UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

ENCLOSURE 1

EXAMINATION REPORT - 50-250/OL-89-02

Facility Licensee: Florida Power and Light Company
Turkey Point Nuclear Plant
P. O. Box 029100
Juno Beach, FL 33102

Facility Name: Turkey Point Units 3 and 4

Facility Docket Nos.: 50-250 and 50-251

Written and operating remedial requalification tests were conducted at the Turkey Point plant site near Florida City, Florida.

Chief Examiner: D. Charles Payne — 8/18/89 Date Signed

Approved By: Charles A. Casto, Chief, Operator Licensing Section 1, Division of Reactor Safety — 8/21/89 Date Signed

Summary:

Examinations were conducted June 1 - 2, 1989.

Written and operating examinations were administered to one Reactor Operator (RO) and five Senior Reactor Operators (SROs). All operators passed the examination.

REPORT DETAILS

1. Facility Employees Attending Exit Meeting

   K. N. Harris, Site Vice President
   J. E. Cross, Plant Manager
   K. E. Beatty, Assistant to Plant Manager
   L. W. Pearce, Operations Superintendent
   T. A. Finn, Training Superintendent
   G. Hollinger, Operator Training Supervisor
   D. C. Henry, Operations Training
   L. Goebel, Operations Training
   W. J. Waylett, Jr., QA Training Coordinator
   B. Ford, Ebasco

2. Examiners

   *C. Payne, NRC, Region II
   C. Casto, NRC, Region II
   M. Morgan, NRC, Region II

   #R. C. Butcher, Senior Resident Inspector

   *Chief Examiner
   #Attended Exit Meeting Only

3. Exit Meeting

   At the conclusion of the site visit, the Chief Examiner met with representatives of the plant staff to discuss the results of the examinations. The following items were covered:

   <u>Operator Weaknesses</u>

   a. During the performance of Job Performance Measures (JPMs) on the simulator there was uncertainty exhibited by the operators as to whether they should actually perform an action or simply talk through it. The confusion apparently stemmed from the manner in which JPMs were covered during training. To save time during training the operators were occasionally told to skip or talk through actions they would normally perform. As a result, they expected the same type of response from the evaluators during this visit and became confused when the expected prompt was not provided.

   b. During the dynamic simulator examination, the operators were placed in a situation where a Steam Generator tube rupture had occurred. Due to plant conditions, the operators were eventually required to enter FR-P.1, "Response to Imminent Pressurized Thermal Shock." Step 22 of this procedure directs the operator to check Reactor Coolant System (RCS) subcooling based on Core Exit Thermocouples (CETs) at 30 degrees F. The Training Department identified this as too stringent a requirement to be met and trained the operators to

maintain a control band of 25-35 degrees F. The use of this band upon transitioning back to E-3, "Steam Generator Tube Rupture" led to procedure conflicts and set the operators up for potential procedure violations. FR-P.1, step 24, directs the operator to perform actions of other procedure in effect (E-3) which do not cool cool down or increase RCS pressure until an RCS temperature soak has been completed. Since E-3 requires subcooling to be greater than 30 degrees F, the use of a band below this temperature almost led the operators to operate Safety Injection (SI) pumps to restore subcooling thus increasing the possibility of PTS (subcooling was below 25 degrees F at the time).

This situation pointed out two significant problems. First, the Training Department was teaching the operators to violate Emergency Operating Procedures (EOPs) by arbitrarily establishing operating bands outside the procedure which were not reviewed and approved by the Plant Nuclear Safety Committee. The facility indicated that this problem would be corrected when the facility incorporated Revision 1A to the EOPs later this year. This concern will remain open (IFI-250,251/OL-92-01) until resolution is confirmed by future inspection. Second, a deficiency in one of the plant's most important emergency procedures was identified by the staff, yet no immediate corrective action was taken to review and mitigate the problem. The facility should ensure that all procedural deficiencies, when identified, are properly addressed in accordance with plant administrative procedures. This concern will remain open (IFI-250,251/OL-92-02) until resolution is confirmed by future inspection.

Improvements

a. Questions for the static simulator and open reference written examinations showed marked improvement over the previous examination. Well written questions that adequately tested the operator's knowledge were available for developing the examinations. Two recommendations were provided by the NRC examiners and incorporated by the facility: (1) improved and standardized the multiple choice question format, and (2) improved distractors on several multiple choice questions.

b. The generic weaknesses displayed by the crews during the March 1989 requalification examinations in the areas of performance of immediate operator actions, communications skills, and use of the Emergency Plan were not evident during this site visit. The performance of the Shift Technical Advisors (STAs) was noted as having improved.

c. Improvement was noted in the area of facility evaluator performance. There was no evidence of prompting or interference by the evaluators during this examination. Additionally, they exhibited improvement in their documentation of observed operator weaknesses.

WRITTEN EXAMINATION COVER SHEET

ANSWER KEY

U.S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: Turkey Point

REACTOR TYPE: PWR - W

DATE ADMINISTERED: 89/06/02

OPERATOR: _____

SECTION		CATEGORY VALUE	OPERATOR'S SCORE	% OF CATEGORY VALUE
A	Plant Proficiency	_____	_____	_____
B	Limits and Controls	_____	_____	_____

_____
Final Grade

Part B ORQ # 20

On Unit #3 determine the amount of reactivity necessary to overcome the power defect change for a power increase from 50% to 100% power by choosing the correct response. (Assume a boron concentration of 500 ppm.). (1.0 pt)

a. +647 PCM

b. +847 PCM

c. +927 PCM

d. +984 PCM

Ans: b. +847 PCM

(100 - 50%)
(1774pcm - 927pcm)
= 847pcm
From curve book

Ref: PCB Unit #3 Section II Figure 6A

K/A: #192004.K1.13(2.9/2.9)

LP#0010, Appendix E, EO 2

Est. Time of Completion 2 min.

Part B ORQ# 34

Unit 3 is operating at 90% reactor power. Due to a temperature increase of the circulating water the condenser backpressure increases from 3.5 to 4.0 inches hg. The output of the main generator would: (1.0 pt)

a. decrease from 608 Mwe to 595 Mwe.

b. increase from 595 Mwe to 608 Mwe.

c. would remain approximately constant because the turbine control valves would respond to maintain load.

d. decrease from 660 Mwe to 595 Mwe.

Ans:

a. decrease from 608 Mwe to 595 Mwe.

Ref: Unit 3 PCB Section I Figure 1

K/A #045000 K5.05(1.9/2.1)

LP#3502077, EO-11

Est. Time of Completion 2 min.

Part B ORQ# 14

Given the following conditions:

- Unit 3 in Mode 4 and on RHR
- 'A' RHR PP is cleared for maintenance
- 'B' RCP is running, 'A' and 'C' RCP's shutdown
- RCS conditions: Tavg = 300°F
  Pressure = 277 psig
  Przr level = 52%
- VCT pressure = 30 psig, VCT level in normal range
- 'A' charging Pp in service ('B' and 'C' charging Pps both cleared for maintenance)
- CCW system in normal Mode 4 lineup with no maintenance in progress.

Based on the above conditions which one of the following is most correct if '3A' 480 Volt Load Center feeder breaker trips open (Breaker fault). All other Load Centers remain energized.

(1.0 pt.)

a. No operator action will result in a RCP Seal Leakoff High Flow Alarm.

b. No operator action will result in 'B' RCP Motor Bearing High Temperature.

c. Above conditions would allow continued operation of 'B' RCP.

d. RCP 'B' should be tripped immediately because of RCP Shaft Seal Water Low ΔP.

Ans:

c.

Part B ORQ# 14 (cont.)

Ref: ONOP-1108.1, 3-OP-041.1, Dwg. 5610T-E-4503 Sh. 1

K/A:	003/000	A2.01	RO-3.5	SRO 3.9
		A2.02	RO-3.7	SRO-3.9
		K1.03	RO-3.3	SRO-3.6
		A1.01	RO-3.4	SRO-3.4

LP#0802056, EO 3

Est. Time of Completion 5 min.

Part B ORQ# 48

Initial Conditions:
The RCS was drained to 3 ft. below the vessel flange. The running RHR pump starts to exhibit oscillations in motor amps and flow.

Which one of the following is correct? (1.0 pts.)

a. Start the standby RHR pump and secure the running RHR pump.

b. Raise RCS level by cycling the ALT Low Head, MOV-872 open and closed.

c. Stop the running RHR pump and restore RCS level.

d. Start the standby RHR pump at minimum flow.

(1 point)
Ans: c.

Ref: ONOP-050, Step 5, 7, 8 & 14

K/A 006020 A4.01 (3.7/3.6)
005000 A4.01 (3.6/3.4)

LP#6902619

Est. Time of Completion 5 min.

Part B ORQ# 64

The unit is operating steady state at 50% power when the controlling pressurizer level channel fails high. Assuming no operator action, which of the following automatic sequence of actions would take place. (The assumption is all systems function as designed.) (1.0 pt)

a.
- Chg. Pp. speed initially increases
- Actual Pzr level increases
- Chg. Pp. speed then decreases
- Actual Pzr level then steadies out at a new level
- Plant operates at steady state

b.
- Chg. Pp. speed decreases
- Actual Pzr level decreases
- Letdown isolation occurs at 14% Pzr level
- Heaters go off
- Actual Pzr level increases
- Reactor Trips on 2/3 Pzr High Level

c.
- Chg. Pp. speed decreases
- All heaters come on due to Pzr level >5% above program
- Actual Pzr level begins to decrease
- Actual Pzr level decreases until Reactor trip on 2/3 low pressurizer level

d.
- Charging Pp. speed remains constant
- Pzr High Level alarm received
- Actual Pzr level steadies out at a new level
- Plant remains steady state

Part B ORQ# 64 (cont.)

Ans: b.
- Chg. Pp. speed decreases
- Actual Pzr level decreases
- Letdown isolation occurs at 14% Pzr level
- Heaters go off
- Actual Pzr level increases
- Reactor Trips on 2/3 Pzr High Level

Ref: 5610-T-D-15

K/A: 012 000 A301, (3.8/3.9)

LP#6902163, EO-2

Est. Time of Completion 2 min.

Part B ORQ# 74

While operating at 100% power, Turbine 1st Stage Pressure transmitter PT-447 fails high followed by a high failure of NIS Power Range Channel 4 (N-44). Which one of the following statements is correct with the given conditions. (1.0 pt)

a. Due to the failures the unit will be removed from service.

b. Even with the failures the rod control system continues to function properly in automatic.

c. Due to the failures, a manual turbine trip can be performed at 8% power and the reactor will not trip.

d. Even with the failures, Permissive P-8 functions normally.

Ans: (1.0 pt)

d. Even with the failures, Permissive P-8 functions normally.

Ref: *-ONOP-59.7, Tech Specs, O-ADM-021, 5610-T-L1, Sh. 17, 16, 5610-T-D-12A

K/A: 015/000, K3.01, (3.9/4.3)
015/000, K1.01, (4.1/4.2)

EO-3, LP-0802057

Est. Time of Completion 5 min.

Part B ORQ# 60

Given the following conditions, which one of the following statements is correct.

Conditions

- Chemistry Technician reports air ejector release concentration is 240 µci/cc.
- PRMS - R-15 has alarmed.
- Waste Gas Tank release is in progress
- Unit is at 75% and load is being increased on a 1% per minute ramp.

(Pts. 1.0)

a. The Waste Gas Tank release should have been terminated due to automatic action.

b. ONOP-041.3 "Excessive RCS leakage" should be entered.

c. ONOP-071 "Steam generator tube leak" should be entered.

d. The Technical Support Center should be activated.

Ans: (1.0 pt)

c. ONOP-071 "Steam generator tube leak" should be entered.

Part B ORQ# 60 (cont.)

Ref: 3-ONOP-067, Tech Specs, EP-20101, 3-ONOP-41.3, 3-ONOP-071

K/A: 000/060 EK3.01, (2.9/4.2)
EK3.03, (3.8/4.2)
EA2.01, (3.1/3.7)
SGA 12.0, (3.3/3.4)

EO-1, EO-2 and EO-3 LP-0802053

Est. Time of Completion 8 min.

Part B ORQ# 67

During normal operations at power you are notified that I&C will be working on the auctioneered Tavg input to the steam dump to condenser system; specifically temperature module TM-408. Which one of the following will best prevent an inadvertant steam dump valve opening, yet not eliminate any steam dump features unnecessarily. (1.0 pt)

a. Leave the switches in their present position.

b. Select Manual on the Mode Selector Switch and Auto on the Pressure Control Station.

c. Select Manual on the Mode Selector Switch and the Pressure Control Station.

d. Select Auto on the Mode Selector Switch and Manual on the Pressure control Station

Ans:

b. (Provides pressure control feature)

Ref: 5610-T-L1 Sh. 22A

K/A: 041 020 K6.03, (2.7/2.9)
041 020 A4.04, (2.7/2.7)
041 G 7, (2.8/3.0)

LP#6902909, EO-3

Est. Time of Completion 3 min.

Part B ORQ# 38

Given both units are operating in Mode 1 and in a normal system alignment. Upon the failure of heat tracing circuits 57A and 57B, which one of the following is the appropriate course of action. (1.0 pt)

a. Initiate a plant work order; maintain Mode 1 operations

b. Within one hour, perform the following realignment and demonstrate flowpath operability.

   1) Open valve 4-376, discharge from 4B BA transfer pump bypassing the Unit 4 BA filter
   2) Close valve 4-348, Unit 4 boric acid filter outlet

c. Within one hour make preparations for Unit 4 shutdown and within 6 hours ensure Unit 4 is in HOT STANDBY.

d. Commence repairs within 24 hours then if necessary enter into Tech Spec. 3.0.1.

Ans: Grader Note :

c. Ckts 57A & 57B service the line running from the Unit 4 BA filter outlet to the blender. Blockage of this line removes all sources of concentrated acid to Unit 4.

Ref: 5610-M-420-300, sheet 6
5610-T-E-4505, sheet 5
Technical Specification 3.6
O-ONOP-046.3, Loss of Boration Flowpaths

K/A004000K1.16 (3.3/3.5)
K/A004000K2.07 (2.7/3.2)
K/A004SGK05(3.1/3.7)

LP#6910233, EO 4, EO 5

Est. Time of Completion 8 min.

Part B ORQ# 190

Select the correct response to the following situation :

At 1:00AM on January 1st the '4C' Vital AC Inverter failed and its bus load automatically swapped to the CVT. At 6:00PM it was decided the '4C' Inverter could not be repaired and the bus load was transferred to the Spare Inverter at that time. Assuming no further equipment failures, how long may the unit operate in this configuration. (1.0 pt)

a. No action required other than a PWO written to repair 4C vital inverter. There are no time restraints associated with this inverter.

b. May operate for 24 hours from the time the transfer to the Spare Inverter was made (6:00 PM) at which time corrective action must be complete or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours and at least COLD SHUTDOWN within subsequent 24 hours.

c. May operate for 48 hours from the time the transfer to the Spare Inverter was made (6:00 PM).

d. May operate for 7 days because one channel of QSPDS will be out of service from the time the transfer to the Spare Inverter was made (6:00 PM).

Ans: d. May operate for 7 days because one channel of QSPDS will be out of service from the time the transfer to the Spare Inverter was made (6:00 PM).

Part B ORQ# 190 (cont.)

Ref: 5610-T-E-1592
0-ADM-021 Table 3.3.6 pg. 3-42
OP-003.1

K/A 012000 K1.01 (3.3/3.7)

LP#6902139 EO13

Est. Time of Completion 12 min.

Part B ORQ# 129

Which one of the following statements is correct? (1.0 pt)

During natural circulation conditions, core exit thermocouples (CET's):.

a. Read higher due to reduced flow past detectors.

b. Provide indication of the adequacy of core cooling.

c. Are faulty if reading >1000°F.

d. Are not accurate due to no flow condition in instrument bypass loop.

Ans:

b. Provide indication of the adequacy of core cooling.

Ref: Basis ES0.2

K/A 002000 A1.13 (3.4/4.0)
000011 EK1.01 (4.1/4.4)
000011 EA2.10 (4.5/4.7)
017020 A1.01 (3.7/3.9)

LP#3502073 EO 7 & 9

Est. Time of Completion 1 min

Part B ORQ# 153

Emergency operating procedures provide guidelines for isolating faulted steam generators and steam generators with tube ruptures. Which one of the following statements about isolation of steam generators is correct. (1.0 pt)

a. All feedwater is isolated to a faulted steam generator regardless of steam generator level.

b. All feedwater is isolated to a steam generator with a tube rupture regardless of steam generator level.

c. If a faulted steam generator is isolated and secondary radiation is abnormal, transition to E-1 "Loss of Reactor or Secondary Coolant" should be made.

d. Do not isolate a faulted steam generator sample lines because subsequent procedural guidance requires steam generator activity samples on the faulted steam generator.

Ans: a. All feedwater is isolated to a faulted steam generator regardless of steam generator level.

Ref: EOP-E2, EOP-E-3

K/A 000037 SGA11 (3.9/4.1)
000037 SGA12 (3.5/3.8)
000040 SGA11 (4.1/4.3)
000040 SGA12 (3.8/4.1)

LP#0063-OL App. Z EO-12
Est. Time of Completion 4 min.

Part B ORQ# 250

A Safety Injection has occurred. After exiting E-0, Reactor trip or Safety Injection, the following set of conditions is observed relevant to the Critical Safety Functions (CSF's):

Subcriticality:	NI-41 through NI-44 are all less than 5%. Intermediate Range startup rate is equal to +0.2 DPM.
Core Cooling:	No RCPs are in service. Core Exit Thermocouples read 700°F RCS subcooling based on Core Exit Thermocouples is 30°F RVLMS indicates a 50% level.
Heat Sink:	All Steam Generator levels are indicating 5% Narrow Range. Main Feed Water Pumps are tripped. Auxiliary feed water flows are: S/G 'A': Total flow 125 GPM S/G 'B': Total flow 125 GPM S/G 'C': Total flow 100 GPM
Containment:	Containment Pressure = 3.0 psig Containment Recir. Sump. Level = 400 inches

The other CSF status trees indicate only green or yellow paths.

Which procedure would you enter and why? (2.0 pts.)

Ans: Procedure FR-H.1, Response to Loss of Secondary Heat Sink, should be entered. This is highest priority Red Path.

Part B ORQ# 250 (cont.)

Ref: EOP F-0

K/A 000054 SGA11 (3.4/3.3)

LP#0063-OL, App. CC, EO10

Est. Time of Completion 5 min.

Part B ORQ# 255

The plant is responding to a small-break LOCA in accordance with EOP E-1, 'Loss of Reactor or Secondary Coolant'. Pressurizer level has risen continuously, even though the RCS pressure has been dropping steadily. All Reactor Coolant Pumps are in operation.

Which one of the following leak locations is consistent with the plant conditions just described? (1.0 pt)

a. Failure of a weld on RCP 'B' discharge piping.

b. Failure of pressurizer PORV in a full open position.

c. Failure of charging header connection to the RCS.

d. Failure of a weld on the pressurizer liquid space sample line.

Ans: b.

Ref: 5610-T-E-4501
EOP E-1

K/A 000008 EA2.20 (3.4/3.6)

LP#0063-OL, App. S, EO8

Est. Time of Completion 1 min.

Part B ORQ# 266

Select the one correct response based on the following : (1.0 pts.)

During FR-C.1 (Response to Inadequate Core Cooling) if attempts to establish adequate core cooling using the HHSI system are ineffective, the intact SGs are depressurized to 90 psig and then to atmospheric pressure.

The SGs are depressurized :

a. To allow use of the condensate pumps to supply feedwater to the SGs.

b. To cause the RCS to depressurize which improves core cooling by heat removal due to boiling.

c. To cause the RCS to depressurize which improves the ability of the RHR Pumps and Accumulators to deliver cooling to the core.

d. To cause the RCS to depressurize which improves the ability of the SI systems to add negative reactivity from borated water to insure adequate shutdown.

Ans: c. To cause the RCS to depressurize which improves the ability of the RHR Pumps and Accumulators to deliver cooling to the core.

Ref: FR-C.1 (Response to Inadequate Core Cooling) Basis Document

K/A 000074 EK1.03 (4.5/4.9)

LP#0063-OL, App. MM, EO9

Est. Time of Completion 3 min.

Part B ORQ# 267

In FR-C.1, Response to Inadequate Core Cooling, if CET temperatures are above 1200 degrees, the operator is directed in Step 18 to "Start available RCPs as necessary until CETs less than 1200 degrees".

Explain whether RCP's should be started if RCP support conditions cannot be established. (1.0 pts.)

Ans:

RCPs should be started even though support conditions cannot be established. The RCPs are sacrificed to save the core.

Ref: FR-C.1/Basis Document

K/A 000074 EK3.07 (4.0/4.4)
000074 EK3.11 (4.0/4.4)

LP#0063-OL, App. MM, EO9

Est. Time of Completion 2 min.

Part B ORQ# 270 Delete

A Primary LOCA has occurred inside of Unit 3 containment. The operating team is implementing E-O, Rx Trip/Safety Injection. The following unit conditions exist:

Containment Pressure is 10 psig
Containment Radiation level is 1.4×10^5 R/hr
RCS temperature is 508 degrees
RCS pressure is 1135 psig
Total HHSI flow to the core is 300 gpm.

Which one of the following is correct? (1.0 pt)

a. RCPs should be tripped because RCS subcooling is 42°F.

b. RCPs should not be tripped because RCS subcooling is >25°F.

c. RCPs should be tripped because Phase "B" isolation has occurred.

d. RCPs should be tripped because subcooling is <25°F.

Ans:

a. RCPs should be tripped because RCS subcooling is 42°F.

Ref: E-O Foldout Page/Steam Tables

K/A 000011 EK3.14 (4.1/4.2)
000009 EA2.01 (4.2/4.8)
017020 A4.02 (3.8/4.1)
000009 SGA 10 (4.3/4.3)

LP#0063-OL, App. M, EO2

Est. Time of Completion 3 min.

Part B ORQ# 237

a. Classify the following event. Specify the highest classifications, the category and part used to make the classification (e.g. classified as "Site Area Emergency" from category 12 'Loss of Power Conditions' Part 2.) (1.0 pt.)

Conditions:

- Large break LOCA in progress
- Containment Pressure had spiked to 30 psig and is being lowered by Containment Spray
- Phase 'A' and Phase 'B' properly activated
- Both CHRRM channels read 3×10^3 R/hr
- Both HHSI and RHR pump flow meters indicate flow to core
- Calculations show dose at site boundary to be 10 Rem Whole Body and 20 Rem Thyroid

b. With regard to the above event, which one of the following PARs is correct? (1.0 pt.)

1. Evacuate all sectors 0-2 miles, evacuate 2-10 miles downwind sectors, shelter remaining sectors.

2. Evacuate all sectors 0-2 and 2-5 miles, evacuate downwind and shelter remaining sectors 5-10 miles.

3. Shelter all sectors 0-10 miles

4. Evacuate downwind sectors and shelter remaining sectors 0-10 miles.

Ans: (2.0 pts)

a. GENERAL EMERGENCY (.075) Category 2 Part 1 (.025)
or Category 9 Part 1, 2, 3 or 4

b. 2

Part B ORQ# 237 (cont.)

Ref: E-Plan 20101 Table 1 & 2

K/A 194001 A1.16 (3.1/4.4)

LP#6902310 EO-10

Est. Time of Completion 10 min.

Part B ORQ# 381

Which one of the following describes a condition in which QPTR Tech. Spec. limitations are exceeded. (1.0 pt)

a. The reactor is at 100% power and QPTR = 1.01

b. While the reactor was at 100% power, QPTR was determined to be 1.08. Ninety (90) minutes after the determination reactor power is reduced to 90% and NIS power range High Flux setpoints reduced to 98%.

c. The reactor is at 45% power. QPTR has been determined to be 1.03 but the OPΔT, OTΔT, and NIS Power Range setpoints have been reduced to 55%.

d. QPTR was determined to be 1.04 but 90 minutes later F_q and $F_\Delta H$ were verified to be within limits and reactor power was reduced from 100% power to 91% power.

Ans:

b.

Ref: ADM-021, T.S. 3.2.6.h LER 87-18

K/A 015000 A1.04 (3.5/3.7)

LP# 0802163 EO-6

Est. Time of Completion 12 min.

Part B ORQ# 383

Classify the following event. Specify the highest classification, category and part used to make the classification (e.g. classified as "Unusual Event" from category 2 Primary Leakage / LOCA Part 3) (1.0 pts.)

Both Units are operating at 100% reactor power. 'A' Emergency Diesel Generator (EDG) is out of service for fuel pump repairs. A loss of all offsite AC power occurs and 'B' EDG fails to auto start. Ten minutes later, the turbine operator is able to locally start and load 'B' EDG.

Ans:

Site Area Emergency (0.75) Category 7 part 4 (0.25)

Ref: EP-20101 Table 1

K/A 000068 K1 (3.3/4.1)

LP# 6902252 EO-7

Est. Time of Completion 4 min.

Part B ORQ# 46

Given the following condition :

It is discovered during the shift that a log entry for entering Mode 1 at 10:35 was not made. It is now 11:15

Which one of the following set of log entries is correct : (1.0 pts.)

a.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
10:35 - Entered Mode 1

b.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
11:15 - Entered Mode 1 at 10:35

c.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
Late Entry 10:35 - Entered Mode 1

d.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
11:15 - Late Entry - Entered Mode 1 at 10:35

Ans:

c.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
Late Entry 10:35 - Entered Mode 1

Part B ORQ# 46 (cont.)

Ref: O-ADM-204, Step 5.1.4

K/A 194001 A1-06 (3.4/3.4)

LP#6902023, EO-1

Est. Time of Completion 5 min.

Part B ORQ# 240

Which of the following would result in a Safety Limit Violation (assume: 3 Loop Operation)? (1.0 pts.)

a. T_{ave} = 640°F
RCS Pressure = 2770 psig
Reactor power = 70%

b. T_{ave} = 630°F
RCS Pressure = 2290 psig
Reactor power = 70%

c. T_{ave} = 620°F
RCS Pressure = 2305 psig
Reactor power = 80%

d. T_{ave} = 615°F
RCS Pressure = 1840 psig
Reactor Power = 50%

Ans:

a. T_{ave} = 640°F
RCS Pressure = 2770 psig
Reactor power = 70%

Ref: Technical Specification Section 2
ADM-021 Section 2

K/A 002 SGK 5.0 (3.6/4.1)
WOG 002-009-002

LP#0802063
Est. Time of Completion 4 min.

WRITTEN EXAMINATION COVER SHEET

U.S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: Turkey Point

REACTOR TYPE: PWR - W

DATE AMINISTERED: 89/06/02

OPERATOR: _____

SECTION	CATEGORY VALUE	OPERATOR'S SCORE	% OF CATEGORY VALUE
A Plant Proficiency	_____	_____	_____
B Limits and Controls	_____	_____	_____

_____
Final Grade

*sf/May 31, 1989 10:36 AM

NRC RULES AND GUIDANCE FOR EXAMINEES

1. Use black ink or dark pencil ONLY to facilitate legible reproductions.

2. Print your name in the blank provided on the cover sheet of the examination.

3. Fill in the date on the cover sheet of the examination, if necessary.

4. Answer each question on the examination. If additional paper is required, use only the lined paper provided by the examiner.

5. Use abbreviations only if they are commonly used in facility literature.

6. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

7. Show all calculations, methods or assumptions used to obtain an answer to a mathematical problem, whether asked for in the question or not.

8. Unless solicited, the location of references need not be stated.

9. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWERS BLANK.

10. Proportional grading will be applied. Any additional wrong information that is provided may count against you. Fore example, if a question is worth one (1) point and asks for four responses, each of which is worth 0.25 points, and you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you got the four correct answers.

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LICENSED OPERATOR REQUALIFICATION

EXAM/QUESTION ANALYSIS WORKSHEET

RO

REQUAL YEAR: 88/89 CYCLE: Remedial

QUESTION NUMBER	POINT VALUE	POINTS MISSED	LESS THAN 70 %	QUESTION NUMBER	POINT VALUE	POINTS MISSED	LESS THAN 70 %
20	1						
34	1						
14	1						
48	1						
64	1						
74	1						
60	1						
67	1						
38	1						
190	1						
129	1						
153	1						
250	2						
255	1						
266	1						
267	1						
270	1						
237	2						
381	1						
383	1						
46	1						
240	1						

STUDENT'S NAME: _____

TOTAL POINTS AVAILABLE: 24

TOTAL POINTS MISSED: _____

TOTAL POINTS EARNED: _____

FINAL EXAM GRADE: _____

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ # 20

On Unit #3 determine the amount of reactivity necessary to overcome the power defect change for a power increase from 50% to 100% power by choosing the correct response. (Assume a boron concentration of 500 ppm.). (1.0 pt)

a. +647 PCM

b. +847 PCM

c. +927 PCM

d. +984 PCM

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 34

Unit 3 is operating at 90% reactor power. Due to a temperature increase of the circulating water the condenser backpressure increases from 3.5 to 4.0 inches hg. The output of the main generator would: (1.0 pt)

a. decrease from 608 Mwe to 595 Mwe.

b. increase from 595 Mwe to 608 Mwe.

c. would remain approximately constant because the turbine control valves would respond to maintain load.

d. decrease from 660 Mwe to 595 Mwe.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 14

Given the following conditions:

- Unit 3 in Mode 4 and on RHR
- 'A' RHR PP is cleared for maintenance
- 'B' RCP is running, 'A' and 'C' RCP's shutdown
- RCS conditions: Tavg = 300°F
  Pressure = 277 psig
  Przr level = 52%
- VCT pressure = 30 psig, VCT level in normal range
- 'A' charging Pp in service ('B' and 'C' charging Pps both cleared for maintenance)
- CCW system in normal Mode 4 lineup with no maintenance in progress.

Based on the above conditions which one of the following is most correct if '3A' 480 Volt Load Center feeder breaker trips open (Breaker fault). All other Load Centers remain energized.

(1.0 pt.)

a. No operator action will result in a RCP Seal Leakoff High Flow Alarm.

b. No operator action will result in 'B' RCP Motor Bearing High Temperature.

c. Above conditions would allow continued operation of 'B' RCP.

d. RCP 'B' should be tripped immediately because of RCP Shaft Seal Water Low ΔP.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 48

Initial Conditions:
The RCS was drained to 3 ft. below the vessel flange. The running RHR pump starts to exhibit oscillations in motor amps and flow.

Which one of the following is correct ? (1.0 pts.)

a. Start the standby RHR pump and secure the running RHR pump.

b. Raise RCS level by cycling the ALT Low Head, MOV-872 open and closed.

c. Stop the running RHR pump and restore RCS level.

d. Start the standby RHR pump at minimum flow.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 64

The unit is operating steady state at 50% power when the controlling pressurizer level channel fails high. Assuming no operator action, which of the following automatic sequence of actions would take place. (The assumption is all systems function as designed.) (1.0 pt)

a.
- Chg. Pp. speed initially increases
- Actual Pzr level increases
- Chg. Pp. speed then decreases
- Actual Pzr level then steadies out at a new level
- Plant operates at steady state

b.
- Chg. Pp. speed decreases
- Actual Pzr level decreases
- Letdown isolation occurs at 14% Pzr level
- Heaters go off
- Actual Pzr level increases
- Reactor Trips on 2/3 Pzr High Level

c.
- Chg. Pp. speed decreases
- All heaters come on due to Pzr level >5% above program
- Actual Pzr level begins to decrease
- Actual Pzr level decreases until Reactor trip on 2/3 low pressurizer level

d.
- Charging Pp. speed remains constant
- Pzr High Level alarm received
- Actual Pzr level steadies out at a new level
- Plant remains steady state

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 74

While operating at 100% power, Turbine 1st Stage Pressure transmitter PT-447 fails high followed by a high failure of NIS Power Range Channel 4 (N-44). Which one of the following statements is correct with the given conditions. (1.0 pt)

a. Due to the failures the unit will be removed from service.

b. Even with the failures the rod control system continues to function properly in automatic.

c. Due to the failures, a manual turbine trip can be performed at 8% power and the reactor will not trip.

d. Even with the failures, Permissive P-8 functions normally.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 60

Given the following conditions, which one of the following statements is correct.

Conditions

- Chemistry Technician reports air ejector release concentration is 240 µci/cc.
- PRMS - R-15 has alarmed.
- Waste Gas Tank release is in progress
- Unit is at 75% and load is being increased on a 1% per minute ramp.

(Pts. 1.0)

a. The Waste Gas Tank release should have been terminated due to automatic action.

b. ONOP-041.3 "Excessive RCS leakage" should be entered.

c. ONOP-071 "Steam generator tube leak" should be entered.

d. The Technical Support Center should be activated.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 67

During normal operations at power you are notified that I&C will be working on the auctioneered Tavg input to the steam dump to condenser system; specifically temperature module TM-408. Which one of the following will best prevent an inadvertant steam dump valve opening, yet not eliminate any steam dump features unnecessarily. (1.0 pt)

a. Leave the switches in their present position.

b. Select Manual on the Mode Selector Switch and Auto on the Pressure Control Station.

c. Select Manual on the Mode Selector Switch and the Pressure Control Station.

d. Select Auto on the Mode Selector Switch and Manual on the Pressure control Station

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 38

Given both units are operating in Mode 1 and in a normal system alignment. Upon the failure of heat tracing circuits 57A and 57B, which one of the following is the appropriate course of action. (1.0 pt)

a. Initiate a plant work order; maintain Mode 1 operations

b. Within one hour, perform the following realignment and demonstrate flowpath operability.

   1) Open valve 4-376, discharge from 4B BA transfer pump bypassing the Unit 4 BA filter
   2) Close valve 4-348, Unit 4 boric acid filter outlet

c. Within one hour make preparations for Unit 4 shutdown and within 6 hours ensure Unit 4 is in HOT STANDBY.

d. Commence repairs within 24 hours then if necessary enter into Tech Spec. 3.0.1.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 190

Select the correct response to the following situation :

At 1:00AM on January 1st the '4C' Vital AC Inverter failed and its bus load automatically swapped to the CVT. At 6:00PM it was decided the '4C' Inverter could not be repaired and the bus load was transferred to the Spare Inverter at that time. Assuming no further equipment failures, how long may the unit operate in this configuration. (1.0 pt)

a. No action required other than a PWO written to repair 4C vital inverter. There are no time restraints associated with this inverter.

b. May operate for 24 hours from the time the transfer to the Spare Inverter was made (6:00 PM) at which time corrective action must be complete or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours and at least COLD SHUTDOWN within subsequent 24 hours.

c. May operate for 48 hours from the time the transfer to the Spare Inverter was made (6:00 PM).

d. May operate for 7 days because one channel of QSPDS will be out of service from the time the transfer to the Spare Inverter was made (6:00 PM).

LOCT 88/89
REMEDIAL RO TRAINING FINAL

Part B ORQ# 129

Which one of the following statements is correct? (1.0 pt)

During natural circulation conditions, core exit thermocouples (CET's): .

a. Read higher due to reduced flow past detectors.

b. Provide indication of the adequacy of core cooling.

c. Are faulty if reading >1000°F.

d. Are not accurate due to no flow condition in instrument bypass loop.

LOCT 88/89
REMEDIAL RO TRAINING FINAL

Part B ORQ# 153

Emergency operating procedures provide guidelines for isolating faulted steam generators and steam generators with tube ruptures. Which one of the following statements about isolation of steam generators is correct. (1.0 pt)

a. All feedwater is isolated to a faulted steam generator regardless of steam generator level.

b. All feedwater is isolated to a steam generator with a tube rupture regardless of steam generator level.

c. If a faulted steam generator is isolated and secondary radiation is abnormal, transition to E-1 "Loss of Reactor or Secondary Coolant" should be made.

d. Do not isolate a faulted steam generator sample lines because subsequent procedural guidance requires steam generator activity samples on the faulted steam generator.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 250

A Safety Injection has occurred. After exiting E-0, Reactor trip or Safety Injection, the following set of conditions is observed relevant to the Critical Safety Functions (CSF's):

Subcriticality: NI-41 through NI-44 are all less than 5%. Intermediate Range startup rate is equal to +0.2 DPM.

Core Cooling: No RCPs are in service.
Core Exit Thermocouples read 700°F
RCS subcooling based on Core Exit Thermocouples is 30°F
RVLMS indicates a 50% level.

Heat Sink: All Steam Generator levels are indicating 5% Narrow Range.
Main Feed Water Pumps are tripped.
Auxiliary feed water flows are:

S/G 'A': Total flow 125 GPM
S/G 'B': Total flow 125 GPM
S/G 'C': Total flow 100 GPM

Containment: Containment Pressure = 3.0 psig
Containment Recir. Sump. Level = 400 inches

The other CSF status trees indicate only green or yellow paths.

Which procedure would you enter and why? (2.0 pts.)

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 255

The plant is responding to a small-break LOCA in accordance with EOP E-1, 'Loss of Reactor or Secondary Coolant'. Pressurizer level has risen continuously, even though the RCS pressure has been dropping steadily. All Reactor Coolant Pumps are in operation.

Which one of the following leak locations is consistent with the plant conditions just described? (1.0 pt)

a. Failure of a weld on RCP 'B' discharge piping.

b. Failure of pressurizer PORV in a full open position.

c. Failure of charging header connection to the RCS.

d. Failure of a weld on the pressurizer liquid space sample line.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 266

Select the one correct response based on the following : (1.0 pts.)

During FR-C.1 (Response to Inadequate Core Cooling) if attempts to establish adequate core cooling using the HHSI system are ineffective, the intact SGs are depressurized to 90 psig and then to atmospheric pressure.

The SGs are depressurized :

a. To allow use of the condensate pumps to supply feedwater to the SGs.

b. To cause the RCS to depressurize which improves core cooling by heat removal due to boiling.

c. To cause the RCS to depressurize which improves the ability of the RHR Pumps and Accumulators to deliver cooling to the core.

d. To cause the RCS to depressurize which improves the ability of the SI systems to add negative reactivity from borated water to insure adequate shutdown.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 267

In FR-C.1, Response to Inadequate Core Cooling, if CET temperatures are above 1200 degrees, the operator is directed in Step 18 to "Start available RCPs as necessary until CETs less than 1200 degrees".

Explain whether RCP's should be started if RCP support conditions cannot be established. (1.0 pts.)

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 270

A Primary LOCA has occurred inside of Unit 3 containment. The operating team is implementing E-O, Rx Trip/Safety Injection. The following unit conditions exist:

Containment Pressure is 10 psig
Containment Radiation level is 1.4×10^5 R/hr
RCS temperature is 508 degrees
RCS pressure is 1135 psig
Total HHSI flow to the core is 300 gpm.

Which one of the following is correct? (1.0 pt)

a. RCPs should be tripped because RCS subcooling is 42°F.

b. RCPs should not be tripped because RCS subcooling is >25°F.

c. RCPs should be tripped because Phase "B" isolation has occurred.

d. RCPs should be tripped because subcooling is <25°F.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 237

a. Classify the following event. Specify the highest classifications, the category and part used to make the classification (e.g. classified as "Site Area Emergency" from category 12 'Loss of Power Conditions' Part 2.) (1.0 pt.)

Conditions :

- Large break LOCA in progress
- Containment Pressure had spiked to 30 psig and is being lowered by Containment Spray
- Phase 'A' and Phase 'B' properly activated
- Both CHRRM channels read 3×10^3 R/hr
- Both HHSI and RHR pump flow meters indicate flow to core
- Calculations show dose at site boundary to be 10 Rem Whole Body and 20 Rem Thyroid

b. With regard to the above event , which one of the following PARs is correct ? (1.0 pt.)

1. Evacuate all sectors 0-2 miles, evacuate 2-10 miles downwind sectors, shelter remaining sectors.

2. Evacuate all sectors 0-2 and 2-5 miles, evacuate downwind and shelter remaining sectors 5-10 miles.

3. Shelter all sectors 0-10 miles

4. Evacuate downwind sectors and shelter remaining sectors 0-10 miles.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 381

Which one of the following describes a condition in which QPTR Tech. Spec. limitations are exceeded. (1.0 pt)

a. The reactor is at 100% power and QPTR = 1.01

b. While the reactor was at 100% power, QPTR was determined to be 1.08. Ninety (90) minutes after the determination reactor power is reduced to 90% and NIS power range High Flux setpoints reduced to 98%.

c. The reactor is at 45% power. QPTR has been determined to be 1.03 but the OPΔT, OTΔT, and NIS Power Range setpoints have been reduced to 55%.

d. QPTR was determined to be 1.04 but 90 minutes later F_q and $F_{\Delta}H$ were verified to be within limits and reactor power was reduced from 100% power to 91% power.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 383

Classify the following event. Specify the highest classification, category and part used to make the classification (e.g. classified as "Unusual Event" from category 2 Primary Leakage / LOCA Part 3) (1.0 pts.)

Both Units are operating at 100% reactor power. 'A' Emergency Diesel Generator (EDG) is out of service for fuel pump repairs. A loss of all offsite AC power occurs and 'B' EDG fails to auto start. Ten minutes later, the turbine operator is able to locally start and load 'B' EDG.

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 46

Given the following condition :

It is discovered during the shift that a log entry for entering Mode 1 at 10:35 was not made. It is now 11:15

Which one of the following set of log entries is correct : (1.0 pts.)

a.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
10:35 - Entered Mode 1

b.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
11:15 - Entered Mode 1 at 10:35

c.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
Late Entry 10:35 - Entered Mode 1

d.
09:30 - Normal Log Entry
09:55 - Normal Log Entry
10:58 - Normal Log Entry
11:15 - Late Entry - Entered Mode 1 at 10:35

LOCT 88/89

REMEDIAL RO TRAINING FINAL

Part B ORQ# 240

Which of the following would result in a Safety Limit Violation (assume: 3 Loop Operation) ? (1.0 pts.)

a. T_{ave} = 640°F
RCS Pressure = 2770 psig
Reactor power = 70%

b. T_{ave} = 630°F
RCS Pressure = 2290 psig
Reactor power = 70%

c. T_{ave} = 620°F
RCS Pressure = 2305 psig
Reactor power = 80%

d. T_{ave} = 615°F
RCS Pressure = 1840 psig
Reactor Power = 50%

NRC REQUALIFICATION EXAM

RAR ADD # 8810139-O-63
RAR IE # _____

I hereby certify that I will take the enclosed exam and abide by the rules and conduct provided by the proctor as specified by Administrative Guideline-017 (Implementation Phase - SAT Guideline)

Name: _____ (*print*) Signature: _____

Date: _____ SS# & CD#: _____

Total Points_____

_____ Instructor _____ Date _____ Program Coordinator/Supervisor _____ Date

SCORE: _____%

_____ Grader (Print Name) _____ Signature _____ Date

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STATIC EXAMINATION INITIAL CONDITIONS

15 MINUTES AGO, PLANT CONDITIONS WERE AS FOLLOWS:

- 3A ICW PUMP O.O.S.
- RCS ACTIVITY INCREASING SLOWLY, CHEMISTRY PERFORMING RCS ACTIVITY SAMPLES ON A ONCE PER 2 HOUR PERIODICITY

ALL OTHER PLANT CONDITIONS CURRENTLY INDICATED ON THE CONTROL BOARD HAVE OCCURRED IN THE LAST 15 MINUTES.

(1.0 pts.)

List the 2 signals which could have caused the auto start of the AFW PUMPS ?

Ans: Lo-Lo S/G level (0.5) and SI (0.5) ALSO ACCEPT S/G HIGH LEVEL F/W ISOLATION

Ref: Control Board Indications

KJ 6-2-89

K/A# 061-000K4.02 (4.5/4.6)
000-054EA2.03 (4.1/4.2)

Static Exam - NRC STAT 11

(1.0 pts.)
List the critical safety functions that are not fully satisfied, (ARE NOT GREEN PATHS).

Ans: Heat Sink, (.5) Core Cooling (0.5)

Ref: Control Board Indications, EOP-F-O

K/A# 000-027GEN.11 (3.4/3.3)

Static Exam - NRC STAT 11

(1.0 pts.)

If SI termination criteria were to be checked now, the criteria would :

a. Not be met

b. Be met for the existing conditions

c. Be met when RCS subcooling based on CET's exceeds 30°F

d. Be met when Pressurizer level exceeds 50%

Ans: a. Not be met (1.0)

Ref: 3-EOP-E-1

K/A# 000-054EK3.04 (4.4/4.6)

Static Exam - NRC STAT 11

(1.0 pts.)

What would be the proper action for the RO to take in order to stop the RCS depressurization?

535 AJ 6-2-89

Ans: Close the PORV block valve, MOV-~~356~~. (1.0)

Ref: 3-EOP-E-O

K/A# 000-008EA2.09 (3.6/3.7)

Static Exam - NRC STAT 11

(1.0 pts.)

Loop B T_{avg} ΔT indication is not consistent with the other RCS loop T_{avg} ΔT indications. Which of the following could cause this indication ?

a. T_h failed low

b. T_c failed low

c. T_h failed high

d. T_c failed high

Ans: d. T_c failed high (1.0)

Ref: Control Board Indications

K/A# 002-020K5.09 (3.6/3.9)

Static Exam - NRC STAT 11

(1.0 pts.)
Prior to the trip, the BOP received the following alarm annunciators C2/2 and C6/2.
What caused S/G level to be higher than program level ?

Ans: Feed regulating valve failed open. (1.0)

Ref: Control Board Indications

K/A# 000-054EA2.05 (3.5/3.7)

Static Exam - NRC STAT 11

(1.0 pts.) OT LJ 6-2-89

An ~~OP~~ DELTA T Turbine runback occurred prior to the trip: What parameter change caused the runback ?

a. Reactor power increasing

b. Delta T Increasing

c. Delta flux decreasing

d. Pressurizer pressure decreasing

Ans: d. Pressurizer pressure decreasing (1.0)

Ref: DDPS Printout, Control Board Indications

K/A# 000-003EK3.03 (3.4/3.7)

Static Exam - NRC STAT 11

(1.0 pts.)

Inward rod motion prior to the trip is indicated on the Bank "D" recorder. The rods were in Auto at that time. What caused the rods to move in ?

Ans: T_{avg} - T_{ref} deviation from instrument failure (1.0)

Ref: Control Board Indications

K/A# 000-003EA1.06 (4.0/4.1)

Static Exam - NRC STAT 111

(1.0 pts.)
What initiating condition (malfunction) caused the low pressure condition ?

Ans: PT-445 fail high (1.0)

Ref: Control Board Indications

K/A# 000-008EA2.03 (3.9/3.9)

Static Exam - NRC STAT 11

(1.0 pts.)

If offsite power were to be lost, and the "B" EDG failed to start, an eventual increase in CCW temperature would occur due to which one of the following conditions :

a. All CCW pumps would be lost

b. All ICW pumps would be lost

c. Intake temperature control valve fails closed

d. RHR system in operation

Ans: b. All ICW pumps would be lost (1.0)

Ref: Control Board Indications

K/A# 000-056EA1.07 / EA1.09 (3.2/3.2), (3.3/3.3)

Static Exam - NRC STAT 11

(1.0 pts.)
If the pressurizer pressure recorder were selected to the alternate channel (PC-444A), what would be the expected reading on the recorder.

Ans: 1700 psig. (off scale low) (1.0)

Ref: Control Board Indication

K/A# 000-008EA2.01 (3.9/4.2)

Static Exam - NRC STAT 11

(1.0 pts.)
Explain whether the Unit 4 SI pumps are required to be in service at this time.

Ans: The Unit 4 pumps are required to be in service, due to the failure of the 3A SI Pump to start (2 pumps required). (1.0)

Ref: Control Board Indication

K/A# 000-008EK3.03 (4.1/4.6)

Static Exam - NRC STAT 11

NRC REQUALIFICATION EXAM

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RAR IE # _____

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Name: _____ (*print*) Signature: _____

Date: _____ SS# & CD#: _____

Total Points _____

_____	_____	_____	_____
Instructor	Date	Program Coordinator/Supervisor	Date

SCORE: _____%

_____	_____	_____
Grader (Print Name)	Signature	Date

SCORE: _____%

_____	_____	_____
QC Grader (Print Name)	Signature	Date

NRC RULES AND GUIDANCE FOR EXAMINEES

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STATIC EXAMINATION INITIAL CONDITIONS

15 MINUTES AGO, PLANT CONDITIONS WERE AS FOLLOWS:

- 3A ICW PUMP O.O.S.
- RCS ACTIVITY INCREASING SLOWLY, CHEMISTRY PERFORMING RCS ACTIVITY SAMPLES ON A ONCE PER 2 HOUR PERIODICITY

ALL OTHER PLANT CONDITIONS CURRENTLY INDICATED ON THE CONTROL BOARD HAVE OCCURRED IN THE LAST 15 MINUTES.

(1.0 pts.)

Assuming that boron concentration has not changed from its value at 100% power, M.O.L., prior to the transient, choose the most correct statement with respect to reactivity effects from T_{avg} changing during the transient

a. Negative net effect due to ~~initial~~ CHANGE IN T_{avg} KJ 6-2-89

b. Positive net effect due to ~~initial~~ CHANGE IN T_{avg} KJ 6-2-89

c. No effect due to change in power defect

d. No effect due to change in MTC

Ans: b. Positive net effect due to initial T_{avg} (1.0)

Ref: Core Physics Manual

K/A # 000-003EK1.03 (3.5/3.8)

Static Exam - NRC STAT 13

(1.5 pts.)

HAVE BEEN LJ 6-2-89

Circle the reactor protection / safeguards trips which ~~are~~ affected by the failure of PT-486.

a. Steamline high ΔP S.I.
b. High steam flow S.I.
c. High steam generator level
d. Steam flow / feed flow reactor trip

Ans: a. Steamline high ΔP S.I. (0.5)
b. High steam flow S.I. (0.5)
d. Steam flow / feed flow reactor trip (0.5)

Ref: 5610-T-0-188 Sht. 1 OF 1

K/A# 012-000K4.02 (3.9 / 4.3)

Static Exam - NRC STAT 13

(1.0 pts.)

List which rods must be placed in disconnect to allow the recovery of the dropped rod.

Ans: H4 (0.25)
D4 (0.25)
6-2-89 M-~~18~~ 8 (0.25)
H8 (0.25)

Ref: Control Rod Disconnect Box

K/A# 000-003EA1.02 (3.6/3.4)

Static Exam - NRC STAT 13

(1.0 pts.)

Chemistry informs you that RCS Activity is 220 µc/cc DE I-131. Classify this event. List applicable level, paragraph and item number.

Ans: Unusual Event (0.5)
EP-20101 Table 1 paragraph 8 - Fuel Element Failure (0.5)

Ref: Control Board Indications, EP-20101

K/A# 000-076EK3.06 (3.2/3.8)

Static Exam - NRC STAT 13

(1.0 pts.)
Given current plant conditions what would be the consequences of PT-466 failing low? (Assume No Operator Actions are Taken)

a. High ΔP SI

b. High Steam Flow SI

c. No effect maintain power operations

d. T/S Condition 3.0.1. met, plant S/D required.

Ans: d. T/S Condition 3.0.1. met, plant S/D required. (1.0)
(Cannot remove the channel from service without initiating protective action)

Ref: 5610-T-0-188 Sht. 1 of 1

K/A# 012-000K4.01 (3.7/4.0)

Static Exam - NRC STAT 13

(1.5 pts.)
Given all current plant conditions, circle each of the below listed ONOP'S which may be applicable at this time.

a. ONOP-028.3 - Dropped RCC

b. ONOP-041.3 - Excessive RCS Leakage

c. ONOP-041.4 - Excessive RCS Activity

d. ONOP-059.4 - Excessive Axial Flux Difference

e. ONOP-059.8 - Power Range NIS Malfunction

f. ONOP-089 - Turbine Runback

(NOTE : More than one answer may be correct)

KJ 6-2-89

Ans: a, c, & f (0.5 each) ALSO ACCEPT "d.", BUT REQUIRE A, C + F
Proportional grading applies for incorrect answers.

Ref: ONOP Books, Plant Conditions

K/A# 000-003EK3.04 (3.8 / 4.1)
015-000A1.04 / A1.05 (3.5 / 3.7) (3.7 / 3.9)

Static Exam - NRC STAT 13

(1.0 pts.)
Explain whether emergency boration is required due to present plant conditions.

Ans: Emergency boration is required due to extra low rod insertion limit (1.0)

Ref: Control Board Indications, ONOP-059.4

K/A# 001-050A2.06 (3.6 / 4.0)

Static Exam - NRC STAT 13

(1.0 pts.)

If two additional rods were to fall into the core, what action would be required ?

Ans: Manually trip the reactor. (1.0)

Ref: ONOP-028.3

K/A# 000-003EK3.04 (3.8 / 4.1)

Static Exam - NRC STAT 13

(0.5 pts.)
What turbine control oil system is currently in control of turbine load ?

Ans: Load Limit (0.5)

Ref: Control Board Indications

K/A# 000-003EK3.03 (3.4 / 3.7)

Static Exam - NRC STAT 13

(1.0 pts.)

All pressurizer heaters are energized. This is as a result of which of the following :.

a. Insurge to the pressurizer

b. Low pressurizer pressure

c. Manual operator action

d. Low pressurizer level deviation

Ans: b. Low pressurizer pressure (1.0)

Ref: Control Board Indications

K/A# 010-000A4.02 (3.6 / 3.4)

Static Exam - NRC STAT 13

(1.0 pts.)
Given current plant conditions, explain the operator actions required to clear annunciator C 8/3.

Ans: Steam dump to condenser mode select switch to reset (1.0)

Ref: Control Board Indications

K/A# 000-003EK3.03 (3.4 / 3.7)

Static Exam - NRC STAT 13

WRITTEN EXAMINATION COVER SHEET

U.S. NUCLEAR REGULATORY COMMISSION

(SENIOR) REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: Turkey Point

REACTOR TYPE: PWR - W

DATE AMINISTERED: 89/06/02

OPERATOR: _____

SECTION		CATEGORY VALUE	OPERATOR'S SCORE	% OF CATEGORY VALUE
A	Plant Proficiency	_____	_____	_____
B	Limits and Controls	_____	_____	_____

_____
Final Grade

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LICENSED OPERATOR REQUALIFICATION

EXAM/QUESTION ANALYSIS WORKSHEET

SRO

REQUAL YEAR: 88/89 CYCLE: Remedial

QUESTION NUMBER	POINT VALUE	POINTS MISSED	LESS THAN 70 %	QUESTION NUMBER	POINT VALUE	POINTS MISSED	LESS THAN 70 %
20	1						
34	1						
14	1						
48	1						
64	1						
74	1						
60	1						
67	1						
38	1						
190	1						
129	1						
153	1						
250	2						
255	1						
266	1						
267	1						
270	1						
237	2						
381	1						
383	1						
297	1						
259	1						

STUDENT'S NAME:

TOTAL POINTS AVAILABLE: 24

TOTAL POINTS MISSED:

TOTAL POINTS EARNED:

FINAL EXAM GRADE:

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ # 20

On Unit #3 determine the amount of reactivity necessary to overcome the power defect change for a power increase from 50% to 100% power by choosing the correct response. (Assume a boron concentration of 500 ppm.). (1.0 pt)

a. +647 PCM

b. +847 PCM

c. +927 PCM

d. +984 PCM

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 34

Unit 3 is operating at 90% reactor power. Due to a temperature increase of the circulating water the condenser backpressure increases from 3.5 to 4.0 inches hg. The output of the main generator would: (1.0 pt)

a. decrease from 608 Mwe to 595 Mwe.

b. increase from 595 Mwe to 608 Mwe.

c. would remain approximately constant because the turbine control valves would respond to maintain load.

d. decrease from 660 Mwe to 595 Mwe.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 14

Given the following conditions:

- Unit 3 in Mode 4 and on RHR
- 'A' RHR PP is cleared for maintenance
- 'B' RCP is running, 'A' and 'C' RCP's shutdown
- RCS conditions: Tavg = 300°F
  Pressure = 277 psig
  Przr level = 52%
- VCT pressure = 30 psig, VCT level in normal range
- 'A' charging Pp in service ('B' and 'C' charging Pps both cleared for maintenance)
- CCW system in normal Mode 4 lineup with no maintenance in progress.

Based on the above conditions which one of the following is most correct if '3A' 480 Volt Load Center feeder breaker trips open (Breaker fault). All other Load Centers remain energized.

(1.0 pt.)

a. No operator action will result in a RCP Seal Leakoff High Flow Alarm.

b. No operator action will result in 'B' RCP Motor Bearing High Temperature.

c. Above conditions would allow continued operation of 'B' RCP.

d. RCP 'B' should be tripped immediately because of RCP Shaft Seal Water Low ΔP.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 48

Initial Conditions:
The RCS was drained to 3 ft. below the vessel flange. The running RHR pump starts to exhibit oscillations in motor amps and flow.

Which one of the following is correct? (1.0 pts.)

a. Start the standby RHR pump and secure the running RHR pump.

b. Raise RCS level by cycling the ALT Low Head, MOV-872 open and closed.

c. Stop the running RHR pump and restore RCS level.

d. Start the standby RHR pump at minimum flow.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 64

The unit is operating steady state at 50% power when the controlling pressurizer level channel fails high. Assuming no operator action, which of the following automatic sequence of actions would take place. (The assumption is all systems function as designed.) (1.0 pt)

a.
- Chg. Pp. speed initially increases
- Actual Pzr level increases
- Chg. Pp. speed then decreases
- Actual Pzr level then steadies out at a new level
- Plant operates at steady state

b.
- Chg. Pp. speed decreases
- Actual Pzr level decreases
- Letdown isolation occurs at 14% Pzr level
- Heaters go off
- Actual Pzr level increases
- Reactor Trips on 2/3 Pzr High Level

c.
- Chg. Pp. speed decreases
- All heaters come on due to Pzr level >5% above program
- Actual Pzr level begins to decrease
- Actual Pzr level decreases until Reactor trip on 2/3 low pressurizer level

d.
- Charging Pp. speed remains constant
- Pzr High Level alarm received
- Actual Pzr level steadies out at a new level
- Plant remains steady state

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 74

While operating at 100% power, Turbine 1st Stage Pressure transmitter PT-447 fails high followed by a high failure of NIS Power Range Channel 4 (N-44). Which one of the following statements is correct with the given conditions. (1.0 pt)

a. Due to the failures the unit will be removed from service.

b. Even with the failures the rod control system continues to function properly in automatic.

c. Due to the failures, a manual turbine trip can be performed at 8% power and the reactor will not trip.

d. Even with the failures, Permissive P-8 functions normally.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 60

Given the following conditions, which one of the following statements is correct.

Conditions

- Chemistry Technician reports air ejector release concentration is 240 µci/cc.
- PRMS - R-15 has alarmed.
- Waste Gas Tank release is in progress
- Unit is at 75% and load is being increased on a 1% per minute ramp.

(Pts. 1.0)

a. The Waste Gas Tank release should have been terminated due to automatic action.

b. ONOP-041.3 "Excessive RCS leakage" should be entered.

c. ONOP-071 "Steam generator tube leak" should be entered.

d. The Technical Support Center should be activated.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 67

During normal operations at power you are notified that I&C will be working on the auctioneered Tavg input to the steam dump to condenser system; specifically temperature module TM-408. Which one of the following will <u>best</u> prevent an inadvertant steam dump valve opening, yet not eliminate any steam dump features unnecessarily. (1.0 pt)

a. Leave the switches in their present position.

b. Select Manual on the Mode Selector Switch and Auto on the Pressure Control Station.

c. Select Manual on the Mode Selector Switch and the Pressure Control Station.

d. Select Auto on the Mode Selector Switch and Manual on the Pressure control Station

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 38

Given both units are operating in Mode 1 and in a normal system alignment. Upon the failure of heat tracing circuits 57A and 57B, which one of the following is the appropriate course of action. (1.0 pt)

a. Initiate a plant work order; maintain Mode 1 operations

b. Within one hour, perform the following realignment and demonstrate flowpath operability.

   1) Open valve 4-376, discharge from 4B BA transfer pump bypassing the Unit 4 BA filter
   2) Close valve 4-348, Unit 4 boric acid filter outlet

c. Within one hour make preparations for Unit 4 shutdown and within 6 hours ensure Unit 4 is in HOT STANDBY.

d. Commence repairs within 24 hours then if necessary enter into Tech Spec. 3.0.1.

LOCT 88/89
REMEDIAL SRO TRAINING FINAL

Part B ORQ# 190

Select the correct response to the following situation :

At 1:00AM on January 1st the '4C' Vital AC Inverter failed and its bus load automatically swapped to the CVT. At 6:00PM it was decided the '4C' Inverter could not be repaired and the bus load was transferred to the Spare Inverter at that time. Assuming no further equipment failures, how long may the unit operate in this configuration. (1.0 pt)

a. No action required other than a PWO written to repair 4C vital inverter. There are no time restraints associated with this inverter.

b. May operate for 24 hours from the time the transfer to the Spare Inverter was made (6:00 PM) at which time corrective action must be complete or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours and at least COLD SHUTDOWN within subsequent 24 hours.

c. May operate for 48 hours from the time the transfer to the Spare Inverter was made (6:00 PM).

d. May operate for 7 days because one channel of QSPDS will be out of service from the time the transfer to the Spare Inverter was made (6:00 PM).

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 129

Which one of the following statements is correct? (1.0 pt)

During natural circulation conditions, core exit thermocouples (CET's):.

a. Read higher due to reduced flow past detectors.

b. Provide indication of the adequacy of core cooling.

c. Are faulty if reading >1000°F.

d. Are not accurate due to no flow condition in instrument bypass loop.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 153

Emergency operating procedures provide guidelines for isolating faulted steam generators and steam generators with tube ruptures. Which one of the following statements about isolation of steam generators is correct. (1.0 pt)

a. All feedwater is isolated to a faulted steam generator regardless of steam generator level.

b. All feedwater is isolated to a steam generator with a tube rupture regardless of steam generator level.

c. If a faulted steam generator is isolated and secondary radiation is abnormal, transition to E-1 "Loss of Reactor or Secondary Coolant" should be made.

d. Do not isolate a faulted steam generator sample lines because subsequent procedural guidance requires steam generator activity samples on the faulted steam generator.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 250

A Safety Injection has occurred. After exiting E-0, Reactor trip or Safety Injection, the following set of conditions is observed relevant to the Critical Safety Functions (CSF's):

Subcriticality:	NI-41 through NI-44 are all less than 5%. Intermediate Range startup rate is equal to +0.2 DPM.
Core Cooling:	No RCPs are in service. Core Exit Thermocouples read 700°F RCS subcooling based on Core Exit Thermocouples is 30°F RVLMS indicates a 50% level.
Heat Sink:	All Steam Generator levels are indicating 5% Narrow Range. Main Feed Water Pumps are tripped. Auxiliary feed water flows are: S/G 'A': Total flow 125 GPM S/G 'B': Total flow 125 GPM S/G 'C': Total flow 100 GPM
Containment:	Containment Pressure = 3.0 psig Containment Recir. Sump. Level = 400 inches

The other CSF status trees indicate only green or yellow paths.

Which procedure would you enter and why? (2.0 pts.)

LOCT 88/89
REMEDIAL SRO TRAINING FINAL

Part B ORQ# 255

The plant is responding to a small-break LOCA in accordance with EOP E-1, 'Loss of Reactor or Secondary Coolant'. Pressurizer level has risen continuously, even though the RCS pressure has been dropping steadily. All Reactor Coolant Pumps are in operation.

Which one of the following leak locations is consistent with the plant conditions just described? (1.0 pt)

a. Failure of a weld on RCP 'B' discharge piping.

b. Failure of pressurizer PORV in a full open position.

c. Failure of charging header connection to the RCS.

d. Failure of a weld on the pressurizer liquid space sample line.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 266

Select the one correct response based on the following : (1.0 pts.)

During FR-C.1 (Response to Inadequate Core Cooling) if attempts to establish adequate core cooling using the HHSI system are ineffective, the intact SGs are depressurized to 90 psig and then to atmospheric pressure.

The SGs are depressurized :

a. To allow use of the condensate pumps to supply feedwater to the SGs.

b. To cause the RCS to depressurize which improves core cooling by heat removal due to boiling.

c. To cause the RCS to depressurize which improves the ability of the RHR Pumps and Accumulators to deliver cooling to the core.

d. To cause the RCS to depressurize which improves the ability of the SI systems to add negative reactivity from borated water to insure adequate shutdown.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 267

In FR-C.1, Response to Inadequate Core Cooling, if CET temperatures are above 1200 degrees, the operator is directed in Step 18 to "Start available RCPs as necessary until CETs less than 1200 degrees".

Explain whether RCP's should be started if RCP support conditions cannot be established. (1.0 pts.)

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 270

A Primary LOCA has occurred inside of Unit 3 containment. The operating team is implementing E-O, Rx Trip/Safety Injection. The following unit conditions exist:

Containment Pressure is 10 psig
Containment Radiation level is 1.4×10^5 R/hr
RCS temperature is 508 degrees
RCS pressure is 1135 psig
Total HHSI flow to the core is 300 gpm.

Which one of the following is correct? (1.0 pt)

a. RCPs should be tripped because RCS subcooling is 42°F.

b. RCPs should not be tripped because RCS subcooling is >25°F.

c. RCPs should be tripped because Phase "B" isolation has occurred.

d. RCPs should be tripped because subcooling is <25°F.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 237

a. Classify the following event. Specify the highest classifications, the category and part used to make the classification (e.g. classified as "Site Area Emergency" from category 12 'Loss of Power Conditions' Part 2.) (1.0 pt.)

Conditions :

- Large break LOCA in progress
- Containment Pressure had spiked to 30 psig and is being lowered by Containment Spray
- Phase 'A' and Phase 'B' properly activated
- Both CHRRM channels read 3×10^3 R/hr
- Both HHSI and RHR pump flow meters indicate flow to core
- Calculations show dose at site boundary to be 10 Rem Whole Body and 20 Rem Thyroid

b. With regard to the above event, which one of the following PARs is correct? (1.0 pt.)

1. Evacuate all sectors 0-2 miles, evacuate 2-10 miles downwind sectors, shelter remaining sectors.

2. Evacuate all sectors 0-2 and 2-5 miles, evacuate downwind and shelter remaining sectors 5-10 miles.

3. Shelter all sectors 0-10 miles

4. Evacuate downwind sectors and shelter remaining sectors 0-10 miles.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 381

Which one of the following describes a condition in which QPTR Tech. Spec. limitations are exceeded. (1.0 pt)

a. The reactor is at 100% power and QPTR = 1.01

b. While the reactor was at 100% power, QPTR was determined to be 1.08. Ninety (90) minutes after the determination reactor power is reduced to 90% and NIS power range High Flux setpoints reduced to 98%.

c. The reactor is at 45% power. QPTR has been determined to be 1.03 but the OPΔT, OTΔT, and NIS Power Range setpoints have been reduced to 55%.

d. QPTR was determined to be 1.04 but 90 minutes later F_q and $F_{\Delta}H$ were verified to be within limits and reactor power was reduced from 100% power to 91% power.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 383

Classify the following event. Specify the highest classification, category and part used to make the classification (e.g. classified as "Unusual Event" from category 2 Primary Leakage / LOCA Part 3)

> Both Units are operating at 100% reactor power. 'A' Emergency Diesel Generator (EDG) is out of service for fuel pump repairs. A loss of all offsite AC power occurs and 'B' EDG fails to auto start. Ten minutes later, the turbine operator is able to locally start and load 'B' EDG.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 297

(1.0 pt.)
The remotely operated steam generator blowdown isolation valve, CV-4-6275B, failed shut. Which one of the following correctly describes the actions required.

a. This valve must be de-energized within 4 hours, then normal operations can continue.

b. Since the valve failed shut, the affected penetration is isolated and, therefore normal operations can continue.

c. Automatic isolation valve 4-CV6278B must be shut within 4 hours.

d. The Key Lock Switch on VPB for 4-6275B must be placed in the Bypass position within 4 hours, then normal operation can continue.

LOCT 88/89

REMEDIAL SRO TRAINING FINAL

Part B ORQ# 259

The reactor is in Mode 3 in preparation for reactor startup. The Intermediate Range Nuclear Instrumentation Analog Channel Operational Test has just been completed. A review of the results indicates that the final 'as-left' setting for the Intermediate Range neutron flux trip for Channel N36 is equivalent to 28% power.

Which one of the following statements describes the required actions for this situation? (1.0 pt)

a. Correct the level trip setpoint to the current equivalent to 25% rated thermal power prior to increasing thermal power above the P-6 setpoint.

b. Correct the level trip setpoint to the current equivalent to 25% rated thermal power prior to exceeding 25% Power.

c. Correct the level trip setpoint to the current equivalent to 25% rated thermal power prior to increasing thermal power above the P-10 setpoint.

d. Place Channel N36 in a tripped condition within 2 hours, then proceed with the startup

WRITTEN EXAMINATION COVER SHEET

ANSWER KEY

U.S. NUCLEAR REGULATORY COMMISSION

(SENIOR) REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: Turkey Point

REACTOR TYPE: PWR - W

DATE AMINISTERED: 89/06/02

OPERATOR: _____

SECTION		CATEGORY VALUE	OPERATOR'S SCORE	% OF CATEGORY VALUE
A	Plant Proficiency	_____	_____	_____
B	Limits and Controls	_____	_____	_____

_____

Final Grade

Part B ORQ # 20

On Unit #3 determine the amount of reactivity necessary to overcome the power defect change for a power increase from 50% to 100% power by choosing the correct response. (Assume a boron concentration of 500 ppm.). (1.0 pt)

a. +647 PCM

b. +847 PCM

c. +927 PCM

d. +984 PCM

Ans: b. +847 PCM

(100 - 50%)
(1774pcm - 927pcm)
= 847pcm
From curve book

Ref: PCB Unit #3 Section II Figure 6A

K/A: #192004.K1.13(2.9/2.9)

LP#0010, Appendix E, EO 2

Est. Time of Completion 2 min.

Part B ORQ# 34

Unit 3 is operating at 90% reactor power. Due to a temperature increase of the circulating water the condenser backpressure increases from 3.5 to 4.0 inches hg. The output of the main generator would: (1.0 pt)

a. decrease from 608 Mwe to 595 Mwe.

b. increase from 595 Mwe to 608 Mwe.

c. would remain approximately constant because the turbine control valves would respond to maintain load.

d. decrease from 660 Mwe to 595 Mwe.

Ans :

a. decrease from 608 Mwe to 595 Mwe.

Ref: Unit 3 PCB Section I Figure 1

K/A #045000 K5.05(1.9/2.1)

LP#3502077, EO-11

Est. Time of Completion 2 min.

Part B ORQ# 14

Given the following conditions:

- Unit 3 in Mode 4 and on RHR
- 'A' RHR PP is cleared for maintenance
- 'B' RCP is running, 'A' and 'C' RCP's shutdown
- RCS conditions: Tavg = 300°F
  Pressure = 277 psig
  Przr level = 52%
- VCT pressure = 30 psig, VCT level in normal range
- 'A' charging Pp in service ('B' and 'C' charging Pps both cleared for maintenance)
- CCW system in normal Mode 4 lineup with no maintenance in progress.

Based on the above conditions which one of the following is most correct if '3A' 480 Volt Load Center feeder breaker trips open (Breaker fault). All other Load Centers remain energized.

(1.0 pt.)

a. No operator action will result in a RCP Seal Leakoff High Flow Alarm.

b. No operator action will result in 'B' RCP Motor Bearing High Temperature.

c. Above conditions would allow continued operation of 'B' RCP.

d. RCP 'B' should be tripped immediately because of RCP Shaft Seal Water Low ΔP.

Ans:

c.

Part B ORQ# 14 (cont.)

Ref: ONOP-1108.1, 3-OP-041.1, Dwg. 5610T-E-4503 Sh. 1

K/A:	003/000	A2.01	RO-3.5	SRO 3.9
		A2.02	RO-3.7	SRO-3.9
		K1.03	RO-3.3	SRO-3.6
		A1.01	RO-3.4	SRO-3.4

LP#0802056, EO 3

Est. Time of Completion 5 min.

Part B ORQ# 48

Initial Conditions:
The RCS was drained to 3 ft. below the vessel flange. The running RHR pump starts to exhibit oscillations in motor amps and flow.

Which one of the following is correct ? (1.0 pts.)

a. Start the standby RHR pump and secure the running RHR pump.

b. Raise RCS level by cycling the ALT Low Head, MOV-872 open and closed.

c. Stop the running RHR pump and restore RCS level.

d. Start the standby RHR pump at minimum flow.

(1 point)
Ans: c.

Ref: ONOP-050, Step 5, 7, 8 & 14

K/A 006020 A4.01 (3.7/3.6)
005000 A4.01 (3.6/3.4)

LP#6902619

Est. Time of Completion 5 min.

Part B ORQ# 64

The unit is operating steady state at 50% power when the controlling pressurizer level channel fails high. Assuming no operator action, which of the following automatic sequence of actions would take place. (The assumption is all systems function as designed.) (1.0 pt)

a.
- Chg. Pp. speed initially increases
- Actual Pzr level increases
- Chg. Pp. speed then decreases
- Actual Pzr level then steadies out at a new level
- Plant operates at steady state

b.
- Chg. Pp. speed decreases
- Actual Pzr level decreases
- Letdown isolation occurs at 14% Pzr level
- Heaters go off
- Actual Pzr level increases
- Reactor Trips on 2/3 Pzr High Level

c.
- Chg. Pp. speed decreases
- All heaters come on due to Pzr level >5% above program
- Actual Pzr level begins to decrease
- Actual Pzr level decreases until Reactor trip on 2/3 low pressurizer level

d.
- Charging Pp. speed remains constant
- Pzr High Level alarm received
- Actual Pzr level steadies out at a new level
- Plant remains steady state

Part B ORQ# 64 (cont.)

Ans: b.
- Chg. Pp. speed decreases
- Actual Pzr level decreases
- Letdown isolation occurs at 14% Pzr level
- Heaters go off
- Actual Pzr level increases
- Reactor Trips on 2/3 Pzr High Level

Ref: 5610-T-D-15

K/A: 012 000 A301, (3.8/3.9)

LP#6902163, EO-2

Est. Time of Completion 2 min.

Part B ORQ# 74

While operating at 100% power, Turbine 1st Stage Pressure transmitter PT-447 fails high followed by a high failure of NIS Power Range Channel 4 (N-44). Which one of the following statements is correct with the given conditions. (1.0 pt)

a. Due to the failures the unit will be removed from service.

b. Even with the failures the rod control system continues to function properly in automatic.

c. Due to the failures, a manual turbine trip can be performed at 8% power and the reactor will not trip.

d. Even with the failures, Permissive P-8 functions normally.

Ans: (1.0 pt)

d. Even with the failures, Permissive P-8 functions normally.

Ref: *-ONOP-59.7, Tech Specs, O-ADM-021, 5610-T-L1, Sh. 17, 16, 5610-T-D-12A

K/A: 015/000, K3.01, (3.9/4.3)
015/000, K1.01, (4.1/4.2)

EO-3, LP-0802057

Est. Time of Completion 5 min.

Part B ORQ# 60

Given the following conditions, which one of the following statements is correct.

Conditions

- Chemistry Technician reports air ejector release concentration is 240 µci/cc.
- PRMS - R-15 has alarmed.
- Waste Gas Tank release is in progress
- Unit is at 75% and load is being increased on a 1% per minute ramp.

(Pts. 1.0)

a. The Waste Gas Tank release should have been terminated due to automatic action.

b. ONOP-041.3 "Excessive RCS leakage" should be entered.

c. ONOP-071 "Steam generator tube leak" should be entered.

d. The Technical Support Center should be activated.

Ans: (1.0 pt)

c. ONOP-071 "Steam generator tube leak" should be entered.

Part B ORQ# 60 (cont.)

Ref: 3-ONOP-067, Tech Specs, EP-20101, 3-ONOP-41.3, 3-ONOP-071

K/A: 000/060 EK3.01, (2.9/4.2)
EK3.03, (3.8/4.2)
EA2.01, (3.1/3.7)
SGA 12.0, (3.3/3.4)

EO-1, EO-2 and EO-3 LP-0802053

Est. Time of Completion 8 min.

Part B ORQ# 67

During normal operations at power you are notified that I&C will be working on the auctioneered Tavg input to the steam dump to condenser system; specifically temperature module TM-408. Which one of the following will best prevent an inadvertant steam dump valve opening, yet not eliminate any steam dump features unnecessarily. (1.0 pt)

a. Leave the switches in their present position.

b. Select Manual on the Mode Selector Switch and Auto on the Pressure Control Station.

c. Select Manual on the Mode Selector Switch and the Pressure Control Station.

d. Select Auto on the Mode Selector Switch and Manual on the Pressure control Station

Ans:

b. (Provides pressure control feature)

Ref: 5610-T-L1 Sh. 22A

K/A: 041 020 K6.03, (2.7/2.9)
041 020 A4.04, (2.7/2.7)
041 G 7, (2.8/3.0)

LP#6902909, EO-3

Est. Time of Completion 3 min.

Part B ORQ# 38

Given both units are operating in Mode 1 and in a normal system alignment. Upon the failure of heat tracing circuits 57A and 57B, which one of the following is the appropriate course of action. (1.0 pt)

a. Initiate a plant work order; maintain Mode 1 operations

b. Within one hour, perform the following realignment and demonstrate flowpath operability.

   1) Open valve 4-376, discharge from 4B BA transfer pump bypassing the Unit 4 BA filter
   2) Close valve 4-348, Unit 4 boric acid filter outlet

c. Within one hour make preparations for Unit 4 shutdown and within 6 hours ensure Unit 4 is in HOT STANDBY.

d. Commence repairs within 24 hours then if necessary enter into Tech Spec. 3.0.1.

Ans: Grader Note :

c. Ckts 57A & 57B service the line running from the Unit 4 BA filter outlet to the blender. Blockage of this line removes all sources of concentrated acid to Unit 4.

Ref: 5610-M-420-300, sheet 6
5610-T-E-4505, sheet 5
Technical Specification 3.6
O-ONOP-046.3, Loss of Boration Flowpaths

K/A004000K1.16 (3.3/3.5)
K/A004000K2.07 (2.7/3.2)
K/A004SGK05(3.1/3.7)

LP#6910233, EO 4, EO 5

Est. Time of Completion <u>8 min.</u>

Part B ORQ# 190

Select the correct response to the following situation :

At 1:00AM on January 1st the '4C' Vital AC Inverter failed and its bus load automatically swapped to the CVT. At 6:00PM it was decided the '4C' Inverter could not be repaired and the bus load was transferred to the Spare Inverter at that time. Assuming no further equipment failures, how long may the unit operate in this configuration. (1.0 pt)

a. No action required other than a PWO written to repair 4C vital inverter. There are no time restraints associated with this inverter.

b. May operate for 24 hours from the time the transfer to the Spare Inverter was made (6:00 PM) at which time corrective action must be complete or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours and at least COLD SHUTDOWN within subsequent 24 hours.

c. May operate for 48 hours from the time the transfer to the Spare Inverter was made (6:00 PM).

d. May operate for 7 days because one channel of QSPDS will be out of service from the time the transfer to the Spare Inverter was made (6:00 PM).

Ans: d. May operate for 7 days because one channel of QSPDS will be out of service from the time the transfer to the Spare Inverter was made (6:00 PM).

Part B ORQ# 190 (cont.)

Ref: 5610-T-E-1592
0-ADM-021 Table 3.3.6 pg. 3-42
OP-003.1

K/A 012000 K1.01 (3.3/3.7)

LP#6902139 EO13

Est. Time of Completion 12 min.

Part B ORQ# 129

Which one of the following statements is correct ? (1.0 pt)

During natural circulation conditions, core exit thermocouples (CET's): .

a. Read higher due to reduced flow past detectors.

b. Provide indication of the adequacy of core cooling.

c. Are faulty if reading >1000°F.

d. Are not accurate due to no flow condition in instrument bypass loop.

Ans:

b. Provide indication of the adequacy of core cooling.

Ref: Basis ES0.2

K/A 002000 A1.13 (3.4/4.0)
000011 EK1.01 (4.1/4.4)
000011 EA2.10 (4.5/4.7)
017020 A1.01 (3.7/3.9)

LP#3502073 EO 7 & 9

Est. Time of Completion 1 min

Part B ORQ# 153

Emergency operating procedures provide guidelines for isolating faulted steam generators and steam generators with tube ruptures. Which one of the following statements about isolation of steam generators is correct. (1.0 pt)

a. All feedwater is isolated to a faulted steam generator regardless of steam generator level.

b. All feedwater is isolated to a steam generator with a tube rupture regardless of steam generator level.

c. If a faulted steam generator is isolated and secondary radiation is abnormal, transition to E-1 "Loss of Reactor or Secondary Coolant" should be made.

d. Do not isolate a faulted steam generator sample lines because subsequent procedural guidance requires steam generator activity samples on the faulted steam generator.

Ans: a. All feedwater is isolated to a faulted steam generator regardless of steam generator level.

Ref: EOP-E2, EOP-E-3

K/A 000037 SGA11 (3.9/4.1)
000037 SGA12 (3.5/3.8)
000040 SGA11 (4.1/4.3)
000040 SGA12 (3.8/4.1)

LP#0063-OL App. Z EO-12
Est. Time of Completion 4 min.

Part B ORQ# 250

A Safety Injection has occurred. After exiting E-0, Reactor trip or Safety Injection, the following set of conditions is observed relevant to the Critical Safety Functions (CSF's):

Subcriticality:	NI-41 through NI-44 are all less than 5%. Intermediate Range startup rate is equal to +0.2 DPM.
Core Cooling:	No RCPs are in service. Core Exit Thermocouples read 700°F RCS subcooling based on Core Exit Thermocouples is 30°F RVLMS indicates a 50% level.
Heat Sink:	All Steam Generator levels are indicating 5% Narrow Range. Main Feed Water Pumps are tripped. Auxiliary feed water flows are: S/G 'A': Total flow 125 GPM S/G 'B': Total flow 125 GPM S/G 'C': Total flow 100 GPM
Containment:	Containment Pressure = 3.0 psig Containment Recir. Sump. Level = 400 inches

The other CSF status trees indicate only green or yellow paths.

Which procedure would you enter and why? (2.0 pts.)

Ans: Procedure FR-H.1, Response to Loss of Secondary Heat Sink, should be entered. This is highest priority Red Path.

Part B ORQ# 250 (cont.)

Ref: EOP F-0

K/A 000054 SGA11 (3.4/3.3)

LP#0063-OL, App. CC, EO10

Est. Time of Completion 5 min.

Part B ORQ# 255

The plant is responding to a small-break LOCA in accordance with EOP E-1, 'Loss of Reactor or Secondary Coolant'. Pressurizer level has risen continuously, even though the RCS pressure has been dropping steadily. All Reactor Coolant Pumps are in operation.

Which one of the following leak locations is consistent with the plant conditions just described? (1.0 pt)

a. Failure of a weld on RCP 'B' discharge piping.

b. Failure of pressurizer PORV in a full open position.

c. Failure of charging header connection to the RCS.

d. Failure of a weld on the pressurizer liquid space sample line.

Ans: b.

Ref: 5610-T-E-4501
EOP E-1

K/A 000008 EA2.20 (3.4/3.6)

LP#0063-OL, App. S, EO8

Est. Time of Completion 1 min.

Part B ORQ# 266

Select the one correct response based on the following : (1.0 pts.)

During FR-C.1 (Response to Inadequate Core Cooling) if attempts to establish adequate core cooling using the HHSI system are ineffective, the intact SGs are depressurized to 90 psig and then to atmospheric pressure.

The SGs are depressurized :

a. To allow use of the condensate pumps to supply feedwater to the SGs.

b. To cause the RCS to depressurize which improves core cooling by heat removal due to boiling.

c. To cause the RCS to depressurize which improves the ability of the RHR Pumps and Accumulators to deliver cooling to the core.

d. To cause the RCS to depressurize which improves the ability of the SI systems to add negative reactivity from borated water to insure adequate shutdown.

Ans: c. To cause the RCS to depressurize which improves the ability of the RHR Pumps and Accumulators to deliver cooling to the core.

Ref: FR-C.1 (Response to Inadequate Core Cooling) Basis Document

K/A 000074 EK1.03 (4.5/4.9)

LP#0063-OL, App. MM, EO9

Est. Time of Completion 3 min.

Part B ORQ# 267

In FR-C.1, Response to Inadequate Core Cooling, if CET temperatures are above 1200 degrees, the operator is directed in Step 18 to "Start available RCPs as necessary until CETs less than 1200 degrees".

Explain whether RCP's should be started if RCP support conditions cannot be established. (1.0 pts.)

Ans:

RCPs should be started even though support conditions cannot be established. The RCPs are sacrificed to save the core.

Ref: FR-C.1/Basis Document

K/A 000074 EK3.07 (4.0/4.4)
000074 EK3.11 (4.0/4.4)

LP#0063-OL, App. MM, EO9

Est. Time of Completion 2 min.

Part B ORQ# 270 Delete

A Primary LOCA has occurred inside of Unit 3 containment. The operating team is implementing E-O, Rx Trip/Safety Injection. The following unit conditions exist:

Containment Pressure is 10 psig
Containment Radiation level is 1.4×10^5 R/hr
RCS temperature is 508 degrees
RCS pressure is 1135 psig
Total HHSI flow to the core is 300 gpm.

Which one of the following is correct? (1.0 pt)

a. RCPs should be tripped because RCS subcooling is 42°F.

b. RCPs should not be tripped because RCS subcooling is >25°F.

c. RCPs should be tripped because Phase "B" isolation has occurred.

d. RCPs should be tripped because subcooling is <25°F.

Ans:

a. RCPs should be tripped because RCS subcooling is 42°F.

Ref: E-O Foldout Page/Steam Tables

K/A 000011 EK3.14 (4.1/4.2)
000009 EA2.01 (4.2/4.8)
017020 A4.02 (3.8/4.1)
000009 SGA 10 (4.3/4.3)

LP#0063-OL, App. M, EO2

Est. Time of Completion 3 min.

Part B ORQ# 237

a. Classify the following event. Specify the highest classifications, the category and part used to make the classification (e.g. classified as "Site Area Emergency" from category 12 'Loss of Power Conditions' Part 2.) (1.0 pt.)

Conditions:

- Large break LOCA in progress
- Containment Pressure had spiked to 30 psig and is being lowered by Containment Spray
- Phase 'A' and Phase 'B' properly activated
- Both CHRRM channels read 3×10^3 R/hr
- Both HHSI and RHR pump flow meters indicate flow to core
- Calculations show dose at site boundary to be 10 Rem Whole Body and 20 Rem Thyroid

b. With regard to the above event, which one of the following PARs is correct? (1.0 pt.)

1. Evacuate all sectors 0-2 miles, evacuate 2-10 miles downwind sectors, shelter remaining sectors.

2. Evacuate all sectors 0-2 and 2-5 miles, evacuate downwind and shelter remaining sectors 5-10 miles.

3. Shelter all sectors 0-10 miles

4. Evacuate downwind sectors and shelter remaining sectors 0-10 miles.

Ans: (2.0 pts)

a. GENERAL EMERGENCY (.075) Category 2 Part 1 (.025)
or Category 9 Part 1, 2, 3 & 4

b. 2

Part B ORQ# 237 (cont.)

Ref: E-Plan 20101 Table 1 & 2

K/A 194001 A1.16 (3.1/4.4)

LP#6902310 EO-10

Est. Time of Completion 10 min.

Part B ORQ# 381

Which one of the following describes a condition in which QPTR Tech. Spec. limitations are exceeded. (1.0 pt)

a. The reactor is at 100% power and QPTR = 1.01

b. While the reactor was at 100% power, QPTR was determined to be 1.08. Ninety (90) minutes after the determination reactor power is reduced to 90% and NIS power range High Flux setpoints reduced to 98%.

c. The reactor is at 45% power. QPTR has been determined to be 1.03 but the OPΔT, OTΔT, and NIS Power Range setpoints have been reduced to 55%.

d. QPTR was determined to be 1.04 but 90 minutes later F_q and $F_\Delta H$ were verified to be within limits and reactor power was reduced from 100% power to 91% power.

Ans:

b.

Ref: ADM-021, T.S. 3.2.6.h LER 87-18

K/A 015000 A1.04 (3.5/3.7)

LP# 0802163 EO-6

Est. Time of Completion 12 min.

Part B ORQ# 383

Classify the following event. Specify the highest classification, category and part used to make the classification (e.g. classified as "Unusual Event" from category 2 Primary Leakage / LOCA Part 3) (1.0 pts.)

Both Units are operating at 100% reactor power. 'A' Emergency Diesel Generator (EDG) is out of service for fuel pump repairs. A loss of all offsite AC power occurs and 'B' EDG fails to auto start. Ten minutes later, the turbine operator is able to locally start and load 'B' EDG.

Ans:

Site Area Emergency (0.75) Category 7 part 4 (0.25)

Ref: EP-20101 Table 1

K/A 000068 K1 (3.3/4.1)

LP# 6902252 EO-7

Est. Time of Completion 4 min.

Part B ORQ# 297

(1.0 pt.)
The remotely operated steam generator blowdown isolation valve, CV-4-6275B, failed shut. Which one of the following correctly describes the actions required.

a. This valve must be de-energized within 4 hours, then normal operations can continue.

b. Since the valve failed shut, the affected penetration is isolated and, therefore normal operations can continue.

c. Automatic isolation valve 4-CV6278B must be shut within 4 hours.

d. The Key Lock Switch on VPB for 4-6275B must be placed in the Bypass position within 4 hours, then normal operation can continue.

Ans:

a.

Ref: TS - 3.3.3

K/A 103/GK5 (3.3/4.1)

LP#0063-OL, App. EE EO-8,3

Est. Time of Completion 5 min.

Part B ORQ# 259

The reactor is in Mode 3 in preparation for reactor startup. The Intermediate Range Nuclear Instrumentation Analog Channel Operational Test has just been completed. A review of the results indicates that the final 'as-left' setting for the Intermediate Range neutron flux trip for Channel N36 is equivalent to 28% power.

Which one of the following statements describes the required actions for this situation? (1.0 pt)

a. Correct the level trip setpoint to the current equivalent to 25% rated thermal power prior to increasing thermal power above the P-6 setpoint.

b. Correct the level trip setpoint to the current equivalent to 25% rated thermal power prior to exceeding 25% Power.

c. Correct the level trip setpoint to the current equivalent to 25% rated thermal power prior to increasing thermal power above the P-10 setpoint.

d. Place Channel N36 in a tripped condition within 2 hours, then proceed with the startup

Ans:

a. Correct the level trip setpoint to the current equivalent to 25% rated thermal power prior to increasing thermal power above the P-6 setpoint.

Ref: 0-ADM-21 2.2.1, 3.3.1

K/A 015000 K3.01 (3.9/4.3)

LP#0802057/6902510

Est. Time of Completion 4 min.

TURKEY POINT PLANT
SIMULATOR EXERCISE GUIDE

Revision: 0

Prepared by: T.A. VEHEC

Reviewed by: L Voelker

Approved by: G Hollinger (Training Supervisor) Date: 5/31/89

Approved by: _____ (Operations Supervisor)

PROGRAM: LICENSED OPERATOR REQUALIFICATION NRC EXAM SCENARIO

EXERCISE TITLE: PRESSURIZER HEATER FAILURE/P1-444 FAILURE/SPRAY VALVE FAIL OPEN

EXERCISE GUIDE NUMBER: 011M-L-E

SCENARIO NUMBER: 62

ESTIMATED TIME: 50 Minutes

REFERENCE BLOCK DIAGRAM(S): ID_PZR, ID_SI

REFERENCES:

3-GOP-103 - POWER OPERATION TO HOT STANDBY (2/11/89)
3-ONOP-41.5 - PRESSURIZER PRESSURE MALFUNCTION (8/26/88)
AP-0103.2 - DUTIES AND RESPONSIBILITIES OF SHIFT OPERATORS (1/5/89)
ONOP-0208 - ANNUNCIATOR LISTS (11/25/88)
3-EOP-E-0 - REACTOR TRIP/SI (3/10/89)
3-OP-0204.2 - PERIODIC TESTS, CHECKS, AND OPERATING EVOLUTIONS (4/26/89)
3-OSP-0201.1 - RCO DAILY LOGS (4/18/89)

TECHNICAL SPECIFICATIONS
PLANT CURVE BOOK

TERMINAL OBJECTIVE: The SRO and the Reactor Control Operator (RCO) will demonstrate the ability to perform reactor power operations, and respond to abnormal plant conditions IAW approved plant procedures ensuring that the health and safety of the public is protected and the integrity of the plant maintained. The RCO will recognize and log entry level conditions to the Technical Specifications.

ENABLING OBJECTIVES:

1.0 Demonstrate the ability to conduct plant power operations, including the ability to diagnose off normal conditions and to take appropriate actions.

1.1 Given procedures for plant power operations, and a Shift relief status sheet:

a. Respond to a Pressurizer heater failure
b. Recognize and respond to a failure of PT-444 IAW plant procedures
c. Recognize and respond to a failure of a pressurizer spray valve to close.
d. Respond to an SI from low pressurizer pressure.
e. Classify the event IAW EP-20101 Table 1.

1.2 Given plant conditions during plant power operations:

a. Identify abnormalities by assessing actual system response with respect to predicted system response.
b. Investigate the cause and effect of abnormalities in system performance.
c. Implement applicable procedures.
d. Perform immediate actions without reference to procedure.

1.3 Given plant conditions during plant power operations, IDENTIFY and IMPLEMENT the applicable on-site and off-site reports and notifications.

1.4 Conduct plant power operations using the following principles for operational effectiveness as they apply to all operators:

a. Plant and control room communication
b. Plant/Control Board Monitoring
c. Plant/Control Board Manipulation
d. Operational Problem solving
e. Use of Normal/Off-normal Procedures/Technical Specification
f. Use of Emergency Procedures IAW Rules of Usage of the EOP's.
g. Annunciator Recognition and Response
h. Written Communications/logs
i. ALARA Awareness

1.5 Conduct plant power operations using the following principles of operational effectiveness as they apply to supervisory personnel:

a. Team Performance management
b. Problem Solving
c. Decision Analysis
d. Action Planning

1.6 The Shift Technical Advisor (STA) will demonstrate the ability to provide diagnositic support to Operations personnel during Off-normal and Emergency events and to advise the PS-N of actions required to terminate or mitigate the consequences of such events.

a. Maintain awareness of plant status, configuration, and alignment of systems/components necessary for safe operation and shutdown.
b. Assess all reactor trips and associated transients with regard to safety.
c. Assist the PS-N in assuring that applicable On/Off-Site notifications are made.
d. Provide technical assistance during the following events:

1. SI

1.7 The control room team shall demonstrate the ability to apply techniques presented during the TEAM WORK AND ACCOUNTABILITY Workshop that contribute to effective team performance.

STIMULATOR LESSON PLAN INSTRUCTOR INFORMATION

TASK NUMBERS AND STATEMENTS:

LS OBJECTIVES	TASKS #	TASK DESCRIPTION
1.5	D-015	DIRECT SHIFT PERSONNEL DURING MAJOR PLANT EVOLUTIONS
1.5,1.6	E-003	CHECK PLANT EQUIPMENT STATUS
1.4,1.6	E-006	REVIEW CONTROL BOARD STATUS
1.5	D-007	AUTHORIZE LOAD CHANGE
1.5	E-015	REVIEW SHIFT LOGS
1.4	F-008	VERIFY CORRECT PERFORMANCE OF EMERGENCY AND OFF-NORMAL PROCEDURES
1.4/1.3	F-009	IDENTIFY/RESPOND TO AN OFF-NORMAL EVENT
1.3/1.6	C-011	INITIATE UNUSUAL EVENT REPORT
1.3/1.6	F-011	REPORT SIGNIFICANT EVENTS
RCO OBJECTIVES	TASKS #	TASK DESCRIPTION
1.2	G-007	BORATE THE RCS
1.4	BB-006	MAKE PLANT WIDE ANNOUNCEMENTS
1.4	L-002	CHANGE POWER LEVEL
1.1d/1.4b	CC-004	ASSURE PROPER SI SYSTEM OPERATION
1.1e/1.4c	CC-010	ISOLATE ACCUMULATORS
1.1d/1.4c	CC-015	RESET SI SYSTEM
1.1c/1.1e	DD-001	DUMP STEAM TO CONDENSER
1.1c/1.4b	DD-004	VERIFY STEAM DUMP/TURBINE BYPASS CONTROL SYSTEM ALIGNMENT ETC.
1.2a/1.4b	DD-005	VERIFY SYSTEM AUTOMATIC OPERATION
1.2b/1.4g	FF-005	INVESTIGATE CONTROL BOARD ALARM
1.1c/1.1e/1.4b	FF-013	MONITOR STEAM GENERATOR OPERATION
1.4c/1.1d/1.1e	G-001	MONITOR CVCS OPERATION
1.4c/1.1d/1.1e	G-002	ADJUST CHARGING FLOWRATE
1.4c/1.1d/1.1e	G-040	PERFORM VALVE LINEUPS OF CVCS
1.1c/1.1d/1.4b	N-001	CHECK ALTERNATING CURRENT (AC) ELECTRICAL DISTRIBUTION SYSTEM INSTRUMENTS
1.1c/1.1c/1.4c	N-004	SHUTDOWN DIESEL GENERATOR
1.1c/1.4b	L-027	MONITOR ROD CONTROL SYSTEM
1.1c/1.4c	O-014	SHUT DOWN CONDENSATE PUMPS
1.1c/1.4c	O-015	SHUT DOWN MAIN FEEDWATER PUMPS
1.1c/1.4c	O-021	STOP HEATER DRAIN PUMP
1.1c/1.4b	O-027	MONITOR MAIN FEEDWATER OPERATION
1.1c/1.4b	O-028	MONITOR CONDENSATE SYSTEM
1.4b/1.4h	R-009	CHECK MINIMUM EQUIPMENT LIST
1.4b/1.4h	R-010	CHECK MINIMUM INSTRUMENTATION LIST
1.2a/1.4b	R-011	CHECK PLANT STATUS

RCO OBJECTIVES	TASKS #	TASK DESCRIPTION
1.4h	R-024	LOG SHIFT ACTIVITIES
1.4h	R-021	LOG EQUIPMENT IN AND OUT OF SERVICE
1.4h	R-025	LOG SHIFT READINGS
1.1e/1.4c	T-002	DUMP STEAM TO ATMOSPHERE
1.4g	T-003	INVESTIGATE CAUSE OF CONTROL BOARD ALARMS
1.1c/1.4c	T-007	REMOVE MOISTURE SEPARATOR/REHEATER FROM SERVICE ETC.
1.1c/1.1e/1.4b	T-013	MONITOR THE MAIN AND REHEAT STEAM SYSTEMS
1.1c/1.4b	U-001	MONITOR TURBINE GENERATOR
1.1c	U-014	PLACE TURBINE TURNING GEAR INTO SERVICE
1.1c	U-019	SHUTDOWN TURBINE GENERATOR
1.1c/1.4b	U-026	VERIFY START/STOP OF TURBINE GENERATOR OIL PUMPS
1.1c	V-002	ACTIVATE NUCLEAR INSTRUMENTATION CHANNEL
1.1c/1.4b	W-004	MONITOR THE PRESSURIZER LEVEL CONTROL SYSTEM
1.4b	W-006	CHANNEL CHECK PRESSURIZER LEVEL
1.1a/1.4g	W-011	INVESTIGATE CONTROL BOARD ALARMS
1.1b/1.2b	W-012	INVESTIGATE PORV MALFUNCTION
1.1b/1.4c	W-013	ISOLATE LEAKING PORV
1.1a/1.4c	W-017	SHIFT PRESSURIZER LEVEL/PRESSURE CONTROL MODE
1.1a/1.1b	W-025	MONITOR PRESSURIZER PRESSURE
1.1e	Y-007	COOLDOWN RCS
1.1/1.4b	Y-018	MONITOR REACTOR COOLANT SYSTEM
1.1c/1.4b	Y-019	MONITOR OPERATION OF THE REACTOR COOLANT PUMPS
1.1c/1.4c	Y-028	SHUTDOWN RCP'S
1.1d/1.1e	Y-029A	STABILIZE PLANT AFTER LOCA
1.1b/1.2d	Y-029B	RESPOND TO SMALL BREAK LOCA
1.1c/1.4g	Z-004	INVESTIGATE CONTROL PANEL ALARMS
1.1c/1.2d	Z-010	RESPOND TO UNIT TRIP

EXERCISE SYNOPSIS

The Plant is in Mode 1 with reactor power at 100% power. A failure of the pressurizer control heaters occurs. Following this failure, a failure of PT-444 occurs causing the pressurizer sprays and PORV 455C to open. The RO takes actions to close the PORV and sprays and reestablish normal pressurizer pressure. One of the spray valves fails to close, causing a continuous RCS depressurization. An OT DELTA T runback occurs. When it becomes apparent to the operators that the depressurization is trending to the low pressurizer pressure reactor trip and SI, the operators trip the reactor and respond IAW E-0. The depressurization is stopped when the RCP's associated with the stuck open spray valves are tripped at VP "A". Recovery actions are IAW E-0, and ES-1.1. Time in core life is determined by the EFPD selected by the instructor. The instructor should ensure that core physics data made available to the operators is consistent with this selection. The exercise is concluded when plant conditions are stabilized in ES-0.1.

10CFR55.59(c)(3)(i) REQUIRED MANIPULATIONS MET BY THIS EXERCISE

( ) - PRESSURIZER PRESSURE MALFUNCTION

INSTRUCTIONAL MATERIALS

Attendance sheet/Black ink pen/ Calculator
Completed Shift Turnover Sheet
Completed appropriate sections of 3-GOP-301
Working copies of:

- AP-0103.2 (FIG 2,3) EDOS LOG
- AP-0103.2 (APPENDIX A-E)
- 3-GOP-301 - HOT STANDBY TO POWER OPERATIONS
- 3-OSP-0200.1 - SCHEDULE OF PLANT CHECKS AND SURVEILLANCES
- 3-OP-0204.2 - PERIODIC TESTS, CHECKS, AND OPERATING EVOLUTIONS
- 3-OSP-0201.1 - RCO DAILY LOGS
- AG-005 - STUDENT EVALUATION ATTATCHMENTS

SEQUENCE OF EVENTS

- Mode 1, 100% power
- Pressurizer heater malfunction
- PT-444 fails high
- PORV/Spray valves open PORV and one spray valve are reset by the RO
- RCS depressurization continues
- OT DELTA T runback
- Reactor trip - manual at console does not function, VP "B" is functional.
- Turbine fails to trip
- SI occurs
- RCP secured
- Depressurization stopped
- Plant conditions stabilized
- End of Simulator Exercise

OUTLINE OF INSTRUCTION	INSTRUCTOR ACTIVITIES	NOTES/STUDENT ACTIVITIES
1. Pre Exercise Briefing	ACTION:	
	Perform the following	
	- Exercise Overview	
	- Student objectives	
	- Conduct of Shift	
	- Team Skills/Communications	
	- Shift Assignment	
II. COMMENCE SIMULATOR EXERCISE	Action: Start simulation	STUDENT ACTIVITIES:
		Perform a control board walkdown, operating log review and verbal shift turnover within 10 minutes of beginning the exercise.
	Shift Turnover - Normal plant lineup for 100% power - No scheduled OSP's - "A" CCW Pump O.O.S. - "A" Charging Pump O.O.S. - R-15 O.O.S. - "B" AFWP O.O.S. - MIMS O.O.S.	

SCENARIO# 62 EVENT# 1

BRIEF DESCRIPTION: PRESSURIZER HEATER CONTROLLER FAILURE

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
		RO	- Acknowledges annunciator B 3/1. - Reports pressurizer control heater loss to APSN. - Calls Nuclear Operator to investigate problem with heater controller.
		ANPS	Directs RO/BOP actions IAW ONOP-0208.14
		ANPS	- Directs that pressure be controlled by cycling backup heaters manually. - Calls I&C to inform them of problem with pressurizer heaters. - Check for required action for malfunction IAW ONOP-041.5.
		RO	Coordinate Nuclear Operator response to the event.

SCENARIO# 62 EVENT# 2
BRIEF DESCRIPTION: PT-444 FAILS HIGH - SPRAY VALVES/PORV OPEN

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
*		RO	- Recognize symptoms of a failure of PT-444. - Attempt manual control at Hagan controller PC-3-444J. - Close PORV 455C
		APSN	- Ensure that subsequent actions of ONOP-41.5 are performed by the RO.
		APSN	- Direct RO actions to recover pressurizer pressure. - Call I&C to inform them of the failed pressurizer pressure instrument.
		RO	Attempt recovery of pressurizer pressure IAW ONOP-041.5
		APSN	When Sprays and PORV indicate closed, direct RO to reestablish normal steady state plant conditions
		RO/BOP	Stabilize plant conditions

SCENARIO# 62 EVENT# 3

BRIEF DESCRIPTION: STUCK OPEN PRESSURIZER SPRAY VALVE/REACTOR TRIP/SI

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	TIME CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
*			RO	Recognizes that RCS pressure is not recovering and informs APSN.
*			APSN	Deduces that one or two pressurizer spray valves is stuck open and directs that the RO trip the reactor and that the immediate actions for E-0 be carried out by the RO and BOP
*			RO	- Attempt to trip the reactor from the console, recognizes that reactor did not trip, goes to VP "B" and manually trips reactor there.
			RO	- Verifies Reactor trip, recognizes that two control rods are stuck out of the core.
			ANPS	Directs RO/BOP to perform immediate actions of E-0.
	*		ANPS/RO/BOP	Perform immediate actions of EOP-E-0.
*		*	BOP	- Verify turbine trip, turbine fails to trip, manually trips turbine
			RO/BOP	- Verify power to 4KV buses. - Check if SI is actuated. - Verify Feedwater Isolation - Verify Containment Isolation Phase A. - Verify AFW pumps running. - Verfiy SI pumps running.
*			RO	- Verify 2 of 3 CCW pumps running, recognizes that only 1 CCW pump is running - manually starts the "B" CCW pump.
			BOP	- Verify 2 of 3 ICW pumps running - Verify 2 of 3 ECCU's and filter fans running. - Check if Main steam lines should be isolated. - Verify Containment Spray not required.
			APSN	Review immediate actions with RO/BOP to ensure that all have been performed

SCENARIO# 62 EVENT# 3 (CONT.)

BRIEF DESCRIPTION: STUCK OPEN PRESSURIZER SPRAY VALVE/REACTOR TRIP/SI

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
		APSN	When immediate action steps are verified, directs the RO to secure the "C" RCP and monitor pressurizer pressure.
		RO	- Trips the "C" RCP and monitors RCS pressure. - Determines that pressurizer pressure is still decreasing - Informs the APSN that pressure is still decreasing.
		APSN	Directs RO to stop the "B" RCP and monitor pressure.
*		RO	Stops the "B" RCP, verifies that pressure decrease has stopped.
		APSN	Directs RO/BOP to stabilize plant conditions.
*		PSN	Classify the event per EP-20101, Table 1 section 7, Loss of Safe Shutdown Functions/ATWS, as an ALERT, Reactor critical <u>AND</u> reactor fails to trip on manual signal.

END OF SCENARIO

OUTLINE OF INSTRUCTION	INSTRUCTOR ACTIVITIES	NOTES/STUDENT ACTIVITIES
SCENARIO END SUMMARY AND POST EXERCISE CRITIQUE		
A. STUDENT REVIEW OF ENABLING OBJECTIVES	Instructor passes out Objective review sheets to the operators and directs the Senior operator to take 15-20 minutes to review them with the crew.	Operators take Objective review sheets to the classroom and review them with the senior operator in the crew.
B. INSTRUCTOR CRITIQUE OPERATOR PERFORMANCE	Critique the students knowledge level and skill in performing the required tasks. Critique individual and team responses by commenting on both the appropriateness and timeliness of individual and team responses. Emphasize good examples of teamwork illustrated during the scenario. Ensure that all T/S compliance actions are entered in the CO's log. Ensure that the students understand the theoretical aspects of performing the tasks Review the Objective review sheet and timeline filled out during exercise and prepare comments for critique of operators.	

OUTLINE OF INSTRUCTION	INSTRUCTOR ACTIVITIES	NOTES/STUDENT ACTIVITIES
C. INSTRUCTOR GUIDED DISCUSSION OF OBJECTIVE REVIEW SHEETS	Instructor asks senior shift crew member to reconstruct exercise sequence of events.	Senior crew member reconstructs exercise sequence.
	Instructor interjects comments and refreshes operators memory, if required, to reconstruct event sequence.	
	Instructor leads operator discussion of Review of Objectives sheets, interjecting comments and rebuttals as necessary.	Operators discuss their comments on the Review of Objectives sheets.
D. AREAS NEEDING IMPROVEMENT	Instructor review areas identified as needing improvement in discussion of Objective Review Sheets.	
E. THEORY REINFORCEMENT		
F. STUDENT FEEDBACK AND CRITIQUE	Instructor requests feedback from the operators on the realism of the simulator exercise.	Operators provide feedback to the instructor on the validity of the simulator exercise.

OBJECTIVE	RATING	COMMENTS/TIMELINE
1.3 Given plant conditions during plant power operations, IDENTIFY and IMPLEMENT the applicable on-site and off-site reports and notifications.		
1.4 Conduct plant power operations using the following principles for operational effectiveness as they apply to all operators:		
a. Plant and control room communication		
b. Plant/Control Board Monitoring		
c. Plant/Control Board Manipulation		
d. Operational Problem solving		
e. Use of Normal/Off-normal Procedures/Technical Specifications		
f. Use of Emergency Procedures IAW Rules of Usage of the EOP's.		
g. Annunciator Recognition and Response		
h. Written Communications/logs		
i. ALARA Awareness		
1.5 Conduct a plant power operations using the following principles of operational effectiveness as they apply to supervisory personnel:		
a. Team performance management		
b. Problem Solving		
c. Decision Analysis		
d. Action Planning		
1.6 The Shift Technical Advisor (STA) will demonstrate the ability to provide diagnositic support to Operations personnel during Off-normal and Emergency events and to advise the PS-N of actions required to terminate or mitigate the consequences of such events.		
a. Maintain awareness of plant status, configuration, and alignment of systems/components necessary for safe operation and shutdown.		

OBJECTIVE	RATING	COMMENTS/TIMELINE

b. Assess all reactor trips and associated transients with regard to safety.
c. Assist the PS-N in assuring that applicable On/Off-Site notifications are made.
d. Provide technical assistance during the following evolutions:

1. SI

1.7 The control room team shall demonstrate the ability to apply techniques presented during the TEAM WORK AND ACCOUNTABILITY Workshop that contribute to effective team performance.

TERMINAL OBJECTIVE EVALUATION CRITERIA

Rate Operations Team performance of the following:

A. Recognized/performed corrective actions during off-normal events

B. Team coordination during load reduction

C. All steps/substeps of procedures performed/initialed

D. Communications were proper.

E. Appropriate off-site notifications (ie: dispatcher/duty call supervisor) without instructor prompting.

F. Instructor assistance was not required.

TURKEY POINT PLANT
SIMULATOR EXERCISE GUIDE

Revision: 0

Prepared by: T.A. VEHEC

Reviewed by: [signature]

Approved by: [signature] (Training Supervisor) Date: 5/31/89

Approved by: _____ (Operations Supervisor)

PROGRAM: LICENSED OPERATOR REQUALIFICATION NRC EXAM SCENARIO

EXERCISE TITLE: STEAM GENERATOR TUBE RUPTURE

EXERCISE GUIDE NUMBER: 013A-L-E

SCENARIO NUMBER: 42

ESTIMATED TIME: 50 MINUTES

REFERENCE BLOCK DIAGRAM(S): ID_SGTL, ID_SGTB, ID_SGTR, ID_RCACT,

REFERENCES:

AP-0103.2	- DUTIES AND RESPONSIBILITIES OF SHIFT OPERATORS (7-12-88)
ONOP-0208	- ANNUNCIATOR LISTS (8-8-88)
0-OSP-0200.1	- SCHEDULE OF PLANT CHECKS AND SURVEILLANCES (8-9-88)
OP-0204.2	- PERIODIC TESTS, CHECKS, AND OPERATING EVOLUTIONS (7-14-88)
3-OSP-0201.1	- RCO DAILY LOGS (5-24-88)
3-ONOP-041.4	- EXCESSIVE RCS ACTIVITY (11/22/85)
EP-20101	- DUTIES OF EMERGENCY COORDINATOR 12/4/87)
3-EOP-E-0	- REACTOR TRIP OR SAFETY INJECTION (6/8/88)
3-OSP-041.1	- RCS LEAK RATE CALCULATION (4/29/88)
3-EOP-E-3	- STEAM GENERATOR TUBE RUPTURE (3/25/88)
3-ONOP-041.3	- EXCESSIVE RCS LEAKAGE (10/28/86)
3-ONOP-071	- STEAM GENERATOR TUBE LEAK (4/27/88)
SEN - 16	- SGTR (NORTH ANNA)
I&E NOTICE	- 87-60 (RCS PRESSURE CONTROL DURING SGTR)
NRC I.N. 88-31	- S/G TUBE RUPTURE ANALYSIS DEFICIENCY

TECHNICAL SPECIFICATIONS
PLANT CURVE BOOK

EXERCISE TITLE: STEAM GENERATOR TUBE RUPTURE

EXERCISE GUIDE NUMBER: 013A-L-E

SCENARIO NUMBER: 42

ESTIMATED TIME: 50 MINUTES

REFERENCE BLOCK DIAGRAM(S): IO_SGTL, IO_SGTB, IO_SGTR, IO_RCACT,

OP-0204.2	- PERIODIC TESTS, CHECKS, AND OPERATING EVOLUTIONS (7-14-88)
3-OSP-0201.1	- RCO DAILY LOGS (5-24-88)
3-ONOP-041.4	- EXCESSIVE RCS ACTIVITY (11/22/85)
EP-20101	- DUTIES OF EMERGENCY COORDINATOR 12/4/87)
3-EOP-E-0	- REACTOR TRIP OR SAFETY INJECTION (6/8/88)
3-OSP-041.1	- RCS LEAK RATE CALCULATION (4/29/88)
3-EOP-E-3	- STEAM GENERATOR TUBE RUPTURE (3/25/88)
3-ONOP-041.3	- EXCESSIVE RCS LEAKAGE (10/28/86)
3-ONOP-071	- STEAM GENERATOR TUBE LEAK (4/27/88)
SEN - 16	- SGTR (NORTH ANNA)
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NRC I.N. 88-31	- S/G TUBE RUPTURE ANALYSIS DEFICIENCY

TECHNICAL SPECIFICATIONS
PLANT CURVE BOOK

e. Isolate/cooldown the affected S/G to minimize secondary system/off-site contamination

1.2 Given plant conditions during plant power operations:

a. Identify abnormalities by assessing actual system response with respect to predicted system response.
b. Investigate the cause and effect of abnormalities in system performance.
c. Implement applicable procedures.
d. Perform immediate actions without reference to procedure.

1.3 Given plant conditions during plant power operations, IDENTIFY and IMPLEMENT the applicable on-site and off-site reports and notifications.

1.6 The Shift Technical Advisor (STA) will demonstrate the ability to provide diagnositic support to Operations personnel during Off-normal and Emergency events and to advise the PS-N of actions required to terminate or mitigate the consequences of such events.

a. Maintain awareness of plant status, configuration, and alignment of systems/components necessary for safe operation and shutdown.
b. Assess all reactor trips and associated transients with regard to safety.
c. Assist the PS-N in assuring that applicable On/Off-Site notifications are made.
d. Provide technical assistance during the following events:

1. RCS Leak Rate calculation
2. Steam Generator Tube Rupture

1.7 The control room team shall demonstrate the ability to apply techniques presented during the TEAM WORK AND ACCOUNTABILITY Workshop that contribute to effective team performance.

SIMULATOR LESSON PLAN INSTRUCTOR INFORMATION

TASK NUMBERS AND STATEMENTS:

LS OBJECTIVES	TASKS #	TASK DESCRIPTION
1.5	D-015	DIRECT SHIFT PERSONNEL DURING MAJOR PLANT EVOLUTIONS.
1.5	E-003	CHECK PLANT EQUIPMENT STATUS
1.4	E-006	REVIEW CONTROL BOARD STATUS
1.4	F-008	VERIFY CORRECT PERFORMANCE OF EMERGENCY AND OFF-NORMAL PROCEDURES
1.4/1.3	F-009	IDENTIFY/RESPOND TO AN OFF-NORMAL EVENT
1.3	F-010	MAKE PROTECTIVE ACTION RECOMMENDATIONS
1.3	F-011	REPORT SIGNIFICANT EVENTS
1.2c	F-012	DIRECT RESPONSE AS EMERGENCY COORDINATOR

RCD OBJECTIVES	TASKS #	TASK DESCRIPTION
1.4b	CC-004	ASSURE PROPER SI OPERATION
1.4e	CC-010	ISOLATE ACCUMULATORS
1.4b	DD-004	VERIFY STEAM DUMP OPERATION PROPER
1.4c	EE-002	BYPASS B/D RECOVERY SYSTEM
1.4b	EE-003	RESPOND TO CONTROL PANEL ALARMS
1.4b	EE-004	ISOLATE B/D
1.4b	FF-005	INVESTIGATE CONTROL PANEL ALARM
1.1c	FF-007	INVESTIGATE S/G TUBE LEAK
1.2c/1.1e	FF-012	RESPOND TO A S/G TUBE RUPTURE
1.2a	FF-013	MONITOR S/G OPERATION
1.4b	G-001	MONITOR CVCS OPERATION
1.1b	G-002	ADJUST CHARGING FLOW RATE
1.1b/1.4c	G-004	ADJUST L/D FLOW RATE & FLOW CONTROLLER FOR MULTIPUMP OPERATION
1.4c	G-007	BORATE THE RCS
1.4c	G-012	CONTROL RCS PRESSURE
1.4c	G-024	MAINTAIN RCP SEAL L.O. RATE
1.4e	G-027	RECOVER FROM LOSS OF LETDOWN
1.4b	G-037	VERIFY BORATION FLOW PATH
1.4c	G-039	SET UP BLENDER FOR AUTO OPERATION
1.4c	H-001	PERFORM LINEUPS ON CCW SYSTEM
1.4c	L-002	CHANGE POWER LEVEL
1.4b	M-002	CHECK AC ELECTRICAL DISTRIBUTION INSTRUMENTATION
1.4b	M-002	CHECK DC ELECTRICAL DISTRIBUTION INSTRUMENTATION
1.4c	N-004	SHUT DOWN EDG
1.4b	O-027	MONITOR MAIN FEEDWATER OPERATION
1.1d	O-028	MONITOR CONDENSATE SYSTEM
1.1	R-011	CHECK PLANT STATUS
1.4	R-013	MAINTAIN LOGS IN THE CONTROL ROOM
1.4h	R-024	LOG SHIFT ACTIVITIES
1.4b	U-001	MONITOR TURBINE GENERATOR OPERATION
1.4c	U-002	ADJUST UNIT LOAD
1.4c	V-002	ACTIVATE NUCLEAR INSTRUMENTATION CHANNEL
1.4c	W-005	CONTROL PRESSURIZER PRESSURE IN MANUAL
1.4b	W-006	CHANNEL CHECK PRESSURIZER LEVEL
1.4g	W-011	INVESTIGATE CONTROL BOARD ALARMS
1.4b	W-025	MONITOR PRESSURIZER PRESSURE
1.1c	X-001	CHECK LIQUID WASTE MONITORING SYSTEM
1.1c	X-004	INVESTIGATE PROCESS RADIATION MONITOR SYSTEM ALARMS
1.1a	Y-004	CALCULATE RCS INVENTORY BALANCE
1.1c/1.4c	Y-007	COOLDOWN THE RCS
1.1a/b	Y-018	MONITOR REACTOR COOLANT SYSTEM
1.4b	Y-019	MONITOR OPERATION OF THE RCP's
1.1c	Y-020	INVESTIGATE RCS HIGH ACTIVITY ALARMS
1.4b	Y-013	EVALUATE RCP SEAL CONDITIONS
1.4b	Y-029C	DETERMINE CAUSE OF SI
1.4c	Z-006	RESET SAFETY INJECTION

EXERCISE SYNOPSIS

The plant is at 100% power. A high RCS activity condition has been identified by Chemistry. Following an increase in charging and letdown due to the increased RCS activity, a small S/G tube leak in "A" S/G causes R-19 PRM to trend upward, then alarm. An RCS leak rate calculation indicates RCS unidentified leakage to be just over 1 gpm. The daily chemistry report shows secondary activity levels to be 10 times greater than normal but well within the Tech Spec limits. The primary inventory loss is through a S/G tube which worsens to the point of a Safety Injection actuation on low RCS pressure or by operator action. The operators respond IAW E-0 and transition to E-3. Following the cooldown in E-3, because the RCP's have been trippped, Tcold in the affected loop decreases to the ORANGE or RED path criteria for integrity. The operators transition to FR-P.1. Time in core life is determined by the EFPD selected by the instructor. The instructor should ensure that core physics data made available to the operators is consistent with this selection. The exercise is concluded at the evaluator's discretion, following the transition to FR-P.1 or when appropriate Post-SGTR cooldown method is selected.

10CFR55.59(c)(3)(i) REQUIRED MANIPULATIONS MET BY THIS EXERCISE

{ G-1 } - SIGNIFICANT S/G LEAKS
{ U } - HIGH ACTIVITY IN REACTOR COOLANT

INSTRUCTIONAL MATERIALS

Attendance sheet/Black ink pen/ Calculator

Completed Shift Turnover Sheet

Completed Nuclear Chemistry Summary Sheet

Working copies of:

AP-0103.2 (FIG 2,3) EOOS LOG	
AP-0103.2 (APPENDIX A-E)	
0-OSP-0200.1	-SCHEDULE OF PLANT CHECKS AND SURVEILLANCES
OP-0204.2	-PERIODIC TESTS, CHECKS, AND OPERATING EVOLUTIONS
3-OSP-0201.1	-RCO DAILY LOGS
3-OSP-041.1	-RCS LEAK RATE CALCULATION
AG-005	- STUDENT EVALUATION ATTATCHMENTS

SEQUENCE OF EVENTS

-100% Steady State Power Operations
-High Iodine 131 activity in Primary
-S/G tube leak
-R-19 in alarm
-S/G Tube Rupture Occurs/SI/Rx Trip
-EOP-E-3 Completed/Plant Stable
-Recovery procedure selected

1. Pre Exercise Briefing

ACTION:

Perform the following

- Exercise Overview
- Student objectives
- Conduct of Shift
- Team Skills/Communications
- Shift Assignment

II. Commence Simulator Exercise

Action:
- Start simulation
- Provide students with shift turnover information

Shift Turnover
- 100% Steady State power operations in progress
- Surveillances scheduled per Red Book
- "A" Charging pump is OOS (packing replacement)
- R-15 is OOS
- "B" AFWP is OOS
- "A" CCW Pump is OOS
- MIMS is OOS

STUDENT ACTIVITIES:
Perform a control board walkdown, operating log review and verbal shift turnover within 10 minutes of beginning the exercise.

SCENARIO# 42 EVENT# 1

BRIEF DESCRIPTION: HIGH RCS ACTIVITY, CHEMISTRY REQUESTS THAT LETDOWN PURIFICATION FLOW BE INCREASED.

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
		APSN	Acknowledges Chemistry request and directs that additional charging pump(s) be placed in service to meet additional letdown requirements.
		RO	Starts additional charging pump after checking with N.O. to ensure it is ready for start.
		RO	Opens an additional orifice in response to Chemistry request to increase letdown purification flow.
		RO	Verifies that charging and letdown flows are balanced.

SCENARIO# 42 EVENT# 2

BRIEF DESCRIPTION: STEAM GENERATOR TUBE LEAK INITIATED, R-19 ALARMS, SECONDARY RADIATION INCREASES.

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
		RO	Acknowledges annunciator H1/4, Process Radiation Alarm.
		BOP	Verifes the validity of the R-19 alarm.
		BOP	Informs ANPS of the validity of the alarm, calls the T.O. to verify that automatic actions associated with the alarm, (blowdown isolation), have occurred.
		APSN	Calls Chemistry to initiate NC-70 calculation, calls HP to initiate surveys of turbine deck and Main Steam lines.
		RO	Checks primary parameters for indication of leak severity.
		BOP	Checks secondary parameters for indication of leak severity.
		APSN	Calls load dispatcher to inform system of tube leakage and possible S/D requirement based on leak determination.

SCENARIO# 42 EVENT# 3

BRIEF DESCRIPTION: STEAM GENERATOR TUBE RUPTURE INITIATED, PRESSURIZER LEVEL AND PRESSURE DECREASE RAPIDLY.

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	TIME CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
			RO	Recognizes that charging pump speed is increasing, pressurizer level/pressure are decreasing. Informs ANPS of these developments.
			BOP	Recognizes increase in "A" S/G level.
	*		RO/BOP	Perform Immediate Actions for Reactor Trip. - Manually trip the reactor. - Verify reactor trip. - Verify turbine trip. - Verify power to 4KV buses.
*			RO	- Check if SI is actuated. Recognize that "3A" SI pump did not start, manually starts pump. - Verify Feedwater Isolation. - Verify Containment Isolation Phase A. - Verify AFW pumps running. - Verfiy SI pumps running. - Verify 2 of 3 CCW/ICW pumps - Verify 2 of 3 ECCU's and filter fans running. - Check if Main steam lines should be isolated. - Verify Containment Spray not required.
			APSN	Review immediate actions with RO/BOP to ensure that all have been performed.
			APSN/RO/BOP	Perform actions of steps 15-24 of E-0.

SCENARIO# 42 EVENT# 3

BRIEF DESCRIPTION: STEAM GENERATOR TUBE RUPTURE ,EOP-E-3 ACTIONS

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	TIME CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
*			APSN	Transition to E-3. Direct RO/BOP actions in E-3 to ensure mitigation of severity of event.
*		*	BOP	Identify Ruptured S/G.
*		*	BOP	Isolate ruptured S/G. Recognize MSIV will not close. Perform RNO
			BOP	Check ruptured S/G pressure > 600 psig.
	*	*	RO/BOP	Initiate RCS Cooldown to temperature requred by step 14 of E-3.
*			APSN	Direct transition to FR-P.1 when conditions requiring transition are identified
			RO	Establish charging flow
*		*	RO	Depressurize RCS to minimize break flow and refill pressurizer.
			RO	Verify SI flow not required and secure SI flow.
			RO	Establish letdown flow, control RCS pressure and charging flow to minimize RCS to secondary leakage
			BOP	Stop EDG's., Ensure that CV-3-1500 is closed.

SCENARIO# 42 EVENT# 3

BRIEF DESCRIPTION: STEAM GENERATOR TUBE RUPTURE (cont.)

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
		RO	Check RCP status.
		RO	Check if Source range NI is energized.
		PSN/APSN/TSC	Determine appropriate Post-SGTR cooldown method.
		APSN/RO/BOP	Transition to selected cooldown procedure, ES ECA 3.1, 3.2, or 3.3
*		PSN	Classify the event as per EP-20101, Table 1, paragraph 3 item 2 (ALERT).

SIMULATOR EXERCISE OBJECTIVE REVIEW SHEET

Based on the objectives for the exercise just completed, rate your performance of the objectives on a scale of 1 - 5, 5 being the highest rating.

OBJECTIVE	RATING	COMMENTS/TIMELINE
1.0 Demonstrate the ability to conduct plant power operations, including the ability to diagnose off normal conditions and to take appropriate actions.		
1.1 Given procedures for plant power operations, and a Shift relief status sheet:		
a. Calculate an RCS leak rate per 3-OSP-041.1		
b. Adjust charging and letdown as necessary to maintain Pzr level in the proper band		
c. Investigate/resolve any ARMS and/or PRMS alarms		
d. Determine, from chemistry results, if secondary activity levels are within Tech Spec limits		
e. Isolate/cooldown the affected S/G to minimize secondary system/off-site contamination		
1.2 Given plant conditions during plant power operations:		
a. Identify abnormalities by assessing actual system response with respect to predicted system response.		
b. Investigate the cause and effect of abnormalities in system performance.		
c. Implement applicable procedures.		
d. Perform immediate actions without reference to procedure.		

OBJECTIVE	RATING	COMMENTS/TIMELINE
1.3 Given plant conditions during plant power operations, IDENTIFY and IMPLEMENT the applicable on-site and off-site reports and notifications.		
1.4 Conduct plant power operations using the following principles for operational effectiveness as they apply to all operators:		
a. Plant and control room communication		
b. Plant/Control Board Monitoring		
c. Plant/Control Board Manipulation		
d. Operational Problem solving		
e. Use of Normal/Off-normal Procedures/Technical Specifications		
f. Use of Emergency Procedures IAW Rules of Usage of the EOP's.		
g. Annunciator Recognition and Response		
h. Written Communications/logs		
i. ALARA Awareness		
1.5 Conduct a plant power operations using the following principles of operational effectiveness as they apply to supervisory personnel:		
a. Team performance management		
b. Problem Solving		
c. Decision Analysis		
d. Action Planning		
1.6 The Shift Technical Advisor (STA) will demonstrate the ability to provide diagnositic support to Operations personnel during Off-normal and Emergency events and to advise the PS-N of actions required to terminate or mitigate the consequences of such events.		
a. Maintain awareness of plant status, configuration,and alignment of systems/components necessary for safe operation and shutdown.		
b. Assess all reactor trips and associated transients with regard to safety.		
c. Assist the PS-N in assuring that applicable On/Off-Site notifications are made.		
d. Provide technical assistance during the following evolutions:		
1. RCS Leak Rate calculation		
2. Steam Generator Tube Rupture		

OBJECTIVE	RATING	COMMENTS/TIMELINE
1.7 The control room team shall demonstrate the ability to apply techniques presented during the TEAM WORK AND ACCOUNTABILITY Workshop that contribute to effective team performance.		

TERMINAL OBJECTIVE EVALUATION CRITERIA

Rate Operations Team performance of the following:

A. Minimized off-site releases.

B. Immediate actions performed without prompting.

C. All substeps of procedures performed.

D. Communications were proper.

E. Appropriate off-site notifications without instructor prompting.

F. Instructor assistance was not required.

TURKEY POINT PLANT
SIMULATOR EXERCISE GUIDE

Revision: 0

Prepared by: T.A. VEHEC

Reviewed by: L Doïlef

Approved by: _____ Date: 5/31/89
(Training Supervisor)

Approved by: _____
(Operations Supervisor)

PROGRAM: LICENSED OPERATOR REQUALIFICATION NRC EXAM SCENARIO

EXERCISE TITLE: LOSS OF ALL FEEDWATER

EXERCISE GUIDE NUMBER: 018C-L-E

SCENARIO NUMBER: 39

ESTIMATED TIME: 1 HOUR

REFERENCE BLOCK DIAGRAM(S): IO_HTSK, IO_TRIP

REFERENCES:

3-GOP-301:	HOT STANDBY TO POWER OPERATIONS (2/28/89)
3-ONOP-0208.3:	ANNUN RESPONSE PROCEDURE (2/28/89)
3-EOP-E-0:	REACTOR TRIP OR SI (3/10/89)
EP-20101:	DUTIES OF THE EMERGENCY COORDINATOR (5/5/89)
3-EOP-FR-H.1:	RESPONSE TO LOSS OF SECONDARY HEAT SINK (9/19/89)
AP-0103.2:	DUTIES AND RESPONSIBILITIES OF OPERATORS ON SHIFT (1/5/89)
ONOP-7308.1	MALFUNCTION OF AFW SYSTEM (3/24/89)
3-OSP-0200.1:	SCHEDULE OF PLANT CHECKS AND SURVEILLANCES (4/28/89)
3-OP-0204.2:	PERIODIC TESTS, CHECKS AND SURVEILLANCES (4/26/89)
IE NOTICE:	85-50 COMPLETE LOSS OF MAIN AND AUX FEED (DAVIS-BESSE INCIDENT)

TECHNICAL SPECIFICATIONS
PLANT CURVE BOOK

TERMINAL OBJECTIVE: The SRO and Reactor Control Operator (RCO) will demonstrate the ability to perform reactor power operations, and respond to abnormal plant conditions IAW approved plant procedures ensuring that the health and safety of the public is protected and the integrity of the plant maintained. The RCO will recognize and log entry conditions to the Technical Specifications

ENABLING OBJECTIVES:

1.0 Demonstrate the ability to conduct plant power operations, including the ability to diagnose off normal conditions and to take appropriate actions.

1.1 Given procedures for plant power operations, and a Shift relief status sheet:

a) Recognize and respond to a failure of Power Range N-44
b) Recognize and respond to a loss of all feedwater outside containment IAW FR-H.1.
c) Attempt to restore feed to the S/G's using the SGFP's.
d) Identify "Red Path" conditions for Heat Sink and implement corrective actions as per plant procedures
f) Classify the event as per EP-20101, Table 1.

1.2 Given plant conditions during plant power operations:

a. Identify abnormalities by assessing actual system response with respect to predicted system response.
b. Investigate the cause and effect of abnormalities in system performance.
c. Implement applicable procedures.
d. Perform immediate actions without reference to procedure.

1.3 Given plant conditions during plant power operations, IDENTIFY and IMPLEMENT the applicable on-site and off-site reports and notifications.

1.4 Conduct plant power operations using the following principles for operational effectiveness as they apply to all operators:

a. Plant and control room communication
b. Plant/Control Board Monitoring
c. Plant/Control Board Manipulation
d. Operational Problem solving
e. Use of Normal/Off-normal Procedures/Technical Specifications
f. Use of Emergency Proceures IAW Rules of Usage of the EOP's.
g. Annunciator Recognition and Response
h. Written Communications/logs
i. ALARA Awareness

1.5 Conduct plant operations using principles of operational effectiveness as they apply to supervisory personnel:

a. Team performance management
b. Problem Solving
c. Decision Analysis
d. Action Planning

1.6 The Shift Technical Advisor (STA) will demonstrate the ability to provide diagnostic support to Operations personnel during Off-normal and Emergency events and to advise the PS-N of actions required to terminate or mitigate the consequences of such events.

a. Maintain awareness of plant status, configuration, and alignment of systems/components necessary for safe operation and shutdown.
b. Assess all reactor trips and associated transients with regard to safety.
c. Assist the PS-N in assuring that applicable On/Off-site notifications are made.
d. Provide technical assistance during the following events:

1) Immediate Bleed and Feed implementation requirements
2) "Red Path" conditions for Heat Sink

1.7 The control room team shall demonstrate the ability to apply techniques presented during the TEAM WORK AND ACCOUNTABILITY Workshop that contribute to effective team performance.

SIMULATOR LESSON PLAN INSTRUCTOR INFORMATION

TASK NUMBERS AND STATEMENTS:

LS OBJECTIVES	TASKS #	TASK DESCRIPTION
1.5	D-015	DIRECT SHIFT PERSONNEL DURING MAJOR PLANT EVOLUTIONS.
1.5, 1.6	E-003	CHECK PLANT EQUIPMENT STATUS
1.4, 1.1	E-006	REVIEW CONTROL BOARD STATUS
1.4	F-008	VERIFY CORRECT PERFORMANCE OF EMERGENCY AND OFF-NORMAL PROCEDURES
1.1, 1.4, 1.3	F-009	IDENTIFY/RESPOND TO AN OFF-NORMAL EVENT
1.3	F-010	MAKE PROTECTIVE ACTION RECOMMENDATIONS
1.3	F-011	REPORT SIGNIFICANT EVENTS
1.2c	F-012	DIRECT RESPONSE AS EMERGENCY COORDINATOR

RCO OBJECTIVES	TASKS #	TASK DESCRIPTION
1.4	BB-006	MAKE PLANT WIDE ANNOUNCEMENTS
1.1, 1.4	CC-003	ALIGN SAFETY INJECTION SYSTEM
1.1, 1.4	CC-004	ASSURE PROPER SAFETY INJECTION SYSTEM OPERATION
1.4	DD-001	DUMP STEAM TO CONDENSER
1.2, 1.4	DD-002	INVESTIGATE CAUSE OF CONTROL BOARD ALARMS
1.2, 1.4	DD-004	VERIFY STEAM DUMP/TURBINE BYPASS CONTROL SYSTEM ALIGNMENT ETC.
1.4	DD-005	VERIFY SYSTEM AUTOMATIC OPERATION
1.2, 1.4	FF-005	INVESTIGATE CONTROL BOARD ALARM
1.2, 1.4	FF-013	MONITOR STEAM GENERATOR OPERATION
1.4	G-001	MONITOR CVCS OPERATION
1.4	G-002	ADJUST CHARGING FLOWRATE
1.4	G-012	CONTROL RCS PRESSURE
1.4	L-027	MONITOR ROD CONTROL SYSTEM
1.4	M-001	CHECK ALTERNATING CURRENT (AC) ELECTRICAL DISTRIBUTION SYSTEM INSTRUMENTS
1.4	M-002	CHECK DIRECT CURRENT (DC) ELECTRICAL SYSTEM INSTRUMENTATION
1.2, 1.4	M-011	INVESTIGATE AC ELECTRICAL DISTRIBUTION SYSTEM ALARMS
1.4	O-001	ACTIVATE FEED TRAIN
1.4	O-006	CONTROL STEAM GENERATOR LEVEL WITH AUXILIARY FEEDWATER
1.4	O-009	FEED STEAM GENERATOR FROM FOSSIL UNIT
1.4	O-010	FEED STEAM GENERATOR WITH AUXILIARY FEEDWATER
1.4	O-018	START THE AUXILIARY FEEDWATER PUMPS
1.4	O-021	STOP HEATER DRAIN PUMP
1.1, 1.4	O-026	RESPOND TO LOSS OF FEEDWATER
1.2, 1.4	O-027	MONITOR MAIN FEEDWATER OPERATION
1.2, 1.4	O-028	MONITOR CONDENSATE SYSTEM
1.1, 1.2, 1.4	R-011	CHECK PLANT STATUS
1.4	R-021	LOG EQUIPMENT IN AND OUT OF SERVICE
1.4	R-024	LOG SHIFT ACTIVITIES
1.4	R-025	LOG SHIFT READINGS
1.4	W-002	ACTUATE SPRAY VALVES
1.2, 1.4	W-025	MONITOR PRESSURIZER PRESSURE
1.4	W-026	MANUALLY OPERATE PORV'S
1.4	T-001	ADJUST MSRH TIMED CONTROL VALVES (ADJUST STEAM REHEATER CONTROL VALVES)
1.4	T-002	DUMP STEAM TO ATMOSPHERE
1.2, 1.4	T-003	INVESTIGATE CAUSE OF CONTROL BOARD ALARMS
1.4	T-007	REMOVE MOISTURE SEPARATOR/REHEATER FROM SERVICE ETC.
1.4	T-011	OPERATE HIGH PRESSURE STEAM DRAINS
1.4	T-012	OPERATE LOW PRESSURE STEAM DRAINS
1.2, 1.4	T-013	MONITOR THE MAIN AND REHEAT STEAM SYSTEMS
1.2, 1.4	U-001	MONITOR TURBINE GENERATOR
1.4	U-014	PLACE TURBINE TURNING GEAR INTO SERVICE
1.4	U-019	SHUTDOWN TURBINE GENERATOR
1.4	U-026	VERIFY START/STOP OF TURBINE GENERATOR OIL PUMPS
1.4	V-002	ACTIVATE NUCLEAR INSTRUMENTATION CHANNEL
1.2, 1.4	W-025	MONITOR PRESSURIZER PRESSURE
1.2, 1.4	Y-001	ASSURE SATURATION MARGIN DISPLAY ON QUALIFIED SAFETY PARAMETER DISPLAY
1.4	Y-007	COOL DOWN RCS
1.2, 1.4	Y-018	MONITOR REACTOR COOLANT SYSTEM
1.2, 1.4	Y-019	MONITOR OPERATION OF THE REACTOR COOLANT PUMPS
1.4	Y-028	SHUT DOWN THE REACTOR COOLANT PUMP
1.4	Z-003	MANUALLY INITIATE SAFETY INJECTION
1.2, 1.4	Z-004	INVESTIGATE CONTROL PANEL ALARMS
1.2, 1.4	Z-010	RESPOND TO UNIT TRIP

EXERCISE SYNOPSIS

The plant is at 100% power Steady state conditions, MOL. 5 minutes into the shift, an NI-44 failure high occurs. Automatic rod motion into the core occurs and is stopped by the operators. N-44 is removed from service IAW ONOP-059.8. Following this failure a rupture occurs on the main feed pump suction headers inside the feed pump room. This causes a loss of all normal feed and a Reactor trip on Low-low S/G level. AFW is disabled and the isolation valves from Unit 1/2 and Unit 4 sources of feedwater to the S/G's are inaccessible due to the hostile environment. The operators attempt to align the Standby S/G Feed pumps to supply the S/G's. The attempt to restore feed using the SSGFP's, is also unsuccessful. This results in no feed flow being available to the S/G's. A decision to initiate feed and bleed is made based on the plant conditions. If necessary the instructor will role play the Plant Superintendent or TSC by recommending that feed and bleed be initiated. Time in core life is determined by the EFPD selected by the instructor. The instructor should ensure that core physics data made available to the students is consistent with this selection. The exercise is concluded at the evaluator's discretion, following the initiation of feed and bleed.

10CFR55.59(c) (3) (i) REQUIRED MANIPULATIONS MET BY THIS EXERCISE

{ K } - LOSS OF FEEDWATER (NORMAL AND EMERGENCY)
{ Y } - REACTOR TRIP

INSTRUCTIONAL MATERIALS

Attendance sheet/Black ink pen/ Calculator
Completed Shift Turnover Sheet
Working copies of:

AP-0103.2 (Fig 2,3)
AP-0103.2 (Appendix A-E)
3-GOP-301
3-OSP-0200.1 - SCHEDULE OF PLANT CHECKS AND SURVEILLANCES
3-OP-0204.2 - PERIODIC TESTS, CHECKS AND SURVEILLANCES
AG-005 - STUDENT EVALUATION ATTACHMENTS

SEQUENCE OF EVENTS

-100% power
- N-44 fails high
-Feed header rupture
-Feed pump trips, Turbine runback
-Reactor trip (low-low S/G level)
-S/G levels decreasing
-Transition to FR-H.1
-SSGFP aligned for service, no flow through DMOS-012.
-Initiate Bleed and Feed
-End of scenario

OUTLINE OF INSTRUCTION	INSTRUCTOR ACTIVITIES	NOTES/STUDENT ACTIVITIES
1. Pre Exercise Briefing	ACTION:	
	Perform the following:	
	- Exercise overview	
	- Student objectives	
	- Conduct of Shift	
	- Team Skills/Communications	
	- Shift Assignments	
II. Commence Simulator Exercise	Action: Start simulation	STUDENT ACTIVITIES:
		Perform a control board walkdown, operating log review and verbal shift turnover within 10 minutes of beginning exercise.
	Shift Turnover - Plant at 100% power, steady state, ARO - "B" AFWP is O.O.S. - "A" CCW Pump is O.O.S. - R-15 is O.O.S. - "A" Charging pump O.O.S. - MIMS O.O.S.	

SCENARIO# 39 EVENT# 1

BRIEF DESCRIPTION: PR N-44 FAILS HIGH

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
		RO	- Recognizes and responds to N1-44 failure.
		APSN	Directs RO actions IAW ONP-059.8 - Ensure that all switches for affected channel on the NIS racks are repositioned. - DROPPED ROD MODE switch for N-44 to BYPASS - ROD STOP BYPASS switch for N-44 to BYPASS - UPPER SECTION comparator defeat switch to N-44 - LOWER SECTION comparator defeat switch to N-44 - Transfer the applicable POWER MISMATCH BYPASS switch to bypass N-44. - Transfer the COMPARATOR CHANNEL DEFEAT switch to N-44. - Trip the power range bistables within one hour by pulling the instrument power fuses.

SCENARIO# 39 EVENT# 2

BRIEF DESCRIPTION: FEED PUMP SUCTION PIPING RUPTURE

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
		BOP	Acknowledges multiple alarms indicating that a loss of feed has occurred.
		RO/BOP	Acknowledges T.O. report of a Main feed line rupture on the feed pump suction header.
	*	APSN/RO/BOP	Respond to lo-lo S/G level reactor trip IAW EOP-E-0.
		PSN/APSN	Direct shift crew actions IAW EOP-E-0.
	*	RO/BOP	Perform immediate actions of EOP-E-0. - Verify reactor trip. - Verify turbine trip. - Verify power to 4KV buses. - Check if SI is actuated.
		PSN/APSN	Transition to ES-0.1

SCENARIO# 39 EVENT# 2

BRIEF DESCRIPTION: LOSS OF HEAT SINK

INDIVIDUAL CRITICAL STEP	CREW CRITICAL STEP	TIME CRITICAL STEP	POSITION	CANDIDATE ACTION/BEHAVIOR
*			BOP	Recognizes that the AFWP's are all tripped on overspeed and that there is no Feed flow to any S/G. Informs ANPS of this situation.
*			ANPS	- Transitions to FR-H.1. - Directs BOP/RO actions IAW FR-H.1.
			BOP/RO	Contact field operators to determine if any feed is available to the S/G's.
			BOP	After acknowledging that feed is available from the Standby S/G feed pumps, directs I.O. to open DWDS-012 in preparation for start of Standby S/G feed pump
			ANPS	Directs BOP to establish 130gpm flow to the S/G's from the SSGFP.
*			BOP	Recognizes that flow to the S/G's with SSGFP is not established, reports this to ANPS.
		*	PSN/ANPS	Makes determination to initiate feed and bleed for heat removal.
*			PSN	Classifies event per EP-2010. (Should be SITE AREA EMERGENCY- PARAGRAPH 7 - LOSS OF SAFE SHUTDOWN FUNCTION. (ITEM 3 UNDER THIS HEADING). - loss of secondary heat sink AND feed and bleed required.

END OF SCENARIO

SCENARIO END SUMMARY AND POST EXERCISE CRITIQUE

A. STUDENT REVIEW OF ENABLING OBJECTIVES	Instructor passes out Objective review sheets to the operators and directs the Senior operator to take 15-20 minutes to review them with the crew.	Operators take Objective review sheets to the classroom and review them with the senior operator in the crew.
B. INSTRUCTOR CRITIQUE OPERATOR PERFORMANCE	Critique the students knowledge level and skill in performing the required tasks. Critique individual and team responses by commenting on both the appropriateness and timeliness of individual and team responses. Emphasize good examples of teamwork illustrated during the scenario. Ensure that all T/S compliance actions are entered in the CO's log. Ensure that the students understand the theoretical aspects of performing the tasks Review the Objective review sheet and timeline filled out during exercise and prepare comments for critique of operators.	

C. INSTRUCTOR GUIDED DISCUSSION OF OBJECTIVE REVIEW SHEETS	Instructor asks senior shift crew member to reconstruct exercise sequence of events.	Senior crew member reconstructs exercise sequence.
	Instructor interjects comments and refreshes operators memory, if required, to reconstruct event sequence.	
	Instructor leads operator discussion of Review of Objectives sheets, interjecting comments and rebuttals as necessary.	Operators discuss their comments on the Review of Objectives sheets.
D. AREAS NEEDING IMPROVEMENT	Instructor review areas identified as needing improvement in discussion of Objective Review Sheets.	
E. THEORY REINFORCEMENT		
F. STUDENT FEEDBACK AND CRITIQUE	Instructor requests feedback from the operators on the realism of the simulator exercise.	Operators provide feedback to the instructor on the validity of the simulator exercise.

SIMULATOR EXERCISE OBJECTIVE REVIEW SHEET

Based on the objectives for the exercise just completed, rate your performance of the objectives on a scale of 1 - 5, 5 being the highest rating.

OBJECTIVE	RATING	COMMENTS/TIMELINE
1.0 Demonstrate the ability to conduct plant power operations, including the ability to diagnose off normal conditions and to take appropriate actions.		
1.1 Given procedures for plant power operations, and a Shift relief status sheet:		
a) Recognize and respond to a failure of N-44 with a failure of rods to move in automatic.		
b) Recognize and respond to a loss of all feedwater outside containment IAW FR-H.1.		
c) Attempt to restore feed to the S/G's using the SGFP's.		
d) Identify "Red Path" conditions for Heat Sink and implement corrective actions as per plant procedures		
e) Classify the event as per EP-20101, Table 1.		
1.2 Given plant conditions during plant power operations:		
a. Identify abnormalities by assessing actual system response with respect to predicted system response.		
b. Investigate the cause and effect of abnormalities in system performance.		
c. Implement applicable procedures.		
d. Perform immediate actions without reference to procedure.		
1.3 Given plant conditions during plant power operations, IDENTIFY and IMPLEMENT the applicable on-site and off-site reports and notifications.		

OBJECTIVE	RATING	COMMENTS/TIMELINE
1.4 Conduct plant power operations using the following principles for operational effectiveness as they apply to all operators: a. Plant and control room communication b. Plant/Control Board Monitoring c. Plant/Control Board Manipulation d. Operational Problem solving e. Use of Normal/Off-normal Procedures/Technical Specifications f. Use of Emergency Procedures IAW Rules of Usage of the EOP's. g. Annunciator Recognition and Response h. Written Communications/logs i. ALARA Awareness		
1.5 Conduct a plant power operations using the following principles of operational effectiveness as they apply to supervisory personnel: a. Team performance management b. Problem Solving c. Decision Analysis d. Action Planning		
1.6 The Shift Technical Advisor (STA) will demonstrate the ability to provide diagnositic support to Operations personnel during Off-normal and Emergency events and to advise the PS-N of actions required to terminate or mitigate the consequences of such events. a. Maintain awareness of plant status, configuration,and alignment of systems/components necessary for safe operation and shutdown. b. Assess all reactor trips and associated transients with regard to safety. c. Assist the PS-N in assuring that applicable On/Off-Site notifications are made. d. Provide technical assistance during the following events: 1. Immediate Bleed and Feed implementation requirements 2. "Red Path" conditions for Heat Sink		
1.7 The control room team shall demonstrate the ability to apply techniques presented during the TEAM WORK AND ACCOUNTABILITY Workshop that contribute to effective team performance.		

TERMINAL OBJECTIVE EVALUATION CRITERIA

Rate Operations Team performance of the following:

A. Performed corrective actions for fire in 4kv bus

B. Proper procedural use and compliance for non-routine evolutions

C. Consultation/compliance with Tech Specs/Plant procedures

D. Communications were proper.

E. Instructor assistance was not required

ENCLOSURE 4

SIMULATOR FIDELITY REPORT

Facility Licensee: Florida Power and Light Company

Facility Docket Nos.: 50-250 and 50-251

Operating Test Administered On: June 1, 1989

This form is used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating test, the following items were observed:

Item	Description
Keys	The key locker had been replaced and moved to replicate that of the control room.
Lighting	The poor lighting existing in the Plant Supervisor's - Nuclear (PSN) office had been improved.
Clock	The simulator clocks have been improved.

The overall performance of the simulator hardware and modeling continued to be exemplary during this examination.