P. O. BOX 14000, JUNO BEACH, FL 33408-0420

ENCLOSURE 3

FEBRUARY 1 1988

L-88-49

Dr. J. Nelson Grace
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta St., N.W., Suite 2900
Atlanta, GA 30323

Dear Dr. Grace:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Senior Reactor Operator Exam Comments

Florida Power & Light Company has reviewed the Senior Reactor Operator Upgrade examination presented to Turkey Point operators on January 26, 1988. As discussed in the exit meeting following the examination period, FPL has prepared comments on questions in the examination for NRC review and consideration prior to grading the examinations. The comments are provided in the attachment.

Should you or your staff have any questions on this information, please contact us.

Very truly yours,

J. K. Hays
for C. O. Woody
Executive Vice President

COW/PLP/gp

Attachment

cc: Document Control Desk, USNRC
Mr. J. A. Arildsen, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

PLP/001.SOL

I have reviewed the NRC Exam Question Review Package and concur with the responses provided.

T.A. Finn Training Department Supt.-N	Jesus Arias Jr. J. Arias PTN Regulation and Compliance
F.H. Southworth for C.J. Baker Plant Manager-N	J.S. Odom J.S. Odom Site Vice President

NRC EXAM QUESTION REVIEW

QUESTION: 5.11

A critical boron calculation has been performed prior to startup. State how the calculated value changes for each of the following. Answer INCREASE, DECREASE or REMAIN THE SAME.

b). The desired critical rod height is increased from 160 steps withdrawn to 180 steps withdrawn.

RESPONSE:

We request that the answer be changed from decrease to increase for the following reason:

If actual critical rod height has been increased by 20 steps this adds more positive reactivity than originally calculated. Critical boron concentration would have to increase by an amount that would insert enough negative reactivity to offset the positive reactivity.

NRC EXAM QUESTION REVIEW

QUESTION: 5:15

For each of the following parameters, state whether a reactor power increase from 50% to 75% will cause the parameter to INCREASE, DECREASE or REMAIN THE SAME. Consider each case separately.

NOTE: Assume that system reaches equilibrium after power change.
c). Shutdown Margin. (power change by rod withdrawal only)

RESPONSE:

We request that the answer be changed from increase to no change for the following reason:

Pulling rods to increase power level will increase RCS temperature. As temperature increases, power defect adds an amount of negative reactivity equal to the positive reactivity added by the control rods. By having power defect alone offset the positive reactivity addition the shutdown margin will remain the same.

NRC EXAM QUESTION REVIEW

QUESTION: 5:17

The following are plant parameters of Reactors "A" and "B" before and after simultaneous reactor trips:

	Reactor"A"	Reactor "B"
Before trips-		
Steady state power	100%	50%
Boron concentration	1500 ppm	1500 ppm
After trips-		
Reactivity from rod insertion	-8100 pcm	-8000 pcm
Average temperature	547 deg	547 deg
Maximum single IRW	-2000 pcm	-2000 pcm

Assuming that no operator action takes place, identify the plant that has the larger shutdown margin {Denote as "REACTOR A" or "REACTOR B"} at each of the following times.

NOTES: 1) Figures 5.1, 5.2, 5.3 and 5.4 are enclosed for reference.
2) Figures 5.1, 5.2, 5.3 and 5.4 are applicable to BOTH REACTORS.

a) One (1) minute after the trip
b) Fifty (50) hours after the trip

RESPONSE:

Part a
We request that you accept "Both Reactor A and Reactor B are the same" as an additional answer for the following reason:

Using the curves supplied the following data was determined-

	Reactor A	Reactor B
Rods	-6100	-6000
Power Defect	+1120	+600
Xenon	-2600	-2200
Total	-7580	-7600

These results are very close and some allowance should be granted for slight differences in reading the curves.

NRC EXAM QUESTION REVIEW

Part c
We request that the answer be changed to Reactor A for the following reason:

Using the curves supplied the following data was determined-

	Reactor A	Reactor B
Rods	-6100	-6000
Power Defect	+1120	+600
Xenon	-900	-400
Samarium	-680	-605
Total	-6560	-6405

NRC EXAM QUESTION REVIEW

QUESTION: 5.18

Answer each of the following statements concerning the count rate (inverse multiplication) plot for rod withdrawal. TRUE or FALSE.

c) A count rate, which is taken before the reactor power level reaches steady state (i.e. count rate is taken shortly after reactivity is added), will result in a HIGHER predicted critical rod height than if a steady state count rate were taken.

d) If the count rate taken prior to the last rod withdrawal did NOT reach steady state, the predicted critical rod height would be HIGHER than if that count rate taken had reached steady state.

RESPONSE:

We request that answer D be changed to True for the following reason:

After reading statement C, which is True, and comparing it to statement D we feel that both statements are saying the same thing.

NRC EXAM QUESTION REVIEW

QUESTION: 6.02

Select the statement below which most correctly describes the effect of overcompensating ONE (1) Intermediate Range Nuclear Instruments (IRNI).

a. The indicated IRNI power level is LOWER than the actual power level, and the Source Range Nuclear Instruments (SNRI) will automatically energize above their reset setpoint during a reactor shutdown.

b. The indicated IRNI power level is LOWER than the actual power level, but the SRNI will NOT automatically energize above their reset setpoint during a reactor shutdown.

c, The indicated IRNI power level is HIGHER than actual power level, and the SRNI will automatically energize above their reset setpoint during a reactor shutdown.

d. The indicated IRNI power level is HIGHER than the actual power level, but the SRNI will NOT automatically energize above their reset setpoint during a reactor shutdown.

RESPONSE:

We request that the answer be changed to B for the following reason:

An overcompensated intermediate range instrument will indicate lower than the actual power level which makes answers C and D incorrect. During a reactor shutdown the overcompensated instrument will decrease faster (indicated) but the logic for auto reset of the source ranges requires that both intermediate instruments have decreased below the auto reset setpoint. The source range instruments will not auto reset until the properly compensated intermediate range instrument has reached the auto reset setpoint. This makes answer B the correct response.

REFERENCE;

SD-4, Excore Nuclear Instrumentation pgs. 33, 38, Figure 23

The lithium and alpha particle resulting from this reaction cause ionization in the outer N2 gas volume. The electrons produced by the ionization are collected on the outer can wall. This produces a signal which is proportional to the neutron flux. Electrons are also collected on the outer can wall from the gamma radiation which interacts with the outer gas volume. This additional signal is proportional to the gamma flux and is additive to the neutron flux signal. The outer chamber operates in the ionization region thus all the charged particles produced in the initial ionizing events are collected on the electrodes.

In the inner can volume, the gamma flux also reacts with the N2 gas producing a signal proportional to the gamma radiation. The inner chamber is operated in the recombination region to permit adjustment of the output current by varying the compensating voltage. If the inner volume compensation voltage is set properly the outer can signal of gammas plus neutrons interacts with the inner can gamma only signal and the gamma signals cancel out. This neutron only signal is then amplified before it is displayed on the meter or sent to the protection and control circuitry.

Gamma Compensation

It becomes necessary to define the term compensation and the effects of under compensation and over compensation to clearly understand the process of neutron detection in the intermediate range.

Compensation is a term applied to the negative voltage signal applied to the inner volume of the CIC which cancels or compensates for the current signal produced by the gamma radiation interacting with the outer volume of the detector. Refer to Figure 22. This becomes very important to the operator because an incorrect setting of compensating voltage i.e., overcompensation or under-compensation would cause an erroneous neutron level indication on the meters. Refer to Figure 22. As noted on the curve undercompensation results in an erroneous high neutron level reading about 10 minutes after shutdown; overcompensation results in an erroneous low neutron level about 12 minutes after shutdown. Compare the two initial ranges and also relate intermediate amps to Source Range count rate (below P-10). If improper compensation is suspected advise I & C promptly.

REMOTE RECORDER

This is the same recorder unit discussed previously with regard to the Source Range. This recorder through selector switches also serves to record the selected IR and PR. It is a two-pen recorder. It will normally record one IR at a time. A 0-50 mv dc signal from the isolation amplifier is supplied to the recorder and is proportional to the IR ion chamber current of 10-11 to 10- 3 amperes.

IR STARTUP-RATE CIRCUITY

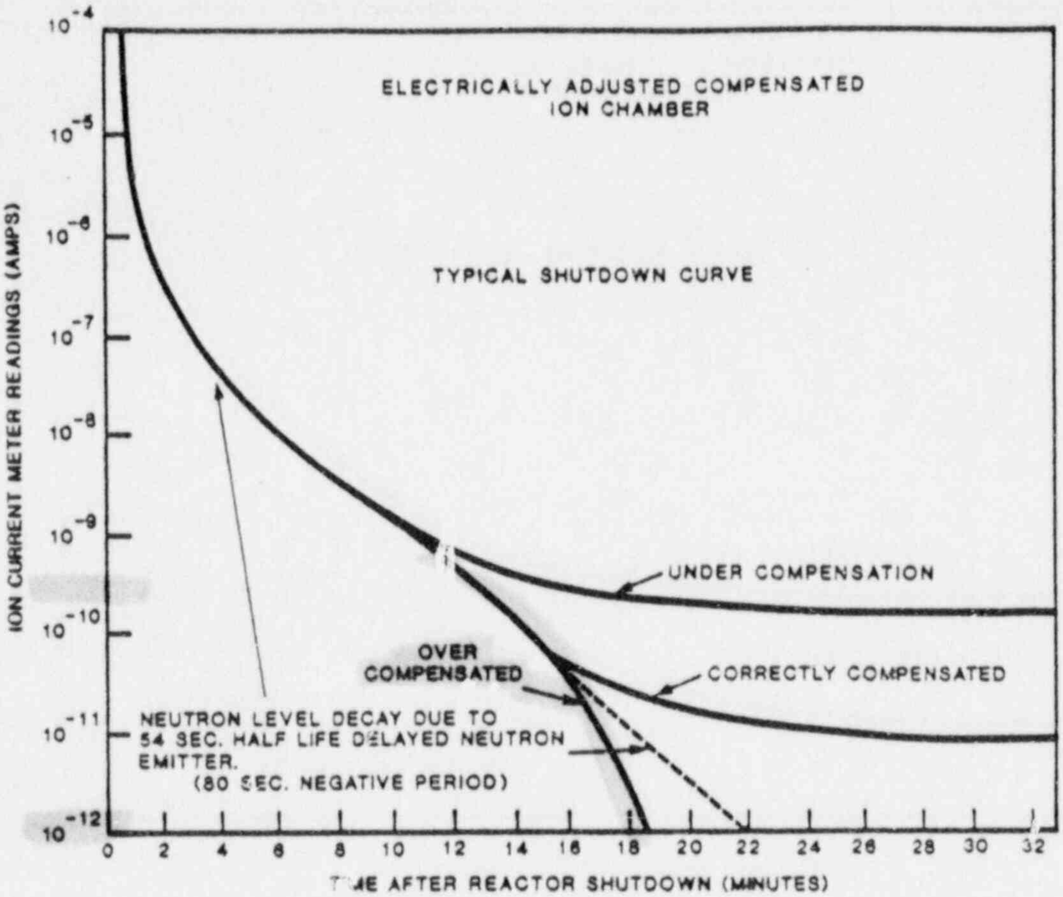
As described earlier for the SR, the startup-rate drawer receives input signals (0-10V DC) from each of the SR and IR channels. Refer to Figures 16 & 29. Four rate amplifier modules condition each of these signals and transmit four respective rate signals to the respective control room startup-rate meters. The remote IR startup-rate indicators for IR channels N-35 and N-36 are located on the console. A test module may be used to inject a test signal into any one of the rate circuits and can be monitored on a test meter mounted on the front panel of the SUR drawer. Two power supplies are installed in such a manner as to ensure rate indication from at least one Source and Intermediate Range channel pair upon the loss of one power supply.

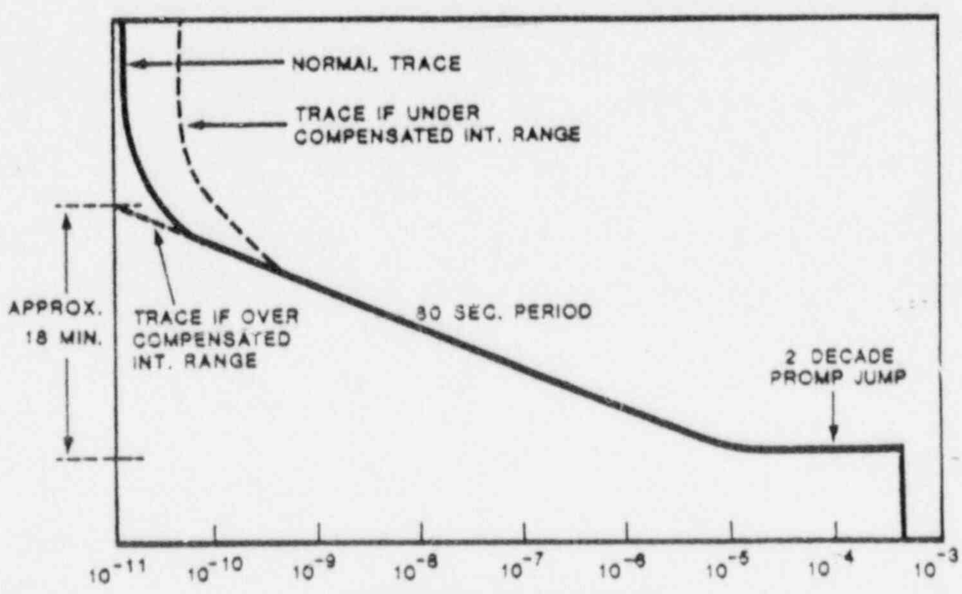
P-6 Permissive

The P-6 permissive is energized during reactor startup when 1 out of the 2 Intermediate Range channels reaches 10-10 amperes. Refer to Figure 25. This is equivalent to a source range count level of approximately 4×10^4 cps. Once the P-6 permissive is satisfied (indicated by a status light on VPB status light panel B, windows 9- 1 and 9-2) the source range high level trip can be manually blocked. By blocking the trip on the console, the high voltage to the source range detector is automatically removed. The provision is only operational below permissive P-10 which is supplied by the Power Range channels. Above P-10 the defeat circuit is automatically bypassed and source range high voltage cutoff is maintained.

When shutting down, source range high level trips and detector high voltage are automatically reactivated when both Intermediate Range channels drop below 10-10 amperes (P-6 light de-energized). If an undercompensation of the IR detector is present preventing clearing of the P-6 permissive, Figure 17,

CIC GAMMA COMPENSATION





TYPICAL RECORDING CHART TRACE (TRIP FROM FULL POWER)

FIGURE 23

REV. 1

NRC EXAM QUESTION REVIEW

QUESTION: 6:06

State three (3) requirements which must be satisfied prior to enabling RHR isolation valve, MOV-750, to open. (Include setpoints as necessary.)

RESPONSE:

We request that you accept additional answers for the following reason:

The question asked for requirements but the answer listed interlocks. For this reason please accept the following as additional answers-

1. Breaker for MOV-750 closed
2. Power supply to MOV-750 available
3. The valve shall not be opened if reactor coolant pressure exceeds 450 psig or reactor coolant temperature exceeds 350°F.

REFERENCE:

SD-21, Emergency Core Cooling System, pg.16, Figure 9
3-GOP-305, Hot Standby to Cold Shutdown, pg.8 step 4.3.1 ,pg.26 step 5.10
3-OP-050, Residual Heat Removal System, pg.7 step 4.1, pg.9 steps 5.1.2.8 and 5.1.2.9

RESIDUAL HEAT REMOVAL SYSTEM

For the purposes of this discussion, the normal and emergency functions of the RHR system will be covered separately.

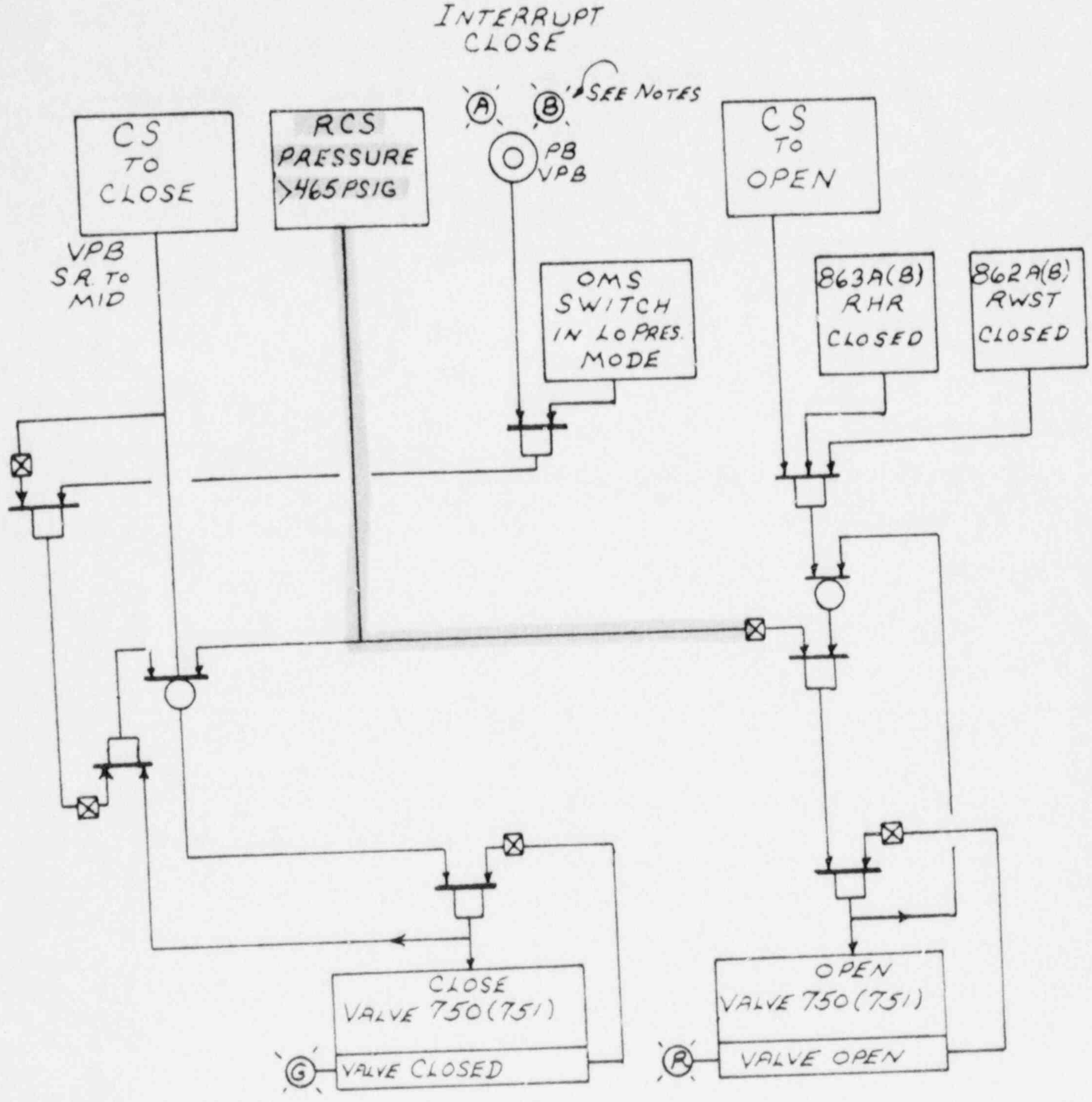
Normal System Functions

As was previously discussed, **the RHR system is normally used to remove the decay and sensible heat from the RCS during plant startup, cooldown, and refueling when RCS pressure is 450 psig and tempe .ture is 350°F.** The 350°F restriction is based on the RHR pump seal limitations. The 450 psig limitation is based on not exceeding RHR system design pressure. The 450 psig system limitation plus the RHR pump shutoff head of about 150 psig equals the system design pressure of 600 psig.

The design heat transfer rate for the RHR system is 58.8×10^6 Btu/hr based on a RHR heat exchanger inlet temperature (RCS) of 140°F and a CCW inlet temperature of 107.4°F. The design heat transfer rate is based on the decay heat removal requirements 20 hours after shutdown from an infinite period of reactor operation. With a total CCW flow of 20,400 gpm to the RHR heat exchangers, the RCS can be cooled down from 350°F to 140°F during the period between 4 hours to 20 hours after reactor shutdown without the CCW leaving the CCW heat exchanger exceeding 125°F. The RHR system is capable of removing more heat than design during a plant cooldown, since design heat transfer is based on the temperature difference of 140°F (reactor coolant) and 107°F (component coolant) as stated above. A rapid cooling rate may result in a significant increase in temperature of component coolant leaving the CCW heat exchanger. (Remember, from SD-8, that the maximum CCW supply temperature to the RCP's is 130°F. During a plant cooldown this becomes a limiting parameter.)

The RHR system consists of two independent, redundant trains. Each train consists of a pump, heat exchanger, piping, valves and attendant instrumentation. See Figure 8. Each pump and the heat exchangers are located in separate compartments on the [-4'] elevation of the auxiliary building.

RHR ISOLATION VALVES 750 OR 751



NOTES.
1. AMBER LIT >525 PSIG AND DURING CLOSE CYCLE OF VALVE
2. BLUE LIT <525 PSIG
3. BOTH AMBER AND BLUE LIT — PERMISSIVE TO INTERRUPT CLOSING CYCLE. OPERATE CS TO FULLY OPEN VALVE
4. WITH RCS >450 PSIG MOV 750 + 751 ARE CLOSED AND BKR 751 IS LOCKED OPEN

SD21

FIG 9
REV 1

4.2 Reactor Coolant Pumps

4.2.1 The precautions and limits listed in 3-OP-041.1, Reactor Coolant Pump, shall be observed.

4.2.2 Operation of reactor coolant pumps should continue until Mode 5, Cold Shutdown, is reached to provide pressurizer spray flow (B or C RCP), to prevent the temperature difference between loops from exceeding 25°F, and to ensure cooldown of steam generators.

4.3 Residual Heat Removal Loop

4.3.1 **The Residual Heat Removal (RHR) Loop Isolation Valves, MOV-3-750 and MOV-3-751 shall not be opened if reactor coolant pressure exceeds 450 psig or reactor coolant temperature exceeds 350°F.**

1. The RHR Isolation Loop Valves, MOV-3-750 and MOV-3-751, will automatically start to isolate when RCS pressure is 515 to 535 psig. The isolation signal is indicated by a yellow light present on VPB, and alarm on the Reactor Coolant Annunciator Panel. When the pressure drops below the isolation setpoint, there will be a blue light present on VPB. The isolation valves travel may be reversed by depressing the push buttons located below the yellow and blue lights VPB. The yellow light should go out and the blue light will remain on until the isolation valves are full open.

4.4 Shutdown Rod Banks

4.4.1 Both shutdown banks of control rods should be at the fully withdrawn position when going from Mode 3, Hot Standby, to Mode 5, Cold Shutdown. However, shutdown rods may be left inserted at the discretion of the Plant Supervisor-Nuclear as long as shutdown margin is maintained. Refer to the Plant Curve Book for applicable boron concentration.

4.4.2 At any time when moving shutdown or control rod banks, closely monitor group step counters, RPIs and all nuclear instrumentation channels.

4.4.3 Control rod drive mechanism cooling fan operation shall continue until RCS temperature is below 350°F, and should continue as long as control rod drive mechanisms are energized.

INIT

_____ 5.10 When RCS loop temperature is less than 350°F and pressurizer pressure is less than 450 psig, place RHR in operation for cooldown in accordance with 3-OP-050, Residual Heat Removal System.

_____ 5.10.1 Refer to Enclosure 1, Valve Exercising Reference, for valves to be exercised. (Mark N/A if exercising is not required per Step 5.1.1)

_____ 5.10.2 If proceeding to Mode 6, Refueling, commence performance of OP-3206.2, Residual Heat Removal System - Refueling Interval. (Mark N/A if not proceeding to Refueling)

CAUTION

Technical Specifications require the Overpressure Mitigating System (OMS) to be in operation when the RCS temperature is less than or equal to 275°F.

_____ 5.11 Prior to RCS cooldown to less than 276°F, when RCS temperature is in the range of 276°F to 285°F and pressurizer pressure is in the range of 325 to 375 psig, establish and verify OMS operation in accordance with 3-OP-041.4, Overpressure Mitigating System.

_____ 5.11.1 Refer to Enclosure 1, Valve Exercising Reference, for valves to be exercised. (Mark N/A if exercising is not required per Step 5.1.1)

NOTE

This step is not part of the OMS test, but is performed at this time to minimize the number of containment entries required.

_____ 5.11.2 In the containment (penetration 34) unlock and close the Containment Service Air Header Drain Regulator Bypass valve, 3-40-208.

_____ 5.11.3 When the OMS system has been placed into operation, inform the Construction Supervisor that construction activities may resume in the following areas:

- Turbine Deck
- Mezzanine Deck
- Turbine Building Ground Level
- EDG Room
- AFW Area
- Blowdown Area
- Feedwater Platform
- 4160V Switchgear Rooms
- 480V Load Center Rooms
- 480V MCC'S
- DC Switchgear and Inverter Room
- Main Steam Platform
- Pipe and Valve Room
- Containment Spray Pump Room
- Safety Injection Pump Room
- RHR Pump and Heat Exchange Room
- BAST Room
- Charging Pump Room
- Condensate Polishers
- Cable Spreading Room
- Auxiliary Building North and South Hallway

4.0 PRECAUTIONS/LIMITATIONS

4.1 The RHR Loop Isolation valves shall not be opened if RCS pressure is greater than 450 psig or RCS temperature is greater than 350°F.

4.2 When the RHR system is not being used for cooldown, it shall be aligned for low head safety injection.

4.3 To prevent overheating and pump damage, do not operate the RHR pumps "dead-headed".

4.4 Each RHR Loop required to be operable during modes 3, 4, 5 and 6 shall be supported by one CCW Heat Exchanger, one CCW Pump and one ICW pump powered from the same electrical source as the associated RHR pump. At least one of the required RHR Loops and its related support components shall be capable of being powered from an operable Emergency Diesel Generator. Only one ICW Header and CCW Basket Strainer are required to be operable to support both RHR Loops.

4.5 When rotating RHR pumps and RCS level is at mid-nozzle, stop the running RHR pump prior to starting the alternate RHR pump to prevent possible RHR pump cavitation due to a decrease in level.

INITIALS
CK'D VERIF

5.1.2 (Cont'd)

4. Verify the following valves indicate closed on the SIS-Recirculation Status - SI Lights panel:

   a. SI Pump Recirc Phase Suct Stop, MOV-3-863A

   b. SI Pump Recirc Phase Suct Stop, MOV-3-863B

5. Open the RHR To CVCS Stop Valve, 3-205B.

6. Slowly open the RHR Letdown to CVCS valve, HCV-3-142, to equalize RHR and letdown pressure.

NOTE

The Low Pressure Letdown Control valve, PCV-3-145 shall be maintained in manual until a RHR pump has been started.

7. Adjust the Low Pressure Letdown Control valve, PCV-3-145 to match the RCS pressure.

8. **Unlock and close the following breakers to energize the RHR Loop 3C Suction Stop valves.**

   a. **30615 (MOV-3-750)**

   b. **30731 (MOV-3-751)**

9. Open the following valves:

   a. RHR Loop 3C Suction Stop, MOV-3-750

   b. RHR Loop 3C Suction Stop, MOV-3-751

10. Unlock and open the HCV-3-758 Air Supply Isolation valve, 3-40-1019.

11. Open the following valves:

   a. RHR Hx B By-Pass Hdr Isolation, 3-757C

   b. RHR Hx A By-Pass Hdr Isolation, 3-757D

NRC EXAM QUESTION REVIEW

QUESTION: 6:11

State the four conditions/switch positions which must exist in order to place the Overpressure Mitigation System (OMS) in operation in the low pressure mode.

RESPONSE:

We request that you accept the following additional answers:

1. Nitrogen (backup air supply) charged to 2070 psig or higher
2. RCS temperature must be in the range of 276°F to 285°F and pressurizer pressure must be in the range of 325 to 375 psig.

REFERENCE:

AP-0103.32, Reactor Cold Shutdown Conditions, pg.3 step 5.1
3-GOP-305, Hot Standby to Cold Shutdown, pg.26 step 5.11

4.11.1 At least one Reactor Coolant Loop is operable; or

4.11.2 The RCS temperature is less than 140°F, boron concentration resulting in a shutdown greater than or equal to 10 percent ΔK/K margin and the refueling cavity is flooded to greater than 23 feet above the reactor vessel flange with the reactor vessel head removed.

4.12 The Pressurizer Safety Valves shall be installed and operable in accordance with 3/4-SMM-041.1, Pressurizer Safety Valve, Setpoint Testing prior to installing the reactor vessel head.

4.13 When using the charging pumps with a flowpath from the BASTs through the RCP seals, flow shall be greater that 15 gpm but less than 45 gpm to prevent RCP seal damage.

5.0 Related System Status:

5.1 When in the Cold Shutdown condition, the OMS shall be operable at the low setpoint range, or there shall be an opening of the RCS with an area of at least 2.2 square inches.

For an Overpressure Mitigating Channel to be operable and meet Technical Specification 3.15.3, the following conditions for the primary and backup channels must be met for each channel:

OMS Primary Aligned:

1. Status light ON, i.e., OMS Primary Aligned
2. MOV-*-535 - OPEN
3. PCV-*-456 - Switch in AUTO position
4. **Nitrogen (Backup Air Supply) Charged to 2070 psig or higher**
5. OMS Low Pressure Setpoint Position selected for *-456

OMS Backup Aligned:

1. Status light ON, i.e., OMS Backup Aligned
2. MOV-*-536 - OPEN
3. PCV-*-455C - Switch in AUTO position
4. **Nitrogen (Backup Air Supply) Charged to 2070 psig or higher**
5. OMS Low Pressure Setpoint Position selected for *-455C

5.2 After reaching Cold Shutdown, one of the two operating CCW pumps should be placed in Standby by performing Table III.

6.0 References/Commitment Documents:

6.1 References

6.1.1 Technical Specification 3.2a

6.1.2 3/4-GOP-103, Power Operation to Hot Standby

6.1.3 3/4-GOP-305, Hot Standby to Cold Shutdown

6.1.4 OP-0209.1, Valve Exercising Procedure

6.1.5 3/4-OP-041.7, Draining the Reactor Coolant System

INIT.

Cont 011

______ 5.10 When RCS loop temperature is less than 350°F and pressurizer pressure is less than 450 psig, place RHR in operation for cooldown in accordance with 3-OP-050, Residual Heat Removal System.

______ 5.10.1 Refer to Enclosure 1, Valve Exercising Reference, for valves to be exercised. (Mark N/A if exercising is not required per Step 5.1.1)

______ 5.10.2 If proceeding to Mode 6, Refueling, perform OP-3206.2, Residual Heat Removal System - Refueling Interval. (Mark N/A if not proceeding to Refueling)

CAUTION

Technical Specifications require the Overpressure Mitigating System (OMS) to be in operation when the RCS temperature is less than or equal to 275°F

______ 5.11 **[Prior to RCS cooldown to less than 276°F, when RCS temperature is in the range of 276°F to 285°F and pressurizer pressure is in the range of 325 to 375 psig, establish and verify OMS operation in accordance with 3-OP-041.4, Overpressure Mitigating System.]**

______ 5.11.1 Refer to Enclosure 1, Valve Exercising Reference, for valves to be exercised. (Mark N/A if exercising is not required per Step 5.1.1)

NOTE

This step is not part of the OMS test, but is performed at this time to minimize the number of containment entries required.

______ 5.11.2 In the containment (penetration 34) unlock and close the Containment Service Air Header Drain Regulator Bypass valve, 3-40-208.

______ 5.11.3 When the OMS system has been placed into operation, inform the Construction Supervisor that construction activities may resume in the following areas:

Turbine Deck
Mezzanine Deck
Turbine Building Ground Level
EDG Room
AFW Area
Blowdown Area
Feedwater Platform
4160V Switchgear Rooms
480V Load Center Rooms
480V MCC'S
DC Switchgear and Inverter Room
Main Steam Platform
Pipe and Valve Room
Containment Spray Pump Room
Safety Injection Pump Room
RHR Pump and Heat Exchange Room
BAST Room
Charging Pump Room
Condensate Polishers
Cable Spreading Room
Auxiliary Building North and South Hallway

NRC EXAM QUESTION REVIEW

QUESTION: 6.12

State five automatic functions which are performed upon receipt of a Emergency Diesel Generator Lockout Relay Signal.

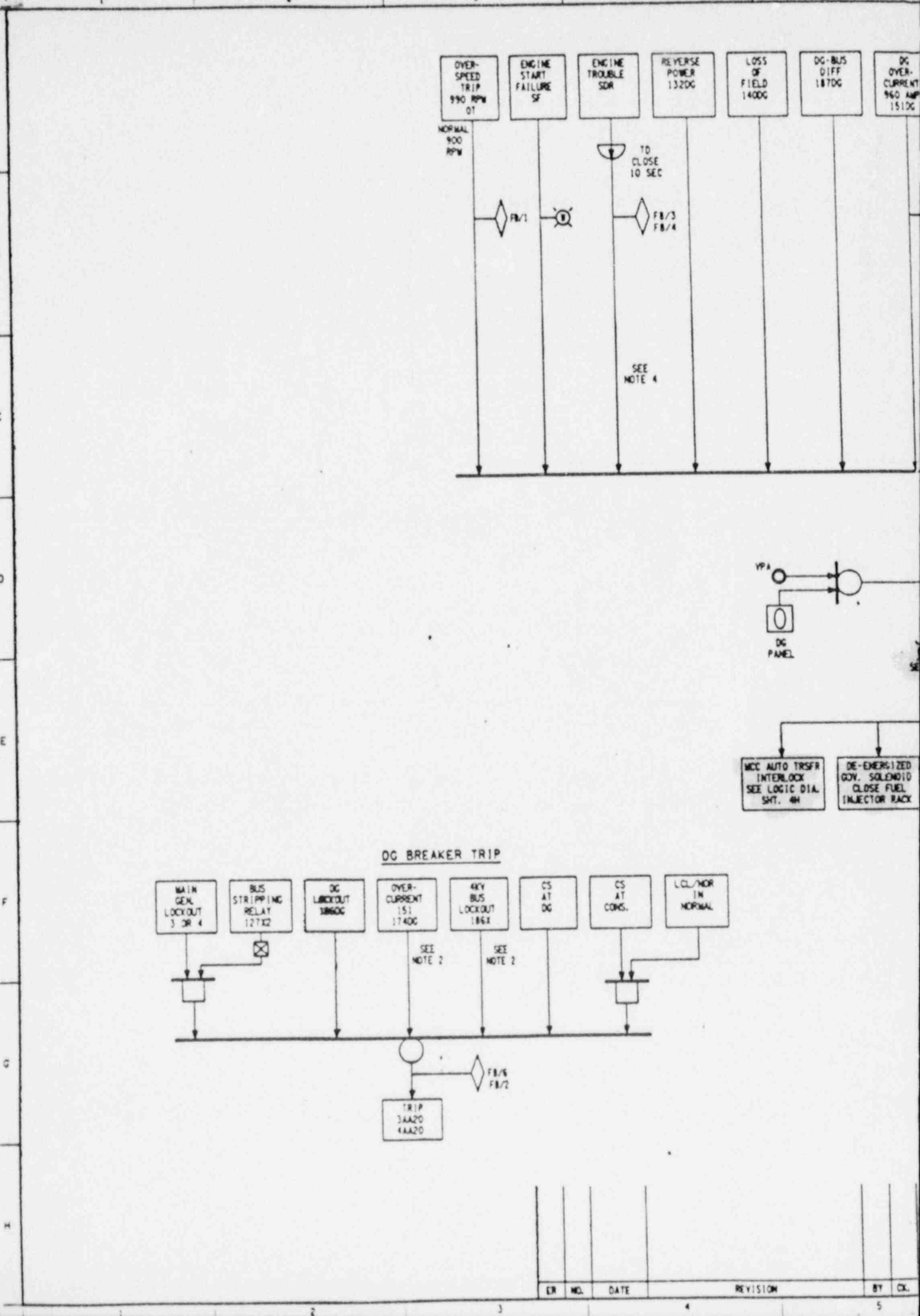
RESPONSE:

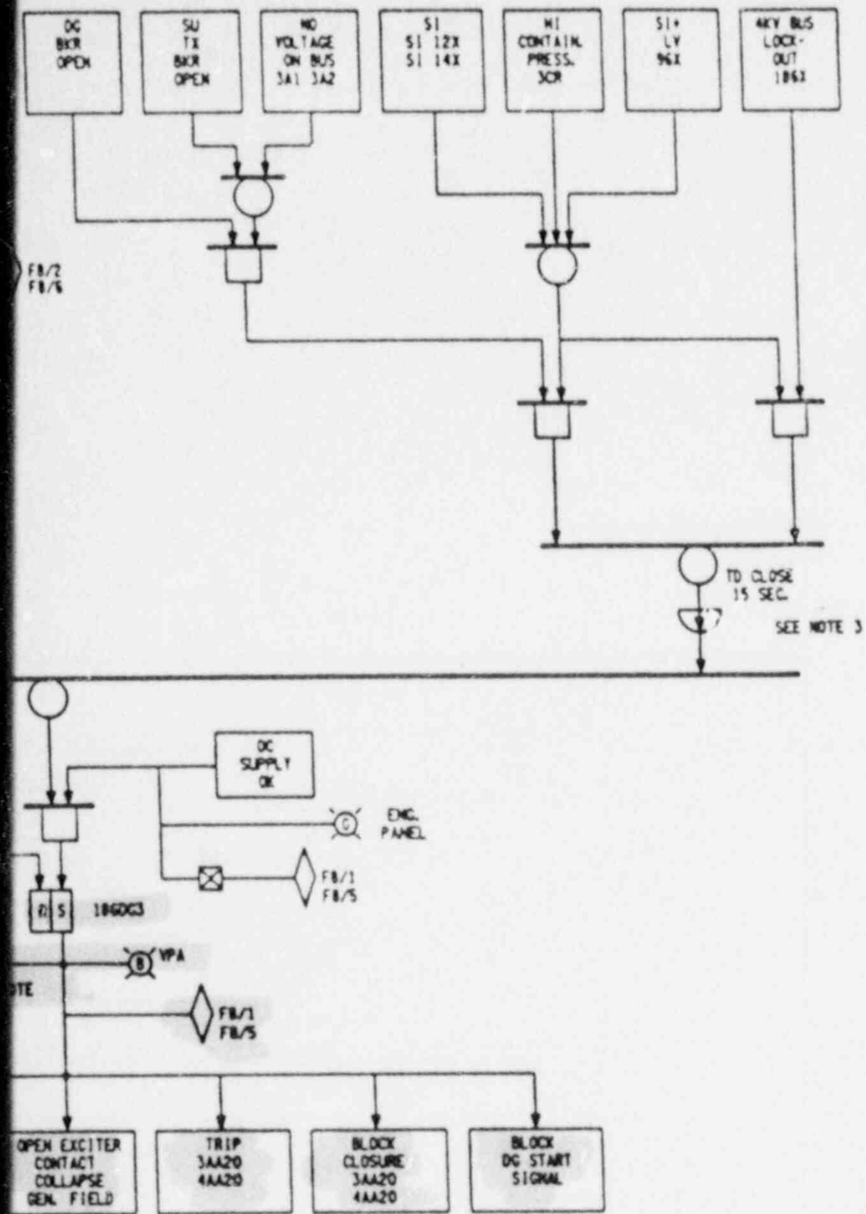
We request that you accept additional answers because this is an open-ended question. Some additional acceptable answers should be:

1. Flashing blue lockout relay light on vertical panel A.
2. Annunciator Targets come in
   a) F 8/1
   b) F 8/5

REFERENCE:

5610-T-L1 sht. 9C, Diesel Generator A Lockout Relay and Breaker Logic
5610-T-L1 sht. 9C1, Diesel Generator B Lockout Relay and Breaker Logic


OVER-SPEED TRIP 990 RPM OT
NORMAL 900 RPM
ENGINE START FAILURE SF
ENGINE TROUBLE SDR
TD CLOSE 10 SEC
REVERSE POWER 132DG
LOSS OF FIELD 140DG
DG-BUS DIFF 187DG
DG OVER-CURRENT 960 AMP 151DG
FB/1
FB/3
FB/4
SEE NOTE 4
YPA
DG PANEL
MCC AUTO TRSFR INTERLOCK SEE LOGIC DIA. SHT. 4H
DE-ENERGIZED GOV. SOLENOID CLOSE FUEL INJECTOR RACK
DG BREAKER TRIP
MAIN GEN. LOCKOUT 3 OR 4
BUS STRIPPING RELAY 127X2
DG LOCKOUT 186DG
OVER-CURRENT 151 174DG
4KV BUS LOCKOUT 186X
CS AT DG
CS AT CONS.
LCL/NOR IN NORMAL
SEE NOTE 2
SEE NOTE 2
FB/6
FB/2
TRIP 3AA20 4AA20
ER
NO.
DATE
REVISION
BY
CK.



NOTES:

1. FOR ANY DG LOCKOUT INSPECT ALL RELAYS ON THE GEN. PANEL AND 4KV DG BREAKERS FOR DROPPED TARGET FLAGS. DOCUMENT AND ADVISE ELECTRICAL DEPARTMENT. RESTART ONLY AFTER CAUSE HAS BEEN FOUND AND CORRECTED.

2. TRIPS THE DG BREAKER ONLY ON THE AFFECTED BUS.

3. THIS LOGIC APPLIES TO 4KV BUSSES 3A AND 4A. LOCKOUT OF A DG FOR PROBLEMS ON 3A OR 4A IS TO INITIATE THE MCC AUTO TRANSFER SCHEME. SEE SHEET 4H.

4. THE SDR RELAY GIVES L.O. BY 2 PATHS:
   A. ENGINE THRU DE-ENERGIZED SDR ACTUATES L.O. RELAY DIRECTLY. THRU A 10 SEC T.D.
   B. SDR ALSO OPENS EXCITER CONTACTS WHICH RESULTS IN LOSS OF FIELD-GEN L.O.

5. LOCKOUT RELAY LAMPS

   NORMALLY GLOWING. NO LIGHT MEANS LOSS OF DC OR BLOWN FUSES.

   LAMP FLASHES WHEN L.O. ACTUATES

8804010181-01

PMOD

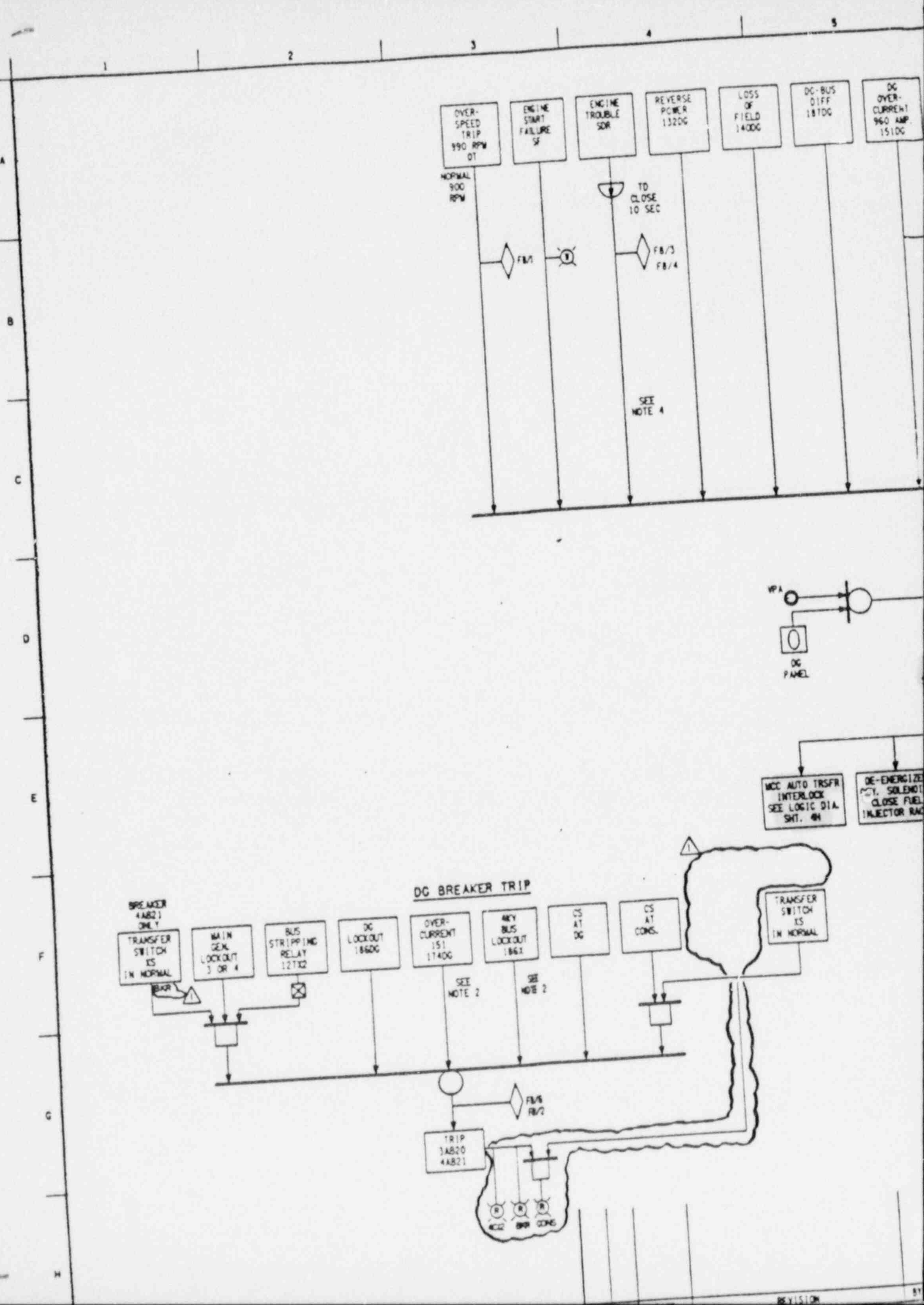
BAYMONT ENGINEERING COMPANY- CAD

FLORIDA POWER & LIGHT CO.

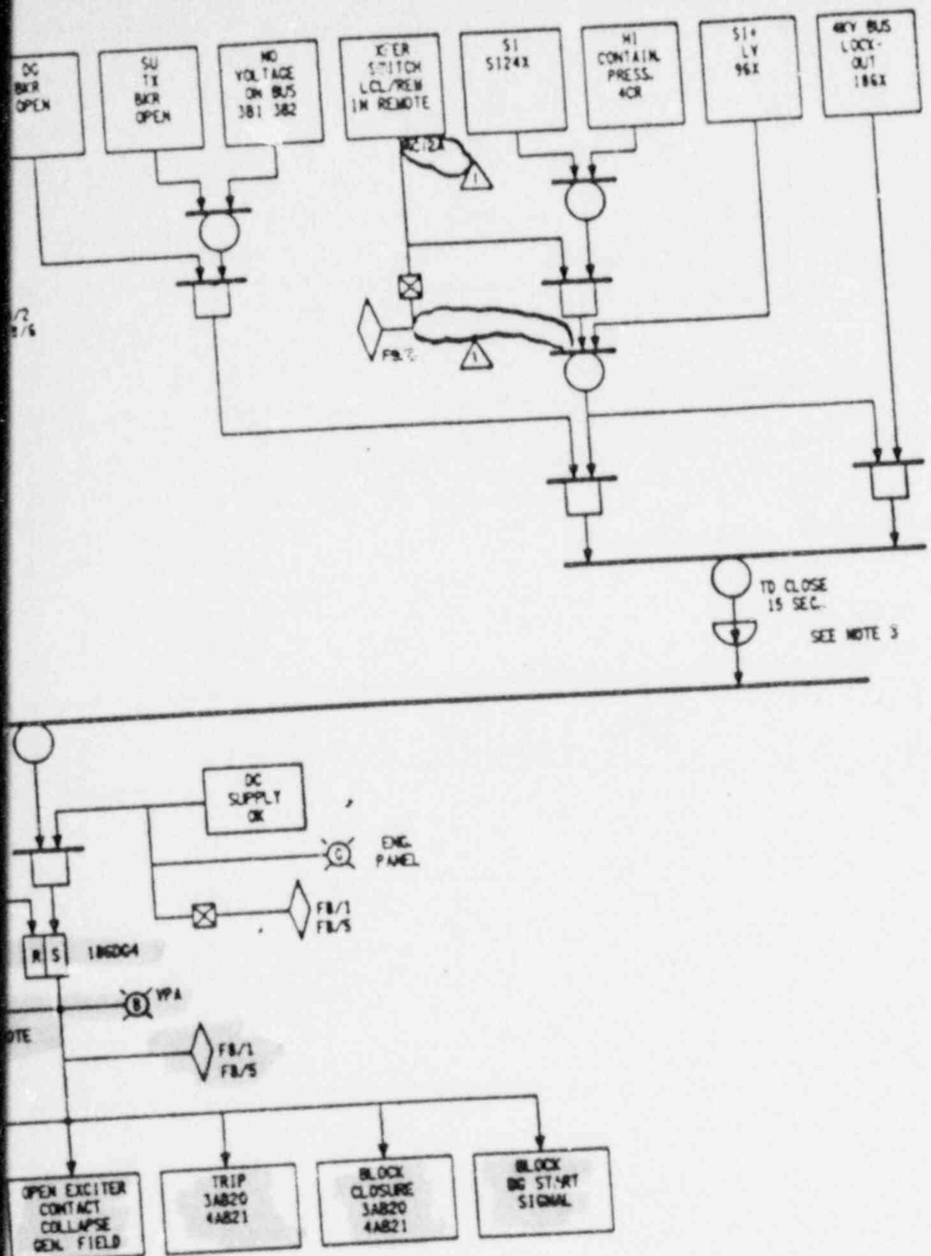
TURKEY POINT PLANT
LOGIC DIAGRAM UNITS 3&4
DIESEL GENERATOR 'A'
LOCKOUT RELAY & BREAKER LOGIC

DRAWN BY: C.L.W	CHECKED BY: JS	APPROVED BY: WCW	DATE: [illegible]	5610-T-L1	SHEET 9C	REV. 1

NO.	DATE	REVISION	BY	CK.	COR.	APP.
1	5-22-74	RELOCATED DG 'B' LOGIC TO 5610-T-L1, SH 9C1	LG	[illegible]	[illegible]	[illegible]



OVER-SPEED TRIP 990 RPM OT
NORMAL 900 RPM
ENGINE START FAILURE SF
ENGINE TROUBLE SDR
TD CLOSE 10 SEC
REVERSE POWER 132DG
LOSS OF FIELD 140DG
DC-BUS DIFF 187DG
DG OVER-CURRENT 960 AMP. 151DG
FB/1
FB/3
FB/4
SEE NOTE 4
VPA
DG PANEL
MCC AUTO TRSFR INTERLOCK SEE LOGIC DIA. SHT. 4H
DG BREAKER TRIP
BREAKER 4AB21 ONLY
TRANSFER SWITCH XS IN NORMAL
BKR
MAIN GEN. LOCKOUT 3 OR 4
BUS STRIPPING RELAY 127X2
DG LOCKOUT 186DG
OVER-CURRENT 151 174DG
SEE NOTE 2
4KV BUS LOCKOUT 186X
SEE NOTE 2
CS AT DG
CS AT CONS.
TRANSFER SWITCH XS IN NORMAL
FB/6
FB/7
TRIP 3AB20 4AB21
4CG2
BKR
CONS
REVISION



NOTES:

1. FOR ANY DG LOCKOUT INSPECT ALL RELAYS ON THE GEN. PANEL AND 4KV DG BREAKERS FOR DROPPED TARGET FLAGS. DOCUMENT AND ADVISE ELECTRICAL DEPARTMENT. RESTART ONLY AFTER CAUSE HAS BEEN FOUND AND CORRECTED.

2. TRIPS THE DG BREAKER ONLY ON THE AFFECTED BUS.

3. THIS LOGIC APPLIES TO 4KV BUS 3B (NOT 4B). LOCKOUT OF B DG FOR PROBLEMS ON 3B IS TO INITIATE THE MCC AUTO TRANSFER SCHEME. PROBLEMS ON 4B DO NOT REQUIRE B DG LOCKOUT SINCE NORMALLY THIS BUS DOES NOT FEED MCC 4A OR D.
SEE SHEET 4H.

4. THE SDR RELAY GIVES L.O. BY 2 PATHS:
   A. ENGINE THRU DE-ENERGIZED SDR ACTUATES L.O. RELAY DIRECTLY, THRU A 10 SEC T.D.
   B. SDR ALSO OPENS EXCITER CONTACTS WHICH RESULTS IN LOSS OF FIELD-GEN L.O.

5. LOCKOUT RELAY LAMPS

   NORMALLY GLOWING. NO LIGHT MEANS LOSS OF DC OR BLOWN FUSES.

   LAMP FLASHES WHEN L.O. ACTUATES

TI APERTURE CARD

Also Available On Aperture Card

8804010181-02

POD

BAYMONT ENGINEERING COMPANY- CAD

FLORIDA POWER & LIGHT CO.

TURKEY POINT PLANT
LOGIC DIAGRAM UNITS 3 & 4
DIESEL GENERATOR 'B'
LOCKOUT RELAY & BREAKER LOGIC

DRAWN BY: ____
CHECKED BY: ____
APPROVED BY: ____
DATE: ____

5610-T-L1
SHEET 9C1

COR.	APP.	ER	NO.	DATE	REVISION	BY	CK.	COR.	APP.
			1	4-3-87	ISSUED AS-BUILT FOR PC/M 83-154	RV	[illegible]	[illegible]	[illegible]
			0	5-22-86	RELOCATED DG'B' LOGIC FROM 5610-T-L1 SH. 9C, CHANGE REQUEST #1 ISSUED AS-BUILT FOR PC/M 83-155	LG	[illegible]	[illegible]	PJS

NRC EXAM QUESTION REVIEW

QUESTION: 6.18

State the two SAFETY design bases for a maximum water inventory in the S/Gs.

RESPONSE:

We request that you accept additional variations to the answer "minimizes the effects of a RCS cooldown in the event of a major Steam Line Break". Please accept a Steam Line Break at EOL.

REFERENCE:

FSAR, Section 14, pgs. 14.2.5-1 thru 14.2.5-8

14.2.5 RUPTURE OF A STEAM PIPE*

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a break in pipe line or due to a valve malfunction. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. **With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin.** If the most reactive control rod is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining control rods inserted. A return to power following a steam pipe rupture is a potential problem only because of the high hot channel factors which may exist when the most reactive rod is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core is ultimately shut down by the boric acid in the refueling water storage tank.

The following systems provide the necessary protection against a steam pipe rupture:

1) Safety Injection System Actuation from any of the following*:

   a. Two out of three pressurizer low pressure signals.

   b. Two out of three differential pressure signals between any steam line and the main steam header.

   c. High steam flow in two out of three lines (one out of two per line) in coincidence with either low reactor coolant system average temperature (two out of three) or low steam line pressure (two out of three).

* The details of the logic used to actuate Safety Injection are discussed in Section 7.

d. Two out of three high containment pressure signals.

2) Reactor trip occurring upon actuation of the Safety Injection System.

3) Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown, thus in addition to the normal control action which will close the main feedwater valves, any safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4) Trip of the fast acting steam line stop valves (designed to close in less than 5 seconds with no flow) on high steam flow in two out of three lines (one out of two per line) in coincidence with either low reactor coolant system average temperature (two out of three) or low steam line pressure (two out of three).

Three cases are presented: Inadvertent opening of a steam generator relief or safety valve and complete (double ended) severance of a steam pipe with and without offsite power available.

Inadvertent Opening of a Steam Generator Relief or Safety Valve

Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system result from an inadvertent opening of a single steam dump, relief, or safety valve.

The steam release as a consequence of this accident results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following criterion is satisfied: assuming a stuck rod cluster control assembly, with offsite power available, and assuming a single failure in the engineered safety features, there will be no consequential damage to the core or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve.

Analysis of Effects and Consequences

A. Method of Analysis

The following analyses of a secondary system steam release are performed for this section:

1. A full plant digital computer simulation using the LOFTRAN code (Reference 1) to determine reactor coolant system temperature and pressure, during cooldown, and the effect of safety injection.

2. Analyses to determine that there is no damage to the core or reactor coolant system.

The following conditions are assumed to exist at the time of a secondary steam system release:

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive rod cluster control assembly stuck in its fully withdrawn position. Operation of rod cluster control assembly banks during core burnup is restricted in such a way that addition of positive reactivity in a secondary system steam release accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive rod cluster control assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included in the LOFTRAN calculations. The K_{eff} versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 14.2.5-1.

3. Minimum capability for injection of concentrated boric acid solution corresponding to the most restrictive single failure in the safety injection system (Failure of one safeguards train). This corresponds to the flow delivered by two safety injection pumps delivering their full contents to the cold leg header. Low concentration boric acid must be swept from the safety injection lines downstream of the refueling water storage tank prior to the delivery of concentrated boric acid (2000 ppm) to the reactor coolant loops. This effect has been allowed for in the analysis. The boron injection tank boron concentration is assumed to be 0 ppm.

4. The case studied is a steam flow of 280 lb/sec at 1100 psia from one steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief, or safety valve. Initial hot standby conditions with minimum required shutdown margin at the no-load T_{avg} is assumed since this represents the most conservative initial condition.

5. Should the reactor be just critical or operating at power at the time of steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the reactor coolant system contains more stored energy than at no load, the average coolant temperature is higher than at no load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam release before the no load conditions of reactor coolant system temperature and shutdown margin assumed in the analysis are reached. After the additional stored energy has been removed, the cooldown and reactivity

insertions proceed in the same manner as in the analysis which assumes no load condition at time zero. However, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the reactor coolant system cooldown are less for steam line release occurring at power.

6. In computing the steam flow, the Moody Curve (Reference 3) for FL/D = 0 is used.

7. Perfect moisture separation in the steam generator is assumed.

B. Results

Figure 14.2.5-2 and 14.2.5-3 show the transient results for a steam flow of 280 lb/sec at 1100 psia from one steam generator.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve.

Safety injection is initiated automatically by low pressurizer pressure. Minimum safety injection capability is that corresponding to two out of four safety injection pumps in operation. Safety injection flow used in the analysis is shown on Figure 14.2.5-11.

Boron solution at 2000 ppm enters the reactor coolant system providing sufficient negative reactivity to prevent core damage. The calculated transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the core transient occurs over a period of about 5 minutes, the neglected stored energy will have a significant effect in slowing the cooldown.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of auxiliary feedwater flow and

afety injection flow as described by plant operating procedures. The operating rocedures would call for operator action to limit reactor coolant system re[illegible]e and pressurizer level by terminating safety injection flow and to ontrol steam generator level and reactor coolant system coolant temperature ising the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following safety injection actuation.

A time sequence of events is given in Table 14.2.5-1.

Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the main steam system, the minimum DNBR remains well above the limiting value and no system design limits are exceeded.

Steam System Piping Failure

Identification of Causes and Accident Description

The steam release arising from a rupture of a main steamline would result in an initial increase in steam flow which decreases during the accident as the steam pressure decreases. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the safety injection system.

Results

The results presented are a conservative indication of the events which would occur assuming a steam line rupture.* The worst case assumes that all of the following occur simultaneously.

1) Minimum shutdown reactivity margin of 1.77%.

2) The most negative moderator temperature coefficient for the rodded core at end of life.

3) The rod having the most reactivity stuck in its fully withdrawn position.

4) The most restrictive failure of engineered safety features, i.e. only two safety injection pumps available and one motor operated valve available to deliver fluid through three cold leg lines.

Core Power and Reactor Coolant System Transient

Figure 14.2.5-3 shows the Reactor Coolant System transient and core heat flux following a steam pipe rupture (complete severence of a pipe) outside the containment, downstream of the flow measuring nozzle at initial no load conditions. The break assumed is the largest break which can occur anywhere outside the containment either upstream or downstream of the isolation valves. Outside power is assumed available such that full reactor coolant flow exists. The transient shown assumes the rods inserted at time 0 (with one rod stuck in its fully withdrawn position) and steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between any steam generator and the main steam header or by high steam flow signals in coincidence with either low reactor coolant system temperature or low steam line pressure will trip the reactor. Steam release from at least two steam generators will be prevented by either the check valves or by automatic trip of the fast acting stop valves in the steam lines by the high steam flow signals in coincidence with either low reactor coolant system temperature or low steam line pressure. Even with the failure of one valve,

* Results have been re-evaluated to assure that criteria are met for current cycle kinetics parameters, as shown in Appendix 14D.

release is limited to no more than 5 seconds for two steam generators while the third generator blows down. (The steam line stop valves are designed to be fully closed in less than 5 seconds with no flow through them. With the high flow existing during a steam line rupture, the valves will close considerably faster).

As shown in Figure 14.2.5-3, the core becomes critical with the rods inserted (with the design shutdown assuming one stuck rod) at 25 seconds. Boron solution at 20,000 ppm enters the Reactor Coolant System from the Safety Injection System at 66 seconds with a delay of 20.5 seconds required to clear the Safety Injection System lines of low concentration boric acid, between the boron injection tank and the coolant loops, after the pressure has fallen to 1350 psia.

The computer calculation used assumes the boric acid is mixed with and diluted by the water flowing in the reactor coolant system prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the Reactor Coolant System and in the Safety Injection System. The variation of mass flow rate in the Reactor Coolant System due to water density changes is included in the calculation as is the variation of flow rate in the Safety Injection System due to changes in the Reactor Coolant System pressure. The Safety Injection System flow calculation includes the line losses in the system as well as the pump head curve.

No credit has been taken for the 2,000 ppm boron which enters the Reactor Coolant System prior to the 20,000 ppm boric acid. The peak core average heat flux for this case is 15.4% of the value at 2200 MWt.

NRC EXAM QUESTION REVIEW

QUESTION: 7.04

For each of the following, provide the Turkey Point guideline for whole body exposure limits in millirem.

c) Yearly limit:____________

RESPONSE:

We request that the answer(s) to this question should read "4,500 mRem/year" or "should not exceed 5,000 mRem/year".

REFERENCE:

0-ADM-600, Health Physics Manual, pg. 30, Bottom note and step 5.18.1.1.b.2)

5.18 Personnel Exposure Control

5.18.1 Radiation Protection Standards (Cont'd)

1. b. Florida Power & Light Company Guideline Values

1) Each individual entering the Radiation Controlled Area shall be limited to 250 mRem per quarter unless one of the following actions have been completed:

a) Individual has signed a statement to the effect that he has no occupational exposure for current quarter. He shall then be allowed to receive up to 800 mRem/quarter in accordance with Section 5.18.1.1.c.

NOTE

Only in an unusual situation (e.g., the only person available for a specialty craft function that will require greater than 250 mRem exposure) should a written statement of exposure, estimated by the individual, be accepted. In such cases, the unusual situation shall be noted and signed by the Health Physics Department Supervisor.

b) Individual has documentation from former employers that gives an estimate of his total occupational exposure for the current quarter. He shall then be allowed to receive a dose that will bring his total up to 800 mRem/quarter in accordance with the Section 5.18.1.1.c.

c) Individual has signed a statement to the effect that he has never received any occupational radiation exposure (completed Form NRC-4). He shall then be allowed to receive up to 2150 mRem/quarter in accordance with Section 5.18.1.1.c.

NOTE

Individual dose should not exceed 5,000 mrem/year from all sources of occupational exposure without prior authorization of the Health Physics Department Supervisor and Plant Manager-Nuclear.

	mrems wk.	mrems qtr.	mrems year
2) Whole body; head and trunk; active blood-forming organs; lens of eyes or gonads	300	See 5.18.1.1C Items 2 & 3	4,500

NRC EXAM QUESTION REVIEW

QUESTION:7.05

While approaching criticality using the 1/M plot, a projected critical rod position must be calculated after the third doubling (i.e., 1/M is approximately .125). State the three times that the result of this critical rod position projection requires that control banks be reinserted.

RESPONSE:

Please expand the answer key to accept such answer as:

"The reactor shall not be made critical with a difference of greater than or equal to 1000 pcm between the projected critical height and the ECC rod position".

REFERENCE:

3-GOP-301, Hot Standby to Power Operation, pg. 22 step 4.28.5

4.23 The operability of the Main Feedwater Control Valves, FCV-3-478, 488, 498 should be checked by stroking through one complete cycle prior to placing the Turbine/Gen on line.

4.24 If at anytime, a Limiting Condition for Operation cannot be met, 10 CFR 50.36 requires that the reactor be shutdown or any remedial action permitted by the Technical Specification be followed; AP-0103.8, Reactor Shutdown Rate Time Limits, provides guidance for the resultant reactor shutdown.

4.25 Serious damage to the Main Generator windings can result from operation of the generator outside of the terminal voltage limits of 20,900 to 23,100 volts.

4.26 Safety Injection Signals shall not be in a blocked status for any reason other than for depressurization and cooldown of the Reactor Coolant System.

4.27 During a Post Trip Recovery at EOL, when startup is within 4 hours of criticality, contact the Reactor Engineering Department for startup guidelines.

4.28 The following guidelines shall be employed while approaching criticality using the 1/m plot.

4.28.1 If after the third (3rd) doubling (i.e., 1/m approximately .125), the projected critical rod position is below the insertion limit (107 steps on Bank C), reinsert the control banks and borate the RCS as necessary.

4.28.2 If after the (3rd) doubling (i.e., 1/m approximately .125), it is projected that the reactor cannot be made critical at the current boron concentration, reinsert the control banks and dilute the RCS as necessary.

4.28.3 If after the (3rd) doubling (i.e., 1/m approximately .125), the projected critical rod position deviates from the ECC rod position by more than 300 pcm but less than or equal to 400 pcm; permission to pull the Reactor critical shall be obtained from the Plant Supervisor - Nuclear or designee.

4.28.4 If after the third (3rd) doubling (i.e., 1/m approximately .125), the projected critical rod position deviates from the ECC rod position by greater than 400 pcm but less than or equal to 500 pcm, permission to pull to criticality shall be obtained from the Reactor Supervisor or his designee.

4.28.5 **If after the third (3rd) doubling (i.e., 1/m approximately .125), the projected critical rod position deviates from the ECC rod position by greater than 500 pcm,** the control banks shall be reinserted and the ECC re-evaluated. If the error cannot be determined, permission of the Operations Superintendent and Reactor Supervisor (or their designee) shall be obtained prior to making the reactor critical under the guidance of the 1/m plot. **The Reactor shall not be made critical with a difference of greater than or equal to 1000 pcm between the projected critical height and the ECC rod position.**

NRC EXAM QUESTION REVIEW

QUESTION:7.06

If the Unit 3 auxiliary feedwater system fails completely but offsite power is still available, the Steam Generator (S/G) feed pumps can be used to supply feedwater to the S/Gs utilizing the feedwater bypass regulating valves. In order of priority, state the four other methods to supply feedwater to the S/Gs under these conditions.

RESPONSE:

This question does not specifically address ONOP-7308.1, Malfunction of the AFW system. For this reason we request that "In order of priority..." be removed from the question. If the question was answered with 3-EOP-FR-H.1,Response to Loss of Secondary Heat Sink in mind, the order is not the same and in fact, at one point, the order is not as important as getting any source of flow to the steam generators established.

REFERENCE:

3-EOP-FR-H.1, Response to loss of secondary Heat sink, pgs. 4-10

STEP	ACTION EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

- *If total feed flow is less than 130 GPM per S/G due to Procedural Guidance, this procedure should not be performed.*
- *If wide range level in any 2 S/Gs is less than 8% [32%] OR PRZ pressure is greater than or equal to 2335 PSIG due to loss of secondary heat sink, RCPs should be tripped and Steps 9 through 15 should be immediately initiated for bleed and feed.*
- *Feed flow should not be reestablished to any faulted S/G if a non-faulted S/G is available.*

STEP	ACTION EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Check If Secondary Heat Sink Is Required:	
	a. RCS pressure - GREATER THAN ANY NON-FAULTED S/G PRESSURE	a. Go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. RCS temperature - GREATER THAN 324°F [324°F]	b. Try to place RHR System in service while continuing in this procedure. Refer to OP-050, RESIDUAL HEAT REMOVAL SYSTEM. IF adequate cooling with RHR System established, THEN return to procedure and step in effect.
2	**Try To Establish AFW Flow To At Least One S/G:**	
	a. Check control room indications for cause of AFW failure: • CST LEVEL • AFW STEAM SUPPLY MOV POWER SUPPLY • AFW VALVE ALIGNMENT	
	b. Check total flow to S/G - GREATER THAN 130 GPM per S/G	b. Perform the following: 1) Reset SI IF necessary 2) Reset FW isolation. 3) OPEN the Unit 3 standby S/G FW pump manual isolation valve DWDS-012 4) **START standby steam generator feedwater pump. A** 5) OPEN feedwater bypass valves to desired flow. **IF** **Feedwater cannot be established to at least one S/G THEN consult TSC staff for possible** use of Unit 2 or 4 feedwater while continuing with step 3
	c. Return to procedure and step in effect	
3	Stop All RCPs	

• CAY dtd

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Try To Establish Main FW Flow To At Least One S/G:**	
	a. Check condensate system - IN SERVICE	a. Try to place condensate system in service. IF NOT, THEN go to Step 8.
	b. Open feedwater bypass valves to desired flow	b. Pe. form the following: 1) Reset SI if necessary. 2) Reset FW isolation. 3) Open feedwater bypass valves. 4) **Start S/G feed pumps** IF no feedwater bypass valve can be opened, THEN go to Step 8.
	c. **Establish main FW flow**	IF feedwater flow cannot be established THEN go to step 6.
5	Check S/G Levels:	
	a. Narrow range level in at least one S/G - GREATER THAN 6% [32%]	a. IF feed flow to at least one S/G verified, THEN maintain flow to restore narrow range level to greater than 6% [32%]. IF NOT verified, THEN go to Step 6.
	b. Return to procedure and step in effect	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Following block of automatic SI actuation, manual SI actuation may be required if conditions degrade.

6 **Try To Establish Feed Flow From Condensate System:**

a. Depressurize RCS to less than 1950 PSIG:

1) Check letdown - IN SERVICE

Response not obtained: 1) Use one PRZ PORV. IF NOT, THEN use auxiliary spray. Go to Step 6b.

2) Use auxiliary spray

Response not obtained: 2) Use one PRZ PORV.

b. Block SI signals:

- LOW PRESSURE SI
- HIGH S/G FLOW COINCIDENT WITH LOW S/G PRESSURE OR LOW TAVG SI

c. Depressurize at least one S/G to less than 430 PSIG:

1) Dump steam to condenser at maximum rate

Response not obtained: 1) Manually or locally dump steam from S/Gs using steam dump to atmosphere valves

d. Establish condensate flow:

Response not obtained: d. Go to Step 8.

1) Start all available condensate pumps

2) Verify flow

7 **Check S/G Levels:**

a. Narrow range level in at least one S/G - GREATER THAN 6% [32%]

Response not obtained: a. IF feed flow to at least one S/G verified, THEN maintain flow to restore narrow range level to greater than 6% [32%]. IF NOT verified, THEN go to Step 8

b. Return to procedure and step in effect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	Check For Loss Of Secondary Heat Sink:	
	a. Parameter - Wide range S/G level in any 2 S/Gs less than 8% [32%] **OR** PZR pressure greater than or equal to 2335 - EXCEEDED	a. Return to Step 1

CAUTION

Steps 9 through 15 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	Actuate SI	
10	Verify RCS Feed Path:	Manually start pumps and align valves as necessary to establish feed path. **IF** a feed path can **NOT** be established, **THEN** continue attempts to establish feed flow. Return to Step 2.
	a. Check high-head SI pumps - AT LEAST ONE RUNNING	
	b. Check valve alignment for operating pumps - PROPER EMERGENCY ALIGNMENT (Amber Lights on VPB)	
11	Reset SI:	
	a. Verify Unit 4 SI pumps - NOT REQUIRED	a. Perform the following: 1) OPEN the following SI pump manual suction intertie valves: a) 870A b) 870B 2) OPEN the following SI pump manual recirculation tie valves: a) 892A b) 892B 3) Unlock and rack in the following Unit 4 RWST outlet isolation valve breakers: a) 40712 b) 40605

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	Reset SI: (Cont'd)	4) CLOSE the following Unit 4 RWST outlet isolation valves: a) MOV-4-864A b) MOV-4-864B 5) Remove the manual gag from valves and CLOSE the following Unit 4 SI test return to RWST valves: a) CV-4-856A b) CV-4-856B 6) Go to Step 12.
	b. Stop the Unit 4 SI pumps and place in standby	
	c. CLOSE the following SI pump discharge header isolation valves: 1) MOV-878A 2) MOV-878B	
12	Reset Containment Isolation Phase A And Phase B	
13	Verify Instrument Air To Containment	Start all diesel powered air compressors
14	Establish RCS Bleed Path:	
	a. Verify power to PRZ PORV block valves - AVAILABLE	a. Restore power to block valves.
	b. Verify PRZ PORV block valves - ALL OPEN	b. Open block valves.
	c. Open all PRZ PORVs	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15	Verify Adequate RCS Bleed Path:	
	a. PRZ PORVs - BOTH OPEN	a. Perform the following: 1) Open all RCS head vents SV-3-6318A SV-3-6318B SV-3-6319A SV-3-6319B SV-3-6320A SV-3-6320B 2) Depressurize at least one intact S/G to atmospheric pressure using S/G steam dump to atmosphere valves. 3) Align any available low pressure water source to the depressurized S/G(s).

CAUTION

The RCS bleed path must be maintained even if RCS pressure remains greater than high-head SI pump shutoff head.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16	Maintain RCS Heat Removal: • MAINTAIN SI FLOW • MAINTAIN PRZ PORVs-BOTH OPEN	
17	Check Power Supply To Charging Pumps - OFFSITE POWER AVAILABLE	Verify adequate diesel capacity to run charging pumps. IF necessary, shed sufficient nonessential loads. (Refer to E-0, Attachment D for component KW load rating)
18	Check If Charging Flow Has Been Established:	
	a. Charging pumps - AT LEAST ONE RUNNING	a. Perform the following: 1) IF CCW flow to RCP(s) thermal barrier is lost, THEN isolate seal injection to affected RCP(s) before starting charging pumps. 2) Start charging pumps.
	b. Establish maximum charging flow:	

STEP	ACTION EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If RWST level decreases to less than [155,000] GAL, the SI system should be aligned for cold leg recirculation using ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

STEP	ACTION EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19	**Continue Attempts To Establish Secondary Heat Sink In At Least One S/G:** • AFW FLOW • STANDBY S/G FEEDWATER PUMPS • MAIN FW FLOW • UNITS 2, OR 4 FEEDWATER FLOW • CONDENSATE FLOW	
20	Check For Adequate Secondary Heat Sink: a. Narrow range level in at least one S/G - GREATER THAN 6% [32%]	a. Return to Step 19
21	Check RCS Temperatures: • CORE EXIT TCs - DECREASING • RCS HOT LEG TEMPERATURES - DECREASING	Return to Step 19

QUESTION 7.12

Concerning the requirement of at least one boric acid flow path to core for boron injection by CVCS, answer the following questions.

c) State the plant conditions which must exist to utilize the acceptable ALTERNATE flow path.

RESPONSE:

We request that other answers be considered correct also. The procedure discusses a secondary alternate flow path which is actually discussed on the page prior to the acceptable alternate flow path. In reality, these are both acceptable flow paths. Another flow path is by using the high head safety injection pumps which take their suction from the RWST. This path is addressed in the Technical Specifications.

REFERENCE:

AP-0103.32, Reactor Cold Shutdown Conditions, pgs. 7 and 8
Technical Specifications, Bases section, pgs. B3.6-1 and B3.4-2

CAUTION: When raising RCS level, verify proper operation of the VCT level makeup system. Ensure that the makeup water contains the proper boron concentration to prevent the possibility of inadvertent RCS dilution.

8.6.2 To raise RCS level, decrease letdown flow by closing HCV-*-142 (VPB) OR by increasing charging flow by increasing speed of running charging pump(s) (console) or opening HCV-*-121 (console).

8.7 MID NOZZLE LEVEL OPERATIONS - When the RCS is drained to mid nozzle, carefully monitor RHR pump amperage and RHR flow. Erratic indications on either requires immediate operator action to raise RCS level. If erratic amperage or flow are indicated, perform the following:

8.7.1 Adjust FCV-*-605 and HCV-*-758 (as necessary) to reduce RHR flow as seen on FI-*-605 (VPB).

THEN

CAUTION: Monitor RCS temperature and RHR pump outlet temperature closely to maintain RCS temperature below 190°F.

8.7.2 Raise RCS level as dictated in Step 8.6.2 of this procedure.

8.8 BORIC ACID FLOW PATHS TO THE CORE: When fuel is in the reactor there shall be at least one flow path to the core for boron injection by the CVCS.

8.8.1 During Cold Shutdown conditions, the Preferred flow path to the core is:

BASTs through Boric Acid Transfer Pumps through *-113A and *-113B (Console) to Charging Pump(s) suction. Charging pump(s) discharge through their normal charging path via HCV-*-121 (Console) and *-310A or *-310B (Console).

If *-113A or *-113B is unavailable, use MOV-*-350 (Console) to supply boric acid to Charging Pump(s) suction. If *-113A is available by *-113B and MOV-*-350 are unavailable, use *-113A and *-356 (local) to supply boric acid to Charging Pump(s) suction.

8.8.2 The preferred alternate flow path to the core is:

RWST through LCV-*-115B (VPB) to Charging Pump(s) suction. Charging Pump(s) discharge through their normal charging path via HCV-*-121 (Console) and *-310A or *-310B. If LCV-*-115B is inoperable, V-*-358 may be used to supply boric acid to the charging pump(s) suction.

8.8.3 A secondary alternate flow path to the core is:

BASTs through Boric Acid Transfer Pumps via *-113A and *-113B to the charging pump(s) suction. Charging Pump(s) discharge through one or more RCP seal water injection lines.

If *-113A and *-113B is unavailable, use MOV-350 to supply boric acid to the charging pump(s) suction. If *-113A is available and *-113B and MOV-350 are unavailable, use *-113A and *-356 (local) to supply boric acid to charging pumps suction.

8.8.4 **An acceptable Alternate flow path to the core** THAT WILL WORK ONLY WHEN THE RCS IS DEPRESSURIZED AND DRAINED TO MID NOZZLE IS:

RWST through *-864A (VPB) and *-864B (VPB) through *-887 (local) and the alternate low head SI flow path, *-872 (VPB).

8.8.5 Due to plant configurations and maintenance activities, the above flow paths may not be available. In this case, an acceptable alternate flow path will need to be identified, and a temporary procedure written identifying the required operator actions to establish a boron injection flow path. An Operator Aid shall be posted to reflect the flow path and then cancelled when the normal flow paths are restored to operable status.

8.9 If less than two Coolant Loops are operable, take immediate corrective action to return at least two Coolant Loops to operation.

8.10 At least one ICW header and basket strainer shall be operable and lined up to supply the CCW Heat Exchangers. The A ICW Pump shall be operable when the A RHR Loop is required to be operable. The B or C ICW Pump shall be operable when the B RHR Loop is required to be operable. At least one ICW Pump shall be capable of being supplied from an operable Emergency Diesel Generator.

8.11 The CCW System has the following operability constraints:

8.11.1 The A or C CCW Pump and one CCW Heat Exchanger shall be operable when the A RHR Loop is required to be operable.

8.11.2 The B CCW Pump and one CCW heat exchanger shall be operable when the B RHR Loop is required to be operable.

8.11.3 Two CCW heat exchangers are required to be operable when 2 RHR Loops are required to be operable.

8.11.4 At least one CCW Pump shall be capable of being supplied from an operable Emergency Diesel Generator.

8.11.5 Two CCW pumps shall be in operation to support the operation of two RHR pumps and heat exchangers. One CCW pump should be in operation for single RHR pump and heat exchanger operation.

NOTE: The motivation for reducing the CCW System to single pump operation is to keep the system flow through each CCW heat exchanger to less than 6840 gpm (while operating above the CCW System low pressure pump start setpoint). This administrative flow rate limit for each CCW heat exchanger is a vendor recommendation as delineated in 3/4-OP-030.

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. There are three sources of borated water available for injection through three different paths:

(1) The boric acid transfer pumps can deliver the boric acid tank contents to the charging pumps.

(2) The charging pumps can take alternate suction from the refueling water storage tank.

(3) The safety injection pumps can take their suction from the refueling water storage tank and inject the boron injection tank contents. *

The quantity of boric acid in storage from either the boric acid tanks or the refueling water storage tank is sufficient for cold shutdown at any time during core life.

One channel of heat tracing is sufficient to maintain the specified temperature limit.

* See reference (11) on Page B3.4-1

Reference

FSAR - Section 9.2

3. Emergency Containment Filtering System

Two of three filter units have capacity to meet the MHA analysis.(7)(8)

4. Component Cooling System

One pump and two heat exchangers meet the requirements of the MHA analysis.(10)

5. Intake Cooling Water System

CONTROLLED DOCUMENT

One pump meets the requirements of the MHA analysis.(6)

References:

(1) FSAR 6.2.2
(2) FSAR 14.2.5
(3) FSAR 14.3.2
(4) FSAR 14.1.9
(5) FSAR 6.2.3
(6) FSAR 14.3.4
(7) FSAR 6.3
(8) FSAR 14.3.5
(9) FSAR 6.4
(10) FSAR 9.3
(11) The requirement for use of the BIT tanks for Mitigation of the Main Steam Line Break accident has been removed following installation of the Model 44F Steam Generators. The required supporting analyses can be found in L-81-(502), dated 11/30/81 The temperature requirement above 145° F is no longer applicable. Therefore, the heat tracing requirement is not necessary. There is no Boron Concentration Requirement in the BIT.

NRC EXAM QUESTION REVIEW

QUESTION 7.13:

List six things you are required to monitor during refueling.

RESPONSE:

This is an open-ended question and as such we request that there should be other acceptable answers such as those highlighted in the attached reference material.

REFERENCE:

3-OP-038.1, Preparation for Refueling Activities, pgs. 9-13
3-OP-040.2, Refueling Core Shuffle, pgs. 7-13

5.2 Requirements Prior to Core Alterations (Excluding Fuel Movement)

INIT

Date/Time Started: ______________________

5.2.1 Initial Conditions

____ 1. The unit is in Mode 6, Refueling.

5.2.2 Procedure Steps

NOTES

- *This section provides the Technical Specification requirements to be completed for any core alteration except fuel movement.*
- *If different types of core alterations are being performed in succession and Technical Specification equipment operability requirements have been maintained, this section need only be completed for the initial start of the core alterations.*

1. Within 100 hours prior to the start of core alterations, perform the following:

____ a. **Establish containment integrity** in accordance with 3-OSP-051.12, Refueling Containment Penetration Alignment. [Commitment - Step 2.3.1]

____ (1) Record date/time 3-OSP-051.12 is completed. __________/__________

____ b. Perform Containment Air Particulate Channels R-3-11 and R-3-12 functional test in accordance with 3-OSP-067.1, Process Radiation Monitoring Operability Test.

____ (1) Record date/time the applicable sections of 3-OSP-067.1 are completed. __________/__________

____ c. Perform the Auxiliary Hoist Load Test in accordance with 3-OSP-038.3, Auxiliary Hoist Operability. (Make N/A if Auxiliary Hoist will not be used for the Core Alterations.

(1) Record date/time 3-OSP-038.3 is completed. __________/__________

INIT

5.2.2 (Cont'd)

2. Within 8 hours prior to the start of core alterations, perform the following:

____ a. Complete 3-OSP-059.1, Source Range Nuclear Instrumentation Analog Operational Test

____ (1) Record the date/time 3-OSP-059.1 is completed. ________ / ________

3. Within 1 hour prior to the start of core alterations perform the following:

____ a. **Establish and check the direct communications between the Control Room and the Refueling Canal area.**

____ (1) Record the date/time direct communications checked. ________ / ________

____ 4. Prior to the start of core alterations, and during core alterations, commence performance of Attachment 3, Core Alterations Minimum Equipment Checklist.

CAUTION

All core alterations shall be directly supervised by a Senior Reactor Operator, in the containment, who has no other concurrent responsibility during the core alterations [Commitment - Step 2.3.2]

____ 5. Prior to the start of core alterations, verify Senior Reactor Operator is stationed in the containment to directly supervise the core alterations.

Date/Time Completed: ________________ / ____

PERFORMED BY (Print)	INITIALS
________________	________
________________	________
________________	________

REVIEWED BY: ______________________________
Plant Supervisor - Nuclear or SRO Designee

5.3 Requirements Prior to Refueling Core Shuffle

INIT

Date Time Started: ____________

5.3.1 Initial Conditions

____ 1. The unit is in Mode 6, Refueling.

5.3.2 Procedure Steps

____ 1. Complete Attachment 2, Refueling Equipment Inventory Checklist.

____ 2. Verify that a satisfactory channel check has been completed for the following **area radiation monitors** in accordance with 3-OSP-201.1, RCO Daily Logs or the monitor has been replaced by a temporary portable monitor equipped with an alarm:

____ a. R-2, Unit 3 Containment Oper Floor

____ b. R-7, Unit 3 Spent Fuel Bldg Canal Area

____ c. R-19, Unit 3 Spent Fuel Pit Exhaust Duct

____ d. R-21, Unit 3 Spent Fuel Bldg North Wall

____ 3. Verify the Unit 3 Spent Fuel Pit SPING High Range Noble Gas Monitor has been determined to be operable by the Chemistry Department personnel in accordance with OP-0204.2, Periodic Tests Checks, and Operating Evolutions.

____ 4. Perform 3-OSP-034.1, Spent Fuel Pit Inlet and Exhaust Damper Operability Test.

5. Within 100 hours prior to the start of fuel movement, perform the following Technical Specification requirements:

____ a. **Establish containment integrity** in accordance with 3-OSP-051.12, Refueling Containment Penetration Alignment. [Commitment - Step 2.3.1]

____ (1) Record date/time 3-OSP-051.12 is completed.
________/________

INIT

5.3.2.5 (Cont'd)

___ b. Perform the manipulator crane load and automatic cutoff test in accordance with 3-OSP-038.2, Manipulator Crane Operability Test.

___ (1) Record date/time 3-OSP-038.2 is completed. ________/________

___ c. Perform the Auxiliary Hoist load test in accordance with 3-OSP-038.3, Auxiliary Hoist Operability Test.

___ (1) Record date/time 3-OSP-038.3 is completed. ________/________

___ d. Perform the Containment Air Particulate Channels **R-3-11 and R-3-12 functional test** in accordance with 3-OSP-067.1, Process Radiation Monitoring Operability Test. [Commitment - Step 2.3.1]

___ (1) Record date/time the applicable section of 3-OSP-067.1 are completed. ________/________

6. Within 8 hours prior to the start of fuel movement, perform the following Technical Specification requirement:

___ a. Complete 3-OSP-059.1, **Source Range** Nuclear Instrumentation Analog Operational Test.

___ (1) Record date/time 3-OSP-059.1 is completed. ________/________

7. Within 2 hours prior to the start of fuel movement, perform the following Technical Specification requirement:

___ a. **Verify the refueling canal water level is greater than or equal to 56 feet, 10 inches.**

___ (1) Record date/time refueling canal water level is verified. ________/________

8. Within 1 hour prior to the start of fuel movement, perform the following Technical Specification requirement:

___ a. **Establish and check the direct communications between the Control Room and the applicable refueling stations.**

___ (1) Record the date/time direct communications checked. ________/________

INIT

5.3.2 (Cont'd)

___ 9. Prior to the start of fuel movement, verify with the HPSS that Health Physics coverage is available.

___ 10. Prior to the start of fuel movement in the reactor vessel, verify the reactor has been subcritical for at least 100 hours:

___ (1) Record date/time reactor was subcritical. ________/________

___ (2) Record date/time reactor verified to be subcritical greater than 100 hours. ________/________

Date/Time Completed: ____________________

PERFORMED BY (Print)	INITIALS
____________________	________
____________________	________
____________________	________
____________________	________
____________________	________
____________________	________

REVIEWED BY: ______________________________
Plant Supervisor - Nuclear or SRO Designee

4.0 PRECAUTIONS/LIMITATIONS

4.1 Minimum equipment operability requirements, requiring immediate suspension of refueling operations, are listed in Attachment 3, Refueling Core Shuffle Minimum Equipment Checklist.

4.2 All personal items, glasses, pencils, personnel monitors, etc., shall be tied or taped to prevent them from falling into the refueling cavity. Each person should inventory personal items he takes into the containment. Do not take any unnecessary items into the containment.

4.3 All applicable radiation protection precautions and procedures shall be observed.

4.4 If, at any time, the Plant Supervisor - Nuclear suspects that continued refueling will involve undue risk to personnel or equipment, or will compromise the Technical Specifications or license provisions, operations shall be suspended pending resolutions.

4.5 Tools and equipment which are withdrawn from the refueling water should be monitored for radiation.

4.6 **A minimum level of 56 feet 10 inches shall be maintained in the Spent Fuel Pit at all times.**

4.7 The manipulator crane operator shall not unlatch from fuel assemblies which have been installed in the core until directed to do so by the individual maintaining and evaluating the inverse count rate ratio plot.

4.8 Access to the refueling work stations shall be restricted to members of the refueling team and observers authorized by the Nuclear Watch Engineer supervising fuel movement.

4.9 Prior to lifting any refueling equipment with a hoist, visually inspect the associated hoist take up drums for correct take up sequence.

4.10 Manning refueling stations:

4.10.1 **All core alterations shall be directly supervised by a licensed SRO stationed in the containment. [Commitment - Step 2.3.1]**

4.10.2 All personnel designated to operate refueling equipment must complete the applicable refueling equipment qualifications prior to operating the refueling equipment.

4.10.3 A Nuclear Watch Engineer (NWE) shall be in the containment to supervise fuel movement.

1. If a problem arises in the Spent Fuel Pit that stops fuel movement in the containment, the NWE should go to the Spent Fuel Pit to help correct the problem.

NOTE

Steps 4.10.4, 4.10.5, and 4.10.6 may be performed by a qualified vendor.

4.10.4 The manipulator crane shall be manned by a designated primary or alternate licensed Reactor Control Operator (RCO).

1. The operators are designated by the Refueling Outage Coordinator and the Plant Supervisor - Nuclear.

2. Prior to operating the Manipulator Crane, non-designated RCO's will be trained until they are sufficiently qualified as determined by the Plant Supervisor - Nuclear and the Refueling Outage Coordinator.

4.10.5 An operator shall be present to visually verify fuel insertion into the core in accordance with Enclosure 3, Observation of Fuel Assembly Loaded into the Reactor Vessel.

4.10.6 The Spent Fuel Pit Bridge Crane shall be manned by a licensed or non-licensed operator.

4.10.7 **Reactor Engineering personnel or a licensed operator shall be stationed in the Control Room to maintain the inverse Count Rate Ratio Plot in accordance with Attachment 1, Inverse Count Rate Monitoring During Refueling.**

4.10.8 **A licensed or non-licensed operator shall maintain communications in the Control Room with the other refueling stations.**

4.10.9 The same non-licensed operators should man their assigned refueling station for both 3 hour refueling shifts during the 12 hour refueling schedule to provide continuity.

1. This may be relaxed if other operating evolutions make it impossible to man a station with the same personnel.

4.11 Lifting of the fuel assemblies by means of tools or adapters attached to the assembly shall be performed with the assembly in the vertical posit[illegible]

4.12 Records of all fuel movements shall be kept in the Control Room. Local records shall be maintained of fuel movements on the operating deck of the containment and in the Spent Fuel Pit area.

4.13 The Control Room copy of the fuel movement record serves as the permanent history of fuel transfer and core loading.

4.14 Any inconsistencies in the loading sequence or in the recording of the fuel movements shall be cause to immediately cease fuel movements and notify the Plant Supervisor - Nuclear and the Reactor Engineering member on site.

4.15 Changes to the Fuel Handling Data Sheet (Form 5712) have no adverse affects on nuclear safety, nor constitute a change to the intent of the Fuel Handling Data Sheets as long as the final core configuration for the cycle has not been changed.

4.15.1 Changes to a Fuel Handling Data Sheet which are a change of intent, are made in accordance with AP-0109.3, On The Spot Changes To Procedures.

4.15.2 Changes to a Fuel Handling Data Sheet which are not a change of intent, are made in accordance with Attachment 2, Fuel Handling Data Sheet Changes.

4.16 Each shift change shall be accompanied by a turnover that fully describes the status of the equipment at that refueling station, including any variations in equipment parameters (e.g., indexing, tape measurements). The oncoming operator shall then evaluate the equipment status and verify that the equipment and lighting are functionally acceptable before initiating any fuel movement.

4.17 The Inverse Count Rate Ratio (ICRR) shall be monitored during fuel movement in the core. If the ICRR falls below .4, the shuffle shall be stopped, the situation analyzed and the Operations Superintendent - Nuclear notified prior to proceeding.

4.18 Do not move a fuel assembly into the transfer canal from the Spent Fuel Pool until the lifting frame is in the full up position and verified.

4.19 Prior to any horizontal moves of fuel assemblies, verify the fuel assembly is in the full up position by visual verification.

NOTE

The thimble plug handling tool may be left on the auxiliary hoist and fitting in the bracket, mounted on handrail during manipulator crane movement.

4.20 Any special equipment or tools used in the reactor cavity or refueling canal area shall be returned to their proper storage place after use to prevent them from interfering with the manipulator crane movement.

4.21 Any small hand tools used to perform maintenance in the reactor cavity or refueling canal areas shall be removed from the cavity or refueling canal after use.

4.22 The fuel loading status board located in the Control Room shall be kept current for each fuel assembly or insert movement.

4.23 In the event of damage to a fuel assembly, refer to 3-ONOP-033.3, Accidents Involving New or Spent Fuel.

4.24 If communication is lost between any refueling station and the Control Room, no fuel movement at that station shall be initiated until communication is restored.

4.24.1 Fuel must be placed in a safe storage location and unlatched from any refueling equipment until communications is restored.

4.25 Attachment 3, Refueling Core Shuffle Minimum Equipment Checklist shall have been completed prior to movement of fuel and each 8 hour shift thereafter, except for the RHR item which shall be done every 4 hours.

4.26 All core alterations shall be directly supervised by a licensed Senior Reactor Operator who has no other concurrent responsibilities during this operation.

4.27 During movement of fuel assemblies within the reactor vessel, a minimum cavity level of 56 feet 10 inches shall be maintained and at least two drop lights in the reactor vessel shall be operable.

4.27.1 Use caution when moving lights near the vessel hot leg nozzles as RHR suction could pull lights into the hot legs.

4.28 If lighting failures occur that hinder any visual monitoring specifically required of the operator at that refueling station, the failures shall be corrected prior to initiating any fuel movement at that location.

4.29 Any problems with lighting or operation of any equipment, directly or indirectly involved with the movement of fuel, shall be brought to the attention of the Plant Supervisor - Nuclear for evaluation and resolution.

4.30 Heavy loads, defined as weight in excess of 2000 pounds, shall be prohibited from travel over irradiated fuel assemblies in the spent fuel pool.

4.31 A copy of Enclosure 3, will be used to remind containment operators of specific observations required during fuel movement.

4.32 The below listed fuel racks do not have their supports directly under the last row of storage cells on the edge of the rack. To prevent these racks from tilting, do not load or unload any fuel storage rack so that it only has fuel stored along its edge listed in the following table:

ROW AND COLUMN IN SFP RACKS WHICH MAY CAUSE RACK TILT	LAST ROW AND COLUMN OF SFP RACKS WHICH MAY BE TILTED
V 1 -- 53	K 1 -- 53
53 A -- V	41 A -- V
53 W -- JZ	45 W -- JZ
75 AA -- LL	68 AA -- LL
76 AA -- LL	85 AA -- LL

Fuel assemblies may be in the rows which might cause rack tilt only if there is at least an equal number of assemblies in the same rack two or move rows in from the edge of the rack.

4.33 Fuel may be moved into Region 2 of the Spent Fuel Racks only if it meets the burnup and enrichment limits specified in the Technical Specifications, Table 3.17-1. Otherwise, it must be moved into Region 1 of the Spent Fuel Racks.

4.34 All personnel engaged in fuel handling activities shall comply with OP-16000.1, Limitations and Precautions for Handling Fuel Assemblies.

5.0 STARTUP/NORMAL OPERATION

5.1 Refueling Core Shuffle

INIT

Date/Time Started: ____________

5.1.1 Initial Conditions

____ 1. The unit is in Mode 6, Refueling.

5.1.2 Procedure Steps

NOTES

- *Fuel Handling Data Sheet (Form 5712) is shown on Enclosure 1.*
- *The Fuel Handling Data Sheet Temporary Procedure format is shown on Enclosure 2.*

____ 1. Prior to commencing fuel movement, verify the Fuel Handling Data Sheets (Form 5712) have been reviewed and approved in accordance with AP-0109.6, Temporary Procedures.

a. Record the Temporary Procedure number.

____ TP-________

____ 2. Prior to commencing fuel movement, verify 3-OP-038.9, Refueling Activities Checkoff List is complete through the QC Hold Point prior to Section 3 of Attachment 1.

REACTOR SUPERVISOR VERIFICATION POINT

Verify the fuel being moved into Region 2 of the spent fuel racks has been verified to meet the burnup and enrichment requirements specified in Technical Specification Table 3.17-1.

Verified by ______________________ ____________

Reactor Supervisor Signature *Date*

____ 3. Prior to commencing fuel movement, commence performance of Attachment 3, Refueling Core Shuffle Minimum Equipment Checklist.

4. Commence fuel movement and perform the following:

____ a. Maintain the inverse count rate rate monitoring in accordance with Attachment 1, Inverse Count Rate Monitoring During Refueling.

INIT 5.1.2.4 (Cont'd)

____ b. Observe fuel assembly loading into the reactor vessel in accordance with Enclosure 3, Observation of Fuel Assembly Loaded into Reactor Vessel.

____ c. Reactor Engineering shall maintain the Fuel Status Record in accordance with AP-0149.1, Special Nuclear Material Accountability.

____ d. If necessary, make changes to the Fuel Handling Data Sheet temporary procedure in accordance with Attachment 2, Fuel Handling Data Sheet Changes.

____ e. Document any fuel handling movement delays in accordance with Attachment 4, Fuel Handling Movement Delays.

5. After fuel movement has been completed, perform the following:

____ a. Perform a core map with an approved copy of the post refueling core pattern in accordance with OP-16900.13, Core Mapping Following Core Loading.

____ b. Stop performance of Attachment 3, Refueling Core Shuffle Minimum Equipment Checklist.

____ c. Notify the on-shift Health Physics Shift Supervisor to perform a radiation survey around the exterior of the Spent Fuel Building to ensure that the radiation levels have not increased greater than 1mr/hr above background due to the increased spent fuel storage.

Date/Time Completed: ____________________

PERFORMED BY (Print)	INITIALS
____________________	________
____________________	________
____________________	________

REVIEWED BY: ______________________________
Plant Supervisor - Nuclear or SRO Designee

NRC EXAM QUESTION REVIEW

QUESTION:7.18

the Natural Circulation Cooldown procedure, EOP-ES-0.2, requires the operator to "verify cold shutdown RCS boron concentration" after boration.

NOTE:
Excerpts of EOP-ES-0.2 are enclosed for reference.

b) State the indication utilized to judge the ultimated shutdown condition.

RESPONSE:

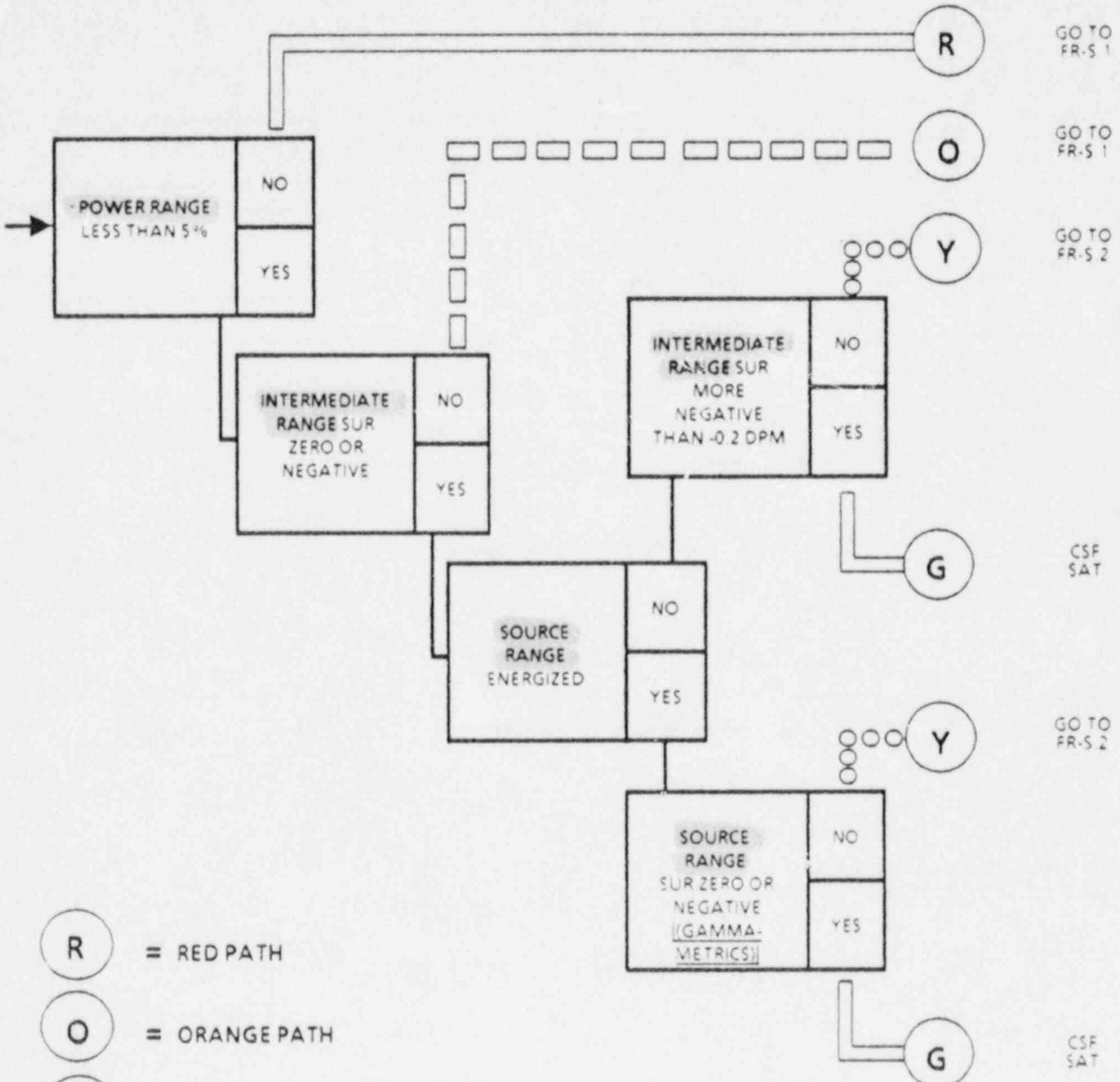
We request that you accept "monitoring critical safety function status trees" as an additional answer. The reason for this request is that the status trees are being monitored at this time and one of the items monitored is nuclear instrumentation.

REFERENCE:

3-EOP-F-O, Critical Safety Function Status Trees, pg. 5

7.18

ENCLOSURE F-0.1 SUBCRITICALITY



R = RED PATH

O = ORANGE PATH

Y = YELLOW PATH

G = GREEN PATH

• HEB/sr/dl

NRC EXAM QUESTION REVIEW

QUESTION: 8.04

Unit 3 and Unit 4 are operating at 100% power when Boric Acid Transfer Pump (BATP) 3A fails. BATP 4A had failed the previous day, and expected time of repairs on both pumps is in excess two weeks. Assuming all other components operable, select the statement below which most correctly describes the actions required by Technical Specifications.

NOTE; Technical Specifications are enclosed for reference.

a. Unit 3 or 4 must be placed in hot shutdown.

b. Units 3 and 4 must be placed in hot shutdown.

c. Unit 3 or 4 must be placed in cold shutdown.

d. Units 3 and 4 must be placed in cold shutdown.

RESPONSE:

We request that the answer be changed to "c" for the following reason:

You need 2 pumps for single unit operation and 3 pumps for dual unit operation. With 2 pumps out of service for more than 2 weeks (exceeds the 24 hour time limit) we have only 2 operable pumps which meet the requirement for single unit operation.

REFERENCE:

Technical specifications, section 3.6, pgs. 3.6-1 and 3.6-2

3.6 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability: Applies to the operational status of the Chemical and Volume Control System.

Objective: To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification: a. When fuel is in the reactor there shall be at least one flow path to the core for boron injection.

b. A reactor shall not be made critical unless the following Chemical and Volume Control System conditions are met:

1. TWO associated charging pumps shall be operable.

2. TWO boric acid transfer pumps shall be operable.

3. The boric acid tanks in service shall contain a total of at least 3,080 gallons of a 20,000 to 22,500 ppm boron solution at a temperature of at least 145 F.

4. System piping, interlocks and valves shall be operable to the extent of establishing one flow path from the boric acid tanks, and one flow path from the refueling water storage tank, to the Reactor Coolant System.

5. TWO channels of heat tracing shall be operable for the flow path from the boric acid tanks.

6. The primary water storage tank contains not less than 30,000 gallons of water.

c. The second reactor shall not be made critical unless the following conditions are met:

2. THREE boric acid transfer pumps shall be operable.*

3. The boric acid tanks in service shall contain a total of at least 6160 gallons of a 20,000 to 22,500 ppm boron solution at a temperature of at least 145 F.

4. System piping, interlocks and valves shall be operable to the extent of establishing one flow path from the boric acid tanks, and one flow path from the refueling water storage tank, to each Reactor Coolant System.

5. TWO channels of heat tracing shall be operable for the flow path from the boric acid tanks.

6. The primary water storage tank contains not less than 30,000 gallons of water.

d. During power operation, the requirements of 3.6.b and c may be modified to allow one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.6b and c within the time period specified, the reactor(s) shall be placed in the hot shutdown condition. If the requirements of 3.6.b and c are not satisfied within an additional 48 hours, the reactor(s) shall be placed in the cold shutdown condition. Specification 3.0.1 applies to 3.6.d.

1. One of the two operable charging pumps may be removed from service provided that it is restored to operable status within 24 hours.

2. One boric acid transfer pump may be out of service provided that it is restored to operable status within 24 hours.

3. One channel of heat tracing may be out of service for 24 hours.

* Only two boric acid transfer pumps need be operable during Unit 3 Low Power Physics testing for Cycle 3. This period shall not exceed 54 hours of testing.

P. O. BOX 14000, JUNO BEACH, FL 33408-0420

FPL

AUG 2 7 1986 8.08 (d)

L-86-348

Office of Nuclear Reactor Regulation
Attention: Mr. Thomas M. Novak, Acting Director
Division of Pressurized Water Reactor Licensing - A
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Novak:

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 & 50-251
Emergency Diesel Generator Technical Specifications

This will confirm our recent discussions with you and your staff regarding the limiting condition for operation (LCO) applicable to the operation of the emergency diesel generators at the Turkey Point plant. The NRC has requested, pending submittal and approval of revised technical specifications for Turkey Point in accordance with our Performance Enhancement Program (Project 10), that an LCO of fixed duration be established by administrative control. **The current Turkey Point Plant Technical Specification 3.7.2.b provides for notification to the NRC if a diesel generator outage is to be seven (7) days or more, but does not require a unit shutdown.**

Accordingly, FPL has established interim administrative controls implementing a seven (7) day LCO in modes 1, 2, 3 and 4 for the Turkey Point plant emergency diesel generators **in addition to the current Technical Specification 3.7.2.b requirement for notification. It is understood that exceeding this administrative LCO would necessitate placing both units in cold shutdown.** This would not preclude a request for emergency or exigent relief, if conditions warrant.

The basis for our interim LCO is attached.

Very truly yours,

C. O. Woody

C. O. Woody
Group Vice President
Nuclear Energy

COW/JKH/cab

L-86-348		NO. OF COPIES: 42	09/03/86
ANDERSON	ARIAS	ARREDONDO	BARROW
BLADOW	BOISSY	BREWER	CRISLER
DANEK	DICKEY	DONIS	FLUGGER
FRANCIS	FRANKLIN	GREEN	GRITZ
HORRELL	KARCH	KERN	KLEIN
MARSH	MILLER	McDONALD	McLAUGHLIN
NUTWELL	PACE	PANIANI	PARKER
POTERALSKI	PREAST	REIS	RICHARDS
SCHOFFMAN	SKELLEY	STEVENS	SYMES
VAULT CUSTODIAN	VILLAR	WILK	WILLIAMS, Jr.
YORK	YOUNG		

JKH1/028/1

NRC EXAM QUESTION REVIEW

QUESTION: 8.08

Concerning the AC electrical operating system requirements, answer each of the following statements TRUE or FALSE.

NOTE: Technical Specifications are enclosed for reference.

d) Power operation may continue for a maximum of 7 days if one diesel is out of service.

RESPONSE:

We request that "TRUE" be accepted as a correct answer also IF the assumption was stated addressing the compensatory action letter which applies to Technical Specification 3.7.2.b.

REFERENCE:

C.O. Woody letter to Mr. Novak (NRC)
Technical Specifications, Section 3, pgs. 3.7-1 and 3.7-2

3.7 ELECTRICAL SYSTEMS

Applicability: Applies to the availability of electrical power for the operation of auxiliaries.

Objective: To define those conditions of electrical power availability necessary (1) to provide for safe reactor operation, and (2) to provide for the continuing availability of engineered safety features.

Specification:

1. Either reactor shall not be started from a cold shutdown without:

   a. The associated 239 KV-4160 volt start-up transformer in service.

   b. 4160-volt busses A and B of the associated unit, and either bus A or B of the second unit, energized.

   c. THREE out of FOUR 480-volt load centers and 480-volt motor control centers A, B or C, and D of the associated unit energized.

   d. TWO diesel generators operable with on site supply of 40,000 gallons of fuel available.

   e. Four batteries and associated DC systems are operable with FOUR out of SIX battery chargers operable.

2. During power operation or restarting from hot shutdown the following components may be inoperable:

   a. ONE start-up transformer may be out of service provided both diesel generators are operable. The NRC shall be notified within 24 hours and be advised of plans to restore the transformer to service.

b. Power operation may continue if ONE diesel generator is out of service provided (1) the remaining diesel generator is tested daily and its associated engineered safety features are operable, and (2) either start-up transformer is operable. If the diesel outage is to be seven (7) days or more the NRC shall be notified.

c. ONE battery may be out of service for a period of twenty four hours.

d. Specification 3.0.1 applies to 3.7.2.

NRC EXAM QUESTION REVIEW

QUESTION: 8.11

State five (5) methods of detecting RCS leakage into containment which would satisfy Technical Specifications.

RESPONSE:

We request that you accept "water inventory change" as an acceptable alternate to D.

REFERENCE:

Technical Specifications, pg. B-3. 1-4

3. Leakage

Any leakage from the reactor coolant system, or from any other system containing potentially radioactive material, is considered to be of major importance as it may indicate a condition is developing that would lead to gross leakage. Gross leakage must be prevented to minimize any remote possibility of release of activity to and from the site. Leakage prevention first of all protects the public and also it prevents potential contamination of the equipment. Prompt maintenance and repair leads to improved reliability, which is an overall operating objective.

Thus any indication of leakage; for example: unbalanced water inventories, radiation monitor reading increases, boric acid crystal deposits, insulation dampness; shall be considered to be the result of a leak and shall require immediate attention with prompt evaluation required.

Action shall be prompt as it is possible that a small leak may propagate and become a major leak. The fact that a leak of 5 gpm, at the maximum allowed reactor coolant activity, released as airborne material without holdup or cleanup, would not exceed 10 CFR 20 limits shall not permit relaxation of the requirement that action be prompt and positive.

When a real or imagined leak is detected, the Plant Supervisor will immediately initiate a detailed investigation as to source and cause after first notifying the Plant Superintendent or his designated alternate. Evaluation will be made by the Plant Superintendent, who will call upon Production Department supervisors, such as the Regional Superintendent and the Superintendent of Generating Stations, as necessary for consultation. This procedure is an established and proven one in operation of fossil fuel fired units when leaks develop, as it brings to bear the judgement of experienced persons.

When the leak has been identified, the plant management will determine by a safety evaluation whether operation may continue. Leakage source (ex. valve stem, pump shaft seal) shall be considered. Make up capability and potential increased demand shall also be one of the evaluation factors.

NRC EXAM QUESTION REVIEW

QUESTION: 8.19

The following events occurred while Unit 3 is operating at 100% power:
(Date: January 26, 1988)

1:00 AM: Accumulator "A" pressure drops below 600 psig (circumstances such that corrective action will not be complete for one week.)
1:10 AM: Commenced bringing Unit 3 to Hot Standby.
4:00 AM: Unit 3 is in Hot Standby.
4:00 PM: Commenced bringing plant to Hot Shutdown.
6:00 PM: Unit 3 is in Hot Shutdown.

Answer the following questions. Consider each case separately.

NOTE: Technical Specifications are enclosed for reference.

a) Were the time limits for any LCOs exceeded by the operators?

b) By what date/time must Unit 3 be in cold shutdown?

c) If Unit 3 were initially in Hot Standby when Accumulator "A" low pressure occurred at 1:00 AM, by what date/time must Unit 3 be in cold shutdown?

d) If a similar Accumulator "B" pressure drop occurred one hour after Accumulator "A" pressure drop occurred, by what date/time must Unit 3 be in cold shutdown.

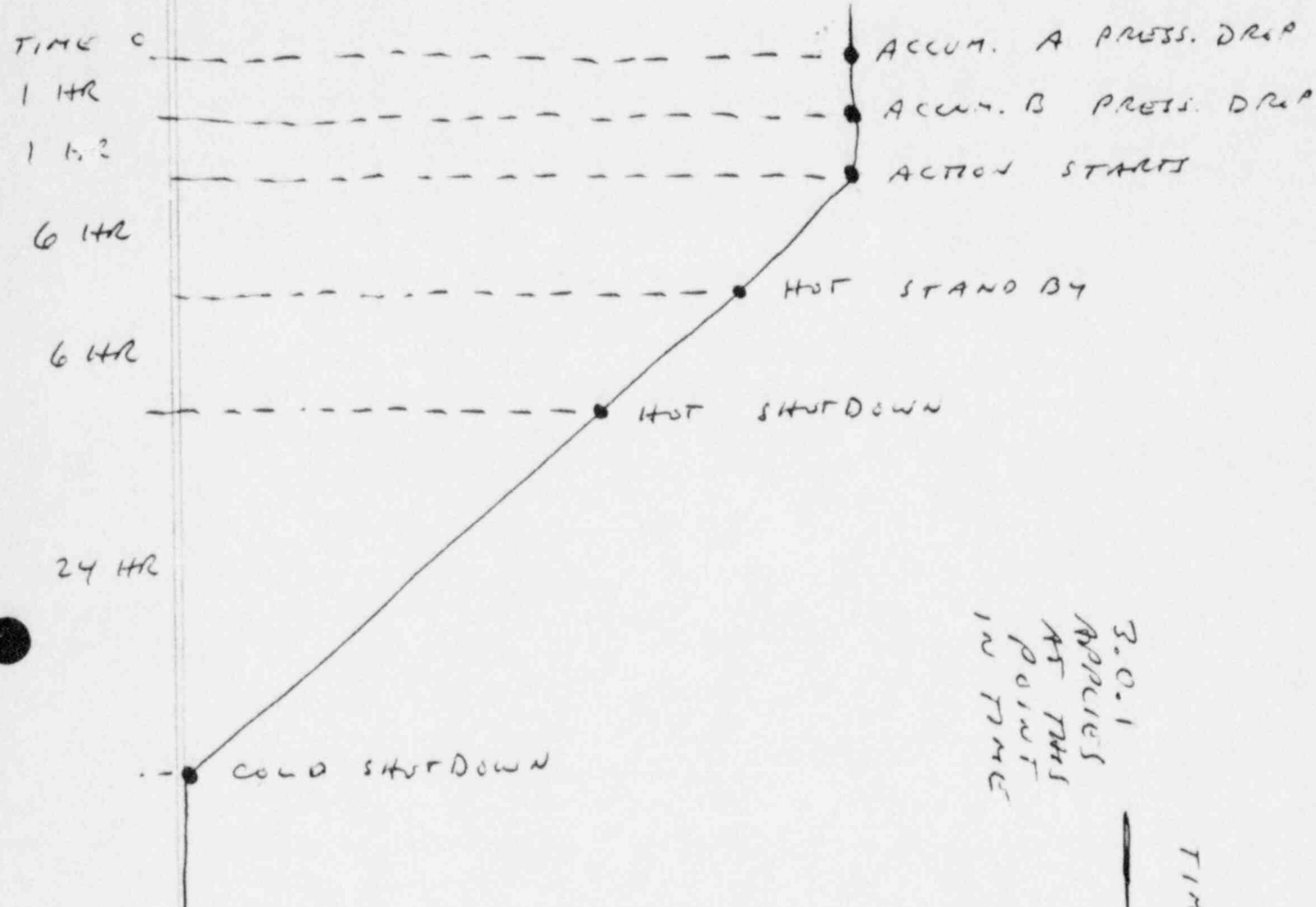
RESPONSE:

We concur with the answers to A, B and C, but we request that answer d) be changed to Jan. 27/3:00 PM for the following reason:

Technical specification 3.0.1 applies only after failure of the second accumulator and you have one hour from that time before taking action. Assuming that a shutdown has not commenced, you have until 3:00 PM on Jan. 27th to be in cold shutdown. (See attached time line)

REFERENCE:

Technical Specifications, section 3 and Bases, pgs. 3.0-1 and B-3.0-1



3.0.1 APPLIES AT THIS POINT IN TIME

TIME ZERO - 1 AM 1-26-88
1 HR 2 AM 1-26-88
1 HR 3 AM 1-26-88
6 HR 9 AM 1-26-88
6 HR 3 PM 1-26-88
24 HR 3 PM 1-27-88

B3.0 BASES - LIMITING CONDITIONS FOR OPERATION

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation within Section 3. In the event of a disagreement between the requirements stated in these Technical Specifications and those stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act shall take precedence and shall be met.

B3.0.1 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.4.2.a requires two containment spray pumps to be OPERABLE and provides explicit ACTION requirements if one spray pump is inoperable. Under the requirements of Specification 3.0.1, if both the required containment spray pumps are inoperable, **within 1 hour, measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours.** It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that provided for operation in subsequent lower OPERATION MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperability is discovered in a mode lower than the applicable mode. For example, if one containment spray pump was discovered to be inoperable while in STARTUP, the ACTION Statement would allow up to 109 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 31 hours, 93 hours would still be available before the plant would be required to be in COLD SHUTDOWN. However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 109 hours to 103 hours.

B3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirements.

B3.0.3 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

B3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with: (1) the full complement of required systems, equipment, or components OPERABLE and (2) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.0 LIMITING CONDITIONS FOR OPERATION - APPLICABILITY

3.0.1 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

a) At least HOT STANDBY within the next 6 hours,*

b) At least HOT SHUTDOWN within the following 6 hours, and*

c) At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.2 Non-compliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

3.0.5 For purposes of determining if a component is operable for LCO considerations, the component need not be considered inoperable due to inoperability of its normal or emergency power supply if all of its redundant components are operable with their normal or emergency power supplies operable.

*NOTE: Until full conversion to STS, when a LCO action statement requires a unit to be placed in HOT SHUTDOWN within 6 hours, refer to Table 1.1 and place the unit on the required status to meet the HOT STANDBY MODE.

FEB 03 '88 16:24 001 FPL-JNL 628 P.02

P. O. BOX 14000, JUNO BEACH, FL 33408-0420

FPL

FEBRUARY 3 1988

L-88-55

Dr. J. Nelson Grace
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta St., N.W., Suite 2900
Atlanta, GA 30323

Dear Dr. Grace:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
<u>Senior Reactor Operator Exam Comments</u>

Florida Power & Light Company has reviewed the Senior Reactor Operator Upgrade examination presented to Turkey Point operators on January 26, 1988. Our comments on questions in the examination were submitted for NRC review and consideration prior to grading the examinations in our letter L-88-49 dated February 1, 1988. The attachment to this letter contains a revised response to Question 8.04.

Should you or your staff have any questions on this information, please contact us.

Very truly yours,

J. K. Hays
for C. O. Woody
Executive Vice President

COW/PLP/gp

Attachment

cc: Document Control Desk, USNRC
Mr. J. A. Arildsen, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

PLP/001.SOL

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NRC EXAM QUESTION REVIEW

QUESTION: 8.04

Unit 3 and Unit 4 are operating at 100% power when Boric Acid Transfer Pump (BATP) 3A fails. BATP 4A had failed the previous day, and expected time of repairs on both pumps is in excess two weeks. Assuming all other components operable, select the statement below which most correctly describes the actions required by Technical Specifications.

NOTE: Technical Specifications are enclosed for reference.

a. Unit 3 or 4 must be placed in hot shutdown.

b. Units 3 and 4 must be placed in hot shutdown.

c. Unit 3 or 4 must be placed in cold shutdown.

d. Units 3 and 4 must be placed in cold shutdown.

RESPONSE:

We request that the answer be changed to "c" for the following reason:

You need 2 pumps for single unit operation and 3 pumps for dual unit operation. With 2 pumps out of service for more than 2 weeks (exceeds the 24 hour time limit) we have only 2 operable pumps which do not meet the requirement for dual unit operation.

During single unit operation, if the BATP Technical Specification requirements cannot be met, the affected unit would eventually be placed in cold shutdown. Turkey Point interprets this requirement to also be applicable to dual unit operation in that if the Technical Specification requirements for dual unit operation are exceeded, one unit will eventually be placed in cold shutdown.

REFERENCE:

Technical Specifications, Section 3.6, pgs. 3.6-1 and 3.6-2.

PLP/001.SOL