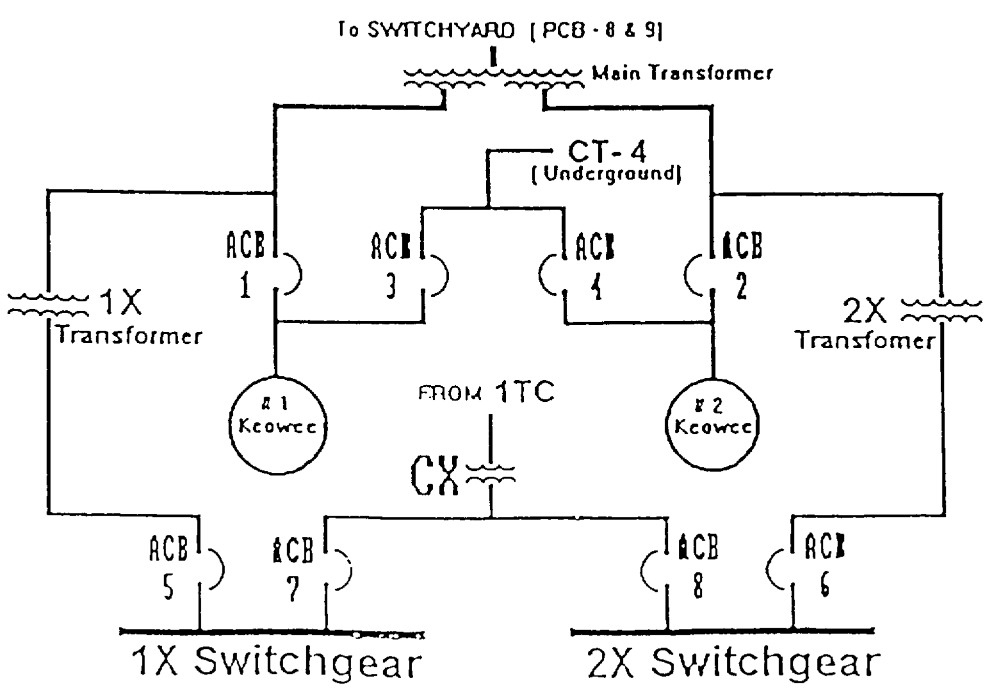
ACB- _____ will be or remain closed even if a Keowee _________________.

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 # 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...

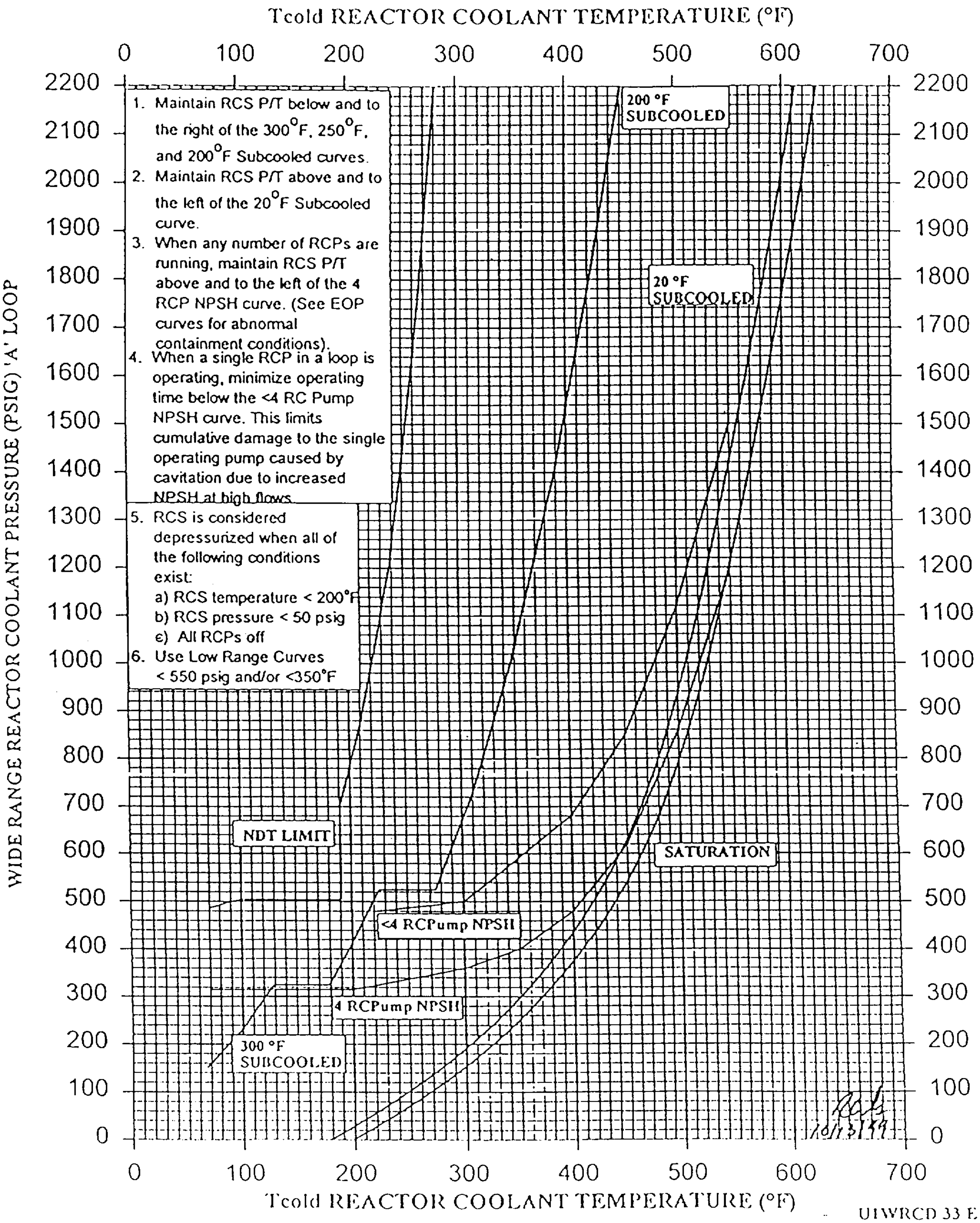
A. high and RCS pressure needs to be increased.

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.

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

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?

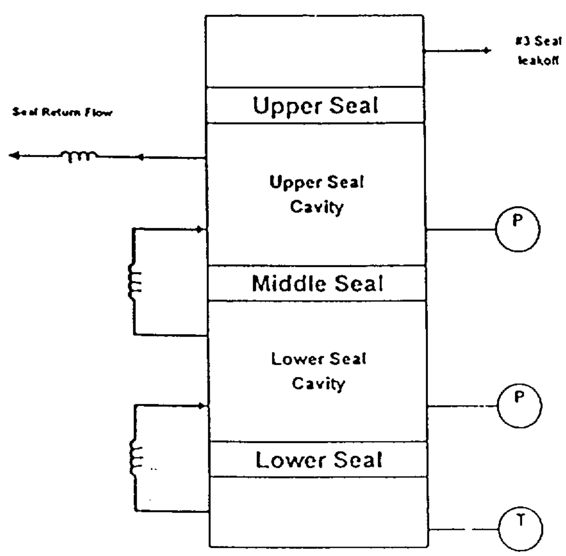
RCP 3A1 _________, 3A2_________, 3B1 _________, 3B2 _________.

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 # 32

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?

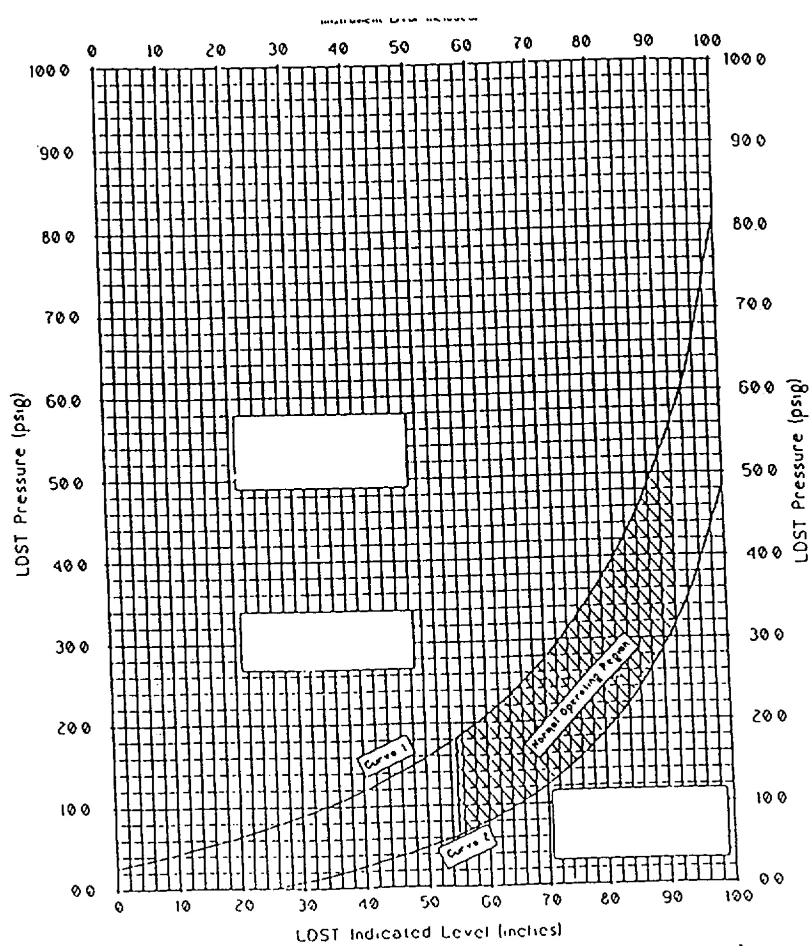
SEE ATTACHMENT

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.

D. Provide HPI suction from the BWST for transients requiring additional HPI flow.



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

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 # 35

Unit 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 # 36

Unit 1 plant conditions:

- MODE 3
- 1A and 1C RBCU's in HIGH
- "A" RB Spray Train is inoperable
- Statalarm 1SA-9/~~C~~ B-9/A 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 _______.

A. has / could result in dilution of the RBES following a LOCA.

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.

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 # 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 Control valve.

C. trips the TD EFDWP, closes "B" FDW Startup Control valve.

D. closes "A" and "B" FDW Startup Control valves.

1 POINT

QUESTION # 39

Unit 1 plant conditions:

- A MS line rupture has occurred
- MSLB circuitry has actuated properly

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.

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.

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 # 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 SFP and RB.

B. Continue fuel movement and no other actions are required.

C. Suspend fuel movement in the 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 # 43

Which ONE of the following is the reason for limiting the pressurization rate of the Core Flood Tanks (CFTs)?

The pressurization rate is limited to ≤ 100 psig/15 min 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 # 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 could cause the increasing RCS 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.

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 # 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 # 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 # 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 remains 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 # 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

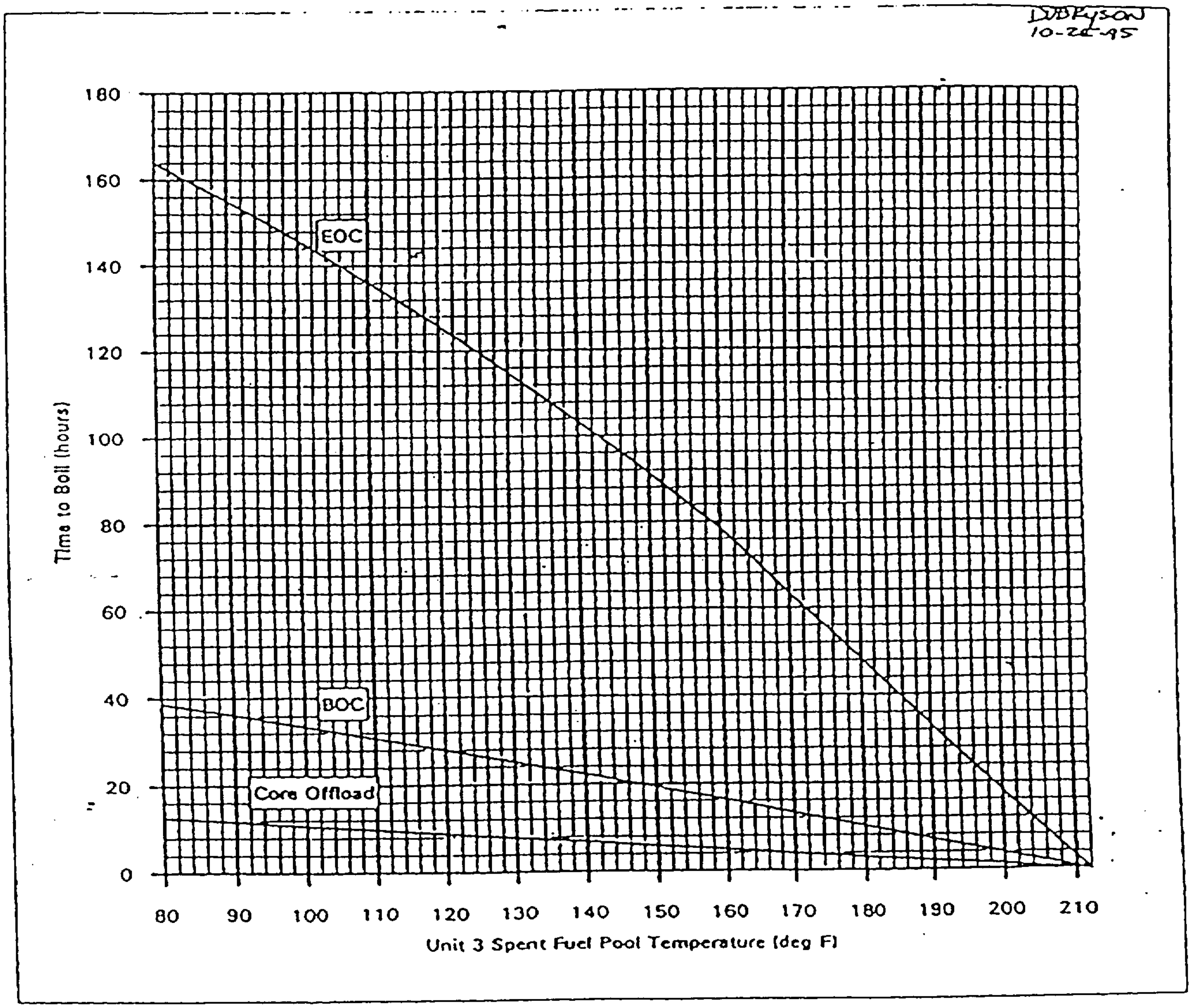
B. 35 hours

C. 30 hours

D. 28 hours

U3 SFP Time To Boil
After Loss Of SF Cooling

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.

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

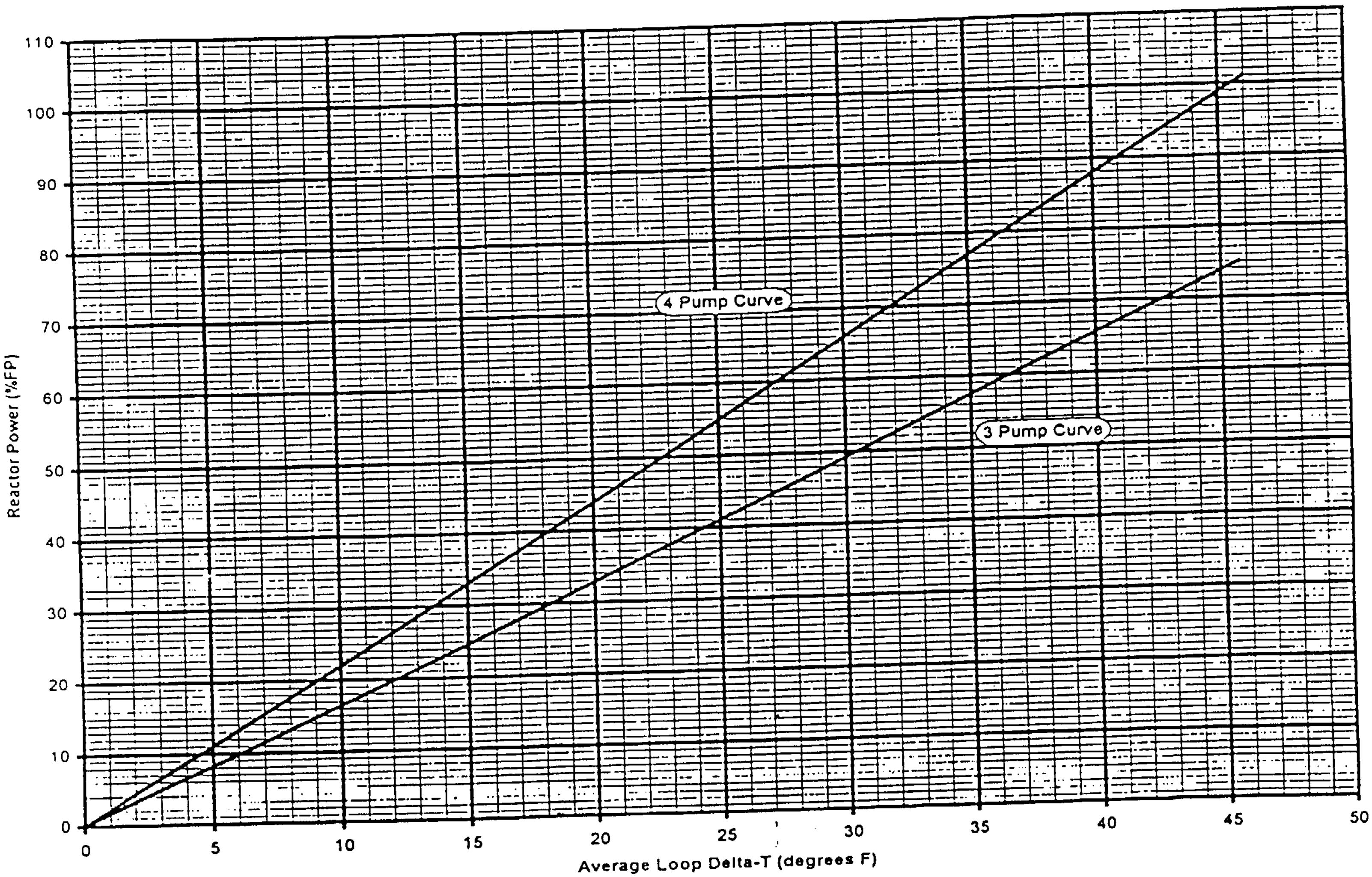
B. 582 / 575

C. 592 / 566

D. 585 / 573

Enclosure 13.12

Unit 2 Cycle 18
Loop ΔT Vs. Reactor Power



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 **19 seconds**

Which ONE of the following is correct?

The MSSVs are ______ because...

A. operable / all four MSSVs are in the closed position.

B. operable / both of the two TSV's Closure Channels have actuated within TS limits.

C. inoperable / one of the two TSV's Closure Channels has closed all the MSSVs within TS limits.

D. inoperable / neither of the two TSV's Closure Channels have closed all the MSSVs within TS limits.

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 <u>first</u> means of detecting the SGTL?

A. CSAE off gas

B. FDW mismatch

C. MS Line monitors

D. Chemistry sampling

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 **<u>first</u>** 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. must be manually transferred to 1XP then re-transferred to 1XO by the NLO as directed by the control room operator.

B. must be manually transferred to 1XP when power is restored then automatically re-transfers to 1XO when power is restored to 1XO.

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 1XP when power is restored to 1XP and remains positioned to 1XP when power is restored to 1XO.

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 Generator Differential 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

1 POINT

QUESTION # 55

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Power = 100%
- ACB-4 closed

CURRENT CONDITIONS:

- 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 # 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.

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.3 Radioactive Effluent Monitoring Instrumentation

COMMITMENT Radioactive Effluent Monitoring Instrumentation shall be OPERABLE as follows:

a. Liquid Effluents

The radioactive liquid effluent monitoring instrumentation channels shown in Table 16.11.3-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of SLC 16.11.1.a are not exceeded.

b. Gaseous Process and Effluents

The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 16.11.3-2 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of SLC 16.11.2.a are not exceeded.

c. The setpoints shall be determined in accordance with the methodology described in the ODCM and shall be recorded.

-----NOTE-----
Correction to setpoints determined in accordance with Commitment c may be permitted without declaring the channel inoperable.

APPLICABILITY: According to Table 16.11.3-1 and Table 16.11.3-2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A Alarm/trip setpoint less conservative than required for one or more effluent monitoring instrument channels.	A.1 Declare channel inoperable.	Immediately
	<u>OR</u>	
	A.2 Suspend release of effluent monitored by the channel.	Immediately

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
	<u>OR</u>	
	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 <u>AND</u> 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
	OR	
	K.2.1 Collect grab sample.	Immediately AND Once per 8 hours
	AND	
	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 # 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 # 58

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

C. Audible alarm at the detector

D. Fire alarm response guide

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 # 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 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 # 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 # 62

Which ONE of the following describes 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 IA is lost.

B. prevents excessive PRV flow during ES actuation.

C. maintains proper PRV flow to prevent PRV system PAC filter channeling.

D. maintains adequate PRV flow to eliminate charcoal filter loading and ignition.

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 # 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 # 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

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

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 due to controller failure

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 on a loss of instrument air.

D. open, close, or throttle the valve upon a loss of the control room auto positioning signal.

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 # 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 <u>initial</u> testing to check motor rotation (pump is uncoupled)
- Following rotation checks, pump coupling will be performed

Which ONE of the following is correct?

Prior to pump coupling, OPEN the breaker and the tag…

A. shall be cleared and reissued.

B. and stub should be marked "VOID", 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.

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 **ONE** 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 ALL 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.

1 POINT

QUESTION # 70

Which ONE of the following conditions **REQUIRES** prior SRO approval/oversight?

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.

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.

1 POINT

QUESTION # 72

Which ONE of the following plant areas is posted **<u>INCORRECTLY</u>** 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 # 73

Which ONE of the following is correct concerning the notification to offsite agencies during E-Plan implementation?

________ notifications shall be made within ________ minutes of declaration of the EAL.

A. Initial / 15

B. Initial / 30

C. Upgrade / 30

D. Upgrade / 60

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

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.

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 # 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 # 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 # 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 # 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 # 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
- All EOP actions are being performed in a timely manner

CURRENT CONDITIONS:

- Time = 0150
- 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 # 82

RO ONLY

Unit 1 plant conditions:

- 350 gpm Steam Generator Tube Rupture is in progress
- 1A1 and 1B2 RCP operating
- 1RC-4, PORV Block 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

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

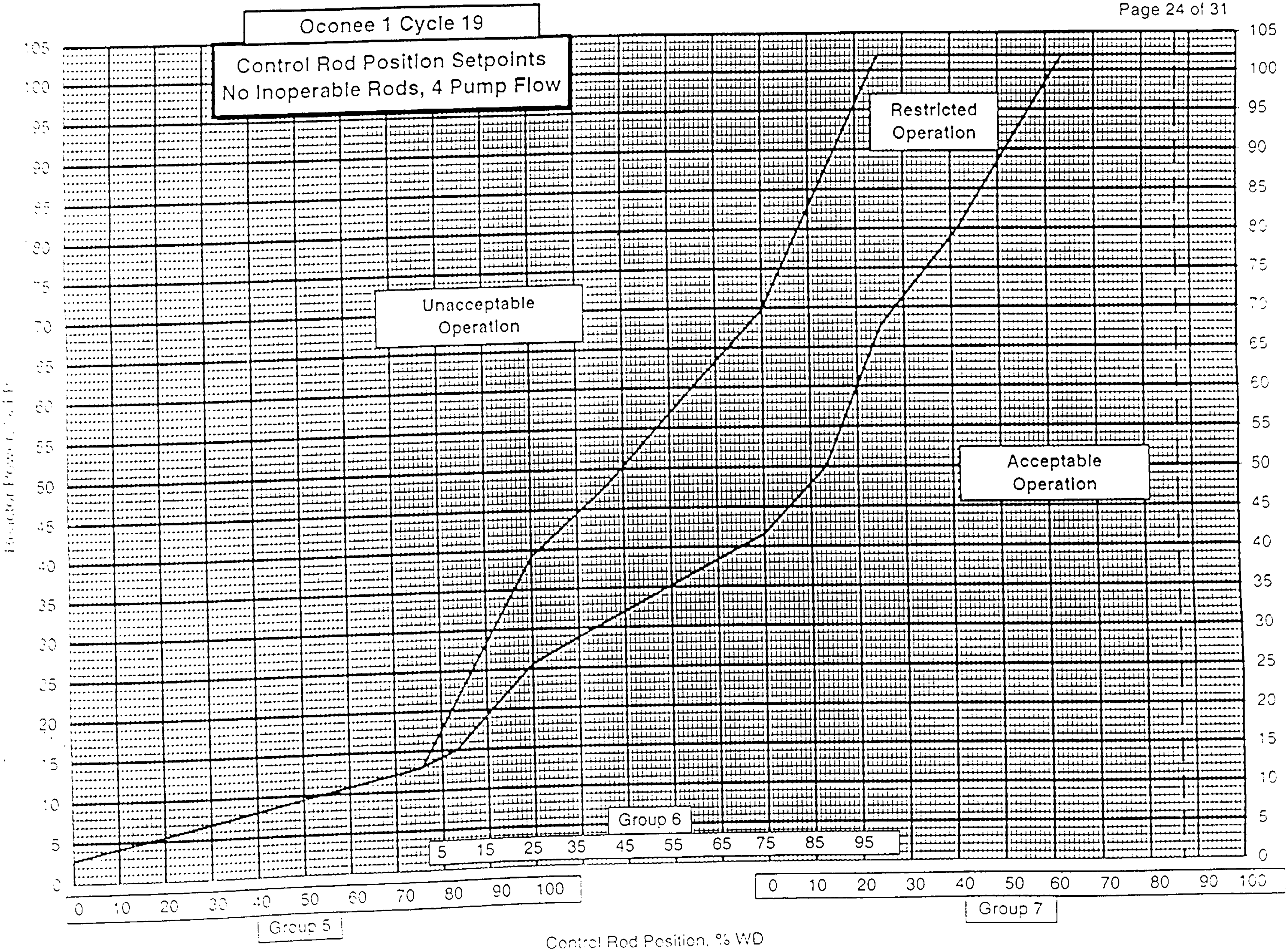
Group _____ / _______ withdrawn.

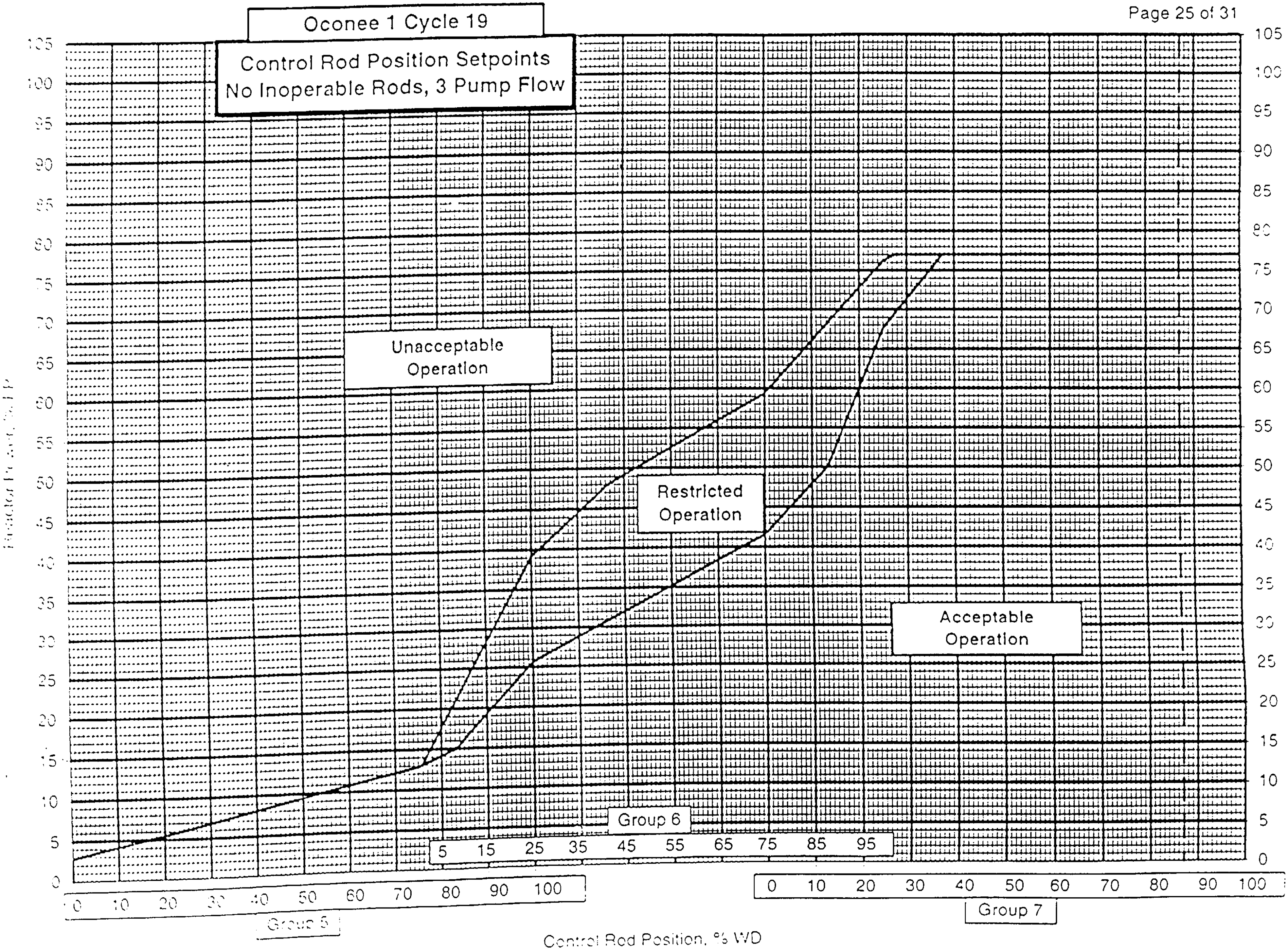
A. 6 / 63%

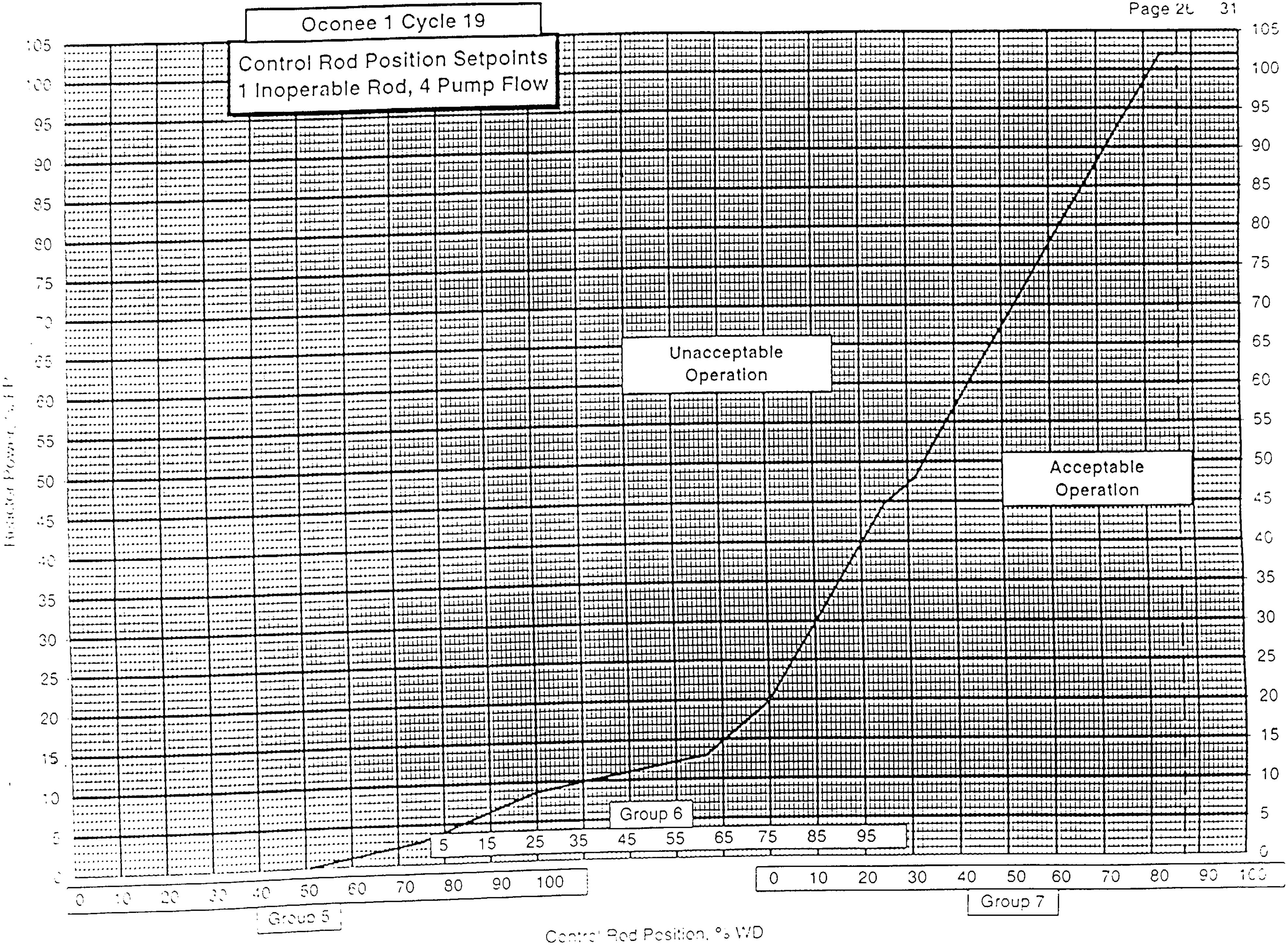
B. 7 / 3%

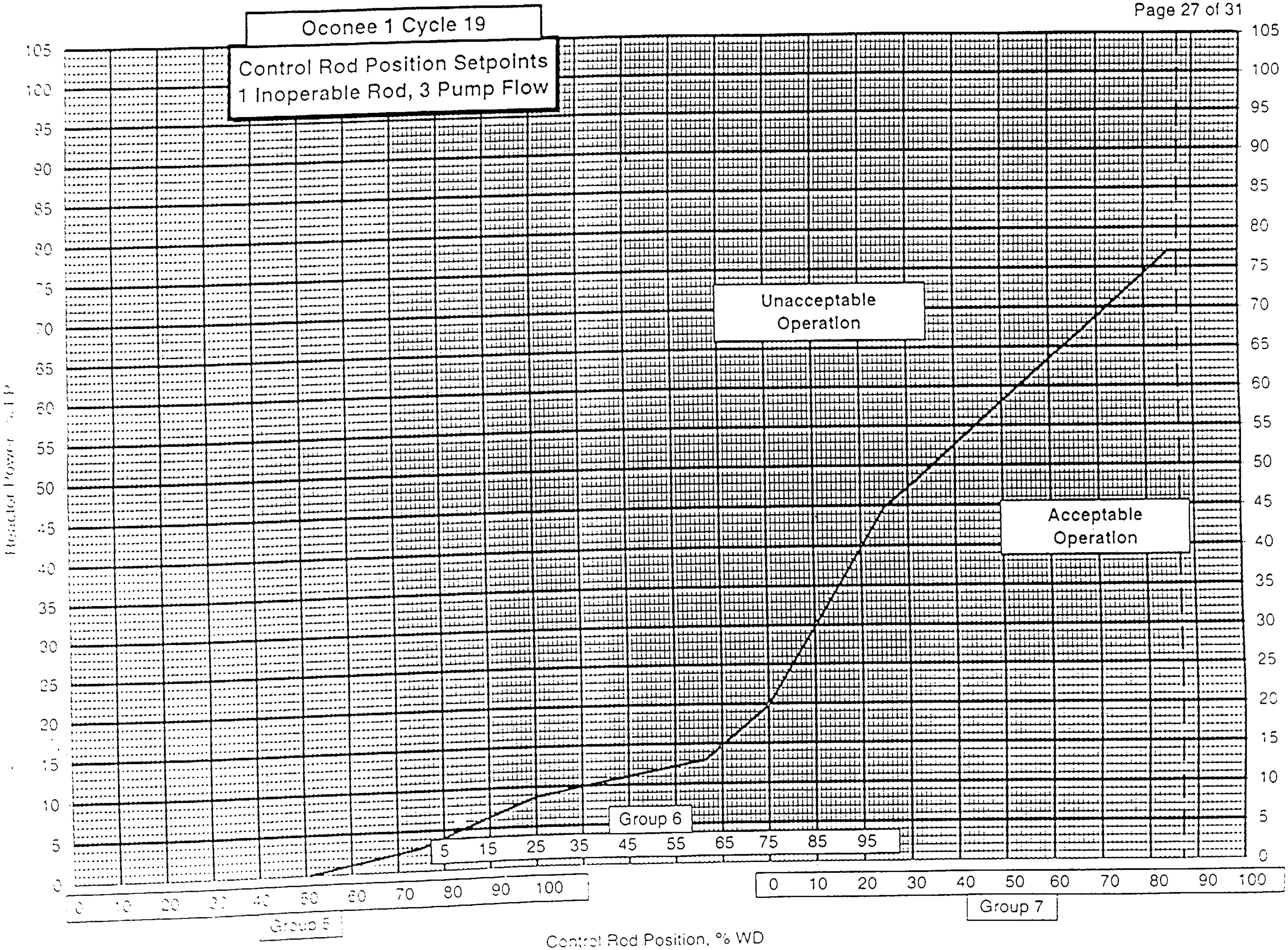
C. 7 / 22%

D. 7 / 55%


Oconee 1 Cycle 19
Control Rod Position Setpoints
No Inoperable Rods, 4 Pump Flow
Restricted Operation
Unacceptable Operation
Acceptable Operation
Group 5
Group 6
Group 7
Control Rod Position, % WD


Oconee 1 Cycle 19
Control Rod Position Setpoints
No Inoperable Rods, 3 Pump Flow
Unacceptable Operation
Restricted Operation
Acceptable Operation
105
100
95
90
85
80
75
70
65
60
55
50
45
40
35
30
25
20
15
10
5
0
0 10 20 30 40 50 60 70 80 90 100
Group 5
5 15 25 35 45 55 65 75 85 95
Group 6
0 10 20 30 40 50 60 70 80 90 100
Group 7
Control Rod Position, % WD


Oconee 1 Cycle 19
Control Rod Position Setpoints
1 Inoperable Rod, 4 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 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
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
Group 7
Control Rod Position, % WD

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 65% 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

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 # 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 <u>not</u> reseated.

B. The PORV is at the lift setpoint and is "chattering".

C. One of the PZR code safety valves is <u>not</u> properly seated.

D. Both of the PZR code safety valves are "chattering".

1 POINT

QUESTION # 87

RO ONLY

Unit 2 plant conditions:

- LOCA in progress
- RB Pressure peaked at 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 digital channels?

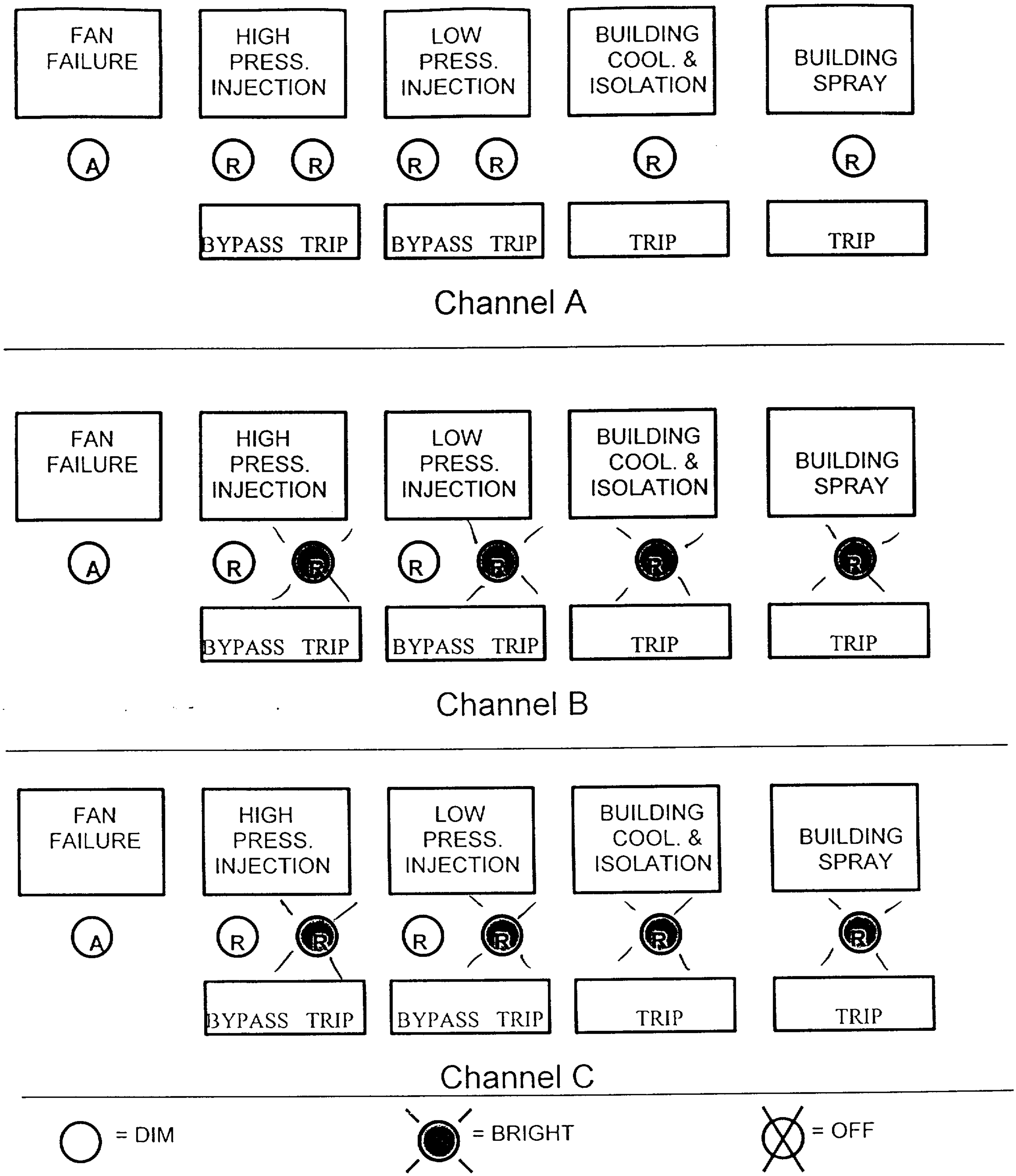
SEE ATTACHMENT

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 ENGINEERED SAFEGUARDS	NOTES: ANALOG CHANNELS A, B, & C	ID#: OC-IC-ES-3A	DATE: 11/10/97
		REF: Design Basis Document	
		DRN by WAH:	APR BY:
		TRAINING USE ONLY	

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 # 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, Reactor Building Cooling System?

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 # 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

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?

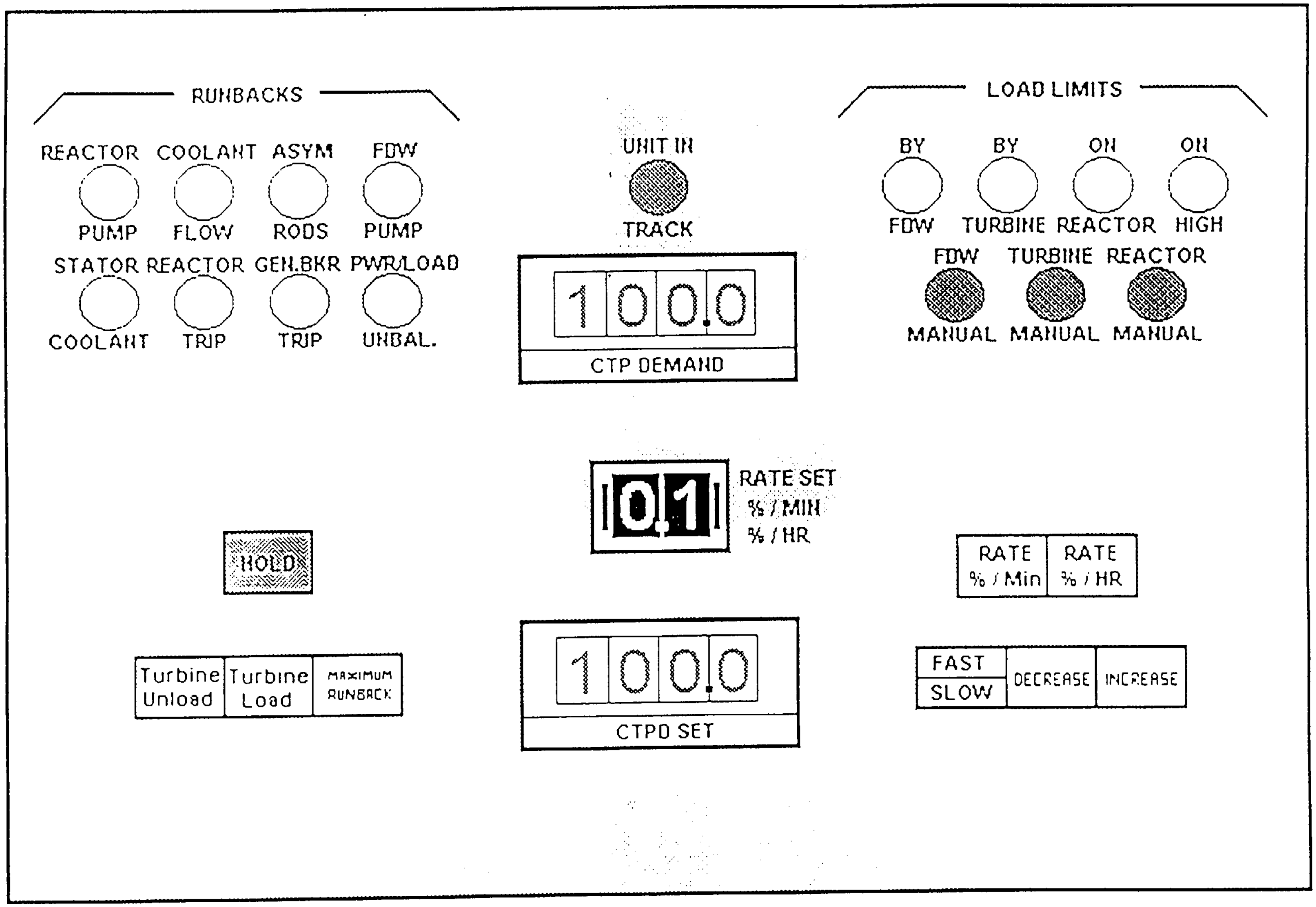
SEE ATTACHMENT

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
		TRAINING USE ONLY	

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

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.

B. OTSG's levels will begin to fill to the Main Steam lines.

C. TD EFDWP Main Steam supply will automatically isolate.

D. TD EFDWP speed control will fail to the low speed stop.

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 # 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 # 95

RO ONLY

Unit 1 Plant conditions:

- LBLOCA has just occurred
- BWST = 38 feet and decreasing
- RB pressure is 12 psig
- "1A" RBS pump is out of service
- "1B" 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 # 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 # 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 # 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...

A. will be manually tripped immediately due to loss of heat tracing.

B. will automatically trip 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 # 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.

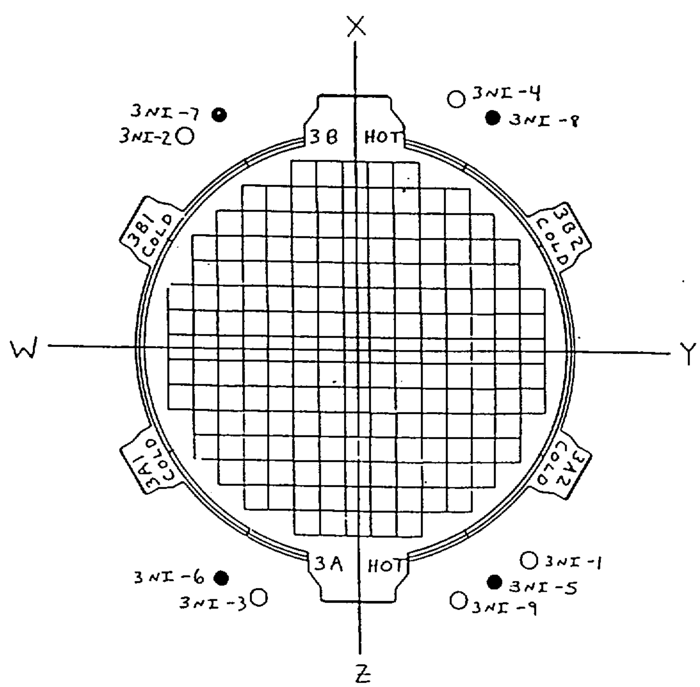
SEE ATTACHMENT

A. outer edge / control

B. outer edge / axial power shaping

C. center / control

D. center / axial power shaping



1 POINT

QUESTION # 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 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.

FINAL SUBMITTAL

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

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

FINAL AS ADMINISTERED

SRO WRITTEN EXAMINATION

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I / II / III / IV
Date:	Facility/Unit: OCONEE 1, 2, 3
License Level: RO / SRO	Reactor Type: W / CE / BW / GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	________ Points
Applicant's Score	________ Points
Applicant's Grade	________ Percent

1 POINT

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 # 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 # 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.

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.

chg #9R

chg #9R

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.

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 # 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

Unit 1 core reactivity will **initially** ____ and the reactor will immediately ____.

A. increase / trip

B. decrease / trip

C. increase / NOT trip

D. decrease / NOT trip

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 # 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 **5910 gpm**
- 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.

D. was positioned to the "fail-open" position and LPSW-4 (LPI Cooler Outlet) will be manually throttled to achieve < 5200 gpm.

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 # 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 ≥ 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 # 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 # 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 # 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 slowly 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 # 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 switch / "2"

B. RESET button / "2"

C. Test toggle switch / "3"

D. RESET button / "3"

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?

1HP-5 (Letdown Isolation) will automatically close as letdown temperature increases to ______ °F which was **designed** to prevent a release of ________ from the purification demineralizer into the RCS.

A. 130 / Boron

B. 135 / Boron

C. 130 / Sulfur

D. 135 / Sulfur

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 # 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 # 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

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 ______ be throttled / _______.

A. can / when PZR level is ≥ 100 inches

B. can / when core SCM increases to ≥ 5°F

C. cannot / until the core and both Loops SCM are ≥ 5°F

D. cannot / until RCS pressure is above the ES 1 and 2 actuation setpoint

1 POINT

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?

A. Shifting RBCUs to high speed.

B. Penetration Room ventilation filters.

C. Reactor Building Purge inlet pre-filter.

D. Triple isolation valves in both intake/exhaust ductwork.

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 shutdown 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 <u>CORE UNCOVERY</u>.

SEE ATTACHMENT

A. 6.9

B. 8.7

C. 9.9

D. 12.3

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

Figure 1: Time To Core Boiling Prior To Refueling

Figure 2: Time To Core Uncovery Prior To Refueling

Figure 3: Time To Core Damage Prior To Refueling

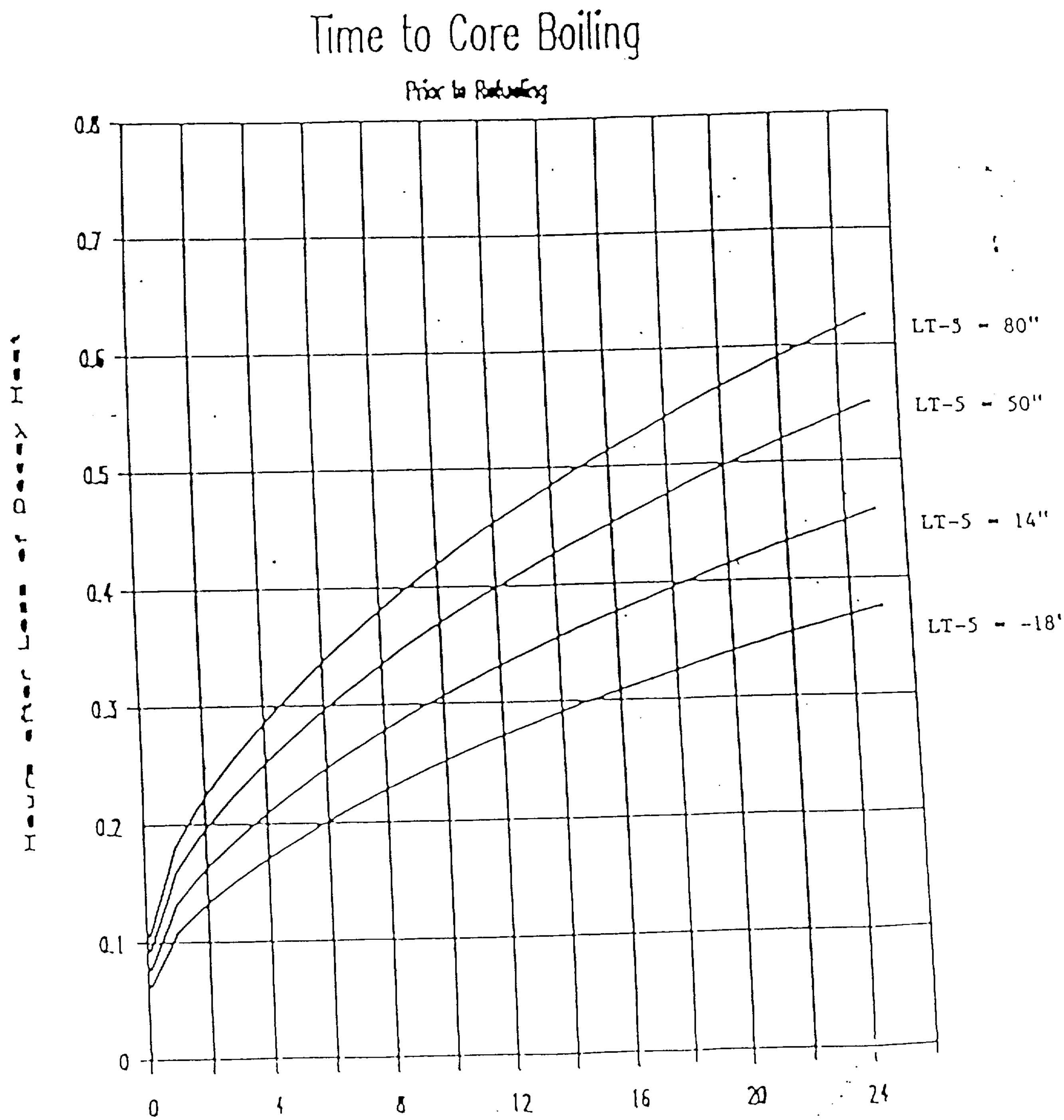
Figure 4: Time To Core Boiling After Refueling

Figure 5: Time To Core Uncovery After Refueling

Figure 6: Time To Core Damage After Refueling

ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 1

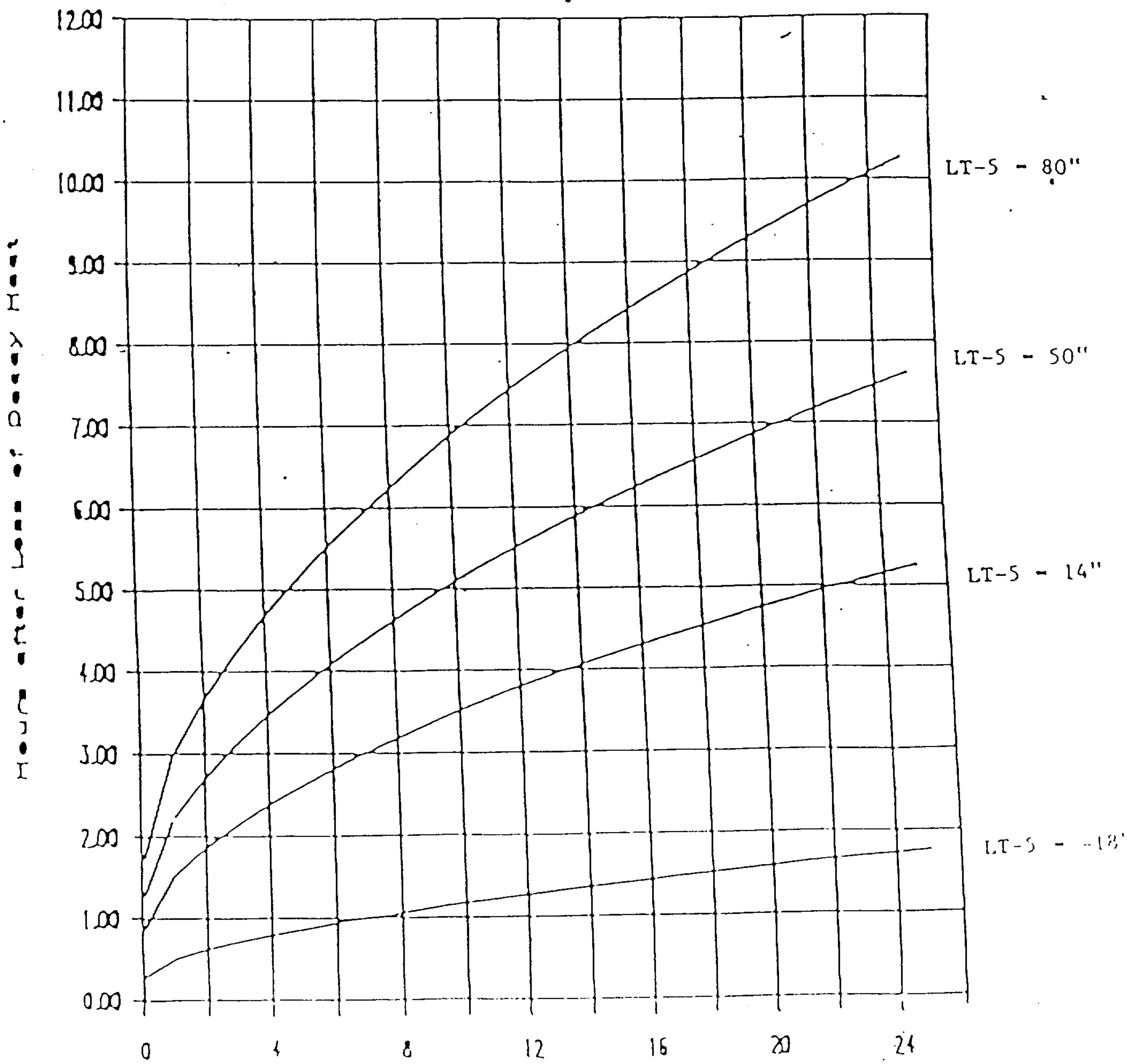


ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 2

Time to Core Uncovery

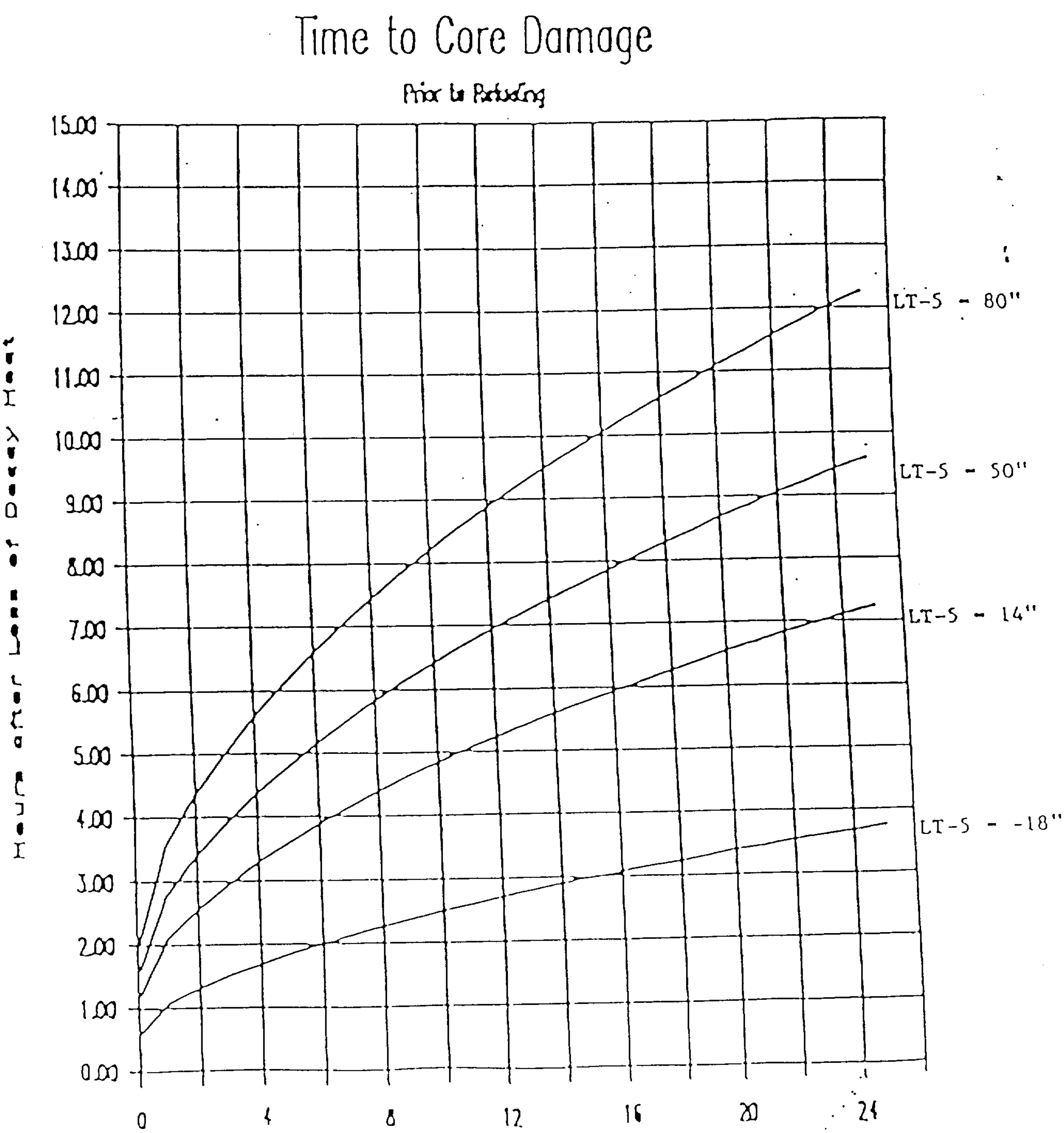
Prior to Refueling



Days after Shutdown

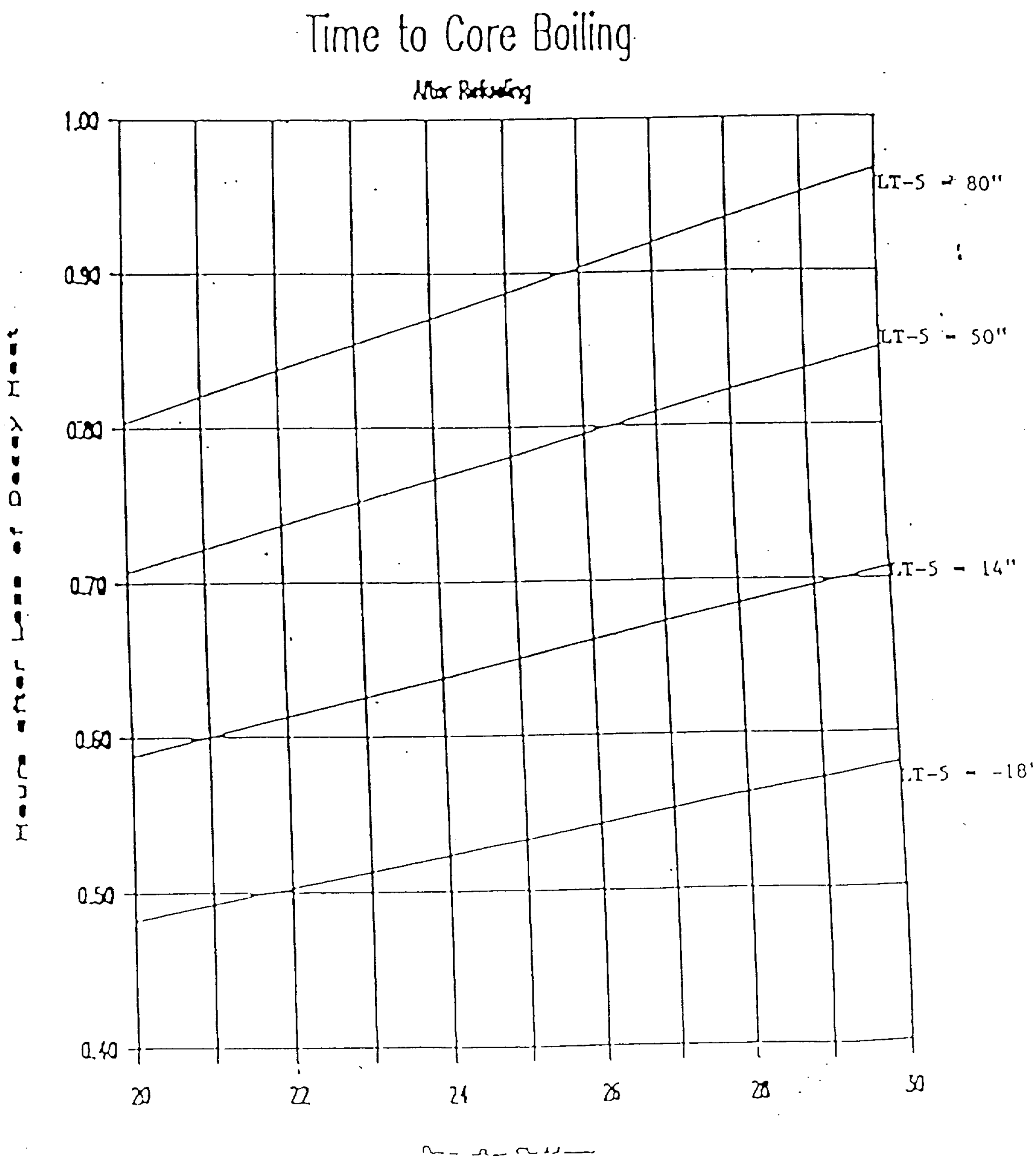
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 3



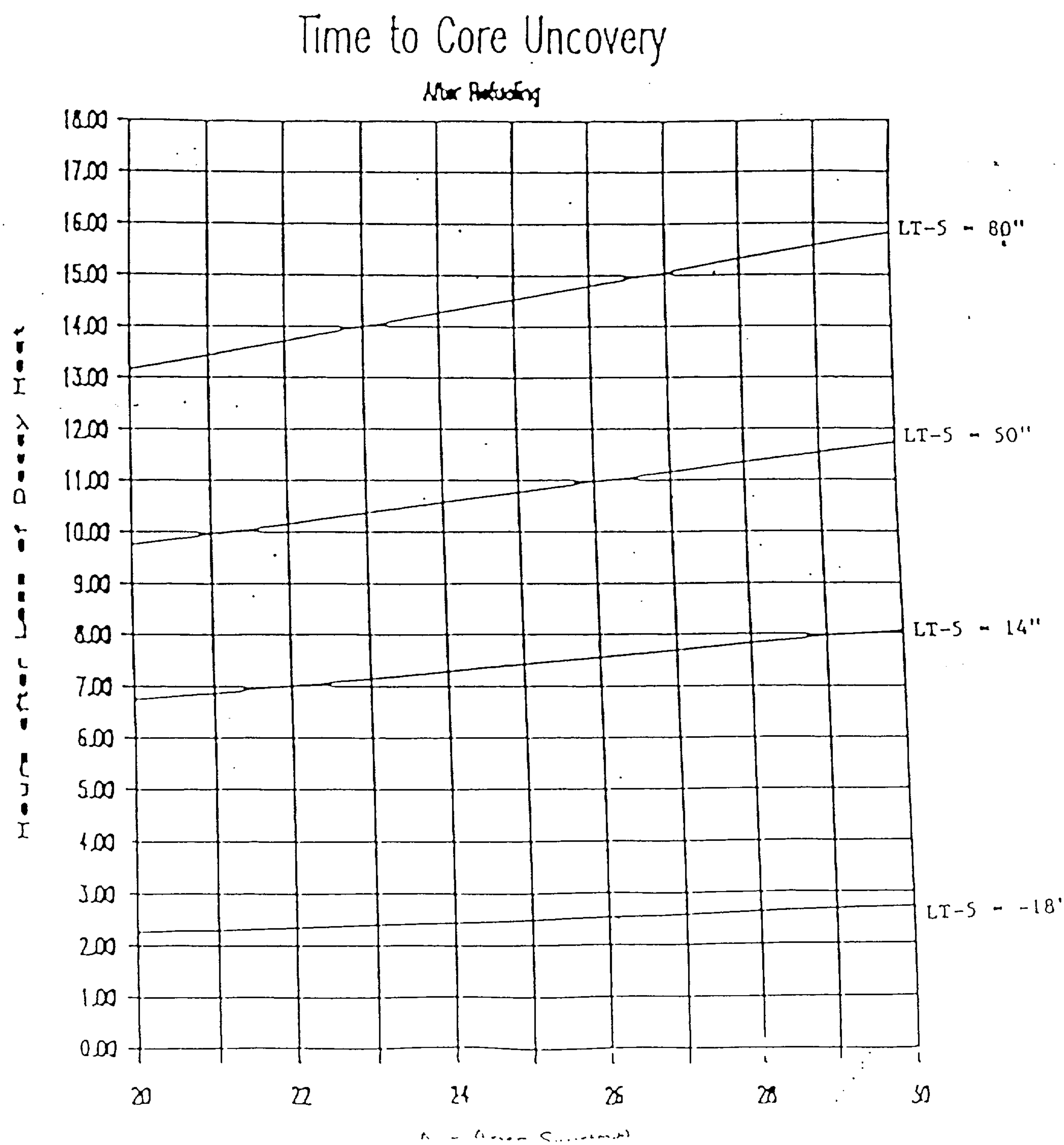
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage Following Loss of DHR Capability

Figure 4



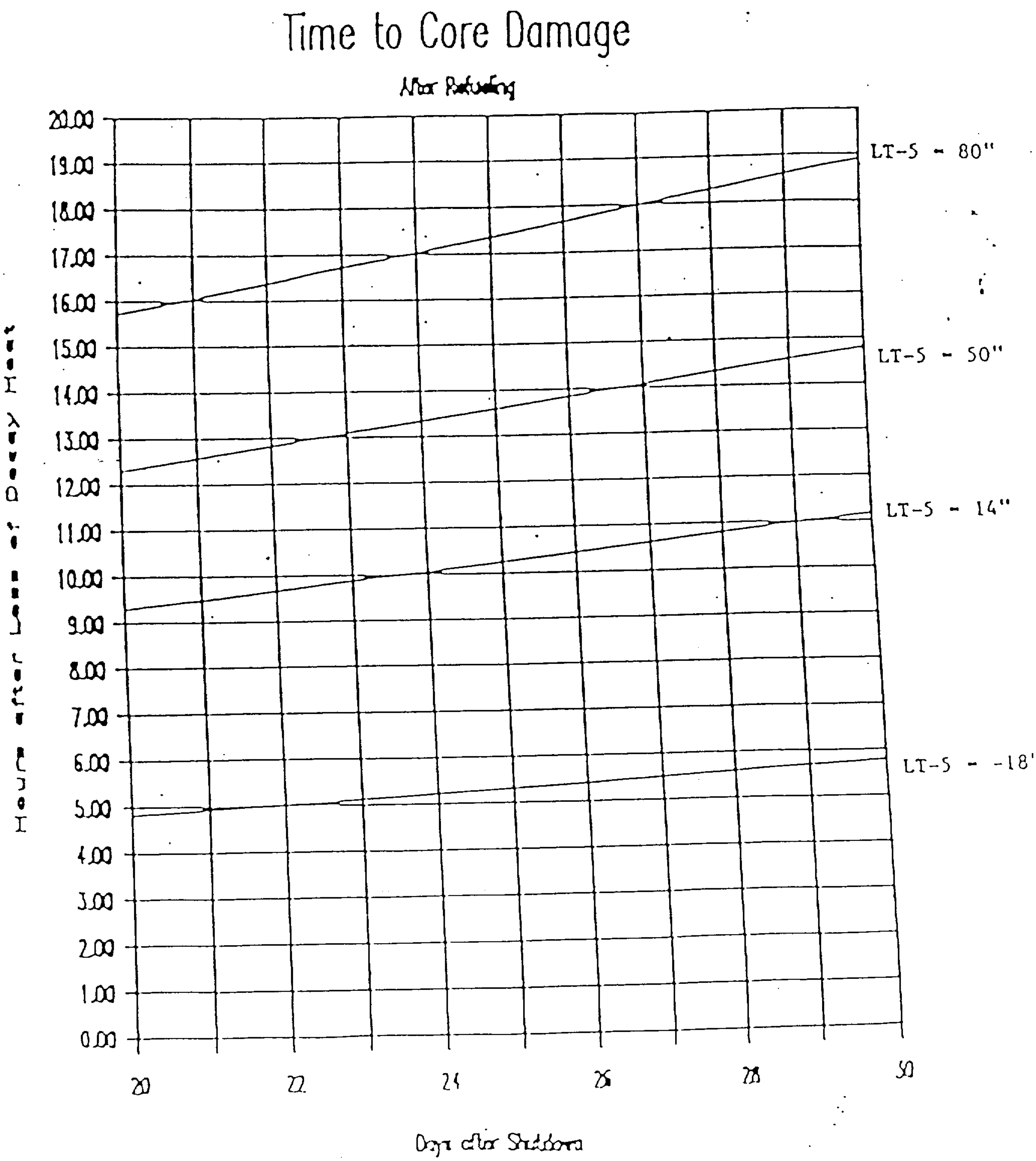
ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 5



ENCLOSURE 6.5
Elapsed Time Until Core Boiling, Core Uncovery, and Core Damage
Following Loss of DHR Capability

Figure 6



1 POINT

QUESTION # 21

Which ONE of the following correctly completes the below statement?

The LPI system design pressure interlock 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 # 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. automatically stop.

B. not stop because the Diamond is in MANUAL.

C. not stop as NI-1 has ONLY failed to mid-scale.

D. stop ONLY if the operator positions the Diamond Control ("joy stick") to "neutral".

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 # 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 # 25

Unit 1 plant conditions:

- Recovery from HPI Cooling is in progress
- EFDW has been restored
- 1TA and 1TB 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 # 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 # 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

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

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.

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

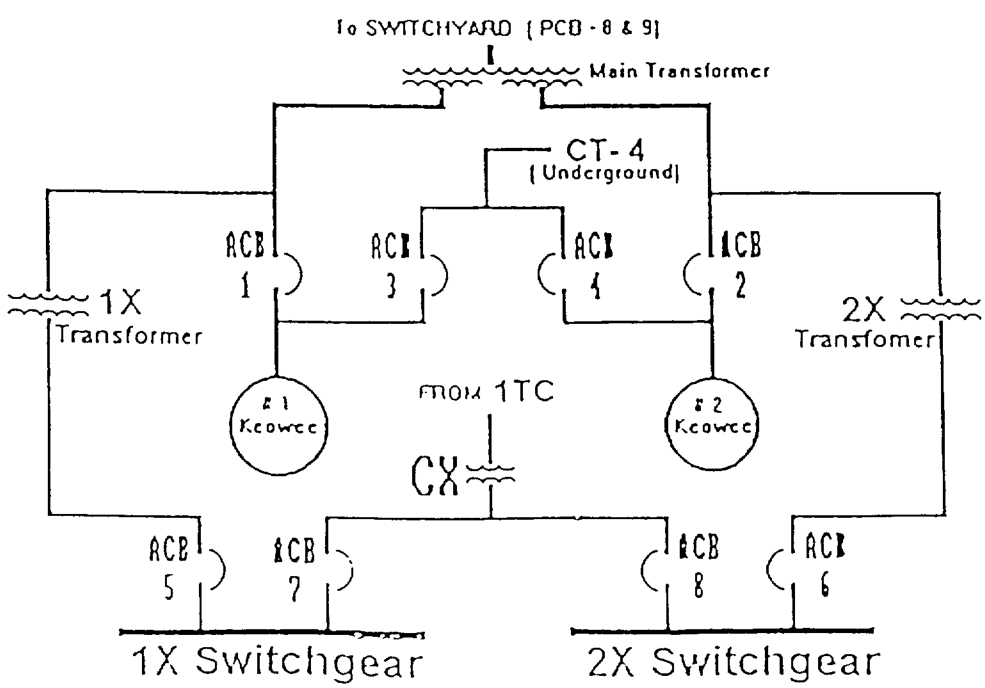
ACB- ______ will be or remain closed even if a Keowee __________________.

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 # 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...

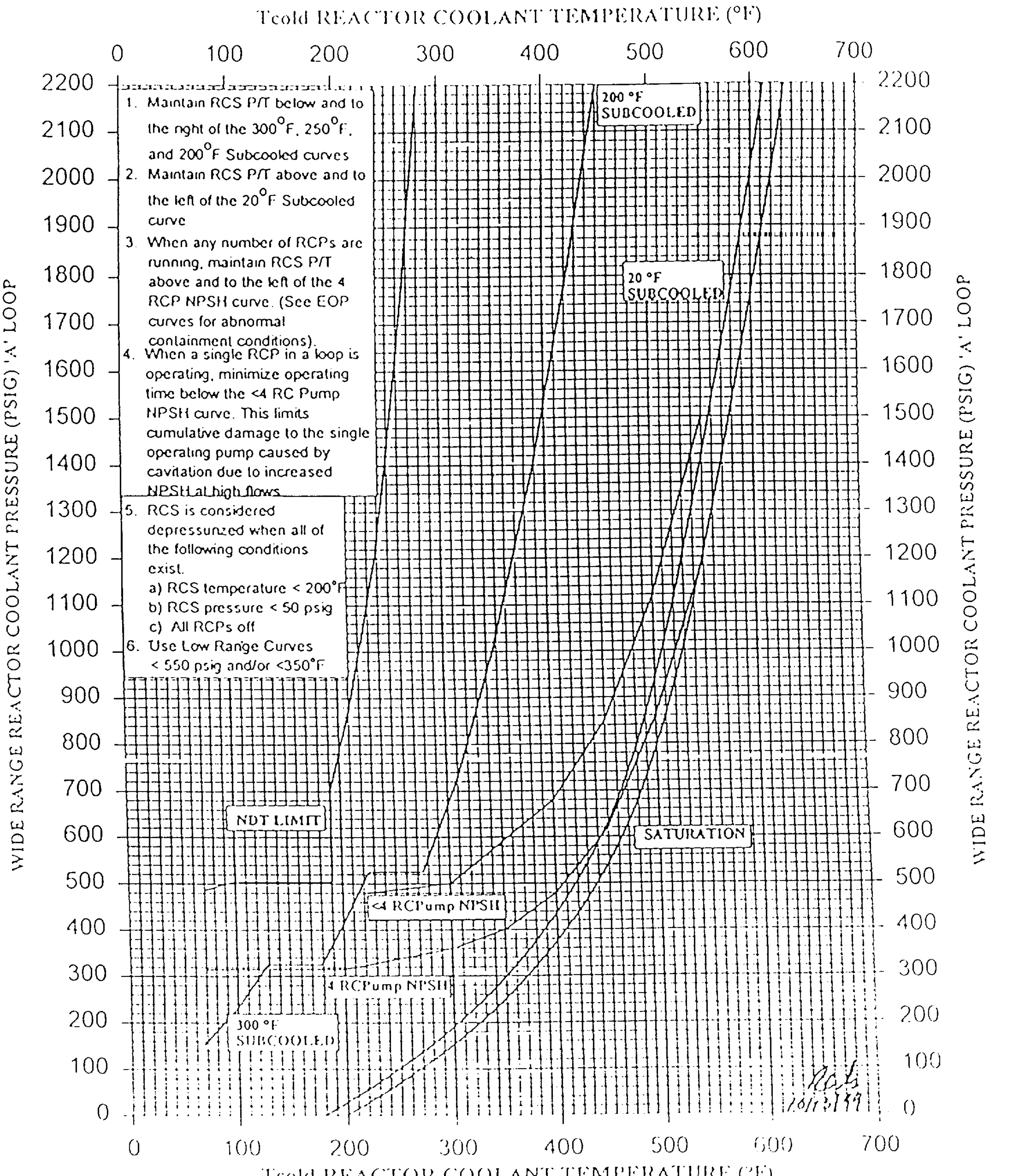
A. high and RCS pressure needs to be increased.

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.

Unit 1 Wide Range Cooldown Curve



01WRCD 33 EFPY, Rev

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

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?

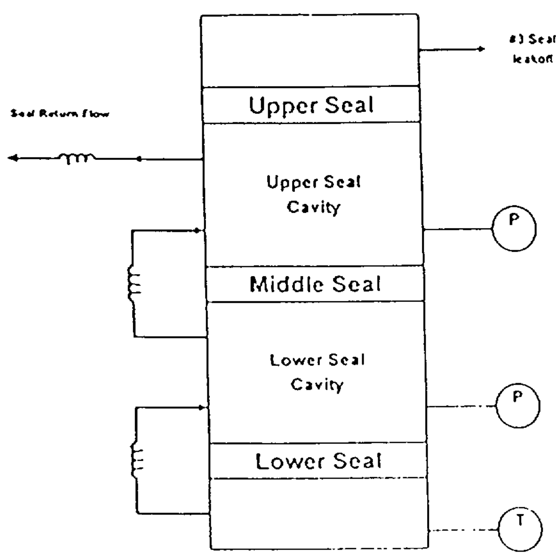
RCP 3A1 ________, 3A2________, 3B1 ________, 3B2 ________.

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 # 32

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?

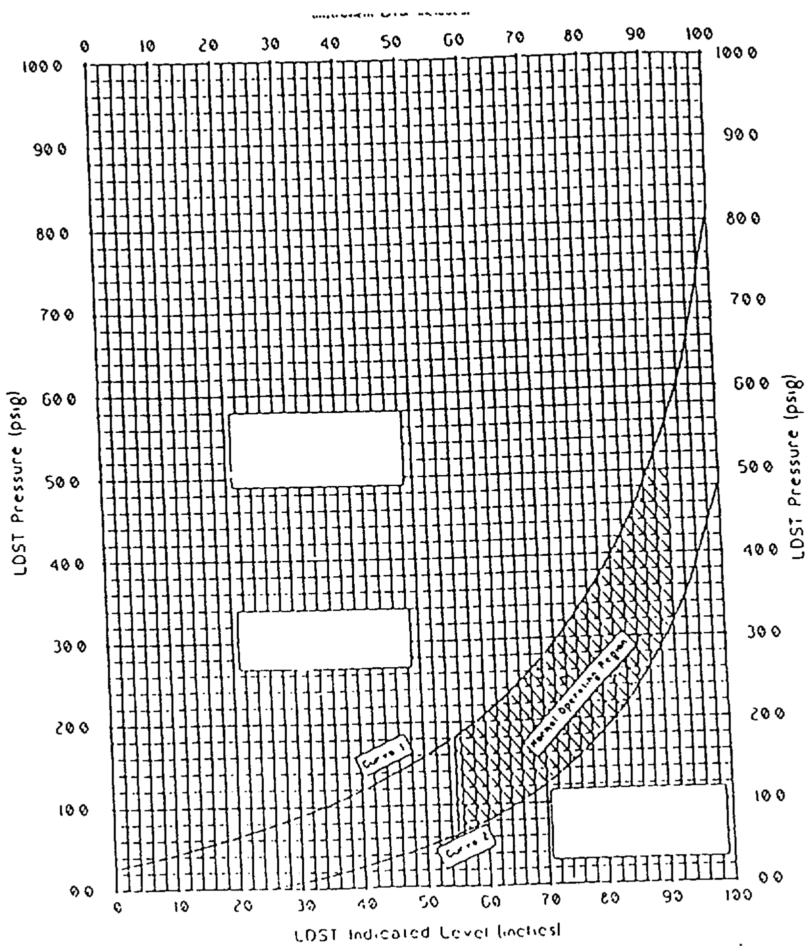
SEE ATTACHMENT

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.

D. Provide HPI suction from the BWST for transients requiring additional HPI flow.



1 POINT

QUESTION # 33

Which ONE of the following describes the designed <u>backup</u> 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

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 # 35

Unit 3 plant conditions:

- RCS pressure = 1100 psig decreasing
- ALL RB temperature <u>indications and functions</u> are inoperable
- Reactor Building Spray has <u>just</u> 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 # 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 _______.

A. has / could result in dilution of the RBES following a LOCA.

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.

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 # 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 Control valve.

C. trips the TD EFDWP, closes "B" FDW Startup Control valve.

D. closes "A" and "B" FDW Startup Control valves.

1 POINT

QUESTION # 39

Unit 1 plant conditions:

- A MS line rupture has occurred
- MSLB circuitry has actuated properly

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.

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.

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 # 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 SFP and RB.

B. Continue fuel movement and no other actions are required.

C. Suspend fuel movement in the 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 # 43

Which ONE of the following is the reason for limiting the pressurization rate of the Core Flood Tanks (CFTs)?

The pressurization rate is limited to ≤ 100 psig/15 min 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 # 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 could cause the increasing RCS 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.

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 # 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 # 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 # 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 remains 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 # 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

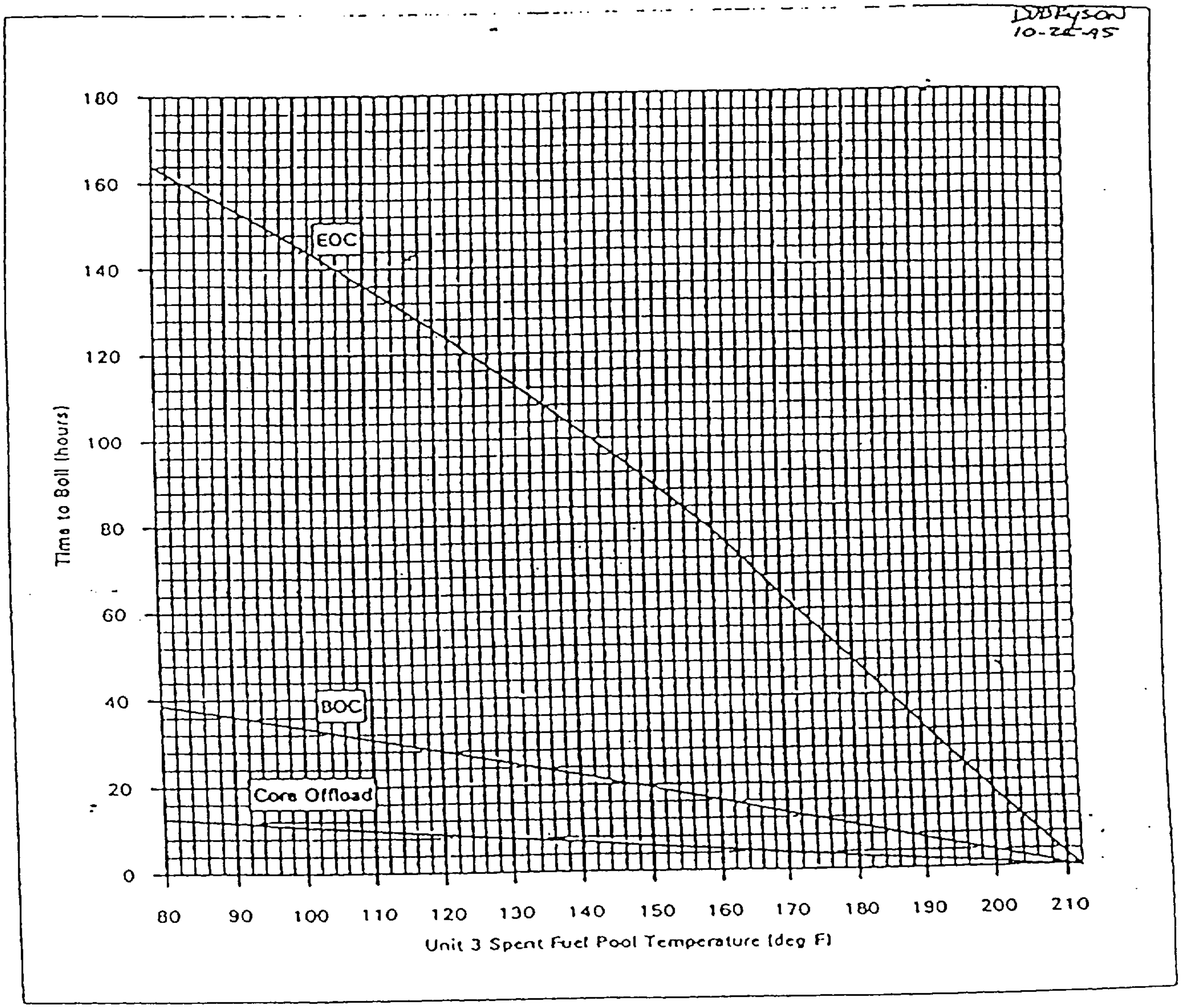
C. 30 hours

D. 28 hours

Enclosure 4.2

U3 SFP Time To Boil
After Loss Of SF Cooling

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

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

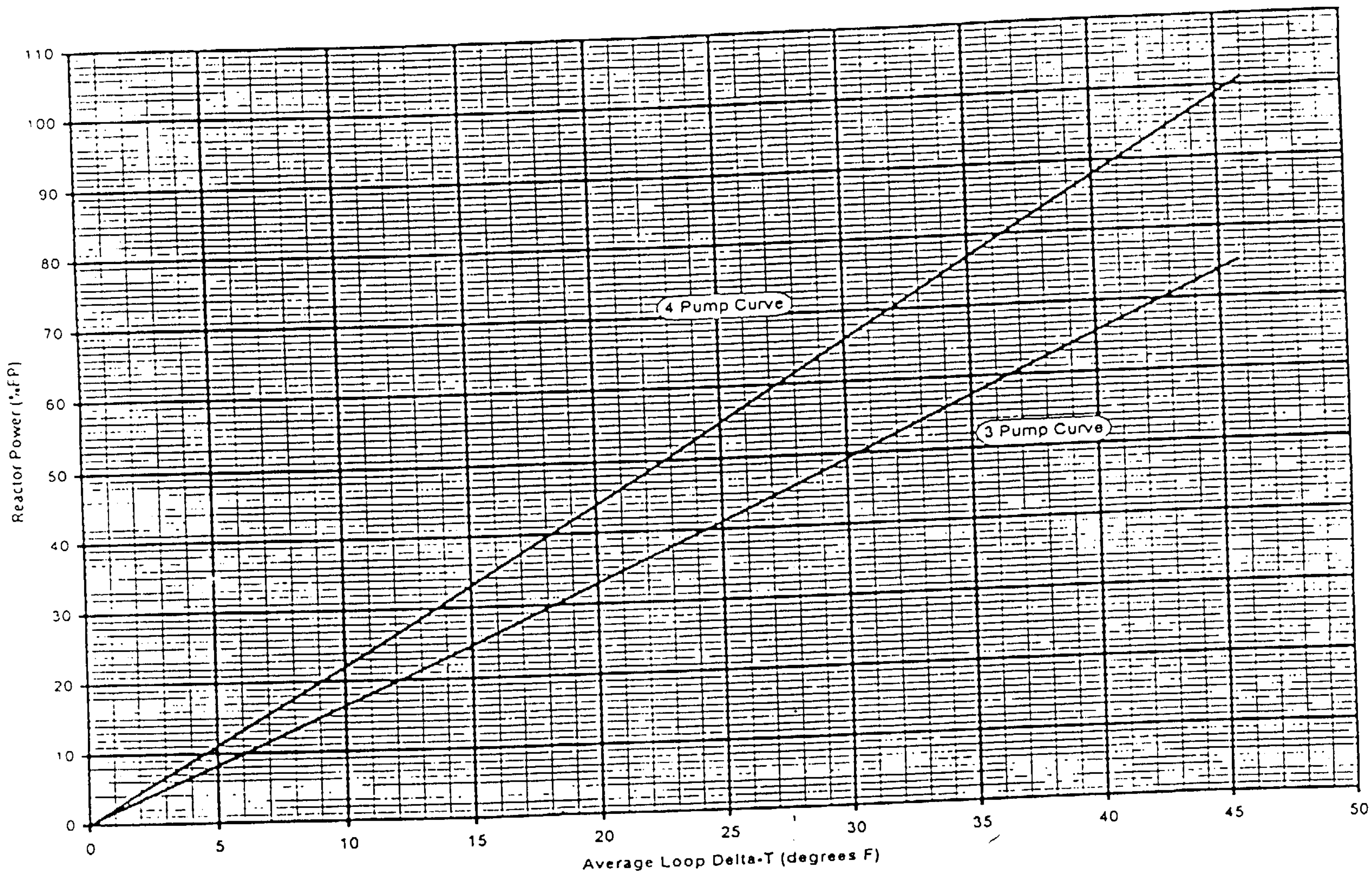
B. 582 / 575

C. 592 / 566

D. 585 / 573

Enclosure 13.12

Unit 2 Cycle 18
Loop ΔT Vs. Reactor Power



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 **19 seconds**

Which ONE of the following is correct?

The MSSVs are ______ because...

A. operable / all four MSSVs are in the closed position.

B. operable / both of the two TSV's Closure Channels have actuated within TS limits.

C. inoperable / one of the two TSV's Closure Channels has closed all the MSSVs within TS limits.

D. inoperable / neither of the two TSV's Closure Channels have closed all the MSSVs within TS limits.

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 <u>first</u> means of detecting the SGTL?

A. CSAE off gas

B. FDW mismatch

C. MS Line monitors

D. Chemistry sampling

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. must be manually transferred to 1XP then re-transferred to 1XO by the NLO as directed by the control room operator.

B. must be manually transferred to 1XP when power is restored then automatically re-transfers to 1XO when power is restored to 1XO.

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 1XP when power is restored to 1XP and remains positioned to 1XP when power is restored to 1XO.

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 Generator Differential 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

1 POINT

QUESTION # 55

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Power = 100%
- ACB-4 closed

CURRENT CONDITIONS:

- 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 # 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.

16.11 RADIOLOGICAL EFFLUENTS CONTROL

16.11.3 Radioactive Effluent Monitoring Instrumentation

COMMITMENT Radioactive Effluent Monitoring Instrumentation shall be OPERABLE as follows

a. Liquid Effluents

The radioactive liquid effluent monitoring instrumentation channels shown in Table 16.11.3-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of SLC 16.11.1.a are not exceeded.

b. Gaseous Process and Effluents

The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 16.11.3-2 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of SLC 16.11.2.a are not exceeded.

c. The setpoints shall be determined in accordance with the methodology described in the ODCM and shall be recorded.

---NOTE---

Correction to setpoints determined in accordance with Commitment c may be permitted without declaring the channel inoperable.

APPLICABILITY: According to Table 16.11.3-1 and Table 16.11.3-2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Alarm/trip setpoint less conservative than required for one or more effluent monitoring instrument channels	A.1 Declare channel inoperable OR A.2 Suspend release of effluent monitored by the channel	Immediately Immediately

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
	<u>OR</u>	
	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 <u>AND</u> 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
	<u>AND</u>	
	I.1.2 Conduct two independent data entry checks for release rate calculations	Prior to initiating subsequent release
	<u>AND</u>	
	I.1.3 Conduct two independent valve lineups of the effluent pathway	Prior to initiating subsequent release
	<u>OR</u>	
	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
	OR	
	J.2 Estimate flow rate	Immediately AND 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
	OR	
	M.2 Suspend release of radioactive effluents by this pathway.	Immediately
	OR	
	M.3.1 Collect grab sample.	Immediately AND Once per 8 hours
	AND	
	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
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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 (*) (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(**) (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) (OVS 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 # 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 # 58

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

C. Audible alarm at the detector

D. Fire alarm response guide

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 # 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 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 # 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 # 62

Which ONE of the following describes 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 IA is lost.

B. prevents excessive PRV flow during ES actuation.

C. maintains proper PRV flow to prevent PRV system PAC filter channeling.

D. maintains adequate PRV flow to eliminate charcoal filter loading and ignition.

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 # 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 # 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

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

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 due to controller failure

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 on a loss of instrument air.

D. open, close, or throttle the valve upon a loss of the control room auto positioning signal.

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 # 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 (pump is uncoupled)
- Following rotation checks, pump coupling will be performed

Which ONE of the following is correct?

Prior to pump coupling, OPEN the breaker and the tag...

A. shall be cleared and reissued.

B. and stub should be marked "VOID", 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.

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 **ONE** 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 ALL 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.

1 POINT

QUESTION # 70

Which ONE of the following conditions <u>REQUIRES</u> prior SRO approval/oversight?

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.

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.

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 # 73

Which ONE of the following is correct concerning the notification to offsite agencies during E-Plan implementation?

________ notifications shall be made within ________ minutes of declaration of the EAL.

A. Initial / 15

B. Initial / 30

C. Upgrade / 30

D. Upgrade / 60

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

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.

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

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

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 # 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 ≥ 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 # 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

Which ONE of the following directions should the SROs provide to the Reactor Operators?

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

* 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 # 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...

A. performs IMAs and then Emergency Boration 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 # 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 # 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 # 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

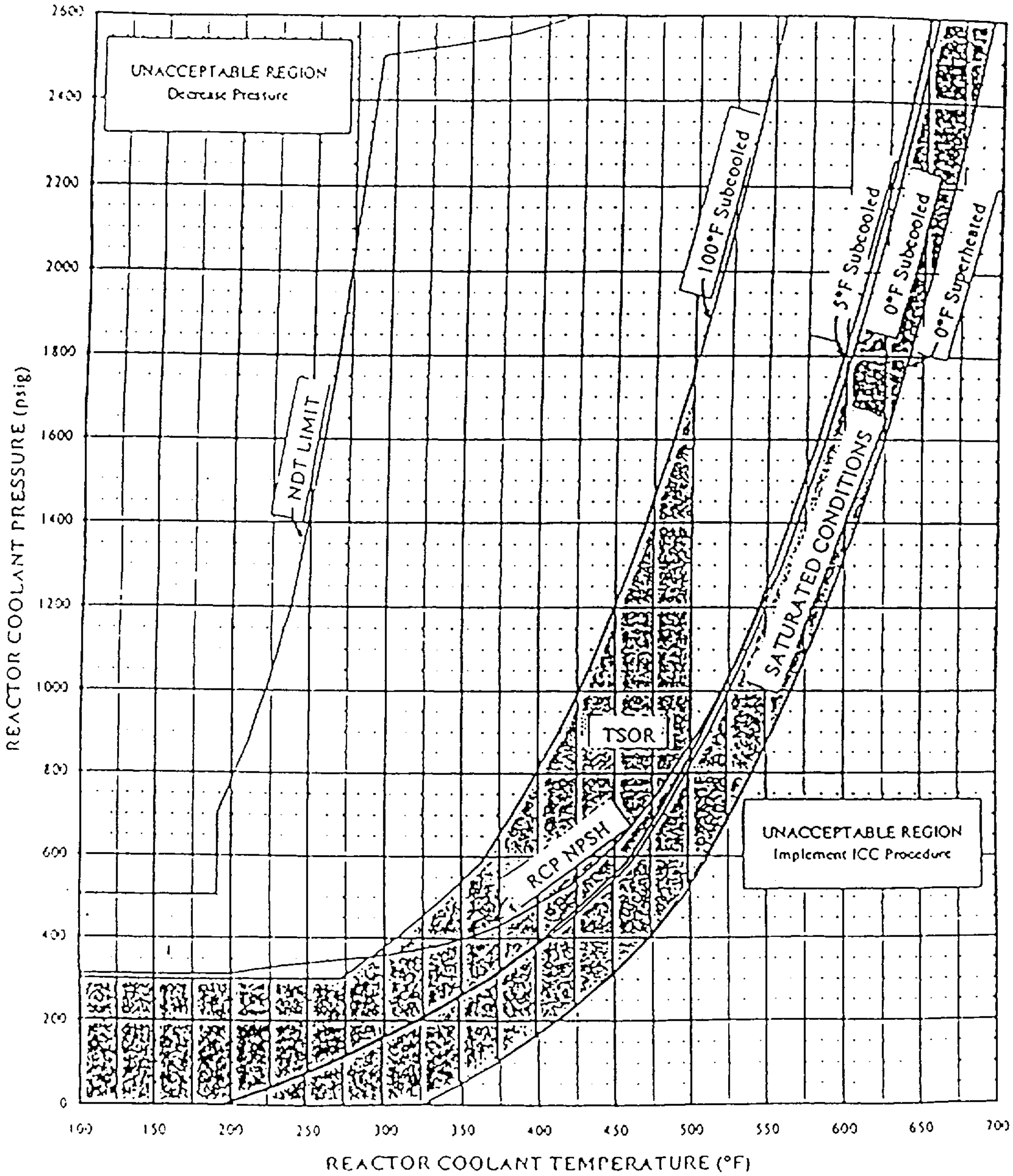
If HPI ____ throttled the ________.

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.


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

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 # 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

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. AND 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. AND 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-l, 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

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 Fire Hose Stations 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 Fire Hose Stations will be required to be OPERABLE at all times.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available for the affected areas until the inoperable equipment is restored to service.

The testing requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression System 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 required.

16.9 AUXILIARY SYSTEMS

16.9.6 Fire Detection Instrumentation

COMMITMENT The provided Fire Detection Instrumentation for each equipment/location shall be OPERABLE as listed in Table 16.9.6-1.

---------------------------------------NOTE---------------------------------------
Fire Detection Instrumentation located within containment is not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

APPLICABILITY: At all times.

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. <u>OR</u> 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

Table 16.9.6-1
FIRE DETECTION INSTRUMENTATION

EL. 796' +6"

Room No.	Equipment	Detectors Provided
300	Unit 1 Work Area	9
310	Unit 1 Equipment Room and Cable Shaft	13
311	Unit 2 Equipment Room and Cable Shaft	15
313	Janitor's Closet (Unit 1)	1
314	Clean Protective Clothing Storage (Unit 1)	1
322	Protective Clothes Storage (Unit 2)	1
329	Hot Lab	1
330	Cold Lab	1
331	Counting Room (Unit 2)	1
333	Health Physics (Unit 2)	1
334	Office (Unit 2)	1
335	Environmental Lab (Unit 2)	1
337	Laundry Sorting (Unit 2)	1
338	Laundry Storage (Unit 2)	1
339	Laundry (Unit 2)	2
347	Work Area (Unit 2)	8
354	Unit 3 Equipment Room and Cable Shaft	21
357	Janitor's Storage (Unit 3)	1
364	Towel Storage (Unit 3)	1
365	Janitor's Storage (Unit 3)	1
366	Protective Clothing (Unit 3)	1
369	HP Office (Unit 3)	1
369A	Supv. Technicians Office	1
369B	Secondary Chemistry Lab	1
369C	I.C. Computer	1
376	Unit 3 Work Area	10

EL. 771' + 0

Room No.	Equipment	Detectors Provided
119	Unit 1 and 2 LPI Hatch Area	3
159	Unit 3 LPI Hatch Area	2

EL. 838"+0

Room No.	Equipment	Detectors Provided
611	Protective Clothing Storage (Unit 2)	1
658	Protective Clothing Storage (Unit 3)	1

EL. 783' + 9"

Room	Equipment	Detectors Provided
204	Storage (Unit 1)	1
207	Chemical Handling and Storage (Unit 1)	1
220	Hot Instrument Shop (Unit 2)	1
224	Storage (Unit 2)	1
264	Storage (Unit 3)	1

Table 16.9.6-1
FIRE DETECTION INSTRUMENTATION

EL. 758' +0

Room No.	Equipment	Detectors Provided
54	Unit 1 High Pressure Injection Pumps	1
56	Unit 1 and 2 High Pressure Injection Pumps	1
58	Unit 2 High Pressure Injection Pumps	1
61	Unit 1 Low Pressure Injection Pumps	2
62	Unit 1 and 2 Low Pressure Injection Pumps	2
63	Unit 2 Low Pressure Injection Pumps	2
76	Unit 3 High Pressure Injection Pumps	1
77	Unit 3 High Pressure Injection Pumps	1
81	Unit 3 Low Pressure Injection Pumps	2
82	Unit 3 Low Pressure Injection Pumps	2

Units 1, 2, and 3 Turbine Buildings
EL. 775'+0

Equipment	Detectors Provided
MCC 1XC, 1XD, 1XE, 1XF; Unit 1 FDW Turbines; Unit 1 Emergency Feedwater Turbine; Unit 1 H2 Panel; Unit 1 EHC Unit	10
MCC 2XB, 2XC, 2XD, 2XE, 2XF; Unit 2 FDW Turbine; Unit 2 Emergency Feedwater Turbine; Unit 2 H2 Panel; Unit 2 EHC Unit	11
MCC 3XC, 3XD, 3XE, 3XF; Unit 3 FDW Turbines; Unit 3 Emergency Feedwater Turbine; Unit 3 H2 Panel; Unit 3 EHC Unit	10

EL. 796' + 6"

Equipment	Detectors Provided
Switchgear 1TA, 1TB, 2TA, 2TB; Load Centers 1X1, 1X2, 1X3, 1X4, 1X5, 1X6, 2X1, 2X2, 2X3, 2X4, 2X5, 2X6	8
Switchgear B1T, B2T; Transformer CT4	5
Switchgear 3B1T, 3B2T	3
MCC 1XA	1
ITTC5 and ITTC6	1
Unit 1 Main Turbine Oil Tank	1
Unit 2 Main Turbine Oil Tank	1
Unit 3 Main Turbine Oil Tank	2
DC Distribution Center IDA; Switchgear 1TC, 1TD, 1TE	7
MCC 1XGA	1
DC Distribution Center 2DA; Switchgear 2TC, 2TD, 2TE	7
MCC 3XGA	1
MCC 2XGB	1
Load Center 3X1, 3X2, 3X3, 3X4; MCC 3XGA; Switchgear 3TC, 3TD, 3TE	5
MCC 3XGB	1

Table 16.9.6-1
FIRE DETECTION INSTRUMENTATION

EL. 822' + 0

Equipment	Detectors Provided
Bearing Oil Lift Pumps for All Units	4 ea unit
High Pressure Unit for All Units	2 ea unit

KEOWEE HYDRO STATION

Equipment	Detectors Provided
Control Room	4
Battery Room	4
Mechanical Equipment Gallery	3
Main Lube Oil Storage Room	1
Generators 1 and 2	6 ea
Operating Floor	6

ESSENTIAL SIPHON VACUUM BUILDING	6

BASES

OPERABILITY of the NRC committed Fire Detection Instrumentation ensures that adequate warning capability is available for the prompt detection of fires in areas containing safety related and important to safety equipment at Oconee and Keowee Facilities. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program. The regulatory requirement is to have NRC committed Fire Detection Instrumentation OPERABLE only when the equipment it is protecting is required OPERABLE for plant safety. However, to also protect the equipment for property conservation and minimize equipment loss due to fire; the Oconee and Keowee NRC committed Fire Detection Instrumentation will be required to be OPERABLE at all times.

In the event that a portion of the Fire Detection Instrumentation is inoperable, the establishment of compensatory actions in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to operability.

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.
5. Oconee Plant Design Basis Specification for Fire Detection, as revised.

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

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

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.

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) 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. 1RC-66 (PZR PORV) failed open.

D. Operator fully opens 1RC-3 (PZR Spray Block).

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?

ASSUME NO OPERATOR ACTIONS

RCS pressure will decrease and…

A. PZR Heaters will energize and continuously cycle on and off between setpoints.

B. PZR Heaters will energize at setpoint and 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 # 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 = 495°F and steady
- RCS pressure = 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...

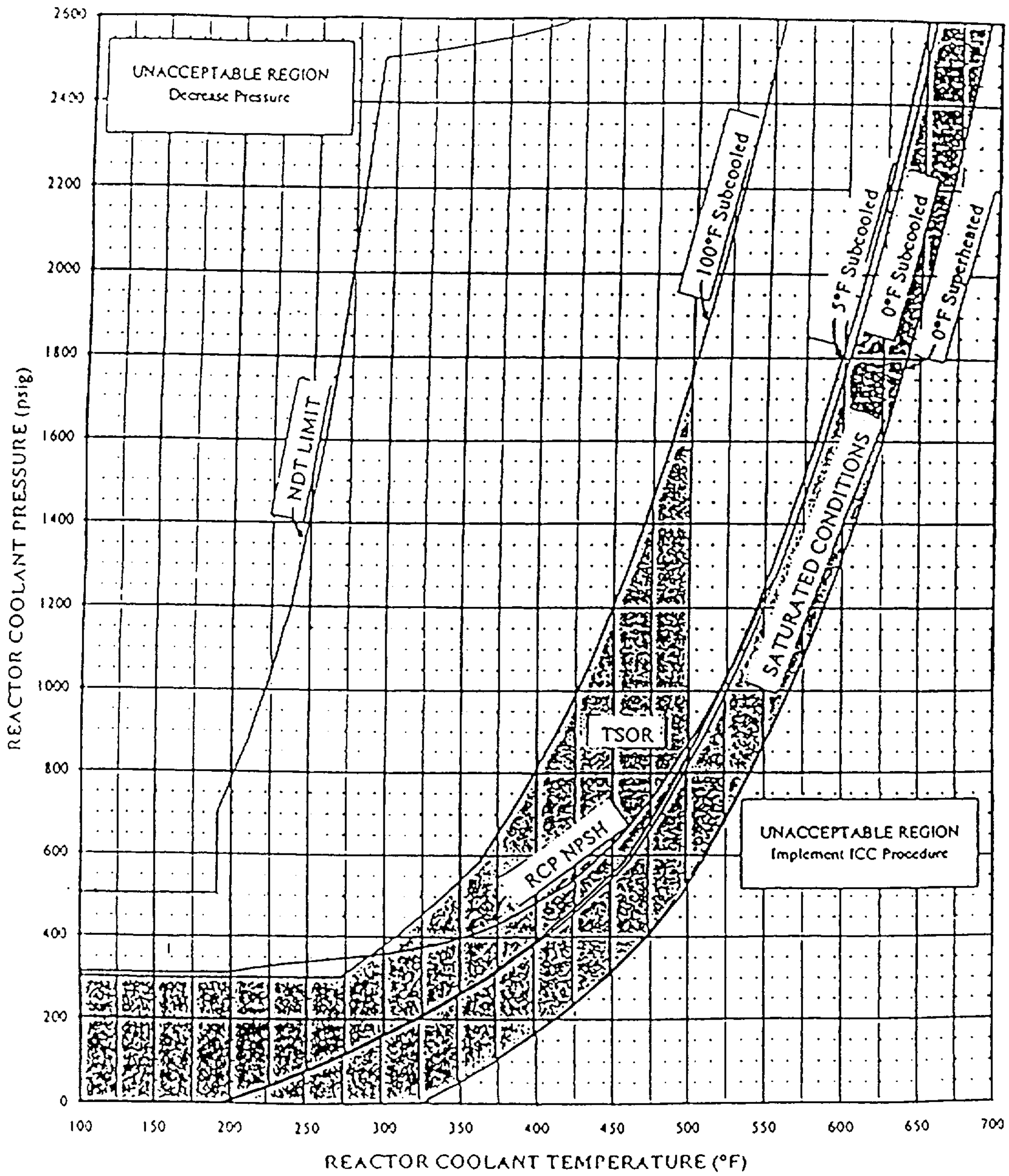
SEE ATTACHMENT

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.


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
2500
100
150
200
250
300
350
400
450
500
550
600
650
700

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 # 91

SRO ONLY

Unit 1 plant conditions:

- Reactor power = 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 # 92

SRO ONLY

Unit 1 plant conditions:

- LOCA in progress
- RCS pressure = 1126 psig
- RB Pressure = 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 immediately secure RB Spray to protect equipment .

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 # 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 not recovered, predict which ONE of the following will be an acceptable FDW condition at **0730**?

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

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.

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. <u>OR</u> Battery cell float voltage < 2.06 volts. <u>OR</u> Electrolyte Temperature < 60°F. <u>OR</u> 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 16.8.3.1	Verify pilot cell float voltage ≥ 2.13 VDC.	7 days
SR 16.8.3.2	Verify pilot cell electrolyte level > minimum and < maximum level indication marks.	7 days
SR 16.8.3.3	Verify each cell float voltage ≥ 2.13 VDC.	92 days
SR 16.8.3.4	Verify each cell electrolyte level > minimum and < maximum level indication marks.	92 days
SR 16.8.3.5	Verify temperature of every sixth connected cell > 60°F.	92 days

BASES

BACKGROUND

This SLC on the 250VDC Power Battery cell parameters utilizes the limits on electrolyte level, float voltage, and temperature for the 250VDC Power Batteries to determine OPERABILITY of the batteries. Float voltage is the voltage that is required to be continuously applied to the battery which is sufficient to maintain a constant state of charge. The limits for the designated pilot cell's float voltage, electrolyte level, and temperature is characteristic of a charged cell with adequate capacity. The limits for each connected cell's float voltage, electrolyte level and temperature ensures the OPERABILITY and capability of the battery.

In addition, the SLC provides the required actions for restoring the system to an OPERABLE status should a battery or charger become inoperable.

APPLICABLE SAFETY ANALYSIS

The 250VDC Power Batteries provide DC power for the Turbine Driven Emergency Feedwater (TDEFW) System. The OPERABILITY of the 250VDC Power Batteries is required to ensure the OPERABILITY and the capability of the TDEFW system. The TDEFW system is required to be OPERABLE in accordance with ITS 3.7.5 In addition, the Anticipated Trip Without Scram (ATWS) system is supported by the power batteries. Selected Licensee Commitment 16.7.2 provides the OPERABILITY requirements of the ATWS system. In order to maintain the required 250VDC Power Batteries OPERABLE, battery cell parameters must be maintained within specific limits.

APPLICABILITY

The Power Battery cell parameters are required to be within limits when the associated DC sources are required to be OPERABLE.

ACTIONS

The pilot cells are monitored closely as a measure of battery performance. Because pilot cells lose more electrolyte than the other cells, the designation of the pilot cell should be rotated among all cells in the battery. The Completion Times are based on engineering judgment considering operating experience, and the time required to complete the Required Actions.

A.1

If the electrolyte level is below the top of the cell plates, the entire battery is conservatively assumed to be inoperable, because the cell's discharge capacity would be reduced, and the plates may suffer permanent damage. The battery may be restored to OPERABLE status by restoring the electrolyte level in accordance with the Required Actions of the SLC.

If the float voltage of a battery cell is < 2.06 volts, the battery is assumed to be inoperable, because battery voltage may not be adequate to carry required loads. The battery may be restored to OPERABLE status by restoring the float voltage to ≥ 2.06 volts in accordance with the Required Actions of the SLC.

If the electrolyte temperature of a connected cell is < 60°F, the associated battery must be declared inoperable and the Required Actions taken as appropriate. With temperature < 60°F, the battery's capability may not be sufficient to meet the design basis load demand.

If no battery charger is available to a battery, then the associated battery shall be declared inoperable. The associated DC buses on all ONS units can be cross-tied to ensure OPERABILITY of the system.

B.1, B.2, and B.3

If a single battery is inoperable, then the associated DC buses (PA or PB) on all ONS units can be cross-tied to ensure OPERABILITY of the system. The TDEFW system and ATWS are considered OPERABLE in this configuration. If the DC buses are not cross-tied then the associated distribution center (1DP, 2DP, or 3DP) is inoperable. The TDEFW system and ATWS on the associated unit are NOT considered OPERABLE in this configuration.

C.1 and C.2

If two or more batteries are inoperable, then the associated distribution centers (1DP. 2DP, or 3DP) are inoperable. The TDEFW system and ATWS on the associated units are NOT considered OPERABLE in this configuration. Inadequate battery capacity is available to operate the PA or PB buses cross-tied with two PA or two PB batteries unavailable. In addition, excessive fault current (greater than protective device ratings) is available with a PA and PB battery unavailable and both PA and PB buses cross-tied.

D.1

If a battery charger is inoperable, then the Standby Charger can be connected to the associated DC bus to ensure OPERABILITY of the system. The TDEFW system and ATWS are considered OPERABLE in this configuration.

E.1

The limits on electrolyte level ensures no physical damage to the plates occurs and adequate electron transfer capability is maintained.

F.1

A float voltage limit of greater than or equal to 2.13 volts will ensure the cell remains fully charged with adequate capacity.

G.1

If the appropriate parameters cannot be restored in accordance with the Required Actions, the associated battery is assumed to be inoperable.

SURVEILLANCE REQUIREMENTS

SR 16.8.3.1

This Surveillance is consistent with the recommendations of Reference 1. The reference indicates that the battery be demonstrated to meet limits on a regularly scheduled interval.

SR 16.8.3.2

This Surveillance is consistent with the recommendations of Reference 1. An adequate electrolyte level ensures that there will be a proper conductivity and capacity of the battery cell.

SR 16.8.3.3

This Surveillance is consistent with the recommendations of Reference 1. A minimum voltage is established to ensure adequate voltage to maintain cells in a constant state of charge.

SR 16.8.3.4

This Surveillance is consistent with the recommendations of Reference 1 and the battery manufacturers. An adequate electrolyte level ensures that there will be a proper conductivity path and capacity of the battery cell.

SR 16.8.3.5

This Surveillance is consistent with the recommendations of Reference 1. The electrolyte must be maintained above a minimum temperature for the battery to deliver designed power.

REFERENCES:

1. A IEEE Standard 450-1975, Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations.
2. ITS 3.7.5, Emergency Feedwater System.
3. Selected Licensee Commitment 16.7.2, Anticipated Transient Without Scram.

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 # 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 # 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

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 **(minimum level of authority)** 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 (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 Manager (Superintendent of Ops) and RP Manager.

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

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
- ☐ (iii) 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

SRO only
#99

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
G.D. Gilbert/02-23-00	M.T. Cash/02-10-00	L. E. Nicholson/02-18-00
Regulatory Compliance Manager	Regulatory Compliance Manager	Regulatory Compliance Manager
Effective Date:	Effective Date:	Effective Date:
03/14/00	03/14/00	03/14/00

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.

Table Of Contents

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 report: "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications."	Licensees shall submit a Licensee Event Report on: "The completion 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

Reportable

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

Reportable

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.

Non-Reportable

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

Reportable

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 actual failures, 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

Reportable

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 before 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

Reportable

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

Non-Reportable

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 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°.

FINAL SUBMITTAL

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

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

NUREG-1021 - ES-501

ES-301-1 - ADMIN TOPICS OUTLINE

ES-301-2 - CONTROL ROOM SYSTEMS AND FACILITY WALK-THROUGH TEST OUTLINE

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

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

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]

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. JPM NRC-006, Restore RBCUs to normal following an ES Channel 5 actuation per RULE #7; (10 min.) EOP RULE #7 [KA: 022A4.01 (3.6/3.6)]	N, 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

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. JPM NRC-006, Restore RBCUs to normal following an ES Channel 5 actuation per RULE #7; (10 min.) EOP RULE #7 [KA: 022A4.01 (3.6/3.6)]	N, 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

System / JPM Title	Type Code*	Safety Function
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

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. JPM NRC-006, Restore RBCUs to normal following an ES Channel 5 actuation per RULE #7; (10 min.) EOP RULE #7 [KA: 022A4.01 (3.6/3.6)]	N, 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

FINAL SUBMITTAL

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

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

NUREG-1021 - ES-501 - F.1.g

FINAL AS-GIVEN JPMs FOR EACH

WALK-THROUGH TEST

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. STANDARD: Obtain copy of PT/1/A/1103/15, Reactivity Balance Calculations. COMMENTS:	___ SAT ___ UNSAT
STEP 2 Determine proper enclosure to use. STANDARD: Enclosure 13.20, Shutdown Margin at Power, is chosen. COMMENTS:	___ SAT ___ UNSAT
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. STANDARD: SDM is determined to be is ≥ 1% ΔK/K. COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
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:

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.

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: ____________________________________ Time Start : ________
NAME Time Finish: ________

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

Examiner: ____________________________ _________________________________/________
NAME SIGNATURE DATE

==

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, through RCS Pressure, Temperature, and Flow DNB Limits.

INITIATING CUE:

The SRO directs you to perform the Axial Power Imbalance Operating Limits verification.

The Reactor calculation package is NOT running.

The Backup Incore detectors will be used for this determination.

Only Imbalance Surveillance is required for this JPM.

Do NOT perform QPTR.

Use Backup Incore recorders refer to PT/0/A/1103/019, Backup Incore Detector System.

- At step 12.1.1, in lieu of Enclosure 13.1, use Backup Incore Chart "A" provided.

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: Refer to Section 12.3. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Review step 12.3.1 which states: Axial Imbalance shall **NOT** exceed appropriate limit curve in COLR. IF axial imbalance limit is exceeded, take immediate corrective action to achieve an acceptable imbalance. 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) COMMENTS: **NOTE: Later in JPM when imbalance calculation is made with the Incores a different enclosure from the COLR will be used.**	___ SAT ___ UNSAT

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STEP 3:

Review step 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/10/A/1103/019 (Backup Incore Detector System).

STANDARD:

Candidate refers to PT/10/A/1103/019 (Backup Incore Detector System).

COMMENTS:

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

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STEP 4:

Verification of minimum Incore operability.

STANDARD:

Candidate receives the information.

COMMENTS:

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___ SAT

___ UNSAT

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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. COMMENTS:	___ SAT ___ UNSAT
STEP 6: Calculate axial imbalance per Enclosure 13.3 using operable recorder points (on Backup Incore Chart "A"). STANDARD: The candidate refers to and obtains a copy of Enclosure 13.3. The candidate performs calculation per Enclosure 13.3. **NOTE: Examiner refer to completed enclosure 13.3 (KEY).** 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 step 12.2.4 cannot be satisfied, notify the Unit Supervisor and take appropriate actions described in the applicable Technical Specifications as listed below Axial Power Imbalance - ITS 3.2.2 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 Surveillance, Enclosures 13.1 has been completed up to page 6, through RCS Pressure, Temperature, and Flow DNB Limits.

INITIATING CUE:

The SRO directs you to perform the Axial Power Imbalance Operating Limits verification.

The Reactor calculation package is NOT running.

The Backup Incore detectors will be used for this determination.

Only Imbalance Surveillance is required for this JPM.

Do NOT perform QPTR.

Use Backup Incore recorders refer to PT/0/A/1103/019, Backup Incore Detector System.

- At step 12.1.1, in lieu of Enclosure 13.1, use Backup Incore Chart "A" provided.

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

NOTE: All points on Backup Incore Chart "A" are operable (no points are off scale or contain a note).

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 through page 20.

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

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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:

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

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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:

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

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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:

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

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</table>

STEP 5: Using the OAC (Main/General/Steam Table) determine saturation temperature for pressure recorded in step 2.2.1 and document below ***Cue: Use OAC to obtain data.*** 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: **(600 °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. **(20)** °F = (638) °F – (600) °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) – 20 = 7** COMMENTS: **Note: 7°F is greater than is ±5°F allowable tolerance.**	**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 through page 20.

INITIATING CUES:

The SRO directs you to complete PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.1.

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 to verify that 3RC-2 (PZR Spray Bypass) is properly back seated.

- Contamination levels on top of the PZR are 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: 06/26/00 13:46

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

FOR INFORMATION ONLY

APPROVED BY: NRW1552	TERMINATED BY: NRW1552
DATE/TIME: 03/30/00 13:35	DATE/TIME: 05/22/00 15:00

RADIATION WORK PERMIT # 1 REV: 11 DATE/TIME: 12/14/99 10:31

OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: ENTRY FOR WALK THROUGH TO OFFICE AREA (NO HANDS ON WORK)

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

A 1. NON-CONTAMINATED AREA. (NO HANDS ON WORK)

SPECIAL DOSIMETRY	RESPIRATORY

COMMENTS

ED (MG) SET POINTS:

DOSE ALARM - 5 MREM

DOSE RATE ALARM - 5 MREM/HR

APPROVED BY: DRW7318 DATE/TIME: 12/14/99 09:57	TERMINATED BY: DATE/TIME:

RADIATION WORK PERMIT # 2 REV: 14 DATE/TIME: 12/14/99 10:32

OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: ENTRY FOR ROUTINE SURVEILLANCE/PLANT AND SYSTEM OPERATION (E.G., ALARA, JOB PLANNING, OPERATIONS, ETC.)

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

A 1. NON-CONTAMINATED AREA.

B 2. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE ONLY CONTACT IS WITH HANDS & NO TEARING OF GLOVES.

C 3. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE THERE IN NO POTENTIAL FOR CONTACT OTHER THAN HAND.

D 4. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

H 5. WORK IN CONTAMINATED AREA.

I 6. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.

M 7. HEAVY WORK IN CONTAMINATED AREA REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

N 8. WORK IN CONTAMINATED AREA WHEN WET CONDITIONS OR POTENTIAL FOR WET CONDITIONS EXIST.

SPECIAL DOSIMETRY	RESPIRATORY

COMMENTS

NOTIFY RP IF JOB SCOPE OR WORK AREA CONDITIONS CHANGE.

UTILIZE MIRRORS AND/OR REMOTE SURVEILLANCE EQUIPMENT TO ELIMINATE ROOM ENTRY WHEN FEASIBLE.

ED (MG) SET POINTS:
DOSE ALARM - 15 MREM
DOSE RATE ALARM - 100 MREM/HR

APPROVED BY: DRW7318 DATE/TIME: 12/14/99 09:58	TERMINATED BY: DATE/TIME:

RADIATION WORK PERMIT # 5 REV: 13 DATE/TIME: 12/14/99 10:34

OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: ENTRY FOR MINOR CORRECTIVE MAINTENANCE (E.G., PHONES, PLUMBING, "HANDS ON WORK")

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

A 1. NON-CONTAMINATED AREA.

B 2. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE ONLY CONTACT IS WITH HANDS & NO TEARING OF GLOVES.

C 3. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE THERE IS NO POTENTIAL FOR CONTACT OTHER THAN HAND.

D 4. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

F 5. LIGHT WORK IN A CONTAMINATED AREA NOT REQUIRING COMPLETE PROTECTION TO THE SKIN AND CLOTHING.

H 6. WORK IN CONTAMINATED AREA.

M 7. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

SPECIAL DOSIMETRY

RESPIRATORY

SPECIAL INSTRUCTIONS/PRECAUTIONS

* NOTIFY RP PRIOR TO START OF WORK
* NOTIFY RP PRIOR TO REACHING/ENTRY INTO OVERHEAD (8 FEET AND ABOVE)

COMMENTS

RESPIRATORY REQUIREMENTS WILL BE BASED ON TEDE/ALARA EVALUATION.

NOTIFY R.P. WHEN JOB SCOPE OR WORK AREA CONDITIONS CHANGE.

PERFORM PREFABRICATION WORK OUTSIDE THE RCA WHEN POSSIBLE.

ED (MG) SET POINTS:
DOSE ALARM - 15 MREM
DOSE RATE ALARM - 100 MREM/HR

APPROVED BY: DRW7318
DATE/TIME: 12/14/99 10:00

TERMINATED BY:
DATE/TIME:

RADIATION WORK PERMIT # 10 REV: 12 DATE/TIME: 12/14/99 10:35

OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: HOUSEKEEPING

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

A 1. NON-CONTAMINATED AREA.

B 2. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE ONLY CONTACT IS WITH HANDS & NO TEARING OF GLOVES.

C 3. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE THERE IS NO POTENTIAL FOR CONTACT OTHER THAN HAND.

D 4. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

H 5. WORK IN CONTAMINATED AREA.

I 6. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.

M 7. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

N 8. 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
* NOTIFY RP PRIOR TO REACHING/ENTRY INTO OVERHEAD (8 FEET AND ABOVE)

COMMENTS

NOTIFY RP IF JOB SCOPE OR WORK AREA CONDITIONS CHANGE.

ED (MG) SET POINTS:
DOSE ALARM - 15 MREM
DOSE RATE ALARM - 100 MREM/HR

APPROVED BY: DRW7318 DATE/TIME: 12/14/99 10:02	TERMINATED BY: DATE/TIME:

RADIATION WORK PERMIT # 11 REV: 13 DATE/TIME: 12/14/99 10:36

OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: ROUTINE SPENT FUEL POOL AREA ACTIVITIES (EXCLUDING REFUELING)

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

A 1. NON-CONTAMINATED AREA.

B 2. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE ONLY CONTACT IS WITH HANDS & NO TEARING OF GLOVES.

C 3. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE THERE IS NO POTENTIAL FOR CONTACT OTHER THAN HAND.

D 4. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

H 5. WORK IN CONTAMINATED AREA.

I 6. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.

M 7. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

SPECIAL DOSIMETRY

RESPIRATORY

SPECIAL INSTRUCTIONS/PRECAUTIONS

* NOTIFY RP PRIOR TO START OF WORK
* NOTIFY RP PRIOR TO MOVING/REMOVING ITEMS IN THE SPENT FUEL POOL

COMMENTS

SPRAY / WIPE DOWN ALL MATERIALS BEITN REMOVED FROM SFP AS DIRECTED BY RP.

RP SHALL BE PRESENT AND PERFORM RADIATION SURVEYS ON ALL MATERIALS BEING REMOVED FROM THE SFP.

SURGICAL CAPS MAY BE WORN IN LIEU OF PROTECTIVE CLOTHING CLOTH HOOD, ONLY WITH PRIOR APPROVAL FROM RP.

ED (MG) SET POINTS:
DOSE ALARM - 15 MREM
DOSE RATE ALARM - 100 MREM/HR

FOR INFORMATION ONLY

APPROVED BY: DRW7318
DATE/TIME: 12/14/99 10:02

TERMINATED BY:
DATE/TIME:

RADIATION WORK PERMIT # 13 REV: 13 DATE/TIME: 12/14/99 10:36

OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: PRIMARY CHEMISTRY ACTIVITIES

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

A 1. NON-CONTAMINATED AREA.

E 2. WORK NOT REQUIRING COMPLETE PROTECTION OF SKIN AND CLOTHING, SUCH AS WORKING ACROSS A RCZ BOUNDARY (I.E. SAMPLING).

F 3. LIGHT WORK IN A CONTAMINATED AREA NOT REQUIRING COMPLETE PROTECTION TO THE SKIN AND CLOTHING.

H 4. WORK IN CONTAMINATED AREA.

SPECIAL DOSIMETRY	RESPIRATORY

COMMENTS

DURING SAMPLING, ENSURE PROTECTIVE CLOTHING COVERS THE ENTIRE ARM AREA.

LAB COAT MAY BE WORN FROM SAMPLING POINT TO PROCESSING LAB.

SLEEVE GUARDS MAY BE WORN, AS NECESSARY, IN ADDITION TO LAB COATS FOR EXTREMITY PROTECTION WHEN CONDUCTING REACH INS.

ED (MG) SET POINTS:
DOSE ALARM - 15 MREM
DOSE RATE ALARM - 100 MREM/HR

FOR INFORMATION ONLY

APPROVED BY: DRW7318 DATE/TIME: 12/14/99 10:03	TERMINATED BY: DATE/TIME:

RADIATION WORK PERMIT # 24 REV: 12 DATE/TIME: 12/14/99 10:39

OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: REMOVAL AND REPLACEMENT OF RADIOACTIVE FILTERS/ STRAINERS (INCLUDING VACUUMS/HEPA'S)

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

A 1. NON-CONTAMINATED AREA.

B 2. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE ONLY CONTACT IS WITH HANDS & NO TEARING OF GLOVES.

C 3. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE THERE IS NO POTENTIAL FOR CONTACT OTHER THAN HAND.

D 4. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

H 5. WORK IN CONTAMINATED AREA.

I 6. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.

M 7. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

N 8. 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

COMMENTS

RESPIRATORY REQUIREMENTS WILL BE BASED ON TEDE/ALARA EVALUATION.

ED (MG) SET POINTS:

DOSE ALARM - 15 MREM

DOSE RATE ALARM - 100 MREM/HR

FOR INFORMATION ONLY

APPROVED BY: DRW7318
DATE/TIME: 12/14/99 10:08

TERMINATED BY:
DATE/TIME:

RADIATION WORK PERMIT # 33 REV: 12 DATE/TIME: 12/14/99 10:41

OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: ENTRY FOR TRAINING AND EMERGENCY DRILLS/CONDITIONS

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

A 1. NON-CONTAMINATED AREA.

B 2. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE ONLY CONTACT IS WITH HANDS & NO TEARING OF GLOVES.

C 3. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE THERE IS NO POTENTIAL FOR CONTACT OTHER THAN HAND.

D 4. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

H 5. WORK IN CONTAMINATED AREA.

I 6. WORK IN CONTAMINATED AREA AND HANDS ON HIGHER CONTAMINATED MATERIAL OR BETA DOSE CONCERN TO HANDS ONLY.

M 7. HEAVY WORK IN CONTAMINATED AREAS REQUIRING ADDITIONAL CONTROLS FOR CONTAMINATION OR SKIN DOSE.

N 8. WORK IN CONTAMINATED AREA WHEN WET CONDITIONS OR POTENTIAL FOR WET CONDITIONS EXIST.

SPECIAL DOSIMETRY	RESPIRATORY

SPECIAL INSTRUCTIONS/PRECAUTIONS

* NOTIFY RP PRIOR TO REACHING/ENTRY INTO OVERHEAD (8 FEET AND ABOVE)

COMMENTS

ED (MG) SET POINTS:

DOSE ALARM - 15 MREM

DOSE RATE ALARM - 100 MREM/HR

FOR INFORMATION ONLY

APPROVED BY: DRW7318 DATE/TIME: 12/14/99 10:10	TERMINATED BY: DATE/TIME:

RADIATION WORK PERMIT # 36 REV: 10 DATE/TIME: 12/14/99 10:42
OCONEE NUCLEAR STATION ACTIVATION DATE: 01/01/00 00:01

Job Title: DECLARED PREGNANT WOMAN

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

A 1. NON-CONTAMAINATED AREA.

B 2. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE ONLY CONTACT IS WITH HANDS & NO TEARING OF GLOVES.

C 3. WORKING FROM A NON-CONTAMINATED AREA WITH CONTAMINATED MATERIAL WHERE THERE IS NO POTENTIAL FOR CONTACT OTHER THAN HAND.

D 4. CONTAMINATED AREA FOR SHORT DURATION WITH NO OBSTRUCTIONS TO CONTRIBUTE TO CONTAMINATION OF UNPROTECTED SKIN / CLOTHING.

E 5. HANDLING AND/OR TRANSPORTING RADIOACTIVE MATERIAL WHERE THE POTENTIAL OF LOOSE SURFACE CONTAMINATION EXIST.

F 6. LIGHT WORK IN CONTAMINATED AREA NOT REQUIRING COMPLETE PROTECTION TO THE SKIN AND CLOTHING.

H 7. WORK IN CONTAMINATED AREA.

Z 8. AS DIRECTED BY R.P.

SPECIAL DOSIMETRY	RESPIRATORY

SPECIAL INSTRUCTIONS/PRECAUTIONS

* NOTIFY RP PRIOR TO START OF WORK

COMMENTS

AVOID ENTRIES INTO AIRBORNE, HIGH RADIATION AND EXTRA HIGH RADIATION AREAS.

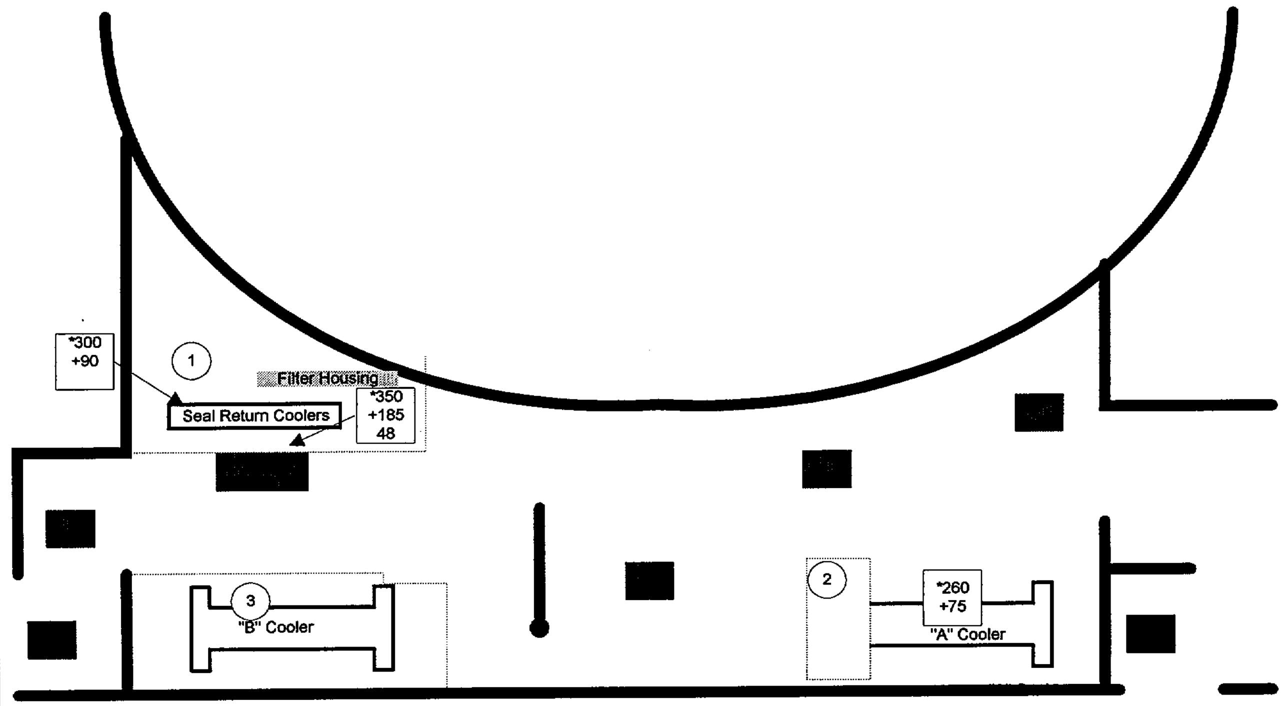
ED (MG) SET POINTS:
DOSE ALARM - 5 MREM
DOSE RATE ALARM - 5 MREM/HR

FOR INFORMATION ONLY

APPROVED BY: DRW7318 DATE/TIME: 12/14/99 10:13	TERMINATED BY: DATE/TIME:

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 of Room 108 (U-1Decay Heat Removal / Seal Return) plan view 1. Describe the type of posting this area should have to warn the radiation worker and why? 2. Identify where and why you should stand while waiting for further direction from the Control Room during a work evaluation.
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 with 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:

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

STEP 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:	___ 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: **Note:** "*" Indicates critical step.	*** CRITICAL STEP** ___ SAT ___ UNSAT

START TIME: _______

STEP 1: Obtain a copy of the appropriate procedure. STANDARD: Operator obtains a copy RP/0/B/1000/15A, Offsite Communications From The Control Room, from the Emergency Procedures Cart. COMMENTS: Emergency Procedures Cart is located in TSC or Unit 3 CR document library in plant. Located in OSM office area in simulator.	___ SAT ___ UNSAT
STEP 2: Obtain portable phone. STANDARD: Portable phone is picked up. COMMENTS: Phone is located on column in Control Room / Simulator.	___ SAT ___ UNSAT
STEP 3: Obtain Yellow Folder from Emergency Procedures Cart. STANDARD: Yellow Folder is obtained from Emergency Procedures Cart. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: Obtain Emergency Action Level Guideline from Emergency Procedures Cart. STANDARD: Emergency Action Level Guidelines is obtained from Emergency Procedures Cart. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

<table>
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STEP 8: Complete Lines 5-10

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: Unidentified RCS leakage is greater than or equal to 10 gpm (Emergency Action Level Description Guidelines 4.2.U.1.a (line 7))***
- ***Plant Condition: "B" STABLE (line 8)***
- ***Reactor Status: 100% power (line 9)***
- ***Emergency Releases: NONE (line 10)***

STANDARD: Lines completed correctly, refer to completed form.

COMMENTS:

Note: Examiner – Do NOT provide line numbers to candidate as part of cue.

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

</td></tr>
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STEP 9: Complete Lines 11-14

STANDARD: Write "Not Applicable" across Lines 11-14 refer to completed form.

COMMENTS:

</td><td>

___ SAT

___ UNSAT

</td></tr>
</table>

STEP 10: Complete Lines 15 ***CUE: When asked, state No Recommended Protective Actions*** 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 state this form has been signed and dated and timed 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: **Note: Faxing only allows offsite agencies to enhance communications therefore this step is NOT critical.**	___ SAT ___ UNSAT
***** SIMULATE ONLY ***** STEP 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 (after the candidate dials *4) 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 ***Cue: Pickens County LEC requests authentication - Code number 60.*** STANDARD: When requested provide authentication by using Authentication Code List. Provide code word "Payload" and record on Line 4. COMMENTS: **NOTE: Examiner Terminate the JPM at this point.** **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

TIME STOP: ________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
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

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

CRO-40A/ADMIN A.1

CALCULATE SDM
WITH A DROPPED CONTROL ROD

CANDIDATE ____

EXAMINER ____

SRO only

**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, has been completed 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:	___ SAT ___ UNSAT
STEP 2 Determine proper enclosure to use. STANDARD: Enclosure 13.20, Shutdown Margin at Power, is chosen. COMMENTS:	___ 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

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
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

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 through step 5.5.4.

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 OVERRIDE pushbutton is located on the Diamond Control panel on 1UB1 and depressed. "SEQ OVERRIDE" 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 and hold the 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

<u>STEP 9</u>: 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. <u>STANDARD</u>: Press selector for the LATCH switch and insert rod for 15 seconds. 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. <u>COMMENTS</u>:	___ 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 6, Group 2, drops 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 through step 5.5.4.

INITIATING CUES:

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

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

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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
The "B" GWD Compressor is tagged out for maintenance.

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: Operate Group 1, 2, 3, and/or 4 as necessary to control PZR temperature increase < 90°F/hr (1.5°F/min). 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 valves are properly positioned:

1N-119 (SG PZR Supply Blk) Closed
Cue: 1N-119 is Closed.

1N-161 (PZR Supply Blk) Closed
Cue: 1N-161 is Closed.

1GWD-146 (PZR Vent to RV Line) Open
Cue: 1N-146 is Open.

Determine if SG Blanketing is required
Cue: SG blanketing is not required

1N-107 (LP Nitrogen Htr Byp) Locked Closed
Cue: 1N-107 is Closed.

1N-110 (Rx Bld Low Press Hdr Supply) Closed
Cue: 1N-110 is Closed.

___ SAT

___ UNSAT

STANDARD:

The following valves are in their proper position:

1N-119 (SG PZR Supply Blk) Closed

1N-161 (PZR Supply Blk) Closed

1GWD-146 (PZR Vent to RV Line) Open

1N-107 (LP Nitrogen Htr Byp) Locked Closed

1N-110 (Rx Bld Low Press Hdr Supply) Closed

COMMENTS:

STEP 6: ENSURE 1GWD-12 (QT Vent inside RB) open STANDARD: Locate 1GWD-12 (QT Vent inside RB) switch and select OPEN. Green OPEN light is used to determine 1GWD-12 is open. COMMENTS: [redacted]	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 7: [redacted]WD-13 (QT Vent outside RB) open STANDARD: RED Locate 1GWD-13 (QT Vent outside RB) switch and select OPEN. Green OPEN light is used to determine 1GWD-13 is open. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 8: Start a second GWD Compressor STANDARD: Verify the second GWD Compressor is operating NOTE: The second GWD compressor is operating COMMENTS:	___ SAT ___ UNSAT

<table>
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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 between 38-45 psig while venting the PZR by throttling open GWD-17. (If QT pressure high alarm (5 psig) (1SA-6/B7) is actuated then the operator closes GWD-17.)

NOTE: When the Examiner has determined that the candidate can control RCS, Quench Tank, and Vent Header pressure he should request that the simulator be placed into freeze and new PZR conditions established.

Freeze simulator at Examiners request.

Recall SNAP 104

COMMENTS:

Critical Parameters:
Below 38 psig the potential to drain the loops exists.
Above 50 psig risk damage to QT.

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

</td></tr>
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Cue: New PZR conditions: all N2 has been vented, GWD-13 and 17 have been open for 35 minutes (Step 2.15)

STEP 10: Closed 1GWD-13 (Quench Tank Vent Outside RB)

STANDARD: 1GWD-13 is closed

COMMENTS:

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

</td></tr>
</table>

		CRITICAL STEP
STEP 11:	Ensure QT level increases with minimal pressure change.	
STANDARD:	Candidate monitors QT pressure, level, and temperature to ensure a steam bubble is established in the PZR. Candidate determines that the N2 has been purged by the following indications: QT pressure remains constant (< 5 psig) __________ QT temperature increase (< 180°F) __________ QT level increase (<90 inches) __________	___ SAT ___ UNSAT
COMMENTS:		
STEP 12:	When PZR venting is complete, ensure the following: 1GWD-17 (PZR Vent) closed 1GWD-13 (QT Vent Outside RB) closed 1GWD-12 (QT Vent inside RB) closed 1GWD-146 (PZR Vent to RV Line) closed ***NOTE: Manual valve located in RB*** ***CUE: 1GWD-146 is closed*** 1GWD-18 (PZR Vent and N2 Isolation) closed ***NOTE: Manual valve located in RB*** ***CUE: 1GWD-18 is closed***	___ SAT ___ UNSAT
STANDARD:	1GWD-17 (PZR Vent) is closed 1GWD-13 (QT Vent Outside RB) is closed 1GWD-12 (QT Vent inside RB) is closed	
COMMENTS:		
	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
The "B" GWD Compressor is tagged out for maintenance.

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.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM NRC-002/SIM
RC-66 (PZR PORV) Stroke 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

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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:

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
All initial conditions have been met
No personnel is in the RB at this time

INITIATING CUES:

The SRO in the Control Room directs you to perform PORV Operability Test (PT/0/A/0201/004) Enclosure 13.1.

START TIME: _______

	CRITICAL STEP*
STEP 1: Cycle 1RC-4 (PORV Block) STANDARD: 1RC-4 (PORV Block) is positioned to close and verified to be closed by the green light on 1RC-4 (PORV Block) is positioned to open and verified to be open by the red light on* ***Reopening 1RC-4 (PORV Block) is critical.** COMMENTS:	___ SAT ___ UNSAT
STEP 2: 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 3: Open 1RC-66 (PORV) STANDARD: RC-66 switch positioned to OPEN The OPEN PERMIT pushbutton is depressed COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 4: 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) Select LOW position on the 1RC-66 select switch COMMENTS:	___ SAT ___ UNSAT

	CRITICAL STEP
STEP 5: Identify failure of 1RC-66 (PORV) to close. Close 1RC-4 (1RC-66 Block) STANDARD: Diagnose 1RC-66 failed to close Then close 1RC-4 (1RC-66 Block) COMMENTS: END TASK	___ SAT ___ UNSAT

STOP TIME: ____________

CRITICAL STEP EXPLANATIONS:

STEP #	Explanation
1	Reopening 1RC-4 establishes a flowpath from the PZR to QT.
3	This step opens 1RC-66.
5	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

INITIAL CONDITIONS:

You are the Unit 1 OATC
Unit 1 startup is in progress
Establishing a PZR bubble is in progress
All initial conditions have been met
No personnel is in the RB at this time

INITIATING CUES:

The SRO in the Control Room directs you to perform PORV Operability Test (PT/0/A/0201/004) Enclosure 13.1.

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

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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:

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 through step 7.0. 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

STEP 5: Dispatch an operator with a safety harness to 1C-573 (MD Suction to UST) to standby until further notice. STANDARD: Dispatches an operator with a safety harness to 1C-573 (MD Suction to UST) ***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, MS to CSAE's Closes 1AS-40, CSAE Aux Steam Supply 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 the following valves as necessary to maintain < 440 gpm/pump. 1FDW-315 (1A SG EFDW Control) and 1FDW-316 (1B SG EFDW Control) 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: ***NOTE: This step is Critical only if UST is less than 1 foot.**	***CRITICAL STEP**

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

STEP 11: Open 1C-391 (TD EFDWP Suction from Hotwell) STANDARD: Open 1C-391(TD EFDWP Suction from Hotwell) 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:

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 through step 7.0. 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.

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

JPM NRC-006/SIM

Restore RBCUs to normal following an ES Channel 5 actuation per RULE #7.

CANDIDATE ______________________________

EXAMINER ______________________________

REGION II
INITIAL LICENSE EXAMINATION
JOB PERFORMANCE MEASURE

Task:

Restore RBCUs to normal after an ES actuation.

Alternate Path:

N/A

Facility JPM #:

N/A

K/A Rating(s):

022 A4.01 (3.6 / 3.6)

Task Standard:

The 1A RBCU is placed in LOW speed.

Preferred Evaluation Location: Simulator __X__ In-Plant ______

Preferred Evaluation Method: Perform __X__ Simulate ______

References:

EOP RULE #7

Validation Time: 5 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:

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:

EOP RULE #7

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 actuation of ES Channels 1-6 has occurred on Unit 1.

EOP RULE #7 has been completed satisfactorily through step #11

INITIATING CUES:

The SRO in the Control Room directs you to complete RULE #7.

START TIME: _______

STEP 1: If RCS pressure is > 550 psig THEN OPEN the following valves: • 1CC-7 (CC Return Inside RB) • 1CC-8 (CC Return Outside RB) • 1LPSW- 15 (Unit 1 LPSW RCP Cooler Outlet) • 1LPSW-6 (Unit 1 RCP LPSW Supply) AND Verify start of the 1A or 1B CC Pump STANDARD: Determine RCS pressure is ≥ 550 psig From the RZ Module, take MANUAL control of the following valves on Channels 5 and 6: • 1CC-7 (CC Return Inside RB) is opened • 1CC-8 (CC Return Outside RB) is opened • 1LPSW- 15 (Unit 1 LPSW RCP Cooler Outlet) is opened • 1LPSW-6 (Unit 1 RCP LPSW Supply) is opened AND 1A or 1B CC Pump is verified started COMMENTS: **NOTE: Valves must be operated from both ES Channels 5 and 6**	___ SAT ___ UNSAT

<table>
<tr><td>

<u>STEP 2</u>:

Check RCS pressure is NOT ≤ 550 psig

<u>STANDARD</u>:

Determine that RCS pressure is NOT ≤ 550 psig

And, that Step 13 is N/A.

<u>COMMENTS</u>:

</td><td>

___ SAT

___ UNSAT

</td></tr>
<tr><td>

<u>STEP 3</u>:

IF RB Pressure > 3 psig

THEN verify all BLUE auto lights on and all WHITE position lights on.

<u>STANDARD</u>:

Determine that RB pressure is > 3 psig

Verify all components on ES Channel 5 and 6 are BLUE and WHITE.

Diagnose that the following indication is not correct

- 1A RBCU (Channel 5) does not have a WHITE position light

<u>COMMENTS</u>:

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

</td></tr>
<tr><td>

<u>STEP 4</u>:

Notify the Control Room crew to perform Enclosure 7.5 .

<u>STANDARD</u>:

The Control Room crew is notified to perform Enclosure 7.5.

<u>COMMENTS</u>:

</td><td>

___ SAT

___ UNSAT

</td></tr>
</table>

STEP 5: Dispatch an operator to perform Enclosure 7.13, Penetration Room Ventilation ES Verification. STANDARD: An operator is dispatched to perform Enclosure 7.13, Penetration Room Ventilation ES Verification. COMMENTS:	___ SAT ___ UNSAT
STEP 6: Place the following ES components in manual: • 1A HPI Pump (ES Channel 1) • 1HP-24 (1A HPI BWST SUCTION) (ES Channel 1) • 1B HPI Pump (ES Channel 1) • 1C HPI Pump (ES Channel 2) • 1HP-25 (1B HPI BWST SUCTION) (ES Channel 2) • 1B HPI Pump (ES Channel 2) STANDARD: The following ES components are placed in manual: • 1A HPI Pump (ES Channel 1) • 1HP-24 (1A HPI BWST SUCTION) (ES Channel 1) • 1B HPI Pump (ES Channel 1) • 1C HPI Pump (ES Channel 2) • 1HP-25 (1B HPI BWST SUCTION) (ES Channel 2) • 1B HPI Pump (ES Channel 2) COMMENTS:	___ SAT ___ UNSAT

STEP 7: Ensure ES channels 1 & 2 components that can be operated from the Control Room are in their desired position. STANDARD: ES channels 1 & 2 components that can be operated from the Control Room are verified to be in their desired position by the white lights indicating on Channel 1 and 2 at the RZ modules. COMMENTS:	___ SAT ___ UNSAT
STEP 8: Notify Chemistry to prepare for caustic addition per CP/1&2/A/2002/005, (Post Accident Caustic Injection Into LPI System). STANDARD: Chemistry has been notified to prepare for caustic addition per CP/1&2/A/2002/005, (Post Accident Caustic Injection Into LPI System). COMMENTS:	___ SAT ___ UNSAT
STEP 9: When ES Channels 3 & 4 have actuated ensure that ES Channel 3 & 4 components are in their desired ES position. STANDARD: Determine that ES Channels 3 & 4 have actuated and that ES Channel 3 & 4 components are in their desired ES position. COMMENTS:	___ SAT ___ UNSAT

STEP 10: Ensure that ES Channel 3 & 4 components are in their desired ES position. STANDARD: The following ES components are PLACED in Manual: 1A LPI Pump 1LP-21 (1A LPI BWST SUCTION) (ES Channel 3) 1B LPI Pump 1LP-22 (1B LPI BWST SUCTION) (ES Channel 4) COMMENTS:	___ SAT ___ UNSAT
STEP 11: Determine that ES Channels 5 & 6 have actuated. STANDARD: Verify that ES Channels 5 & 6 have actuated. COMMENTS:	___ SAT ___ UNSAT
STEP 12: Manually place the 1A RBCU to LOW speed to prevent RBCUs from running in mixed speed. STANDARD: At the RZ Module the 1A RBCU is placed in MANUAL At 1AB3 1A RBCU switch is place in the LOW speed position ***Cue: This JPM is complete.*** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

END TASK

TIME STOP: ________

CRITICAL STEP EXPLANTIONS:

STEP #	Explanation
3	This step is required, because components must be placed in MANUAL to be able to reposition them.
12	This step is required to place the RBCU in the ES position (LOW), and prevents RBCUs operating in a mixed speed configuration.

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

INITIAL CONDITIONS:

You are the Unit 1 BOP

An actuation of ES Channels 1-6 has occurred on Unit 1.

EOP RULE #7 has been completed satisfactorily through step #11

INITIATING CUES:

The SRO in the Control Room directs you to complete RULE #7.

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

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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:

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 through step 2.1.3.

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

<u>STEP 3</u>: 2.1.6 Energize the STBY BUSES from CT-5. <u>STANDARD</u>: 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.** <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT
<u>STEP 4</u>: 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 <u>STANDARD</u>: 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 <u>COMMENTS</u>: **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 through step 2.1.3.

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.

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

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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:

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 3 is at 100% power

3RIA-57 has just been returned to service following detector replacement.

INITIATING CUES:

The Control Room Supervisor directs you to check the operation of 3RIA-57 from the individual monitor per PT/0/A/0230/001.

START TIME: ________

STEP 1: VERIFY alarm setpoints. STANDARD: • Locate 3RIA-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

*****Italicized Cues Are To Be Used Only If JPM Performance Is Being Simulated.***

STEP 3*:	CRITICAL STEP
Perform Check source. • Depress "C/S" button. ***CUE: C/S depressed*** • Check readings between 5.0 E-1 and 1.0 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 3RIA-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… • Refers to Operability Criteria for RIA-57 Enclosure 13.3 and determines that RIA-57 is NOT operable. COMMENTS:	____ 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 ITS section. ***CUE: RIA-57 listed on ITS 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 ITS.*** ***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 RO and SRO** **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)

INITIAL CONDITIONS:

Unit 3 is at 100% power

3RIA-57 has just been returned to service following detector replacement.

INITIATING CUES:

The Control Room Supervisor directs you to check the operation of 3RIA-57 from the individual monitor per PT/0/A/0230/001.

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

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COMMENTS

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____

____

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 through 1.2.4

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 2 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. ***Cue: Two red lights are on.*** STANDARD: "AUX. SER. WTR. SWGR. 4160V FDR B1T-UNIT 10" breaker indicates closed on the ASW SWGR 600V LOAD CENTER. COMMENTS:	___ SAT ___ UNSAT
STEP 2: Ensure closed the "AUX. SER. WTR. SWGR TRANSFORMER" breaker **Located: ASW SWGR 600V LC Unit 5** ***CUE: Inform the student that the red light is ON and that the green light is off at the control switch for Aux. Ser. Wtr. Swgr. Xfrmr. Bkr.*** STANDARD: Student verifies the "AUX. SER. WTR SWGR TRANSFORMER" breaker is Closed. COMMENTS:	___ SAT ___ UNSAT

	CRITICAL STEP
STEP 3: Rack in "AUXILIARY SERVICE WATER PUMP" breaker at the ASW SWGR 600V LOAD CENTER Unit 6. ***CUE: After breaker is racked in, inform student that the AUX SERVICE WATER PUMP MOTOR breaker green indicating light is ON.*** **STANDARD:** Student opens shutter, inserts 600v breaker rackout tool, and turns tool clockwise to rack breaker in. **COMMENTS:** Student is expected to follow (simulate) all applicable safe electrical work practices.	___ SAT ___ UNSAT

STEP 4: CLOSE CCW-309 (ASWP Disch Drain). ***Cue: Indicate that CCW-309 is closed.*** STANDARD: CCW-309 (ASWP Disch Drain) is manually CLOSED. COMMENTS:	___ SAT ___ UNSAT
STEP 5 OPEN CCW-99 (ASWP Suction). ***Cue: Indicate that CCW-99 is open.*** STANDARD: CCW-99 (ASWP Suction) is manually opened. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 6 OPEN CCW-101 (ASWP Disch). ***Cue: Indicate that CCW-101 is open.*** STANDARD: CCW-101 (ASWP Disch) is manually opened. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 7 OPEN CCW-247 (ASWP Recirc). ***Cue: Indicate that CCW-247 is open.*** 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). ***Cue: Indicate that CCW-308 is throttled open and water is issuing from vent, then indicate that CCW-308 is closed.*** 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. **CUE: After switch is rotated, inform student that the AUX SERVICE WATER PUMP MOTOR breaker RED indicating light is ON.** STANDARD: Student locates AUX SERVICE WATER PUMP MOTOR control switch and rotates switch to the CLOSE position. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 10: VERIFY adequate HPIP motor cooler flow indication locally (>1 gpm) by local flow indication. (Meter # 2LPSP1013) **Location: Aux. Bldg. 1st – HPI Pump Room** STANDARD: Student verifies flow located AB-1st HPI pump Room. COMMENTS: **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.

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: 001345501

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: ______________________________ 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

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 1 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 1 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 Open Sw #1 (left switch) ***Cue: Indicate that Sw #1 is positioned to OPEN,*** Open Sw #3 (right switch) ***Cue: Indicate that Sw #3 is positioned to OPEN,*** Close Sw #2 (center switch) ***Cue: Indicate that Sw #2 is positioned to CLOSED.*** STANDARD: Inverter is MANUALLY BYPASSED by positioning the following switches in sequence: Sw #1 (left switch) is OPENED And Sw #3 (right switch) is OPENED Then Sw #2 (center switch) is CLOSED COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

<table>
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<u>STEP 3</u>:

OPEN KU Inverter Bypass Switch cabinet door.

<u>STANDARD</u>:
Student locates the KU Inverter Bypass Switch cabinet and opens cabinet door. (Located in Equipment Room)

<u>COMMENTS</u>:
</td><td>
CRITICAL STEP

___ SAT

___ UNSAT
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<tr><td>
<u>STEP 4</u>:

MANUALLY BYPASS KU Inverter

Open Sw #1 (left switch)

Cue: Indicate that Sw #1 is positioned to OPEN,

Open Sw #3 (right switch)

Cue: Indicate that Sw #3 is positioned to OPEN,

Close Sw #2 (center switch)

Cue: Indicate that Sw #2 is positioned to CLOSED.

<u>STANDARD</u>:
Inverter is MANUALLY BYPASSED by positioning the following switches in sequence:

Sw #1 (left switch) is OPENED

<u>And</u>

Sw #3 (right switch) is OPENED

<u>Then</u>

Sw #2 (center switch) is CLOSED

<u>COMMENTS</u>:
</td><td>
CRITICAL STEP

___ SAT

___ UNSAT
</td></tr>
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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. Reset breaker KRA #13 (175A 2P, Power Panelboard 1KU) ***Cue: Breaker #13 (175A 2P, Power Panelboard 1KU)) is RESET.*** Reset breaker #21 (30A 1P, ICS/NNI Hand Power) ***Cue: Breaker #21 (30A 1P, ICS/NNI Hand Power) is RESET.*** STANDARD: Breaker KRA #13 (175A 2P, Power Panelboard 1KU) and Breaker #21 (30A 1P, ICS/NNI Hand Power) have been reset. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 7: Notify the Control Room ***Cue: BOTH ICS AUTO and HAND power has been restored.*** STANDARD: Control Room has been notified.	

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.

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

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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:

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 "A" Desiccant Air Filter Outlet). (TB5 L-39) ***Cue: Indicate that IA-2730 is CLOSED.*** 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. COMMENTS: **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.**	___ SAT ___ UNSAT
STEP 2: Position the following valves: Close IA-2731 (Primary IA "B" Desiccant Air Filter Outlet). (TB5 L-39) ***Cue: Indicate that IA-2731 is CLOSED.*** 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. COMMENTS: **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.**	___ SAT ___ UNSAT

STEP 3: At the Primary IA Dryer "A" Control Panel, position the (ON/OFF) switch to OFF. ***Cue: Indicate that the Primary IA Dryer "A" Control Panel, switch is positioned to OFF.*** STANDARD: On the "A" Dryer control panels the student REMOVES the Primary IA Dryers from service by rotating the following switch, 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. ***Cue: Indicate that the Primary IA Dryer "B" Control Panel, switch is positioned to OFF.*** STANDARD: On the "B" Dryer control panels the student REMOVES the Primary IA Dryers from service by rotating the following switch, 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) ***Cue: Indicate that IA-2735 is CLOSED.*** 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. COMMENTS: **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.**	___ SAT ___ UNSAT
STEP 6: Position the following valves: Close IA-2736 (Primary Air Filter "B" Outlet). (TB5 L-39) ***Cue: Indicate that IA-2736 is CLOSED.*** 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. COMMENTS: **NOTE: The valve is located on the Turbine floor between the Primary IA Compressor Dryer Complexes.**	___ SAT ___ UNSAT

STEP 7: Open HPSW-771 (Primary IA Comp. Disc. Block) (TB5 M-39) ***Cue: Indicate that HPSW-771 is OPEN.*** STANDARD: The student LOCATES and OPENS HPSW-771 (Primary IA Compressor Cooling Discharge Block) by rotating the switch to the "Open" position. COMMENTS: **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.**	**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. **Cue: Using a pointing device, indicate to the student the following readings:** - **OHPS-PG-0823 = 64 psig.** STANDARD: The student VERIFIES adequate cooling water flow by monitoring the following gauges: - OHPS-PG-0823 (Primary IA Comp. Cooling Water Inlet Pressure). COMMENTS:	___ SAT ___ UNSAT

STEP 9: Verify HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) in Locked Open Position. ***Cue: Indicate that HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) in Locked Open Position*** STANDARD: HPSW-767 (Primary IA Comp. Disch. Control) (TB5 M-39) is verified in the Locked Open Position. COMMENTS:	___ SAT ___ UNSAT
STEP 10: Depress (RESET/LAMP TEST) pushbutton. • Verify all alarm indicators light. ***Cue: While RESET/LAMP TEST pushbutton is depressed, inform student that all alarm indicators are lit.*** • Release (RESET/LAMP TEST) pushbutton and verify all alarm indicator lamps extinguish. ***Cue: When RESET/LAMP TEST pushbutton is released, inform student that all alarm indicator lamps extinguish.*** STANDARD: The candidate tests the alarm indicators by depressing the black RESET/LAMP TEST pushbutton on the compressor control panel located on the north side of the compressor housing. COMMENTS:	___ SAT ___ UNSAT

STEP 11: • Depress (START) Primary IA Compressor pushbutton. ***Cue: Indicate that the green "Machine Run" light has illuminated.*** • Start the Primary Air Compressor by depressing the "Start" pushbutton on the control panel located on the north side housing of the compressor. 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. COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 12:
 	
Verify 0HPS-PG-0824 (Primary IA Compressor Cooling Water Outlet Pressure) is with in the range specified (in procedure).
 	
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.
 	
Cue: When HPSW-767 is throttled closed, indicate with the pointing device that flow is 98 gpm and outlet pressure is 18 psig
 	
STANDARD:
 	
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).
 	
COMMENTS:
	CRITICAL STEP
	
	___ SAT
	
	___ UNSAT

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STEP 13:

VERIFY selected Enclosure fan is running.

Cue: Inform the student that the selected Enclosure Fan is running properly.

Verify all door panels are installed on the Primary IA Compressor Enclosure.

Cue: Inform student that all door panels are installed on enclosure.

STANDARD:

The student determines that selected enclosure fan is operating and all door panels located on the compressor enclosure are installed.

COMMENTS:

</td><td>

___ SAT

___ UNSAT

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STEP 14:

Throttle open IA-2735 (Primary Air Filter "A" Outlet) or IA-2736 (Primary Air Filter "B" Outlet) (TB5 L-39) and SLOWLY pressurize the Dryer tanks to system pressure (100-110 psig).

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.

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)

COMMENTS:

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CRITICAL STEP

___ SAT

___ UNSAT

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STEP 15: At the Primary IA Dryer A panel, position the (ON/OFF) switch to ON. ***Cue: Indicate that Primary IA Dryer A panel is positioned 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 16: At the Primary IA Dryer B panel, position the (ON/OFF) switch to ON. ***Cue: Indicate that Primary IA Dryer B panel is positioned 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 17: CONNECT the Primary IA Compressor to the IA Header. Open IA-2735 (Primary Air Filter A Outlet). (TB5 L-39) ***Cue: Indicate that IA-2735 is OPEN*** 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) COMMENTS: **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.**	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 18: CONNECT the Primary IA Compressor to the IA Header. Open IA-2736 (Primary Air Filter B Outlet). (TB5 L-39) ***Cue: Indicate that IA-2736 is OPEN*** 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) COMMENTS: **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.**	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 19: CONNECT the Primary IA Compressor to the IA Header. ***Cue: Indicate that the valves are fully open.*** 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) COMMENTS: **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.**	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 20: CONNECT the Primary IA Compressor to the IA Header. Slowly open IA 2731 (Primary Desiccant Air Filter B Outlet). (TB5 L-39) ***Cue: Indicate that the valves are fully open.*** 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) COMMENTS: **NOTE: The valve is fully open when the position indicator arrows are parallel to the piping.**	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 21: As system pressure increases check for air leaks on the Primary IA Compressor, Air Dryers, and Air Filters. ***Cue: No air leaks are found.*** **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. 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.

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

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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:

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: _______

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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:

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___ SAT

___ UNSAT

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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:	___ 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: STANDARD: Candidate determines BWST Level is ≤ 9 feet (decreasing) **IF BWST level is not <9 feet then,** **CUE: BWST is < 9 feet.** COMMENTS:	___ SAT ___ UNSAT

<u>STEP 6</u>: Simultaneously open the following valves 1LP-19 ('1A' RX. BLDG. SUCTION) 1LP-20 ('1B' RX. BLDG. SUCTION) <u>STANDARD</u>: The student locates the control switch for the following valves on 1UB2. Rotates the switches in the OPEN direction. Verify Green CLOSED light ON, Red OPEN light OFF. 1LP-19 ('1A' RX. BLDG. SUCTION) Rotates the switches in the OPEN direction. Verify Green CLOSED light ON, Red OPEN light OFF. AND 1LP20 ('1B' RX. BLDG. SUCTION) Rotates the switches in the OPEN direction. Verify Green CLOSED light ON, Red OPEN light remains 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 <u>all</u> attempts to open 1LP-20 are unsuccessful.** <u>COMMENTS</u>:	**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 fails to OPEN 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

<u>STEP 10</u>: <u>IF</u> LP-20 fails to open IF BWST level is ≤ 6 feet, <u>THEN</u> 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** <u>STANDARD</u>: 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 <u>COMMENTS</u>: **NOTE: RZ Module 5 and 7 for 1LP-21 must be in MANUAL to operate 1LP-21.**	**CRITICAL STEP** ___ SAT ___ UNSAT
<u>STEP 11</u>: Stop the following pumps 1B LPI pump 1B RBS Pump <u>STANDARD</u>: 1B LPI pump is stopped 1B RBS Pump is stopped <u>COMMENTS</u>:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 12: 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 throttle 1LP-12 (1A LPI COOLER OUTLET) to maximize in each LPI header flow ≤1100 gpm. STANDARD: Candidate determines LPI Cooler outlet flow has NOT been locally throttled COMMENTS: **NOTE: LP-12 and LP-14 must be throttled simultaneously to achieve balanced flow.**	___ SAT ___ UNSAT
STEP 13: GO TO step 7 of Enclosure 7.11 STANDARD: Transitions to step 7 Enclosure 7.11 COMMENTS:	___ SAT ___ UNSAT

STEP 14: 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) STANDARD: Verification of ≈1000 gpm flow is indicated in the 1A RB Spray header **NOTE: "A" and "B" RBS flow was throttled in Step 4.** **AND** **"B" RBS Pump was secured in Step 11** COMMENTS:	___ 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: **NLO is dispatched to close 1LP-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 OPEN position 1LPSW-252 FAIL SWITCH in FAIL OPEN 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 that ALL three (3) LPSW pumps are operating. COMMENTS:	___ SAT ___ UNSAT

STEP 20: Open the following valves _____ 1LPSW-4 (1ALPI CLR SHELL OUTLET) _____ 1LPSW-5 (1BLPI CLR SHELL OUTLET) STANDARD: 1LPSW-4 (1ALPI CLR SHELL OUTLET) is opened 1LPSW-5 (1BLPI CLR SHELL OUTLET) is opened COMMENTS:	___ SAT ___ UNSAT
STEP 21: GO TO step 10.8 of Enclosure 7.11 STANDARD: Transitions to step 10.8 COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 22: IF only one LPI cooler is available STANDARD: Determines that BOTH LPI coolers are available. COMMENTS:	___ SAT ___ UNSAT

STEP 23: WHEN 1LP-28 (BWST Outlet) is closed THEN perform the following . . . ***CUE: 1LP-28 was closed in step 16.*** STANDARD: **Candidate determines that steps 11.1 and 11.2 quoted below are N/A.** 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. 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
4	Decreases RBS flow to prevent pump runout when suction is swapped 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

FINAL SUBMITTAL

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

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

NUREG-1021 - ES-501

FINAL AS GIVEN
OPERATOR ACTIONS

F.1.g - FORM ES-D-2
OPERATOR ACTIONS

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 / BOP	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

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	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
	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:

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 / A12: • Determine cause of alarm. • Refer to ARG
	BOP / SRO	Determine that "1B" CFT level is low and decreasing slowly. Refer to OP/1104/01, Core Flooding System, Enclosure 4.10, Makeup to the CFT's Using the HP Boric Acid Pump and DW Initial Conditions: 1. Boric acid supply available from the BAMT 2. DW header in service 3. Determine amount of DW and BAMT necessary for addition to maintain CFT between 2700 ppm and 3750 ppm per Enclosure Boron Concentration Calculation: 1B CFT = 2556 ppm BAMT = 15850 ppm 1B CFT = 5 gals/.01ft (per OP/1108/01, Encl 3.26) (1ft=500gals) CfVf = C1V1 + C2V2 2700 (6600) = 2556 (6350) + 15850 (X) 17820000 = 16230600 + 15850X 17820000 – 16230600/15850 = X ≈100 gals = (BAMT addition) and ≈150 gals of DW After volume is calculated then CFT leakage will increase to > makeup capacity

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 / SRO	Diagnose that "1B" CFT water level is decreasing at a rate > makeup capability. The probable source of leakage into the RB is the CFT. (No RB RIA in alarm). • RBNS level high 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 indication with Loop B Th 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.

Op-Test No.: 01	Scenario No.: 01	Event No.: 5, 6, 7

Event Description:
Turbine Header Pressure transmitter FAILS HIGH.
"1B" TBV FAIL 50% OPEN.
1MS-26 ("B" TBV Block) FAILED OPEN (Breaker failure).
(Steam 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

Op-Test No.:	01	Scenario No.:	01	Event No.:	5, 6, 7

Event Description:
Turbine Header Pressure transmitter FAILS HIGH.
"1B" TBV FAIL 50% OPEN.
1MS-26 ("B" TBV Block) FAILED OPEN (Breaker failure).
(Steam 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 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.
	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 • Subsequent Actions • Section 503, Extended EFDW Operation (SEE attached procedure section for BOP actions)

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

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	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 • 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.

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	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 • Maintain Tave ≥ 532°F • Dispatch operators to operate the Atmospheric Dump Valves • Dispatch operators to the SSF
	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)
BOP - Refer to ARG
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)

NOTE: The leak will start small and after the crew determines the proper makeup procedure durection then the leak will increase to > makeup capacity.

"1B" CFT level and pressure decreases.
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 return 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 FDWP 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

TIME=1-15 minutes TOTAL 70-75 min.

E-Plan Classification:

SRO Admin A.4

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?
ANSWER – upgrade to SAE based on Fission Product Matrix total = 7 (HPI Cooling (4)+ SGTL > 10 gpm with direct opening to the environment (3).

Unit Status Assessment

1. Ensure the following breakers are closed:

	_____ "1TC INCOMING FDR BUS 1"
1TC	_____ "1TC INCOMING FDR BUS 2"
	_____ "1TD INCOMING FDR BUS 1"
1TD	_____ "1TD INCOMING FDR BUS 2"
	_____ "1TE INCOMING FDR BUS 1"
1TE	_____ "1TE INCOMING FDR BUS 2".

_____ 2. **IF** NONE of the 4160v Switchgear (1TC, 1TD, or 1TE) are energized,

THEN continue efforts to regain power to the 4160v Switchgear,

AND **GO TO** Section 504, Blackout. {1}

Unit Status Assessment

3. Verify the following Statalarms Off:

_____ 1X5 "EL 600V LC 1X5 POWER FAILURE" (1SA-04/ A-5)

_____ 1X6 "EL 600V LC 1X6 POWER FAILURE" (1SA-04/ A-6)

_____ 1X7 "EL 600V LC 1X7 POWER FAILURE" (1SA-04/ A-7)

_____ 1X8 "EL 600V LC 1X8 POWER FAILURE" (1SA-04/ A-8)

_____ 1X9 "EL 600V LC 1X9 POWER FAILURE" (1SA-04/ A-9)

_____ 1X10 "EL 600V LC 1X10 POWER FAILURE" (1SA-04/ C-3)

_____ 1XS1, 1XS2, 1XS3 "EL ENG SAFEGUARDS MOTOR CONTROL CTR TROUBLE" (1SA-04/ D-6)

_____ 1CA Control Battery Charger

"EL DC SYSTEM 1CA TROUBLE" (1SA-05/ A-11)

_____ 1CB Control Battery Charger

"EL DC SYSTEM 1CB TROUBLE" (1SA-06/ A-1).

_____ 4. Ensure Pressurizer Heaters in "AUTO".

_____ 5. **IF** a 230KV Switchyard Isolation has occurred as indicated by either of the following Statalarms being on,

_____ "EL SWYD ISOLATION CONFIRMED CHNL A LOGIC" (1SA-15/ E-6)

_____ EL SWYD ISOLATION CONFIRMED CHNL B LOGIC" (1SA-14/ E-6),

AND any Oconee Unit is receiving power from its Normal Source (1T, 2T, 3T),

THEN perform the following:

CAUTION 5.1:	Any Oconee Unit receiving power from its Normal Source (1T, 2T, 3T) could rapid transfer (out of sync) from the Normal to the Startup source unless the "AUTO/MAN" transfer switches are in "MANUAL" for both the 6900v and 4160v Switchgear.

_____ 5.1 Place the following transfer switches in "MANUAL" for those Units receiving power from its Normal source (1T, 2T, 3T):

	Unit 1	Unit 2	Unit 3
"MFB1 AUTO/MAN"	_____	_____	_____
"MFB2 AUTO/MAN"	_____	_____	_____
"TA AUTO/MAN"	_____	_____	_____
"TB AUTO/MAN"	_____	_____	_____

NOTE 6: Continue with this procedure while awaiting feedback from the Operator.

_____ 6. **IF** CT-1 is energizing Main Feeder Bus(s),

AND only one Startup Feeder Breaker is closed:

- "$E1_1$ MFB1 STARTUP FDR"
- "$E2_1$ MFB2 STARTUP FDR",

THEN dispatch an Operator to verify NO targets actuated on the open Startup Feeder Breaker.

_____ 6.1 **IF** NO targets are found on the open Startup Feeder Breaker,

THEN close the open Startup Feeder Breaker:

_____ "$E1_1$ MFB1 STARTUP FDR"

_____ "$E2_1$ MFB2 STARTUP FDR".

_____ 7. **IF** Load Shed was initiated (statalarms on):

- "EL LOAD SHED CHNL A LOGIC INITIATE" (1SA-15/ D-4)
- "EL LOAD SHED CHNL B LOGIC INITIATE" (1SA-14/ D-4)

THEN reset the Main Feeder Bus Monitor Panel Load Shed circuitry as follows:

NOTE 7.1:	If power is supplied from CT-5, load centers 1X5 AND 1X6 will Load Shed for 30 seconds when Step 7.1 is performed.

_____ 7.1 **IF** ES has occurred,

THEN press "MANUAL" on the Load Shed ES modules:

_____ "LOAD SHED & STBY BKR 1"

_____ "LOAD SHED & STBY BKR 2".

7.2 Simultaneously press "RESET" on both of the following switches, to reset the Main Feeder Bus Monitor Panel Load Shed Circuitry:

_____ "MFB UNDERVOLTAGE CHANNEL 1 RESET"

_____ "MFB UNDERVOLTAGE CHANNEL 2 RESET".

_____ 7.3 **IF** the Main Feeder Bus(s) are being powered from a Lee Gas Turbine,

THEN notify Lee Steam Station to monitor and adjust frequency as additional plant loads are added.

NOTE 8: Coordinate efforts concerning Main Vacuum Pumps and Backup Instrument Air Compressors with other affected Units.

_____ 8. Dispatch an Operator
to perform Enclosure 6.5, "Restoring Loads Outside The Control Room".

_____ 8.1 **IF** **AT ANY TIME** notification is received that the following Load Centers are energized,

THEN perform the following:

8.1.1 Verify the associated statalarms are off:

_____ 1X1 "EL 600V LC 1X1 POWER FAILURE" (1SA-04/ A-1)

_____ 1X2 "EL 600V LC 1X2 POWER FAILURE" (1SA-04/ A-2)

_____ 1X3 "EL 600V LC 1X3 POWER FAILURE" (1SA-04/ A-3)

_____ 1X4 "EL 600V LC 1X4 POWER FAILURE" (1SA-04/ A-4)

_____ SWYD FDR 'A' "SY-1 BATT. TROUBLE" (SA-05/ A-1).

_____ 8.1.2 **IF** condenser vacuum **CANNOT** be maintained,

THEN start all available Main Vacuum Pumps,

AND notify the Operator to tie into the affected Unit(s) per Enclosure 6.5, "Restoring Loads Outside The Control Room".

_____ 8.1.3 **IF** IA Header pressure < 90 psig:

- "Aux Bldg IA Hdr Press"
- "Turb Bldg IA Hdr Press",

THEN notify Operator to ensure all Backup IA Compressors are operating per Enclosure 6.5, "Restoring Loads Outside The Control Room".

CAUTION 9: Normal plant loads can overload the Auxiliary Transformer, CT-4, or CT-5.

_____ 9. **IF** Main Feeder Bus(s) is energized from CT-4 or CT-5,

OR Main Feeder Bus(s) is energized via backcharge,

THEN verify, as electrical loads are added, that the transformer supplying power is within limits.

- **REFER TO** Enclosure 6.1, "CT-4, CT-5, And Backcharge Electrical System Overload Limits".

_____ 10. Ensure LDST or BWST suction available.

_____ 11. Ensure 1A or 1B HPI Pump running.

_____ 12. Ensure 1A or 1B CC Pump operating.

_____ "CC Total Flow" > 575 gpm.

_____ 13. **IF** 1HP-21 (RCP SEAL RETURN BLOCK) was **NOT** closed,

THEN ensure RCP "Seal Inlet Hdr Flow" is ≈ 32 gpm.

13.1 Ensure open the following valves:

_____ 1HP-228 (1A1 RCP SEAL RETURN STOP)

_____ 1HP-226 (1A2 RCP SEAL RETURN STOP)

_____ 1HP-232 (1B1 RCP SEAL RETURN STOP)

_____ 1HP-230 (1B2 RCP SEAL RETURN STOP).

_____ 14. **IF** 1HP-21 (RCP SEAL RETURN BLOCK) was closed,

AND SSF RC Makeup Pump is supplying Unit 1 RCP seals,

THEN reestablish normal RCP seal injection and seal return flow by performing the following:

14.1 Ensure closed the following valves:

_____ 1HP-228 (1A1 RCP SEAL RETURN STOP)

_____ 1HP-226 (1A2 RCP SEAL RETURN STOP)

_____ 1HP-232 (1B1 RCP SEAL RETURN STOP)

_____ 1HP-230 (1B2 RCP SEAL RETURN STOP).

_____ 14.2 Open 1HP-21 (RCP SEAL RETURN BLOCK).

14.3 Open the following valves:

_____ 1HP-228 (1A1 RCP SEAL RETURN STOP)

_____ 1HP-226 (1A2 RCP SEAL RETURN STOP)

_____ 1HP-232 (1B1 RCP SEAL RETURN STOP)

_____ 1HP-230 (1B2 RCP SEAL RETURN STOP).

CAUTION 14.4:	Slowly re-establish normal RCP Seal Injection flow to prevent thermal shock of the RCP seals. Allow seal return and radial bearing temperatures to stabilize prior to RCP restart.

NOTE 14.4:	Continue with this procedure while re-establishing normal RCP Seal Injection flow.

_____ 14.4 Slowly throttle open 1HP-31 (RCP SEAL FLOW CONTROL) to establish ≈ 32 gpm RCP Seal Injection Header Flow.

_____ 14.5 Place 1HP-31 (RCP SEAL FLOW CONTROL) in "AUTO".

_____ 14.6 Notify CR SRO that SSF RC Makeup Pump is NO longer required.

_____ 15. **IF** 1HP-21 (RCP SEAL RETURN BLOCK) was closed,

AND SSF RC Makeup Pump is NOT supplying RCP seal flow,

THEN reestablish RCP seal injection and seal return flow.

- **REFER TO** AP/1/A/1700/016 (Abnormal Reactor Coolant Pump Operation) for guidance on Abnormal Seal Injection.

CAUTION 16: If the ESV System is NOT operating properly,
LPSW Pump NPSH problems may develop if forced CCW flow is NOT attained on two (2) Oconee Units following a loss of power.

If power was lost and the ESV System is NOT operating properly, consider starting CCW Pumps (Step 22) or valving in the second Condensate Cooler (Step 23), prior to restarting LPSW Pumps.

_____ 16. Ensure at least two (2) Unit 1 & 2 LPSW Pumps are in operation.

NOTE:
- ESV Pumps automatically restart two (2) minutes after power is restored to 1XS1, 1XS2, and 1XS3.
- Only one (1)ESV Pump with its associated receiver tank and header is required to maintain CCW siphon flow.

_____ 17. Ensure two (2) ESV Pumps in operation on Unit 1.

NOTE: If the computer point is NOT available, verification that an ESV Pump is operating and that ESV Tank vacuum is being maintained is sufficient to ensure proper operation of the system.

_____ 18. Monitor CCW header levels to ensure proper ESV System operation:

- "1A CCW Header Level" (O1E2712)
- "1B CCW Header Level" (O1E2717)
- "1C CCW Header Level" (O1E2728)
- "1D CCW Header Level" (O1E2734).

_____ 18.1 IF > two (2) CCW Header Levels indicate ≤ 85" and level is decreasing or NOT increasing,

THEN the ESV System is NOT functioning properly. (4)

_____ 19. Start the HPSW Jockey Pump.

20. Clear HPSW Pump Breaker Logic by performing the following:

_____ 20.1 Place 'A' HPSW Pump switch in "OFF".

_____ 20.2 Place 'A' HPSW Pump switch in "BASE" or "STANDBY".

_____ 20.3 Place 'B' HPSW Pump switch in "OFF".

_____ 20.4 Place 'B' HPSW Pump switch in "BASE" or "STANDBY".

_____ 21. Monitor EWST Level indication.

- Ensure HPSW Pump(s) automatically start, if needed.

CAUTION 22:	Transformer CT-4 <u>or</u> CT-5 limits may be exceeded if more than two (2) CCW Pumps are started for the Station. If the ESV System is <u>NOT</u> in operation, operation of one (1) CCW Pump on two (2) <u>different</u> Oconee Units is required to satisfy LPSW Pump NPSH requirements. Coordinate with Units 2 and 3 as required.

_____ 22. <u>**IF**</u> any Unit 1 CCW Pump is available,

<u>**AND**</u> it is desired to establish CCW forced flow on Unit 1,

<u>**THEN**</u> start <u>one</u> Unit 1 CCW Pump as follows:

_____ 22.1 <u>**IF**</u> Instrument Air has been lost,

<u>**AND**</u> <u>any</u> of the following valves <u>**CANNOT**</u> be opened:

_____ 1CCW 1-6 (WATERBOX EMER DISCH)

_____ CCW-8 (EMERGENCY CCW DISCHARGE TO TAILRACE),

<u>**THEN**</u> dispatch an Operator to valve in CCW to both Condensate Coolers by ensuring open, the following valves:

_____ 1CCW-76 (1A Condensate Cooler CCW Inlet) (TB1/F-25)

_____ 1CCW-78 (1B Condensate Cooler CCW Inlet) (TB1/F-25)

_____ 1CCW-75 (Condensate Coolers CCW Supply) (TB1/F-25)

_____ 1CCW-77 (1A Condensate Cooler CCW Outlet) (TB1/E-25)

_____ 1CCW-79 (1B Condensate Cooler CCW Outlet) (TB1/E-25)

_____ 1CCW-86 (Condensate Cooler CCW Flow Control Bypass). (TB1/E-26)

_____ 22.2 **IF** <u>all</u> CCW Pump discharge valves are open,

THEN place <u>ALL</u> the CCW Pump switches in the "TRIP" position:

- _____ 1A CCW Pump
- _____ 1B CCW Pump
- _____ 1C CCW Pump
- _____ 1D CCW Pump.

_____ 22.2.1 Locally close
the CCW Pump discharge valve on the pump to be started:

- _____ 1CCW-10 BKR (1A CCW PUMP DISCH) (1XS1-F2C)
- _____ 1CCW-13 BKR (1D CCW PUMP DISCH) (1XS1-F3C)
- _____ 1CCW-11 BKR (1B CCW PUMP DISCH) (1XS2-F2D)
- _____ 1CCW-12 BKR (1C CCW PUMP DISCH). (1XS3-2E)

_____ 22.3 Verify SSW is available to the CCW Pump that will be started:

- "CCW Pump 1A Bearing Seal Flow" (O1E2713)
- "CCW Pump 1B Bearing Seal Flow" (O1E2718)
- "CCW Pump 1C Bearing Seal Flow" (O1E2729)
- "CCW Pump 1D Bearing Seal Flow" (O1E2735).

Unit Status Assessment

_____ 22.4 Verify SSW is available to the CCW Pump Motor that will be started:

- "CCW Pump 1A Motor Cooler Flow" (O1E2714)
- "CCW Pump 1B Motor Cooler Flow" (O1E2719)
- "CCW Pump 1C Motor Cooler Flow" (O1E2730)
- "CCW Pump 1D Motor Cooler Flow" (O1E2736).

_____ 22.5 Start a CCW Pump with a closed discharge valve.

_____ 22.6 **IF** IA is available,

THEN verify that the following valves open:

_____ "1CCW-20 Condenser 1A Outlet 1" (O1D0272)

_____ "1CCW-21 Condenser 1A Outlet 2" (O1D0274)

_____ "1CCW-22 Condenser 1B Outlet 1" (O1D0276)

_____ "1CCW-23 Condenser 1B Outlet 2" (O1D0278)

_____ "1CCW-24 Condenser 1C Outlet 1" (O1D0280)

_____ "1CCW-25 Condenser 1C Outlet 2" (O1D0282).

NOTE 23:	In order for a Unit to be considered an adequate supply to the CCW cross-over header for supplying LPSW Pump NPSH, one of the following must be true: 1) At least one (1) CCW Pump in operation **OR** 2) One (1) ESV Pump and Siphon Header combination functioning properly as determined by Step 18.

_____ 23. **IF** at least two (2) Units are available to provide adequate flow to the CCW cross-over header,

THEN **GO TO** Step 24.

_____ 23.1 **IF** **NO** Unit 1 CCW Pumps are operating,

AND Forebay elevation > 91 ft,

THEN locally valve in CCW to both Condensate Coolers by ensuring open the following valves:

_____ 1CCW-76 (1A Condensate Cooler CCW Inlet) (TB1/F-25)

_____ 1CCW-78 (1B Condensate Cooler CCW Inlet) (TB1/F-25)

_____ 1CCW-75 (Condensate Coolers CCW Supply) (TB1/F-25)

_____ 1CCW-77 (1A Condensate Cooler CCW Outlet) (TB1/E-25)

_____ 1CCW-79 (1B Condensate Cooler CCW Outlet) (TB1/E-25)

_____ 1CCW-86 (Condensate Cooler CCW Flow Control Bypass) (TB1/E-26).

Unit Status Assessment

24. Ensure RCW Pumps are operating as required:

_____ 'A' RCW Pump

_____ 'B' RCW Pump

_____ 'C' RCW Pump

_____ 'D' RCW Pump.

25. Ensure SF Cooling Pumps are operating as required:

_____ 'A' SF Cooling Pump

_____ 'B' SF Cooling Pump

_____ 'C' SF Cooling Pump.

_____ 25.1 **IF** starting 'C' SF Cooling Pump,

THEN **REFER TO** OP/1&2/A/1104/006 (Spent Fuel Cooling System).

_____ 26. **IF** either Main FDWP suction valve is open,

_____ 1FDW-1 (1A FDWP SUCTION)

_____ 1FDW-6 (1B FDWP SUCTION),

THEN verify ≥ 4 psig "FWPT Brg Lube Oil" pressure on the associated Main FDWP(s):

_____ 1A FDWPT

_____ 1B FDWPT.

_____ 26.1 **IF** ≥ 4 psig "FWPT Brg Lube Oil" pressure CANNOT be established,

THEN ensure closed the Main FDWP suction valve on the affected FDWP(s):

_____ 1FDW-1 (1A FDWP SUCTION)

_____ 1FDW-6 (1B FDWP SUCTION).

CAUTION 27:	If it has been > 25 minutes since a loss of all Condensate flow, a steam induced water hammer may occur when the first Hotwell Pump is restarted. Unless needed to immediately restore feed to a SG, a HWP should NOT be restarted until an Engineering evaluation has been performed.

_____ 27. **IF** NO HWPs are operating,

THEN align the Condensate system for recirculation per AP/1/A/1700/019 (Loss Of Main Feedwater).

CAUTION 28:	CSAE(s) steam supplies should be isolated to minimize Condensate system heatup and steam void formation.

_____ 28. **IF** HWP restart has been delayed,

THEN perform the following:

_____ 28.1 Start and align the Main Vacuum Pumps to Unit 1 per Enclosure 6.5, "Restoring Loads Outside The Control Room".

28.2 Close the following Unit 1 CSAE valves:

_____ 1V-7 (CSAE A1 Inlet) (TB3/G-25/26)

_____ 1V-10 (CSAE B1 Inlet) (TB3/G-25/26)

_____ 1V-13 (CSAE C1 Inlet) (TB3/H-25/26).

28.3 Close the following Unit 1 CSAE drains:

_____ 1C-388 (CSAE Inner Cond Drn Hdr To Cond) (TB1/G-24)

_____ 1C-385 (CSAE After Cond Drn Hdr Block) (TB1/F-25).

28.4 Ensure closed the following:

_____ 1AS-40 (AS TO CSAE)

_____ 1MS-47 (MS TO CSAE).

_____ 28.5 Stop the CSAE Off Gas Blower.

_____ 29. **IF** IA Header Pressure < 90 psig:

- "Aux Bldg IA Hdr Press"
- "Turb Bldg IA Hdr Press",

OR the Backup IA Compressors **CANNOT** be started,

THEN start the Primary IA Compressor as follows:

_____ 29.1 Verify B3T is energized from either the 230 KV RED BUS or the 525 KV Switchyard:

_____ 230 KV RED BUS:

- "PCB-4 4T FEEDER" closed
- "B3T NORMAL FEEDER" closed,

OR

_____ 525 KV Switchyard:

- "OCB-40 5T FEEDER" closed
- "B3T ALTERNATE FEEDER" closed.

_____ 29.2 **IF** B3T is **NOT** energized:

- "4KV SWITCHGEAR B3T/B4T TROUBLE" (SA-17/ C-2) on,

THEN notify the TSC to have I&E Electrical Technical Support investigate why the B3T power supply did **NOT** automatically swap. {3}

_____ 29.3 **IF** **AT ANY TIME** B3T is energized,

THEN dispatch an Operator to start the Primary IA Compressor.

- **REFER TO** OP/0/A/1106/027 (Compressed Air System).

_____ 30. **IF** Unit 1 and Unit 2 are carrying any portion of the GWD Vent Header,

THEN ensure at least one (1) GWD Compressor is operating:

_____ 'A' GWD Compressor

_____ 'B' GWD Compressor.

_____ 30.1 Ensure ≈ 0 inches Vent Header pressure.

31. Ensure RB Aux Fans are operating:

_____ 1A RB Aux Fan

_____ 1B RB Aux Fan

_____ 1C RB Aux Fan

_____ 1D RB Aux Fan.

32. Ensure at least one (1) Continuous Vacuum Priming Pump is operating:

_____ 'A' Priming Pump

_____ 'B' Priming Pump.

> **CAUTION:** If the MFDWP and Main Turbine were taken off the Turning Gear due to a Station Blackout, shaft bowing may have occurred which could cause turbine damage if placed on the Turning Gear.

_____ 33. **IF** the FDWP Auxiliary Oil Pump is operating:

_____ 1A FDWP Auxiliary Oil Pump

_____ 1B FDWP Auxiliary Oil Pump,

THEN secure the following pump(s):

_____ 1A FDWP Emergency Bearing Oil Pump

_____ 1B FDWP Emergency Bearing Oil Pump.

_____ 33.1 **IF** **AT ANY TIME** the 1A FDWPT reaches zero (0) speed,

THEN ensure the following:

_____ 1A FDWP Turning Gear Motor starts

_____ 1A FDWP Turning Gear engages.

_____ 33.2 **IF** **AT ANY TIME** the 1B FDWPT reaches zero (0) speed,

THEN ensure the following:

_____ 1B FDWP Turning Gear Motor starts

_____ 1B FDWP Turning Gear engages.

34. Restore lube oil to the Turbine-Generator:

_____ 34.1 Ensure Turbine Turning Gear Oil Pump starts.

_____ 34.2 Ensure Turbine Bearing Oil Lift Pumps start.

_____ 34.3 Ensure Turbine Motor Suction Pump starts.

_____ 34.4 Dispatch an Operator to verify proper operation of the oil pumps.

_____ 34.5 Secure Turbine Emergency Bearing Oil Pump.

_____ 34.6 **IF** **AT ANY TIME** the Turbine-Generator rotor stops,

THEN place Turbine on the Turbine Turning Gear.

_____ 34.6.1 Stop Turbine Motor Suction Pump.

_____ 35. Dispatch an Operator to verify the Auxiliary Building Ventilation operability.

- **REFER TO** OP/0/A/1104/041 (Auxiliary Building Ventilation).

NOTE 36:	A different individual must be used to verify proper operation of the Control Room Ventilation system than originally performed Enclosure 6.5, "Restoring Loads Outside The Control Room".

_____ 36. Within 18 hours of the initiating event, locally ensure that the following components are operating/aligned properly using Enclosure 6.5, "Restoring Loads Outside The Control Room": {5}

- Air Handling Units:

  _____ AHU 1-11

  _____ AHU 1-12

  _____ AHU-22

  _____ AHU-23

  _____ AHU-34

  _____ AHU-35.

- Dampers:

  _____ 1VS-DA-CD13 (Unit 1 Equip. Rm N alcove in overhead)

  _____ 1VS-DA-CD14 (Unit 1 Equip. Rm N alcove in overhead)

  _____ 2VS-DA-CD13 (Unit 2 Equip. Rm S alcove in overhead)

  _____ 2VS-DA-CD14 (Unit 2 Equip. Rm S alcove in overhead)

  _____ VS-DA-FD01 (Unit 2 Cable Room North Wall)

  _____ VS-DA-FD02 (Unit 2 Cable Room North Wall)

  _____ VS-DA-FD03 (Unit 2 Equipment Room North Wall)

  _____ VS-DA-FD04 (Unit 2 Equipment Room North Wall).

NOTE 37:	Any Unit's computer indication is acceptable to verify ECCW discharge piping full. Unit specific computer point "CCW Emer Discharge Level" (D2673) will read "NOT LOW" when the line is primed.

_____ 37. **IF** **AT ANY TIME** any Unit 1 CCW Pump is operating,

AND **NO** other Unit requires CCW Siphon Flow,

THEN perform the following to secure the Station CCW Siphon Flow lineup:

_____ 37.1 Throttle 1CCW 1-6 (WATERBOX EMER DISCH) to the intermediate position.

37.2 Coordinate with Units 2 & 3 and perform the following:

_____ 37.2.1 Ensure 2CCW-7 (WATER BOX EMERGENCY DISCHARGE) throttled.

_____ 37.2.2 Ensure 3CCW-93 (WATER BOX EMERGENCY DISCHARGE) throttled.

_____ 37.2.3 Close CCW-8 (EMERGENCY CCW DISCHARGE TO TAILRACE).

_____ 37.2.4 Close 2CCW-7 (WATER BOX EMERGENCY DISCHARGE).

_____ 37.2.5 Close 3CCW-93 (WATER BOX EMERGENCY DISCHARGE).

_____ 37.3 **IF** **AT ANY TIME** the Emergency Discharge Line is primed,

THEN close 1CCW 1-6 (WATERBOX EMER DISCH).

_____ 38. **IF** a 230 KV Switchyard Isolation has occurred (any of the following Statalarms are on):

- "CHANNEL #1 UNDERFREQUENCY" (SA-15/ A-2)
- "CHANNEL #1 UNDERVOLTAGE" (SA-15/ C-1)
- "CHANNEL #2 UNDERFREQUENCY" (SA-15/ A-4)
- "CHANNEL #2 UNDERVOLTAGE" (SA-15/ C-3),

THEN perform the following:

- Verify the following statalarms are on:

_____ "EL SWYD ISOLATION CONFIRMED CHNL A LOGIC" (1SA-15/ E-6)

_____ "EL SWYD ISOLATION CONFIRMED CHNL B LOGIC" (1SA-14/ E-6).

- Verify the following PCBs are open:

_____ PCB-8	_____ PCB-12	_____ PCB-15
_____ PCB-17	_____ PCB-21	_____ PCB-24
_____ PCB-26	_____ PCB-28	_____ PCB-33.

- Verify the following PCBs are closed:

_____ PCB-9

_____ PCB-18

_____ PCB-27

_____ PCB-30.

38.1 Verify both Keowee Units Emergency Start:

_____ "UNIT #1 EMERG START INITIATED" (2SA-17/ C-1)

_____ "UNIT 2 EMERGENCY START INITIATED" (2SA-18/ C-1).

- Voltage indicates ≈ 13.8 KV on:

  _____ "KEOWEE 1 OUTPUT VOLTS" meter

  _____ "KEOWEE 2 OUTPUT VOLTS" meter.

_____ 38.2 Verify "CT-4 VOLTS" meter indicates ≈ 4.16 KV.

_____ 38.3 IF Switchyard Isolation is NOT complete (NEITHER statalarm on):

- "EL SWYD ISOLATION CONFIRMED CHNL A LOGIC" (1SA-15/ E-6)
- "EL SWYD ISOLATION CONFIRMED CHNL B LOGIC" (1SA-14/ E-6),

AND power is required from Keowee through the Overhead path,

THEN coordinate with Units 2 & 3 to recover from Switchyard Isolation per Enclosure 6.7, "Recovery From Switchyard Isolation".

_____ 38.4 IF AT ANY TIME conditions permit recovery from Switchyard Isolation,

THEN coordinate with Units 2 & 3 to recover from Switchyard Isolation per Enclosure 6.7, "Recovery From Switchyard Isolation".

_____ 39. **IF** the Main Feeder Bus(s) is energized from the Standby Bus,

AND the Startup Transformer (CT-1) is energized from the 230 KV Switchyard,

THEN transfer the Main Feeder Bus loads to the Startup Transformer using either of the following two procedures:

- **REFER TO** OP/0/A/1106/019 (Keowee Hydro At Oconee)
- **REFER TO** OP/0/A/1107/003 (100 KV Power Supply).

CAUTION 40: A thermal overload trip of RBCU(s) may occur when power is regained if switch position is in "HIGH" and NO ES signal exists.

_____ 40. **IF** an ES Actuation has **NOT** occurred,

THEN ensure proper operation of RB Cooling Units.

- **REFER TO** OP/1/A/1104/015 (Reactor Building Cooling System).

_____ 41. **IF** a Load Shed of the Inverters was performed due to a Station Blackout,

THEN shutdown and isolate AC and DC power to the 1KI, 1KU, 1KX, and 1KOAC Inverters per OP/1/A/1107/004 (Operation Of The Vital Bus, Computer, ICS, And Auxiliary Inverters).

_____ 41.1 Restore Inverters to normal operation when desired.

- **REFER TO** the following procedures:

  _____ OP/1/A/1107/004 (Operation Of The Vital Bus, Computer, ICS, And Auxiliary Inverters)

  _____ OP/0/A/1103/020 (Loss Of Computer).

_____ 42. Notify Operators to perform complete rounds at each watch station and monitor equipment affected by the Loss of Power (Control, Power, and Switchyard Battery Chargers, Inverters, Regulated Power Supply 1KRA/1KRB, Cable and Equipment Room temperatures, etc.).

- **REFER TO** OMP 2-16 (Shift Turnover).

43. Restore electrical systems to normal alignment.

_____ 43.1 Perform the enclosure for restoration of electrical loads after Load Shed per OP/1/A/1107/002 (Normal Power).

END

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).
- 1RIA-17 (B MS Line Rad Monitor) OOS
- 1A GWD Tank release in progress per GWR#00-200
- 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	Override	I,BOP	1RIA-37 and 38 fails to terminate GWR

3		N, ALL	De-Lithiation with the deborating Demineralizer
4	Override	C,BOP	1HP-14 fails in the "bleed" position (IPE – PIP O-99-05270)
5	Override FDW03	C,OATC	ICS STAR module failure (FDW)
6		R,OATC	Unit/reactor shutdown
7	MPI171, MPI500	I, OATC	RC T-Hot "A" (1) fails LOW (median select with MPI 500) RC T-Hot "A" (2) fails LOW
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
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: 1RIA-37 and 38 fails to terminate 1A GWD Tank release.		

Time	Position	Applicant's Actions or Behavior
	BOP	Respond to statalarm 1SA-8/E-10 RM GWD Discharge Radiation Inhibit • Refer to ARG for 1SA-8/E-10 RM GWD Discharge Radiation Inhibit
	SRO / BOP	Determine that the automatic action of 1SA-8/E-10 RM GWD Discharge Radiation Inhibit did not terminate the release • Manually terminate the release: • Close GWD-4 (1A GWD Tank Discharge) • Refer to in progress gas release procedure OP/1104/18 • Inform the CR SRO

Op-Test No.:	01	Scenario No.:	03	Event No.:	3

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

Op-Test No.: 01	Scenario No.: 03	Event No.: 3	
Event Description: De-Lithiation with the deborating Demineralizer.			

Time	Position	Applicant's Actions or Behavior
	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) • Close 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.:	4

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. • LDST level decreasing • Increased rate of decrease in the LDST • OAC alarm 1HP-14 BLEED • 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

Op-Test No.: 01	Scenario No.: 03	Event No.: 4

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
	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.
	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.:	5
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.:	6

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.:	7

Event Description:
RC T-Hot "A" (1) FAILS LOW (media 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.:	8/9

Event Description:

Steam Generator tube LEAK OTSG "A" (200 gpm)
RIAs FAIL. 1RIA-16 – LOW ("as-is")
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:** 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.
- **Second:** 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.

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
	OATC / BOP / SRO	Perform AP/1/A/1700/002, Excessive RCS Leakage • Determine tube leak size • Refer to ITS • Diagnose tube 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.
	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
	SRO / OATC / BOP	Calculate SGTL to be > than TS (150 gpd) and enter the EOP Section 504, SGTL • Maintain PZR level via RULE #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 tube leak size • Continue the unit shutdown (Do not trip the reactor unless needed to prevent opening the MSRV) • Trip at 15 % Reactor Power (35 MWE) • Maintain SG pressures < 950 psig • Depressurize the RCS to lower SCM (5° - 10°F) • Cooldown the RCS to 532°F • Prepare to isolate the "A" SG
	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

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	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 ≥ 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:

- 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).
- 1RIA-17 (B MS Line Rad Monitor) OOS
- 1A GWD Tank release in progress per GWR#00-200
- GWD Vent header cross-connected – Unit 1 has the GWD header

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. GWR#00-200

   Determine that the automatic action of 1SA-8/E-10 RM GWD Discharge Radiation Inhibit did not terminate the release. The crew will manually terminate the release by closing GWD-4 (1A GWD Tank Discharge) and close out release per in progress gas release procedure OP/1104/18.

3. 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

4. 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.

5. 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 nor occur.

When the crew diagnoses a star module failure and continues the shutdown this event is complete

TIME = 5 minutes TOTAL = 15-20 minutes

6. Unit shutdown per OP/1102/04 Operation at Power with the ICS FDW in manual

7. 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.

8 and 9. OTSG tube leak (≈ 200gpm) with faulted RIA-16 (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.

10. "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.

Alarm Response Guide 1SA-08

E-10

RM

GWD DISCHARGE RADIATION INHIBIT

1. Alarm Setpoint

1.1 Set by Operators per HP calculation.

2. Automatic Action

2.1 1RIA-37 and/or 1RIA-38 will close valves 1GWD-4, -5, -6, and -7 and GWD-206, -207 and stop the WG Exhauster if the high alarm is received.

3. Manual Action

3.1 Verify that the automatic action has taken place.

3.2 Refer to OP/1&2/A/1104/18 (Gaseous Waste Disposal System) for direction on continuing or terminating release.

3.3 Refer to AP/1/A/1700/18 (Abnormal Release of Radioactivity).

4. Alarm Sources and References

4.1 OEE-188-17 & 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]).

Continuous Use

1. Initial Conditions

✓ 1.1 Waste Gas discharge flow instrument operable.

✓ 1.2 Initiate Enclosure "GWD Tank Sample Request."

✓ 1.3 Review Limits and Precautions.

2. Procedure

✓ 2.1 Sample results are received:

✓ 2.1.1 IF sample results allow release as determined by CR SRO, continue to Step 2.2.

N/A 2.1.2 IF sample results are too high in activity for release, complete Enclosure "GWD Tank Sample Request" AND return to RP.

N/A • Stop this Enclosure.

> **NOTE:**
> - GWRs should be coordinated with favorable meteorological conditions.
> - Unfavorable meteorological conditions are described as follows:
>   - Temperature inversion indicated by (+) on charts.
>   - Low wind speeds.

✓ 2.2 Meteorological conditions acceptable for release.

2.3 Determine if other GWRs are in progress at station.

	Releases in Progress ☑	Release Rate of Station Limit
Unit 1	☐ Yes ☑ No	
Unit 2	☐ Yes ☑ No	
Unit 3	☐ Yes ☑ No	

2.4 IF release will be made at 1/3 station release limit, verify 1RIA-45 High AND Alert Alarm setpoints set per PT/0/A/0230/001 (Radiation Monitor Check).

✓ ✓ • 1RIA-45 High Alarm set.

✓ ✓ • 1RIA-45 Alert Alarm set.

2.5 **IF** release will be made at 2/3 station release limit:

N/A 2.5.1 Verify other two units **NOT** releasing.

___ 2.5.2 Multiply Alert **AND** High setpoints for 1RIA-45 per PT/0/A/0230/001 (Radiation Monitor Check) by two.

___ ___ • Reset 1RIA-45 High Alarm.

___ ___ • Reset 1RIA-45 Alert Alarm.

___ 2.5.3 Notify Unit 2 CR to perform the following:

- Multiply 2RIA-45 Alert **AND** High setpoints per PT/0/A/0230/001 (Radiation Monitor Check) by 0.5.
- Reset 2RIA-45 High Alarm.
- Reset 2RIA-45 Alert Alarm.

___ 2.5.4 Notify Unit 3 CR to perform the following:

- Multiply 3RIA-45 Alert **AND** High setpoints per PT/0/A/0230/001 (Radiation Monitor Check) by 0.5.
- Reset 3RIA-45 High Alarm.
- Reset 3RIA-45 Alert Alarm.

3. GWD Release

✓ 3.1 Record background readings for 1RIA-37 & 1RIA-38 on Enclosure "GWD Tank Sample Request".

NOTE:	The RIA required to terminate release (RIA within range) must be operable. **OR** Two independent samples must be taken.

3.2 Recommended 1RIA-37 and 38 High and Alert setpoints:

✓ 1RIA-37 1E3 cpm above background.

✓ 1RIA-38 1E3 cpm above background.

NOTE:	• RIA- 38 continuous source check fulfills SLC 16.11-3 requirements for "source check". • Operable RIAs should be source checked.

✓ 3.3 **IF** required, perform source check on 1RIA-37.

NOTE:	Release setpoint = background cpm + recommended cpm above background.

3.4 Adjust 1RIA-37 High **AND** Alert setpoints for release as follows:

✓ ✓ • **IF** 1RIA-37 must be overriden, set High **AND** Alert setpoints at zero.

OR

✓ • **IF** required, set alarms:

✓ ✓ • 1RIA-37 Alert setpoint at 1000 cpm.

✓ ✓ • 1RIA-37 High setpoint at 1000 cpm.

3.5 Adjust 1RIA-38 High **AND** Alert setpoints for release as follows:

N/A ____ • **IF** 1RIA-38 must be overriden, set High **AND** Alert setpoints at zero.

OR

____ • **IF** required, set alarms:

____ ____ • 1RIA-38 Alert setpoint at ______________cpm.

____ ____ • 1RIA-38 High setpoint at ______________cpm.

✓ 3.6 Reset integrator to zero.

✓ 3.7 Place **OR** verify GWR Discharge Flow Controller in "HAND".

✓ • Verify GWR Discharge Flow Controller "CLOSED".

✓ 3.8 Open GWD-100 (Decay Tanks Discharge Header Block). (A-2-209/E 13' N of Door)

3.9 **IF** releasing GWD Tank 1A perform the following:

N/A ____ • Open GWD-98 (Decay Tank 1A Discharge Block). (A-2-209/E 14' N of Door)

____ ____ • Place GWD-4 ('A' GWD TANK DISCHARGE) switch to AUTO.

3.10 **IF** releasing GWD Tank 1B perform the following:

✓ ✓ • Open GWD-99 (Decay Tank 1B Discharge Block). (A-2-209/E 14' N of Door)

✓ ✓ • Place GWD-5 ('B' GWD TANK DISCHARGE) switch to AUTO.

3.11 **IF** releasing GWD Tank 1C perform the following:

N/A ____ • Open GWD-204 (Decay Tank 1C Discharge Block). (IRW Building)

____ ____ • Place GWD-206 ('C' INTERIM GWD TANK DISCH) switch to OPEN.

3.12 **IF** releasing GWD Tank 1D perform the following:

N/A ____ • Open GWD-205 (Decay Tank 1D Discharge Block). (IRW Building)

____ ____ • Place GWD-207 ('D' INTERIM GWD TANK DISCH) switch to OPEN.

NOTE:
- Station Limit release rates, as described by SLC 16.11, will NOT be exceeded if recommended release rates per Enclosure "GWD Tank Sample Request" are followed.
- L&P has required approval levels for release.

3.13 Approval granted for release: ✓ Approval ✓ / ✓ Date Time

3.14 IF releasing at 2/3 Station Limit:

N/A • Verify 2RIA-45 High and Alert Alarms set per step 2.5.3.

• Verify 3RIA-45 High and Alert Alarms set per step 2.5.4.

✓ 3.15 Recommended Release Rate from Enclosure "GWD Tank Sample Request":

35 cfm.

✓ 3.16 Adjust GWR Discharge Flow Controller to obtain desired release rate.

NOTE: Meteorological instrumentation may be inoperable during GWR.

✓ 3.17 IF available, mark Meteorological charts (Temperature, Wind Direction and Wind Speed) at beginning of GWR with "Begin GWR # 00-200 ".

✓ 3.18 Record "Begin GWR # 00-200 " in Unit Log.

✓ 3.19 Notify Unit 2 CR to place the following note on turnover sheet:

• IF 2RIA-45 alarms, notify Unit 1 CR.

✓ 3.20 Notify Unit 3 CR to place the following note on turnover sheet:

• IF 3RIA-45 alarms, notify Unit 1 CR.

NOTE: IF RIA High Alarm is received:

- GWD Tank discharge valve should automatically close.
- "Control Enable Screen/s" on RIA VIEW NODE should show "INHIBIT".

3.21 IF automatic termination occurs:

_____ 3.21.1 Close GWR Discharge Flow Controller.

_____ 3.21.2 Investigate cause of alarm.

_____ 3.21.3 IF required, request an independent sample on GWD tank.

_____ 3.21.4 Manually reset valve inhibit on "Control Enable Screens" of RIA VIEW NODE.

_____ 3.21.5 IF required, perform GWD Tank Release using revised setpoints.

_____ 3.21.6 Record maximum cpm of RIA-37 or 38: ____________ cpm

_____ 3.21.7 IF required, terminate release.

_____ 3.22 WHEN GWD Tank is ≈ 5 psig, close GWR Discharge Flow Controller.

_____ 3.23 Add N^2 to affected GWD tank to increase pressure to 15-20 psig per Enclosure "Adding N^2 To GWD Tank".

_____ 3.24 Adjust GWR Discharge Flow Controller to obtain desired release rate.

_____ 3.25 WHEN GWD Tank is ≈ 5 psig, close GWR Discharge Flow Controller.

3.26 Continue with Section 4, "GWD Termination".

4. GWD Termination:

4.1 Perform the following:

_____ • IF available, mark Meteorological charts (Temperature, Wind Direction and Wind Speed) at termination of GWR with "Stop GWR # _______________."

_____ • Record "Stop GWR # _________" in Unit Log.

_____ 4.2 Complete Enclosure "GWD Tank Sample Request" AND route to RP.

4.3 IF GWD Tank 1A released, isolate as follows:

_____ _____ • Close GWD-98 (Decay Tank 1A Discharge Block). (A-2-209/E 14' N of Door)

_____ _____ • Close GWD-100 (Decay Tanks Discharge Header Block). (A-2-209/E 13' N of Door)

_____ • Place GWD-4 ('A' GWD TANK DISCHARGE) switch to "CLOSED".

4.4 IF GWD Tank 1B released, isolate as follows:

_____ _____ • Close GWD-99 (Tank 1B Discharge Block). (A-2-209/E 14' N of Door

_____ _____ • Close GWD-100 (Decay Tanks Discharge Header Block). (A-2-209/E 13' N of Door)

_____ • Place GWD-5 ('B' GWD TANK DISCHARGE) switch to "CLOSED".

4.5 IF GWD Tank 1C released, isolate as follows:

_____ _____ • Close GWD-204 (Decay Tank 1C Discharge Block). (IRW Building)

_____ _____ • Close GWD-100 (Decay Tanks Discharge Header Block). (A-2-209/E 13' N of Door)

_____ • Place GWD-206 ('C' INTERIM GWD TANK DISCH) switch to "CLOSED".

4.6 IF GWD Tank 1D released, isolate as follows:

_____ _____ • Close GWD-205 (Decay Tank 1D Discharge Block). (IRW Building)

_____ _____ • Close GWD-100 (Decay Tanks Discharge Header Block). (A-2-209/E 13' N of Door)

_____ • Place GWD-207 ('D' INTERIM GWD TANK DISCH) switch to "CLOSED".

4.7 IF released, check GWD Tank 1A for accumulation of water.

4.7.1 Throttle the following AND close WHEN no water passes through sightglass: (A-2-209)

_____ _____ • LWD-240 (Waste Gas Tank 1A Drain).

_____ _____ • LWD-352 (Waste Gas Tank 1A Drain Block).

_____ _____ • LWD-350 (Waste Gas Tank 1A Drain).

_____ _____ • LWD-351 (Waste Gas Tank 1A Drain Block).

4.8 IF released, check GWD Tank 1B for accumulation of water.

4.8.1 Throttle the following AND close WHEN no water passes through sightglass: (A-2-209)

_____ _____ • LWD-241 (Waste Gas Tank 1B Drain).

_____ _____ • LWD-355 (Waste Gas Tank 1B Drain Block).

_____ _____ • LWD-353 (Waste Gas Tank 1B Drain).

_____ _____ • LWD-354 (Waste Gas Tank 1B Drain Block).

4.9 IF released, check GWD Tank 1C for accumulation of water.

4.9.1 Throttle the following AND close WHEN no water passes through sightglass: (Interim Bldg)

_____ _____ • GWD-197 (Waste Gas Tank 1C Drain).

_____ _____ • GWD-198 (Waste Gas Tank 1C Drain).

4.10 IF released, check GWD Tank 1D for accumulation of water.

4.10.1 Throttle the following AND close WHEN no water passes through sightglass: (Interim Bldg)

_____ _____ • GWD-200 (Waste Gas Tank 1D Drain Isolation).

_____ _____ • GWD-201 (Waste Gas Tank 1D Drain Isolation).

4.11 Perform the following to drain water from carbon **AND** absolute filters: (A-2-210)

_____ _____ • Throttle LWD-242 (Waste Gas Absolute Filter Drain).

_____ _____ • Throttle LWD-243 (Waste Gas Charcoal Filter Drain).

NOTE:	No level change in HAWT indicates filter is drained.

4.11.1 **WHEN** filters are drained: (A-2-210)

_____ _____ • Close LWD-242 (Waste Gas Absolute Filter Drain).

_____ _____ • Close LWD-243 (Waste Gas Charcoal Filter Drain).

_____ 4.12 Perform purge on 1RIA-37 **AND** 1RIA-38 per Enclosure "1RIA-37 and 38 Purge."

4.13 **IF** release was at 2/3 Station Limit:

4.13.1 Reset 1RIA-45 Alarms per PT/0/A/0230/001 (Radiation Monitor Check).

_____ _____ • Reset 1RIA-45 High Alarm.

_____ _____ • Reset 1RIA-45 Alert Alarm.

_____ 4.13.2 Notify Unit 2 CR to reset 2RIA-45 Alarms per PT/0/A/0230/001 (Radiation Monitor Check).

- Reset 2RIA-45 High Alarm.
- Reset 2RIA-45 Alert Alarm.

_____ 4.13.3 Notify Unit 3 CR to reset 3RIA-45 Alarms per PT/0/A/0230/001 (Radiation Monitor Check).

- Reset 3RIA-45 High Alarm.
- Reset 3RIA-45 Alert Alarm.

Enclosure 3.10

GWD Tank Sample Request

GWR # 00-200

Gaseous analysis request for GWD Tank # 1B

Tank Isolated: Date 6-27-00 Time 1146 Volume 6300 (ft³)

Independent Sample Analysis Required (☑) ☐ Yes ☑ No

Sample Analysis Results

Iodine 2.06 E -5 uci/ml

Gaseous 6.17 E -4 uci/ml

RP Representative ✓ Date/Time ✓

Independent Sample Analysis Results

Iodine N/A

Gaseous

RP Representative Date/Time

Independent Sample agrees with initial sample N/A

RP Representative

Recommended Release Rates

☑ 1/3 Station Limit 35 cfm

☐ 2/3 Station Limit ______ cfm

NOTE:	**IF** 1RIA-37&38 are inoperable, set High and Alert setpoints to zero.

Recommended High and Alert setpoints for:

1RIA-37 1E 3 cpm above background

1RIA-38 1E 3 cpm above background

Enclosure 3.10

GWD Tank Sample Request

GASEOUS RELEASE VOLUME UPDATE

1B GWD Tank Release

Started: Date ✓ Time ✓

Tank Pressure: Start 70 psig

RIA Background Reading:

IF available, 1RIA-37 25

IF available, 1RIA-38 0

Actual Release Rate: 30 cfm

RIA Equilibrium Readings during Release:

IF available, 1RIA-37 ______

IF available, 1RIA-38 ______

Terminated: Date ______ Time ______

Integrator Reading: ______ ft^3

Tank Pressure: Finish ______ psig

Comments: ______

Route this GWD Tank Sample Request to RP.

______ ______/______
CRO Date Time