Draft Submittal
(Pink Paper)

Senior Reactor Operator Written Exam

TURKEY POINT JULY/AUGUST 2005 EXAM

50-250/2005-301 AND 50-251/2005-301

JULY 18 - 22, 2005 & AUGUST 1 - 5, 2005
JULY 15, 2005 (WRITTEN)

2/9/06

2005

Turkey Point

Written Exam

SRO

BANK INFORMATION REPORT
for Draft TP05-301-SRO

Item Type	#Items	Title
MCS	25	Multiple choice: single

Category 2 (SRO Tier)	#Items	Title
T1G1	7	
T1G2	5	
T2G1	4	
T2G2	2	
T3	7	

	Category 4 (Cognitive Level)	#Items	Title
	C/A(2.1/3.1)	1	
	C/A(2.2/3.6)	1	
	C/A(2.3/3.3)	1	
	C/A(2.5/3.7)	2	
	C/A(2.6/3.0)	1	
	C/A(2.9/3.3)	1	
13	C/A(3.3/3.4)	1	
	C/A(3.4/4.0)	1	
	C/A(3.5/4.1)	1	
	C/A(3.6/4.2)	1	
	C/A(3.7/4.3)	1	
	C/A(4.0/4.0)	1	
	M(2.2/3.3)	1	
	M(2.5/3.3)	1	
	M(2.5/3.7)	1	
	M(2.9/3.3)	1	
	M(3.1/4.0)	1	
12	M(3.2/3.3)	1	
	M(3.3/3.9)	1	
	M(3.4/3.8)	1	
	M(3.8/3.6)	1	
	M(3.9/4.7)	1	
	M(4.0/4.0)	1	
	M(4.0/4.3)	1	

Category 5 (Source)	#Items	Title
B	8	
M	5	
N	12	

RESEARCH RANDOMIZER

Close
For best results, close this window after each use.

Print

Download in Excel

Research Randomizer Results

1 Set of 25 Non-unique Numbers Per Set
Range: From 1 to 4 -- Unsorted

Job Status: Finished

Set #1:

11 13 15 17 19 21 23 25

4, 2, 3, 2, 4, 2, 2, 3, 2, 4, 1, 2, 1, 1, 1, 4, 4, 3, 1, 4, 2, 1, 2, 2, 2

1 2 3 4 5 6 7 8 9 10 12 14 16 18 20 22 24

ANSWER BREAK DOWN

RO			SRO
B	A	B	D
C	C	B	B
B	C	D	C
C	C	C	B
A	A	A	B
A	B	B	B
B	A	D	D
B	A	D	B
A	B	A	D
B	B	D	A
C	C	B	B
C	A	B	B
B	C	D	A
C	C	D	A
C	D	B	D
C	C	A	D
C	B	C	C
D	B	B	A
D	A	B	D
B	C	B	B
B	A	B	A
C	B	D	B
A	A	D	B
D	D	D	B
C	C	A	C

	RO		SRO		
A =	16	+	5	=	21
B =	24	+	11	=	35
C =	21	+	3	=	24
D =	14	+	6	=	20
	75		25		

QUESTIONS REPORT

for DRAFT TP05-301-SRO with TP&NRC changes

1. 003AG2.2.25 001//T1G2/DROPPED ROD/M(2.5/3.7)/N/TP05301/S/MC

Unit 4 had been operating at 100% power when a Bank "D" rod dropped. The crew entered ONOP-028.3, DROPPED RCCA. The rod was declared inoperable and power was reduced to < 50%.

Which ONE of the following describes the basis for the Technical Specifications LCO requiring the crew to reduce reactor power to 75% within one hour?

A. Ensures minimum DNBR in the core remains less than or equal to the applicable design limit for continued operation and in short-term transients.

B. Provides adequate protection against $F_Q(Z)$,Heat Flux Hot Channel Factor, in the event of a subsequent Loss of All AC Power (LOAAC) event.

C. Provides assurance that the effects of residual xenon redistribution impact from past operation near EOL is minimal.

D✓ Ensures that design margins to core limits will be maintained under both steady-state and anticipated transient conditions.

Feedback

REFERENCES:

1. Technical Specifications, 3.1.3.1, pages 3/4 1-17 & 1-18, Amendment Nos. 186 & 216
2. ADM-536, TECHNICAL SPECIFICATION BASES CONTROL PROGRAM, pages 32-36,41, rev 05/01/03

DISTRACTORS:

A Incorrect. The goal is to maintain DNBR greater than applicable design limits.

B Incorrect. The LOAAC accident is not an anticipated transient condition as described in Chapter 14 of the FSAR.

C Incorrect. This is the basis for Tech Spec 3.2.1, AXIAL FLUX DIFFERENCE, but had nothing to do with EOL.

D Correct. IAW ref 2, page 33, paragraph 3.

K/A CATALOGUE QUESTION DESCRIPTION:

- Dropped Control Rod; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

2. 006G2.1.28 001//T2G1/ECCS/M(3.2/3.3)/N/TP05301/S/MC

Unit 4 entered FR-C.2, RESPONSE TO DEGRADED CORE COOLING, after an ORANGE condition was identified. IAW FR-C.2 all Accumulator Discharge MOVs were closed.

Which ONE of the following describes why the Accumulator Discharge MOVs are shut?

A. To minimize subsequent RCS cool down and vessel thermal shock.

B✓ To minimize subsequent nitrogen introduction into the RCS.

C. To prevent Accumulator injection flow from hindering HHSI or RHR cooling flow.

D. To prevent the loss of Accumulator water which will be needed if conditions degrade to a RED condition.

Feedback

REFERENCES:
1. BD-EOP-FR-C.2, RESPONSE TO DEGRADED CORE COOLING, page 35, rev 12/14/02
2. EOP-FR-C.2, RESPONSE TO DEGRADED CORE COOLING, page 12, rev 12/14/02
3. SD 021/SYS.050,062,064, page 19, rev 04/22/04

DISTRACTORS:

A Incorrect. Thermal shock is a lower concern than degraded core cooling. Actions are taken in FR-C.2 to inject accumulators.

B Correct. IAW Basis.

C Incorrect. Accumulator injection will supplement and not hinder HHSI or RHR flow to the core.

D Incorrect. Accumulator injection is indeed called upon in the RED condition (FR-C.1) but it is also called for in this ORANGE condition to preclude a RED condition.

K/A CATALOGUE QUESTION DESCRIPTION:
- Emergency Core Cooling System (ECCS); Knowledge of the purpose and function of major system components and controls.

3. 014A2.02 001//T2G2/RPIS/C/A(2.6/3.0)/M/TP05301/S/MC

Unit 4 is at 100% power with all control rods in the fully withdrawn position when power is lost from the RPI inverter.

Which ONE of the following describes the effect on the RPI system and correct operator response to this event?

A. RPI indication has been lost. Reduce thermal power to less than 75% within 8 hours.

B. RPI indication has been lost. Restore the failed inverter to operable status within 1 hour or place Unit 4 in Hot Standby within the following 6 hours.

C.✓ RPI power has auto swapped to the CVT. Initiate PWOs to repair the failed Inverter and calibrate the RPI system to compensate for changes in power supply voltage.

D. RPI power has auto swapped to the CVT. Restore the failed inverter to operable status within 1 hour or place Unit 4 in Hot Standby within the following 6 hours.

Feedback

REFERENCES:

1. ONOP-028.2, RCC POSITION INDICATION MALFUNCTION, pages 5,6, rev 04/12/02
2. SD-006/SYS.028B, ROD POSITION INDICATION SYSTEM, pages 9,10, fig 5, rev 04/13/04
3. ONOP-028.1, RCC MISALIGNMENT, pages 6,7, rev 04/12/02
4. Tech Spec 3.1.3.3, REACTIVITY CONTROL SYSTEMS, page 1-23, Amendment Nos. 149 & 144

DISTRACTORS:

A Incorrect. Power has not been lost to the RPI System.

B Incorrect. Power has not been lost to the RPI System.

C Correct. IAW ONOP-028.2.

D Incorrect. The shutdown statement would be correct if all RPI indication had been lost. Since it has not been lost, the shutdown requirement does not apply.

K/A CATALOGUE QUESTION DESCRIPTION:

- Rod Position Indication System (RPIS); Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of power to the RPIS.

4. 015AG2.4.49 001//T1G1/RCP/C/A(4.0/4.0)/M/TP05301/S/MC

Unit 3 has been operating at 100% power for three days when the following conditions occur:

- Annunciator A-1/5, RCP SEAL LEAK-OFF HI FLOW, actuates
- Annunciator G-2/2, RCP B STANDPIPE HI LEVEL, actuates
- RCP "B" seal injection flow is 7.8 gpm
- RCP "B" seal leak-off flow is 6.1 gpm
- Seal return temperature is 150^{O}F and rising steadily

Based on the above indications, the crew enters ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL. IAW ONOP-041.1, the SRO should direct the operating crew to perform which ONE of the following?

A. RCP "B" operation may continue for up to 24 hours due to number two seal sticking.

B.✓ Manually trip the reactor, then stop RCP "B" and close its seal leakoff valve after the pump stops.

C. Commence unit shutdown using ONOP-100, FAST LOAD REDUCTION, when the turbine is tripped, then trip the reactor, when the reactor is tripped, then stop RCP "B".

D. Begin preparations to shutdown and stop RCP "B" using GOP-103, POWER OPERATION TO HOT STANDBY and contact plant management for further guidance.

Feedback

REFERENCES:
1. ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL, pages 7,12,13,23,30, rev 06/14/99C1
2. ARP-097CR, CONTROL ROOM ANNUNCIATOR RESPONSE, pages 25,352, rev 06/09/03

DISTRACTORS:

A Incorrect. This would be true on initial startup (or for 8 hours during normal operation).

B Correct. IAW ref 1.

C Incorrect. This would be true if seal flow was less than 6 gpm but greater than 5.5 gpm.

D Incorrect. This would be true if seal flow was less than 5.5 gpm.

K/A CATALOGUE QUESTION DESCRIPTION:
- Reactor Coolant Pump Malfunction; Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

5. 027AG2.2.25 001//T1G1/PZR PRESS/C/A(2.5/3.7)/M/TP05301/S/MC

Unit 4 is operating at 100% power on February 25, 2005, at 0930, when pressurizer pressure instrument PT-455 fails low. At 1045 pressure instrument PT-456 fails to 2300 psig. No bistables on either channel have been tripped.

Based on the above plant conditions, which ONE of the following describes the actions(s) that must be performed to satisfy Technical Specifications and why?

A. Be in at least Mode 3 within 6 hours; permits the time limits of the ACTION requirements to be reset to the point in time where the plant entered the new Mode to allow completion of remedial measures and a return to POWER.

B✓ Be in at least Mode 3 within 7 hours; permits shutdown in a controlled and orderly manner.

C. Be in at least Mode 4 within 12 hours; permits shutdown in a controlled and orderly manner.

D. Be in at least Mode 4 within 13 hours; permits the time limits of the ACTION requirements to be reset to the point in time where the plant entered the new Mode to allow completion of remedial measures and a return to POWER.

Feedback

REFERENCES:
1. Tech Specs 3/4.3.1, REACTOR TRIP SYSTEM INSTRUMENTATION, Table 3.3-1, Amendment NOS. 140 & 135
2. Tech Specs 3/4.0, APPLICABILITY, 3.0.3, Amendment NOS. 137 & 132
3. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, pages 22,23, rev 05/01/03

DISTRACTORS:

A Incorrect. Wrong time requirement and reason.

B Correct. IAW 3.0.3, within 1 hour action shall be initiated to place the unit in: (1) at least HOT STANDBY within the next 6 hours (for a total of 7), (2) at least HOT SHUTDOWN within the following 6 hours (for a total of 13), (3) at least COLD SHUTDOWN within the subsequent 24 hours (for a total of 37).

C Incorrect. Wrong time requirement.

D Incorrect. Wrong reason.

K/A CATALOGUE QUESTION DESCRIPTION:
- Pressurizer Pressure Control System Malfunction; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

6. 033GG2.4.30 001//T2G2/SFPCS/C/A(2.2/3.6)/N/TP05301/S/MC

Unit 4 is in Mode 6. On January 29th at 0900 the water level in the Spent Fuel Pool was lowered to 56' to perform maintenance. All movement of fuel assemblies and crane operation in the fuel storage area had been suspended at 0700 the same day. Maintenance was scheduled for completion with level restored by February 5th at 0900.

- Maintenance was completed and level restored on February 6th at 1000

Which ONE of the following correctly completes the statement:

This event is ___(1)___ to the NRC because ___(2)___.

A. Reportable; SFP level remained below the minimum for more than 24 hours beyond originally scheduled.

B.✓ Reportable; SFP level remained below the minimum for more than 7 days.

C. Not Reportable; SFP level was not lowered below the minimum level required by Tech Specs.

D. Not Reportable; the amount of time the SFP level remained below the minimum did not exceed 7 days.

Feedback

REFERENCES:

1. Tech Specs, 3.9.11, REFUELING OPERATIONS – WATER LEVEL – STORAGE POOL, page 9-12, Amendment Nos. 224 & 219

DISTRACTORS:

A Incorrect. The time exceeded 7 days.

B Correct. SFP level shall be maintained greater than or equal to elevation 56' – 10". If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

C Incorrect. Level was lowered below the minimum required by Tech Specs.

D Incorrect. The amount of time the SFP level remained below the minimum exceeded the maximum time allowed by Tech Specs.

K/A CATALOGUE QUESTION DESCRIPTION:

- Spent Fuel Pool Cooling System (SFPCS); Knowledge of which events related to system operations/status should be reported to outside agencies.

7. 039GG2.1.32 001//T2G1/MRSS/M(3.4/3.8)/N/TP05301/S/MC

Unit 4 is operating at 100% power when "A" MSIV is found to be inoperable in the OPEN position at 1400, 02-21-05.

Which ONE of the following describes the actions to be taken and why?

A. Be in MODE 3 by 2000, 02-21-05. Limits the pressure rise within containment in the event a steam line break occurs within containment.

B. Be in MODE 3 by 0200, 02-22-05. Minimizes the negative reactivity effects of the Reactor Coolant System cooldown associated with the blowdown in the event of a steam line break.

C. Be in MODE 3 by 1400, 02-22-05. Minimizes the negative reactivity effects of the Reactor Coolant System cooldown associated with the blowdown in the event of a steam line break.

D✓ Be in MODE 3 by 2000, 02-22-05. Limits the pressure rise within containment in the event a steam line break occurs within containment.

Feedback

REFERENCES:

1. Tech Specs, 3.7.1.5, MAIN STEAM ISOLATION VALVES, page 7-10, Amendment Nos. 137 & 132
2. 0-ADM-536, TECHNICAL SPECIFICATION BASES CONTROL PROGRAM, page 85, rev 05/01/03

REFERENCE PROVIDED: TS 3.7.1.5

DISTRACTORS:

A Incorrect. Action is incorrect. Basis is correct.

B Incorrect. Action is incorrect. Basis is incorrect.

C Incorrect. Action is incorrect. Basis is incorrect.

D Correct. "With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. The OPERABILITY of the MSIVs ensures that no more than one steam generator will blow down in the event of a steam line rupture." "This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment."

K/A CATALOGUE QUESTION DESCRIPTION:

- Main and Reheat Steam; Ability to explain and apply all system limits and precautions.

8. 055EA2.03 001//T1G1/STA BLACKOUT/M(3.9/4.7)/B/TP05301/S/MC

A loss of all AC power has occurred. The STA reports the status of the CSFs are as follows:

- Subcriticality – RED
- Core Cooling – RED
- Heat Sink – RED
- Integrity – GREEN
- Containment – GREEN
- Inventory - YELLOW

Which ONE of the following procedures should be used FIRST to mitigate these conditions?

A. EOP-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS

B.✓ EOP-ECA-0.0, LOSS OF ALL AC POWER

C. EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK

D. EOP-FR-C.1, RESPONSE TO INADEQUATE CORE COOLING

Feedback

REFERENCES:
1. EOP-ECA-0.0, LOSS OF ALL AC POWER, page 6, rev 02/22/02

DISTRACTORS:

A Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

B Correct. Note that precedes Step 1 of EOP-ECA-0.0 states that "CSF Status Trees are required to be monitored for information only. FRPs shall NOT be implemented."

C Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

D Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Offsite and Onsite Power (Station Blackout); Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.

9. 056AA2.51 001//T1G1/OFFSITE PWR/C/A(3.3/3.4)/B/TP05301/S/MC

Unit 3 was operating at 100% power when a Loss Of Off-site Power (LOOP) occurred.

Which ONE of the following describes the response of the reactor CORE delta T from the time the LOOP occurred until one hour later in the event?

Delta T ________ as natural circulation is being established, then ________.

A. Lowers; remains constant as heat removal is established with the atmospheric steam dumps.

B. Lowers; rises as decay heat load diminishes and heat removal is controlled by the atmospheric steam dumps.

C. Rises; remains constant as heat removal is established with the atmospheric steam dumps.

D✓ Rises; lowers as decay heat load diminishes and heat removal is controlled by the atmospheric steam dumps.

Feedback

REFERENCES:
1. FSAR

DISTRACTORS:

A Incorrect. Delta T has to become higher to establish a driving head for natural circulation.

B Incorrect. Delta T has to become higher to establish a driving head for natural circulation. Does not take decay heat into account. Distractor provides opposite of actual effect.

C Incorrect. Does not take decay heat into account.

D Correct.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Offsite Power; Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Delta T, (core, heat exchanger, etc.).

10. 058AG2.2.25 001//T1G1/DC PWR TS/C/A(2.5/3.7)/N/TP05301/S/MC

Unit 3 is operating at 100% power and Unit 4 is in COLD SHUTDOWN. For the past two hours, Unit 3 operators have been unable to raise float charge voltage above 124 volts on battery bank 3A. Checks of individual cell voltages have confirmed this value.

Based on the above plant conditions, which of the following describes the Technical Specification (TS) requirements and the reason for the requirements?

A.✓ Place Unit 3 in at least HOT STANDBY within 34 hours. This time allows the orderly shutdown of one unit at a time so as not to jeopardize the stability of the electrical grid by imposing a dual unit shutdown.

B. Place Unit 3 in at least HOT STANDBY within 12 hours. This time allows restoring the battery to within limits and not jeopardizing the stability of the electrical grid by imposing a dual unit shutdown.

C. Place Unit 3 in at least HOT STANDBY within 34 hours. This time allows the orderly shutdown of one unit at a time in order to avoid dual unit natural circulation cooldown in the event of a loss of both startup transformers.

D. Place Unit 3 in at least HOT STANDBY within 12 hours. This time allows restoring the battery to within limits and avoiding dual unit natural circulation cooldown in the event of a loss of both startup transformers.

Feedback

REFERENCES:

1. Tech Specs, 3/4.8.2, DC SOURCES, pages 8-13 – 8-16, Amendment Nos. 138 & 133
2. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, pages 92-95, rev 05/01/03

PROVIDE REFERENCE: TECH SPEC 3/4.8.2, pages 8-13 thru 8-16.

DISTRACTORS:

A Correct. IAW note (3) of Table 4.8-2, the "allowable value" for float voltage for each connected cell must be = 2.07 or the battery is considered INOPERABLE. IAW ACTION step "b," operators have 2 hours* *(which can be extended to 24 hours with Unit 4 in Mode 5) to correct the problem or have both Units in Mode 3 within the next 12 hours.

B Incorrect. 34 hours would be correct.

C Incorrect. The time is correct but the reason is incorrect.

D Incorrect. The time and reason are incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of DC Power; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

11. 064A2.06 001//T2G1/EDG/C/A(2.9/3.3)/N/TP05301/S/MC

You are the SRO on Unit 4. Following recovery from a Loss Of Off-site Power (LOOP), the 4A 4KV Bus has been transferred to the SU Transformer. 4A EDG is running unloaded at 900 RPM.

Which ONE of the following describes a procedural requirement regarding subsequent operation of the 4A EDG and consequences of failing to adhere to that requirement?

A. Perform a Normal Stop of the EDG within 4.5 hours. Running the EDG unloaded beyond 4.5 hours can result in overheating the EDG due to a lack of air flow across the radiator cooling fins.

B.✓ Perform a Normal Stop of the EDG within 4.5 hours. Running the EDG unloaded beyond 4.5 hours can result in an accumulation of oil in the exhaust which can lead to a fire.

C. Perform an Emergency Stop of the EDG within 4.5 hours. Running the EDG unloaded beyond 4.5 hours can result in overheating the EDG due to a lack of air flow across the radiator cooling fins.

D. Perform an Emergency Stop of the EDG within 4.5 hours. Running the EDG unloaded beyond 4.5 hours can result in an accumulation of oil in the exhaust which can lead to a fire.

Feedback

REFERENCES:

1. 4-OP-023, EMERGENCY DIESEL GENERATOR, step 4.3, rev 09/19/03
2. 4-ONOP-004.1, step 9, rev 02/26/99

DISTRACTORS:

A Incorrect. Normal Stop is the correct method of EDG shutdown. Reason stated is a potential low speed operation concern on Unit 3 but not on Unit 4 which has electric auxiliary cooling fans.

B Correct. Correct per ONOP-004.1 and 4-OP-023.

C Incorrect. Normal Stop is the correct method of EDG shutdown. Reason stated is a potential low speed operation concern on Unit 3 but not on Unit 4 which has electric auxiliary cooling fans.

D Incorrect. Normal Stop is the correct method of EDG shutdown.

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Diesel Generator (EDG) System; Ability to (a) predict the impacts of the following malfunctions or operations on the EDG system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Operating unloaded, lightly loaded, and highly loaded time limit.

12. 065AA2.06 001//T1G1/INST AIR/C/A(3.6/4.2)/M/TP05301/S/MC

Units 3 and 4 are both operating at 100% power when the following occurs on both Units:

- The Lag CM starts followed shortly by startup of both CD air compressors.
- Annunciator I-6/1, INSTR AIR HI TEMP/LO PRESS, actuates
- Annunciator G-1/2, CHARGING PUMP HIGH SPEED, actuates
- Annunciators C-1/1 – 1/3, SG A,B,C, LO/LO-LO LEVEL ALARMS, actuate
- SG levels are 25% and decreasing
- Unit 3 instrument air pressure is 56 psig
- Unit 4 instrument air pressure is 64 psig

Which ONE of the following describes the correct operator response?

A. Trip Unit 3 IAW ONOP-013, LOSS OF INSTRUMENT AIR, and perform a Fast Load Reduction on Unit 4 IAW ONOP-100, FAST LOAD REDUCTION.

B.✓ Trip both Units and enter EOP-E-0, REACTOR TRIP OR SAFETY INJECTION on both Units.

C. Perform a Fast Load Reduction on both Units IAW ONOP-100, FAST LOAD REDUCTION and establish AFW flow.

D. Allow both Units to trip and enter EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

Feedback

REFERENCES:
1. ONOP-013, LOSS OF INSTRUMENT AIR, page 6 & Foldout, rev 12/23/02

DISTRACTORS:

A Incorrect. Both Units are either below or at the trip criterion.

B Correct. IA to trip the Unit when its instrument air pressure drops below 65 psig. Both Units are either below or at the trip criterion.

C Incorrect. IA to trip the Unit when its instrument air pressure drops to 65 psig.

D Incorrect. IA to trip the Unit when its instrument air pressure drops to 65 psig.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Instrument Air; Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing.

13. 076GG2.4.49 001//T2G1/SWS/M(4.0/4.0)/N/TP05301/S/MC

Unit 3 is operating at 100% power when the following occurs:

- I-4/1, ICWP A/B/C MOTOR OVERLOAD, actuates
- H-8/5, CCW HX OUTLET HI TEMP, actuates
- I-5/4, TPCW HI TEMP/LO PRESS, actuates
- E-9/4, GEN EXCITER AIR HI TEMP, actuates
- TE-3414 (det #31) cold air (fan discharge) as read on R-347 Pts 5 and 6 is 60^{O}C
- TE-3416 (det #33) hot air (exciter armature outlet) as read on R-347 Pts 7 and 8 is 87^{O}C
- Reactive generator load = 150 MVAR in the Lag

Which ONE of the following describes the correct operator responses?

A.✓ Start standby ICW pump then stop affected ICW pump, reduce reactive load on the generator.

B. Start standby ICW pump then stop affected ICW pump, initiate fast load reduction.

C. Stop affected ICW pump then start standby ICW pump, initiate fast load reduction.

D. Stop affected ICW pump then start standby ICW pump, reduce reactive load on the generator.

Feedback

REFERENCES:

1. ONOP-019, INTAKE COOLING WATER MALFUNCTION, pages 5,12 rev 10/24/02C

DISTRACTORS:

A Correct. Actions IAW ref 1.

B Incorrect. Reduce reactive load vice FAST LOAD REDUCTION.

C Incorrect. Start standby ICW pump THEN stop affected ICW pump. Reduce reactive load vice FAST LOAD REDUCTION.

D Incorrect. Start standby ICW pump THEN stop affected ICW pump.

K/A CATALOGUE QUESTION DESCRIPTION:

- Service Water System (SWS); Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

14. G2.1.14 001//T3/OPS/M(2.5/3.3)/N/TP05301/S/MC

Unit 3 has been in Mode 3 for three days to facilitate the performance of maintenance. Prior to this, Unit 3 had been operating at 100% power for an extended period of time. Following completion of the work the unit will be returned to full power operation.

Which ONE of the following identifies two plant personnel that are required to be notified to review the requirements of 0-ADM-529, UNIT RESTART READINESS, prior to entering Mode 2 and again prior to entering Mode 1?

A.✓ Site Vice President — Plant General Manager

B. Reactor Engineering — Work Control Center Supervisor

C. Chemistry Department — Security Department

D. Health Physics — Shift Manager

Feedback

REFERENCES:

1. GOP-301, HOT STANDBY TO POWER OPERATION, pages 16,19,62, rev 04/22/04

DISTRACTORS:

A Correct. IAW GOP-301, sections 3.1.19 & 3.2.17 the following personnel are to be notified prior to entry into Modes 1 & 2: Site VP, Plant GM, and Operations Shift Manager.

B Incorrect. See A.

C Incorrect. See A.

D Incorrect. See A.

K/A CATALOGUE QUESTION DESCRIPTION:

- Knowledge of system status criteria which require the notification of plant personnel.

15. G2.1.33 001//T3/OPS/C/A(3.4/4.0)/N/TP05301/S/MC

Plant conditions for Unit 4 are as follows:

- Mode 2
- Low Power Physics testing is in progress
- All shutdown bank rods are fully withdrawn

Which ONE of the following will require Unit 4 to enter a Technical Specification Action Condition?

A. K_{eff} = .99 and MTC = -3.0 x 10^{-4} delta k/k/^{O}F Lowest T_{avg} = 611^{O}F

B. K_{eff} = 1 and MTC = -3.0 x 10^{-4} delta k/k/^{O}F Highest T_{avg} = 610^{O}F

C. K_{eff} = .99 and MTC = +5.5 x 10^{-5} delta k/k/^{O}F Lowest T_{avg} = 531^{O}F

D.✓ K_{eff} = 1 and MTC = +5.5 x 10^{-5} delta k/k/^{O}F Highest T_{avg} = 530^{O}F

Feedback

REFERENCES:

1. Tech Spec 3.1.1.3, MODERATOR TEMPERATURE COEFFICIENT, pages 1-5 & 1-6, Amendment Nos. 137 & 132
2. Tech Spec 3.10.3, PHYSICS TESTS, page 10-3, Amendment Nos. 137 & 132
3. Tech Spec 2.1.2, SAFETY LIMITS – REACTOR CORE, pages 2-1 & 2-2, Amendment Nos. 137 & 132

DISTRACTORS:

A Incorrect. Applicable at EOL, not BOL.

B Incorrect. Applicable at EOL, not BOL.

C Incorrect. Applicable only with K_{eff} greater than or equal to 1.

D Correct. IAW TS 3.10.3.c given the other plant conditions, action step "b" applies.

K/A CATALOGUE QUESTION DESCRIPTION:

- Conduct of Operations; Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

16. G2.2.20 001//T3/EQUIP CONT/M(2.2/3.3)/B/TP05301/S/MC

Unit 4 is at 100% power. Mechanical Maintenance is planning to erect a scaffold over redundant safety related equipment to perform trouble shooting activities.

Which ONE of the following identifies the highest level of approval required for the erection of this scaffolding?

A. Work Control Manager

B. Unit Supervisor

C. Shift Manager

D✓ Assistant Operations Manager

Feedback

REFERENCES:

1. ADM-012, step 3.3.2

DISTRACTORS:

A Incorrect.

B Incorrect.

C Incorrect.

D Correct.

K/A CATALOGUE QUESTION DESCRIPTION:

- Equipment Control; Knowledge of the process for managing troubleshooting activities.

17. G2.2.32 001//T3/EQUIP CONT/C/A(2.3/3.3)/B/TP05301/S/MC

Reactor engineering had designed a core loading pattern that will be performed during the next refueling outage. The CHANGE will result in placing the "twice-burned" fuel assemblies more toward the periphery and the new fuel assemblies more toward the center of the core. Based on engineering calculations, it has been determined that Kexcess will be the same at the beginning of both fuel cycles.

Based on the above information, which ONE of the following describe the affect the new loading pattern will have on the unit?

A. The expected full power loop delta-T value should be significantly LOWER for this fuel cycle when compared to the value of full power loop delta-T for the previous fuel cycle.

B. The expected full power loop delta-T value should be significantly HIGHER for this fuel cycle when compared to the value of full power loop delta-T for the previous fuel cycle.

C✓ If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly BELOW actual power when the 1st calorimetric is performed after the refueling outage.

D. If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly ABOVE actual power when the 1st calorimetric is performed after the refueling outage.

Feedback

DISTRACTORS:

A Incorrect.

B Incorrect.

C Correct.

D Incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:
– Equipment Control; Knowledge of the effects of alterations on core configuration.

18. G2.3.6 001//T3/RAD CONTROL/C/A(2.1/3.1)/M/TP05301/S/MC

The following is a time-line of activities associated with Waste Monitoring:

- 1430: Waste Monitoring Tank (WMT) "A" is placed on mini-flow recirc for sampling
- 1620: Chemistry completes sampling of WMT "A." Result = 5.5×10^{-5} μCi/ml
- 1730: Chemistry submits a Radiological Liquid Waste Discharge Permit for WMT "A"
- 1735: The Shift Manager authorizes the release of the Radiological Liquid Waste Permit without the approval of the Radiochemist or Health Physics Supervisor
- 1745: Operators align WMT "A" for discharge and start the release
- 1746: R-18, Waste Disposal System Liquid Effluent Monitor, fails low. The release is terminated and WMT "A" is restored to a normal lineup
- 1930: The R-18 monitor is repaired and restored to service
- 1935: The Shift Manager re-authorizes the release of WMT "A" on the same Radiological Liquid Waste Permit
- 1940: Operators re-align WMT "A" for discharge and start the release

Based on the above information, which ONE of the following represents the problem associated with these actions?

A✓ The sample taken for the Radiological Liquid Waste Permit may not be representative of the contents of WMT "A" now being released.

B. A Radiological Liquid Waste Permit approved for one shift may NOT be used for initiation of a release on the next shift.

C. The discharge required the approval of the Health Physics Supervisor in addition to the Shift Manager.

D. The contents of WMT "A" must first be transferred to the Waste Holdup Tanks for further processing prior to release.

Feedback

REFERENCES:

1. SD 049/SYS.061A, pages 27-29, rev 02/01/02
2. OP-061.11, CONTROLLED LIQUIED RELEASE TO THE CIRCULATING WATER
3. NCOP-003, PRÉPARATION OF LIQUID RELEASE PERMIT

DISTRACTORS:

A Correct. IAW NCOP-003, PREPARATION OF LIQUID RELEASE PERMIT, Attachments 1 & 6.

B Incorrect. Not required.

C Incorrect. As long as the specific activity of the tank contents is less than or equal to 1×10^{-4} Ci/ml, only the SM's approval is required.

D Incorrect. Not required.

K/A CATALOGUE QUESTION DESCRIPTION:

– Radiation Control; Knowledge of the requirements for reviewing and approving release permits.

19. G2.4.34 001//T3/E PRO/PLAN/M(3.8/3.6)/N/TP05301/S/MC

Turkey Point has experienced a fire in the North/South Breezeway.

- The Fire Suppression System in the N-S Breezeway has activated
- Operators are in the process of carrying out the actions of ONOP-105, CONTROL ROOM EVACUATION

IAW ONOP-105, which ONE of the following correctly describes an action a particular operator is required to take given existing plant conditions?

A. From the Unit 3 480 Volt Load Center Room the Unit 3 RO will verify LC 3D Supply to LC 3H Breaker 30402 – CLOSED; and verify 3B Load Center Supply Breaker 30210 - CLOSED.

B. From the Unit 4 480 Volt Load Center Room the Unit 4 RO will OPEN LC 4D Supply to LC 4H Breaker 40402; and verify 4B Load Center Supply Breaker 40210 - CLOSED.

C. From the Unit 3 480 Volt Load Center Room the Third RO will verify LC 3D Supply to LC 3H Breaker 30402 – CLOSED; and verify 3B Load Center Supply Breaker 30210 - CLOSED.

D✓ From the Unit 4 480 Volt Load Center Room the Third RO will OPEN LC 4D Supply to LC 4H Breaker 40402; and verify 4B Load Center Supply Breaker 40210 - CLOSED.

Feedback

REFERENCES:

1. ONOP-105, CONTROL ROOM EVACUATION, pages 93,94, rev 10/15/04

DISTRACTORS:

A Incorrect. The Third RO would take this action, NOT the Unit RO and it would be taken only if the fire was NOT in the N-S Breezeway.

B Incorrect. This is the action taken by the Third RO, NOT the Unit RO.

C Incorrect. This is the action the Third RO would take if there was NO fire in the N-S Breezeway.

D Correct. IAW ref 1, this action is to be taken by the Third RO in the event of a fire in the N-S Breezeway.

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Procedures/Plan; Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.

20. G2.4.6 001//T3/E PRO/PLAN/M(3.1/4.0)/B/TP05301/S/MC

Given the following information:

- An event has occurred in the plant that has resulted in a radioactive release in the containment
- The Safety Parameter Display System (SPDS) indicates Critical Safety Function Status Tree display of YELLOW priority for Containment
- FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL, has been entered

Which ONE of the following indicates the mitigation strategy for operator actions directed by FR-Z.3?

A. Allow a controlled release through the containment filtration system prior to exceeding design pressure limits on the containment.

B✓ Verify containment ventilation isolation and attempt to reduce activity by containment filtration.

C. Reduce containment activity levels with dilution flow using the Containment Purge System.

D. Verify containment isolation Phase "A" and place all containment coolers in slow speed.

Feedback

REFERENCES:

1. WOG, FR-Z.3, page 2, rev 1C
2. EOP-F-O, CRITICAL SAFETY FUNCTION STATUS TREES, page 11, rev 04/15/99C

DISTRACTORS:

A Incorrect. See B.

B Correct. IAW WOG.

C Incorrect. See B.

D Incorrect. See B.

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Procedures/Plan; Knowledge of symptom based EOP mitigation strategies.

21. WE01EA2.2 001//T1G2/REDIAGNOSIS/M(3.3/3.9)/N/TP05301/S/MC

Following a plant event, operators entered the EOP network. The control room crew transitioned to, and completed, the FINAL step in ES-0.0, REDIAGNOSIS.

Which ONE of the following describes the plant conditions that resulted in reaching the end point in ES-0.0?

A.✓ A ruptured S/G had to have been identified and SI is required.

B. A ruptured S/G had to have been identified and SI is not required.

C. A faulted S/G had to have been identified and SI is required.

D. A faulted S/G had to have been identified and SI is not required.

Feedback

REFERENCES:
1. EOP-ES-0.0, REDIAGNOSIS, pages 3,5, rev 12/14/02
2. BD-EOP-ES-0.0, REDIAGNOSIS BASIS, pages 8,12, rev 12/14/02

DISTRACTORS:

A Correct. ES-0.0, REDIAGNOSIS, should only be used if SI is in service or in required. To reach step 4, the final step, in ES-0.0, a ruptured S/G had to have been identified in the previous step.

B Incorrect. SI should either be in service or is required.

C Incorrect. A ruptured S/G had to have been identified.

D Incorrect. A ruptured S/G had to have been identified and SI should either be in service or is required.

K/A CATALOGUE QUESTION DESCRIPTION:
- Rediagnosis; Ability to determine and interpret the following as they apply to the (Reactor Trip or Safety Injection Rediagnosis): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

22. WE03EA2.2 001//T1G2/LOCA C/D/C/A(3.5/4.1)/N/TP05301/S/MC

Following a LOCA with a concurrent Loss Of Offsite Power (LOOP), Unit 3 entered E-1, LOSS OF REACTOR OR SECONDARY COOLANT. Currently, the following plant conditions exist:

- T_{avg} is 345°F
- RCS pressure is 350 psig
- Containment temperature is 178°F
- RWST level = 255,000 gallons
- A mechanical failure of one train of SI has just occurred

Which ONE of the following describes the required operator actions in accordance with E-1?

A. Transition to ES-1.1, SI TERMINATION. Stop the running HHSI and RHR pumps and place in standby.

B✓ Transition to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. Stop the running RHR pump and place in standby.

C. Transition to ES-1.3, TRANSITION TO COLD LEG RECIRCULATION. Align RHR suction to containment recirc sump.

D. Transition to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION. Establish makeup to the Unit 3 RWST.

Feedback

REFERENCES:

1. E-1, LOSS OF REACTOR OR SECONDARY COOLANT, pages 18,21, rev 04/03/02
2. BD-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, page 35, rev 04/03/02
3. TECHNICAL SPECIFICATION, 3.5.3, page 3/4 5-9, Amendment Nos. 138 & 133
4. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, page 72, rev 05/01/03
5. ES-1.2, POST LOCA COOLDOWN & DEPRESSURIZATION, pgs 3,6,12, rev 04/03/02
6. BD ES-1.2, POST LOCA COOLDOWN AND DEPRESS, page 8, rev 04/03/02
7. TECHNICAL SPECIFICATION, 3.5.4, page 3/4 5-10, Amendment Nos. 138 & 133

DISTRACTORS:

A Incorrect. The transition is not correct. The stated action would be correct if the transition were made to ES-1.1.

B Correct. IAW E-1, Step 19, if RCS pressure is > 250 psig, go to ES-1.2, step 1 which directs stopping RHR pumps.

C Incorrect. While this is a valid transition from E-1, it occurs when RWST drops below 155,000 gallons so this transition would be incorrect.

D Incorrect. While this is a valid transition from E-1, it occurs if neither RHR pump is available to support cold leg recirculation so this transition would be incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:

- LOCA Cooldown and Depressurization; Ability to determine and interpret the following as they apply to the (LOCA Cooldown and Depressurization): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

23. WE05EA2.2 001//T1G1/HEAT SINK/C/A(3.7/4.3)/B/TP05301/S/MC

Operators are performing EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, and have successfully initiated Bleed and Feed. The BOP subsequently announces secondary heat sink is restored using "A" Standby SG Feed Pump.

Which ONE of the following describes the correct operator response?

A. Return to procedure and step in effect when feed flow is verified to be > 345 gpm.

B✓ Continue performing FR-H.1 to completion.

C. Return to procedure and step in effect when narrow range level in any S/G is > 6%[32%].

D. Return to procedure and step in effect only when narrow range levels in all S/Gs are > 6%[32%].

Feedback

REFERENCES:

1. FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, steps 7,27, rev 04/30/02

DISTRACTORS:

A Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because 345 gpm is a normal indicator of adequate heat sink.

B Correct. After Bleed and Feed is established, FR-H.1 must be completed to ensure SI reduction/termination and PORV closure are completed. Restoration of secondary heat sink is not enough to transition from FR-H. beyond Step 12 of the procedure.

C Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because 6% in any S/G is a normal indicator of adequate heat sink.

D Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because >6% in all S/Gs is a goal of the procedure.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of Secondary Heat Sink; Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

24. WE08EG2.4.4 001//T1G2/PTS/M(4.0/4.3)/B/TP05301/S/MC

Which ONE of the following conditions would require entering FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, on an orange or red path?

A. Cooldown of cold leg GREATER THAN 100°F in 60 minutes, cold leg = 325°F, RCS pressure = 450 psig

B✓ Cooldown of cold leg GREATER THAN 100°F in 60 minutes, cold leg = 280°F, RCS pressure = 460 psig

C. Cooldown of cold leg LESS THAN 100°F in 60 minutes, Tavg = 280°F, RCS pressure = 460 psig

D. Cooldown of cold leg LESS THAN 100°F in 60 minutes, Tavg = 270°F, RCS pressure = 450 psig

Feedback

REFERENCES:

1. EOP-F-0, CRITICAL SAFETY SYSTEM STATUS TREES, page 10, rev 08/03/01

DISTRACTORS:

A Incorrect. Cold leg temperature is NOT less than 320°F, but is less than 350°F, returning a Yellow path. Pressure is of no consequence.

B Correct. Cold leg temperature is less than 290°F, returning a red path.

C Incorrect. Tavg is NOT less than 275°F, returning a green path.

D Incorrect. Tavg is NOT less than 275°F and pressure is NOT greater than 460 psig.

K/A CATALOGUE QUESTION DESCRIPTION:

- Pressurized Thermal Shock; Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

25. WE16EA2.1 001//T1G2/HIGH RAD/M(2.9/3.3)/B/TP05301/S/MC

Which ONE of the following describes an entry criteria for FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL?

A. Greater than 2.6 E 3 on R-11/12.

B. Greater than 6.1 E 5 on R-11/12.

C✓ Greater than 1.3 E 4 on CHRRMS.

D. Greater than 1.3 E 3 on CHRRMS.

Feedback

REFERENCES:
1. FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL, page 3, rev 04/15/99
2. EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES, page 11, rev 04/15/99C
3. ARP-097.CR, CONTROL ROOM ANNUNCIATOR RESPONSE, page 424, rev 07/23/02

DISTRACTORS:

A	Incorrect.	FR-Z.3 is not entered based on this reading.
B	Incorrect.	FR-Z.3 is not entered based on this reading.
C	Correct.	This is the value that will initiate entry into FR-Z.3.
D	Incorrect.	FR-Z.3 is not entered based on this reading.

K/A CATALOGUE QUESTION DESCRIPTION:
- High Containment Radiation; Ability to determine and interpret the following as they apply to the (High Containment Radiation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

1. Need to add more to the stem. Where is power reduced too? Why was power reduced to 75%? How did we get here? c

2. Change stem:

The crew on Unit 4 entered FR-C.2, RESPONSE TO......, after an ORANGE condition was identified. The crew closes all Accumulator Discharge MOVs as directed by procedure........... (Was this a result of the orange condition being identified or did the crew enter FR-C.2 as a result of?) c

3. Change stem:

Unit 4 is in Mode 3 with all Shutdown Rods withdrawn and MCC4B is tagged out for maintenance when power is subsequently lost to 4D01. c

For the current plant conditions, which ONE of the following describes the impact on the plant and the actions necessary to mitigate the impact of these conditions?

(What position is the rod motion selector switch in for this mode?) c AUTO

Look at wording of distractor "C" c

4. Stem:

Unit 3 has been operating at 100% power for three days and the following conditions exist: c

Based on the above conditions, the SRO should direct the...... c

Are we going to give a procedure with this. Is this something we expect them to do without reference? None of the ARP direct or indicate a need to trip. The both direct you to ONOPs.

We can give procedure if necessary.

please re-check stem.

5. Stem:

Unit 4 is operating at 100% power, on February 25, 2005, at 0930, when......... At 1045 the crew verifies that all procedural and Technical Specifications (TS) requirements associated with the failure of PT-455 has been completed. At 11:45 the crew determines that pressurizer pressure instrument PT-456 fails to 2300 psig. c

Based on the above plant conditions, which ONE of the following.....

10. Rewrite the stem: Please recheck

Only B is correct

11. Look at distractors

Based on the information given there may be two correct answers. B & D. Could the DG not be secured using applicable procedure? Not without violating Precaution/Limit

?

12. Change stem

Look at answer

Unit 3 is operating at 100% power when the following annunciators actuate and plant conditions are observed by the crew: OK as is

....

....

Which ONE of the following describes the crews response based on the above conditions?

Why is B not correct? Define the difference between the two method of load reductions and its affect on the generator.

14. Stem and:

Unit 3 is in Mode 3 and will performing maintenance on .... Following the completion of the work the unit will be returned to full power operation. Needed to keep part of this as is

Which ONE of the following identifies the plant personnel that are required to be notified prior to entering Mode 2 and again prior to entering Mode 1? C

17. Stem

Reactor engineering had designed a core loading pattern that will be performed during the next refueling outage. The CHANGE will result in placing the "twice-burned" fuel assemblies more toward the periphery and the new fuel assemblies more toward the center of the core. Based on engineering calculations, it has been determined that K excess will be the at the beginning of both fuel cycles. C

Based on the above information, which ONE of the following describe the affect the new loading pattern will have on the unit? C

18. Stem:

Rewrite the stem. ? confusing. Please re-check.

22. Stem C

The crew entered E-1, LOSS......., following a LOCA and loss of offsite power that occurred on Unit 3. Currently the following plant conditions exist:

....

....

C

1. 003AG2.2.25 1

Unit 4 had been operating at 100% power when a Bank "D" rod dropped. The crew entered ONOP-~~0~~028.3, DROPPED RCCA. The rod was declared inoperable and power was reduced to < 50%.

Which ONE of the following describes the basis for the Technical Specifications LCO requiring the crew to reduce reactor power to 75% within one hour?

A. Ensures minimum DNBR in the core remains less ~~greater~~ than or equal to the applicable design limit for continued operation and in short-term transients.

B. Provides adequate protection against $F_Q(Z)$, Heat Flux Hot Channel Factor, in the event of a subsequent Loss of All AC Power (LOAAC) event. ~~loss-of-coolant accident~~.

C. Provides assurance that the effects of residual xenon redistribution impact from past operation near EOL is minimal.

D.✓Ensures that design margins to core limits will be maintained under both steady-state and anticipated transient conditions.

REFERENCES:

1. Technical Specifications, 3.1.3.1, pages 3/4 1-17 & 1-18, Amendment Nos. 186 & 216
2. ADM-536, TECHNICAL SPECIFICATION BASES CONTROL PROGRAM, pages 32-36,41, rev 05/01/03

DISTRACTORS:

A Incorrect. The goal is to maintain DNBR greater than applicable design limits. ~~This is the basis for Tech Spec 3.2, POWER DISTRIBUTION LIMITS~~

B Incorrect. The LOAAC accident is not an anticipated transient condition as describe in Chapter 14 of the FSAR. ~~This is part of the basis for POWER DISTRIBUTION LIMITS but has nothing to do with a LOCA.~~

C Incorrect. This is the basis for Tech Spec 3.2.1, AXIAL FLUX DIFFERENCE, but had nothing to do with EOL.

D Correct. IAW ref 2, page 33, paragraph 3.

K/A CATALOGUE QUESTION DESCRIPTION:

- Dropped Control Rod; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Basis for Change: "D" is correct as written. "A" and "B" are also correct because they fall under the "design margins to core limits" umbrella stated in "D". DNBR is a design margin which has a core limit that is applicable during continued operation and in short term transients. F(Q)Z is also a design margin which has a core limit and LOCA is an "anticipated transient condition."

Answer: D

C

2. 006G2.1.28 1

Unit 4 entered FR-C.2, RESPONSE TO DEGRADED CORE COOLING, after an ORANGE condition ~~was identified. IAW FR-C.2~~ all Accumulator Discharge MOVs were closed when HHSI or RHR flow is established. why, only opened once in FR-C.2

Which ONE of the following describes why ~~power is not subsequently removed from~~ the Accumulator Discharge MOVs ~~once they~~ are shut?

A. To minimize subsequent RCS cool down and vessel thermal shock. ~~Check valves are installed to prevent back leakage from the RCS into the accumulators.~~

B. ✓ To minimize subsequent nitrogen introduction into the RCS. ~~Leaving the MOVs energized ensures the operator does not delay actions to restore core cooling by de-energizing the MOVs.~~

C. To prevent Accumulator injection flow from hindering HHSI or RHR cooling flow. ~~The circuit breakers supplying the MOVs are normally open and "racked out" and are returned to this condition during the course of completing FR-C.2.~~

D. To prevent the loss of Accumulator water which will be needed if conditions degrade to a RED condition. ~~The Accumulators are designed to prevent the introduction of nitrogen into the RCS upon equalizing with RCS pressure.~~

REFERENCES:
1. BD-EOP-FR-C.2, RESPONSE TO DEGRADED CORE COOLING, page 35, rev 12/14/02
2. EOP-FR-C.2, RESPONSE TO DEGRADED CORE COOLING, page 12, rev 12/14/02
3. SD 021/SYS.050,062,064, page 19, rev 04/22/04

DISTRACTORS:

A Incorrect. Thermal shock is a lower concern than degraded core cooling. Actions are taken in FR-C.2 to inject accumulators. ~~See B.~~

B Correct. IAW Basis.

C Incorrect. Accumulator injection will supplement and not hinder HHSI or RHR flow to the core. ~~See B.~~

D Incorrect. Accumulator injection is indeed called upon in the RED condition (FR-C.1) but it is also called for in this ORANGE condition to preclude a RED condition. ~~See B.~~

K/A CATALOGUE QUESTION DESCRIPTION:
- Emergency Core Cooling System (ECCS); Knowledge of the purpose and function of major system components and controls.

Basis for change: As asked the question does not reflect the stated KA (Knowledge and function of major system components) when it asks the basis for actions <u>not</u> taken. The question has been refocused on the primary reason for performing the referenced step. The revised question supports the KA by asking the basis for actions directed to be performed in the EOP.

Answer: B

3. 014A2.072 1 ✓

l✓

l✓
Don't

Unit 4 is at 100% power with all control rods in the fully withdrawn position when power is lost to the RPI inverter.

Which ONE of the following describes the effect on the RPI system and correct operator response to this event?

~~Unit 4 is in Mode 3 with all Shutdown Rods withdrawn, Rod Motion Selector Switch in AUTO, and MCC 4B tagged out for maintenance. Power is subsequently lost to 4D01.~~

~~For the current plant conditions, which ONE of the following describes the impact on the plant and the actions necessary to mitigate the impact of these conditions?~~

A. RPI indication has been lost. Reduce thermal power to less than 75% within 8 hours. ~~Power has been lost to Demand Position Indication. Open the reactor trip breakers IAW Tech Spec 3.1.3.3, REACTIVITY CONTROL SYSTEMS.~~

B. RPI indication has been lost. Restore the failed inverter to operable status within 1 hour or place Unit 4 in Hot Standby within the following 6 hours.
~~Power has been lost to Demand Position Indication. Commence a boration of the RCS to ensure adequate shutdown margin IAW ONOP-028.1, RCC MISALLIGNMENT.~~

C.✓ RPI power has auto swapped to the CVT. Initiate PWOs to repair the failed Inverter and calibrate the RPI system to compensate for changes in power supply voltage.
~~Power has been lost to Digital Rod Position Indication. Place the Rod Motion Selector Switch to the MAN position and notify the I&C Supervisor IAW ONOP-028.2, RCC POSITION INDICATION MALFUNCTION.~~

D. RPI power has auto swapped to the CVT. Restore the failed inverter to operable status within 1 hour or place Unit 4 in Hot Standby within the following 6 hours.
~~Power has been lost to Digital Rod Position Indication. Place the Rod Selector Switch to the MAN position and verify that all Shutdown Bank rods are fully withdrawn by performing an incore flux trace IAW ONOP-028.2, RCC POSITION INDICATION MALFUNCTION.~~

REFERENCES:

1. ONOP-028.2, RCC POSITION INDICATION MALFUNCTION, pages 5,6, rev 04/12/02
2. SD-006/SYS.028B, ROD POSITION INDICATION SYSTEM, pages 9,10, fig 5, rev 04/13/04
3. ONOP-028.1, RCC MISALIGNMENT, pages 6,7, rev 04/12/02
4. Tech Spec 3.1.3.3, REACTIVITY CONTROL SYSTEMS, page 1-23, Amendment Nos. 149 & 144

DISTRACTORS:

A Incorrect. Power has not been lost to the RPI System. ~~Demand Position Indication~~.

B Incorrect. Power has not been lost to the RPI System. ~~Demand Position Indication and the purpose of boration is to maintain Tavg/Tref within 3 degrees.~~

C Correct. IAW ONOP-028.2.

D Incorrect. The shutdown statement would be correct if all RPI indication had been lost. Since it has not been lost, the shutdown requirement does not apply. ~~This is the response for a loss of Demand Position Indication.~~

K/A CATALOGUE QUESTION DESCRIPTION:

- Rod Position Indication System (RPIS); Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of power to the RPIs. ~~Loss of LVDT~~.

Basis for changes: KA does not apply to Turkey Point which utilizes an analog RPI system, not reed switches associated with digital RPI. Additionally the original question was flawed because the unit would never be in Mode 3 with control rods in AUTO. Also a loss of 4D01 would result in an immediate automatic reactor trip making the original responses incorrect. Finally the question requires knowledge of power supplies for components at the MCC level which is not part of required operator knowledge.

Answer: C

C

4. 015AG2.4.49 1

Unit 3 has been operating at 100% power for three days when the following conditions occur:

1. Annunciator A-1/5, RCP SEAL LEAK-OFF HI FLOW, has lit
2. Annunciator G-2/2, RCP B STANDPIPE HI LEVEL, has lit
3. RCP B seal injection flow is 7.8 gpm
4. RCP B seal leak-off flow is 6 gpm
5. Seal return temperature is 150°F and rising steadily

Based on the above indications, the crew enters ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL. IAW ONOP-041.1, the SRO should direct the operating crew to perform which ONE of the following?

A. RCP B operation may continue for up to 24 hours due to number two seal sticking.

B.✓ Manually trip the reactor, then stop RCP B and close its seal leakoff valve after the pump stops.

C. Commence unit shutdown using ONOP-100, FAST LOAD REDUCTION, when the turbine is tripped, then trip the reactor, when the reactor is tripped, then stop RCP B.

D. Begin preparations to shutdown and stop RCP B using GOP-103, POWER OPERATION TO HOT STANDBY and contact plant management for further guidance.

REFERENCES:
1. ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL, pages 7,12,13,23,30, rev 06/14/99C1
2. ARP-097CR, CONTROL ROOM ANNUNCIATOR RESPONSE, pages 25,352, rev 06/09/03

DISTRACTORS:

A Incorrect. This would be true on initial startup (or for 8 hours during normal operation).

B Correct. IAW ref 1.

C Incorrect. This would be true if seal flow was less than 6 gpm but greater than 5.5 gpm.

D Incorrect. This would be true if seal flow was less than 5.5 gpm.

K/A CATALOGUE QUESTION DESCRIPTION:
- Reactor Coolant Pump Malfunction; Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

5. 027AG2.2.25 1

Unit 4 is operating at 100% power on February 25, 2005, at 0930, when pressurizer pressure instrument PT-455 fails low. At 1045 the crew verifies that all procedural and Technical Specifications (TS) requirements associated with the failure of PT-455 have been completed. At 1145 the crew determines that pressurizer pressure instrument PT-456 fails to 2300 psig.

Based on the above plant conditions, which ONE of the following describes the actions(s) that must be performed to satisfy Technical Specifications and why?

A. Be in at least Mode 3 within 6 hours; permits the time limits of the ACTION requirements to be reset to the point in time where the plant entered the new Mode to allow completion of remedial measures and a return to POWER.

B. ✓ Be in at least Mode 3 within 7 hours; permits shutdown in a controlled and orderly manner.

C. Be in at least Mode 4 within 12 hours; permits shutdown in a controlled and orderly manner.

D. Be in at least Mode 4 within 13 hours; permits the time limits of the ACTION requirements to be reset to the point in time where the plant entered the new Mode to allow completion of remedial measures and a return to POWER.

REFERENCES:

1. Tech Specs 3/4.3.1, REACTOR TRIP SYSTEM INSTRUMENTATION, Table 3.3-1, Amendment NOS. 140 & 135
2. Tech Specs 3/4.0, APPLICABILITY, 3.0.3, Amendment NOS. 137 & 132
3. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, pages 22,23, rev 05/01/03

DISTRACTORS:

A Incorrect. Wrong time requirement and reason.

B Correct. IAW 3.0.3, within 1 hour action shall be initiated to place the unit in: (1) at least HOT STANDBY within the next 6 hours (for a total of 7), (2) at least HOT SHUTDOWN within the following 6 hours (for a total of 13), (3) at least COLD SHUTDOWN within the subsequent 24 hours (for a total of 37)..

C Incorrect. Wrong time requirement.

D Incorrect. Wrong reason.

K/A CATALOGUE QUESTION DESCRIPTION:

- Pressurizer Pressure Control System Malfunction; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Answer: B

C

6. 033GG2.4.30 1

Unit 4 is in Mode 6. On January 29th at 0900 the water level in the Spent Fuel Pool was lowered to 56' to perform maintenance. All movement of fuel assemblies and crane operation in the fuel storage area had been suspended at 0700 the same day. Maintenance was scheduled for completion with level restored by February 5th at 0900.

- Maintenance was completed and level restored on February 6th at 1000

Which ONE of the following correctly completes the statement:

This event is ___(1)___ to the NRC because ___(2)___.

A. Reportable; SFP level remained below the minimum for more than 24 hours beyond originally scheduled.

B.✓ Reportable; SFP level remained below the minimum for more than 7 days.

C. Not Reportable; SFP level was not lowered below the minimum level required by Tech Specs.

D. Not Reportable; the amount of time the SFP level remained below the minimum did not exceed 7 days. ~~the maximum time allowed by Tech Specs.~~

REFERENCES:

1. Tech Specs, 3.9.11, REFUELING OPERATIONS – WATER LEVEL – STORAGE POOL, page 9-12, Amendment Nos. 224 & 219

DISTRACTORS:

A Incorrect. The time exceeded 7 days.

B Correct. SFP level shall be maintained greater than or equal to elevation 56' – 10". If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

C Incorrect. Level was lowered below the minimum required by Tech Specs.

D Incorrect. The amount of time the SFP level remained below the minimum exceeded the maximum time allowed by Tech Specs.

K/A CATALOGUE QUESTION DESCRIPTION:

- Spent Fuel Pool Cooling System (SFPCS); Knowledge of which events related to system operations/status should be reported to outside agencies.

Basis for Change: Operators are not required to know TS Action Times that exceed 1 hour (ref. KA G.2.1.11)

C

Done

7. 039GG2.1.32 1

TS

↑

D

Unit 4 is operating at 100% power when "A" MSIV is found to be inoperable in the OPEN position at 1400, 02-21-05.

Which ONE of the following describes the actions to be taken and why?

B. ~~A.~~ Be in MODE 3 by 0200, 02-22-05. Minimizes the negative ~~positive~~ reactivity effects of the Reactor Coolant System cooldown associated with the blowdown in the event of a steam line rupture.

C. ~~B.~~ Be in MODE 3 by 1400, 02-22-05. Minimizes the negative ~~positive~~ reactivity effects of the Reactor Coolant System cooldown associated with the blowdown in the event of a steam line rupture.

D ~~C.~~ Be in MODE 3 by 2000, 02-22-05. Limits the pressure rise within containment in the event a steam line rupture occurs within containment.

A ~~D.~~ Be in MODE 3 by 2000, 02-21-05. Limits the pressure rise within containment in the event a steam line rupture occurs within containment.

REFERENCES:

1. Tech Specs, 3.7.1.5, MAIN STEAM ISOLATION VALVES, page 7-10, Amendment Nos. 137 & 132
2. 0-ADM-536, TECHNICAL SPECIFICATION BASES CONTROL PROGRAM, ~~page 85~~, rev ~~05/01/03~~

REFERENCE PROVIDED : TS 3.7.1.5

DISTRA~~CTORS:~~

B Incorrect. Action is incorrect. Basis is incorrect.

C Incorrect. Action is incorrect. Basis is incorrect.

D Correct. "With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. The OPERABILITY of the MSIVs ensures that no more than one steam generator will blow down in the event of a steam line rupture." "This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment."

A Incorrect. Action is incorrect. Basis is correct.

K/A CATALOGUE QUESTION DESCRIPTION:

- Main and Reheat Steam; Ability to explain and apply all system limits and precautions.

Basis for Changes: The second half of all four responses is correct and supported by the Tech. Spec. basis and therefore does not offer a basis for discrimination. The TS should be provided because the question is asking for memorization of a TS Action Statement that exceeds one hour (in this case 30 hour AS). Responses should be reordered to chronological order as shown.

Answer: D ~~C~~

8. 055EA2.03 1

A loss of all AC power has occurred. The STA reports the status of the CSFs are as follows:

- Subcriticality – RED
- Core Cooling – RED
- Heat Sink – RED
- Integrity – GREEN
- Containment – GREEN
- Inventory - YELLOW

Which ONE of the following procedures should be used to mitigate these conditions?

A. EOP-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS

B.✓ EOP-ECA-0.0, LOSS OF ALL AC POWER

C. EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK

D. EOP-FR-C.1, RESPONSE TO INADEQUATE CORE COOLING

REFERENCES:

1. EOP-ECA-0.0, LOSS OF ALL AC POWER, page 6, rev 02/22/02

DISTRACTORS:

A Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

B Correct. Note that precedes Step 1 of EOP-ECA-0.0 states that "CSF Status Trees are required to be monitored for information only. FRPs shall NOT be implemented."

C Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

D Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of Offsite and Onsite Power (Station Blackout); Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.

Answer: B

C

9. 056AA2.51 1

Unit 3 was operating at 100% power when a loss of offsite power occurred.

- ~~Subsequently, a loss of CCW occurs and all RCPs are tripped~~

Which ONE of the following describes the response of the reactor CORE delta T from the time the LOOP occurred ~~RCPs are tripped~~ until one hour later ~~later~~ in the event?

Delta T ________ as natural circulation is being established, then ________.

A. Lowers; remains constant as heat removal is established with the atmospheric steam dumps.

B. Lowers; rises as decay heat load diminishes and heat removal is controlled by the atmospheric steam dumps.

D ~~C~~. Rises; lowers as decay heat load diminishes and heat removal is controlled by the atmospheric steam dumps.

C ~~D~~. Rises; remains constant as heat removal is established with the atmospheric steam dumps.

REFERENCES:
1. FSAR

DISTRACTORS:

A Incorrect. Delta T has to become higher to establish a driving head for natural circulation.

B Incorrect. Delta T has to become higher to establish a driving head for natural circulation. Does not take decay heat into account. Distractor provides opposite of actual effect.

C ~~D~~ Incorrect. Does not take decay heat into account.

D ~~C~~ Correct.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Offsite Power; Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Delta T, (core, heat exchanger, etc.).

Basis for Changes: RCPs will trip immediately as a direct result of the LOOP. Deleted second "later". "C" Response is correct. Delta T decreases as decay heat decreases. Answer key appears to be incorrect.

Answer: C ~~D~~

10. 058AG2.2.25 1

Unit 3 is operating at 100% power and Unit 4 is in COLD SHUTDOWN. For the past two hours, Unit 3 operators have been unable to raise float charge voltage above 124 volts on battery bank 3A. Checks of individual cell voltages have confirmed this value.

Based on the above plant conditions, which of the following describes the Technical Specification (TS) requirements and the reason for the requirements?

A. Have Unit 3 in at least HOT STANDBY within 34 hours. This time allows the orderly shutdown of one unit at a time so as not to jeopardize the stability of the electrical grid by imposing a dual unit shutdown.

B. Have Unit 3 in at least HOT STANDBY within 12 hours. This time allows restoring the battery to within limits and not jeopardizing the stability of the electrical grid by imposing a dual unit shutdown.

C. Have Unit 3 in at least HOT STANDBY within 34 hours. This time allows the orderly shutdown of one unit at a time in order to avoid dual unit natural circulation cooldown in the event of a loss of both startup transformers.

D. Have Unit 3 in at least HOT STANDBY within 12 hours. This time allows restoring the battery to within limits and avoiding dual unit natural circulation cooldown in the event of a loss of both startup transformers.

REFERENCES:
1. Tech Specs, 3/4.8.2, DC SOURCES, pages 8-13 – 8-16, Amendment Nos. 138 & 133
2. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, pages 92-95, rev 05/01/03

REFERENCE PROVIDED : TS 3.8.2.1 & 3.8.2.2

DISTRACTORS:

A Correct. IAW note (3) of Table 4.8-2, the "allowable value" for float voltage for each connected cell must be = 2.07 or the battery is considered INOPERABLE. IAW ACTION step "b," operators have 2 hours* *(which can be extended to 24 hours with Unit 4 in Mode 5) to correct the problem or have both Units in Mode 3 within the next 12 hours.

B Incorrect. 34 hours would be correct.

C Incorrect. The time is correct but the reason is incorrect.

D Incorrect. The time and reason are incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of DC Power; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

C Don TS ↑ C ?

Basis for Change: The TS should be provided because the question is asking for memorization of a TS Action Statement that exceeds one hour (in this case 34 hour AS, further complicated by the answer source is a footnote based on an asterisk).

Answer: A

C

Done

11. 064A2.06 1

You are the SRO on Unit 4. Following recovery from a loss of offsite power, the 4A 4KV Bus has been transferred to the SU Transformer. 4A EDG is running unloaded at 900 RPM.

Which ONE of the following describes a procedural requirement regarding subsequent operation of the 4A EDG and consequences of failing to adhere to that requirement?

~~Diesel Generator continued to run unloaded for 4.5 hours at 900 rpm. The BOP discovered this and placed a 500KW load on the diesel for 30 minutes with the EDG Air Box drain fully open and has requested to secure the EDG.~~

~~Considering the conditions given above, which ONE of the following describes your response and why?~~

A. Perform a Normal Stop of the EDG within 4.5 hours. Running the EDG unloaded beyond 4.5 hours can result in overheating the EDG due to a lack of air flow across the radiator cooling fins.

~~Have him run the diesel an additional hour with a minimum of 2500KW load with the air box drain shut to remove the fuel that accumulated in the crankcase to prevent the possibility of a crankcase explosion if the diesel loads suddenly on its next start.~~

B. Perform a Normal Stop of the EDG within 4.5 hours. Running the EDG unloaded beyond 4.5 hours can result in an accumulation of oil in the exhaust which can lead to a fire.

~~Have him run the diesel an additional half hour with a minimum of 1250KW load with the air box drain shut to clean out the oil that accumulated in the exhaust stack to prevent the possibility of an exhaust fire if the diesel loads suddenly on its next start.~~

C. Perform an Emergency Stop of the EDG within 4.5 hours. Running the EDG unloaded beyond 4.5 hours can result in overheating the EDG due to a lack of air flow across the radiator cooling fins.

~~He should not have run the diesel partially loaded. Have him run the diesel an additional half hour with no load and with the air box drain open 25 percent to effectively drain the air box and to ensure the diesel is sufficiently cooled down prior to securing.~~

D. Perform an Emergency Stop of the EDG within 4.5 hours. Running the EDG unloaded beyond 4.5 hours can result in an accumulation of oil in the exhaust which can lead to a fire.

~~His actions were correct. The diesel may be secured with no adverse impact.~~

REFERENCES:

1. 4-OP-023, EMERGENCY DIESEL GENERATOR, Step 4.3 Rev. 12/19/03 ~~pages 14,50,120,126, rev 08/29/02~~
2. 4-ONOP-004.1, Step 9 rev. 02/26/99 ~~ONOP-004.3, LOSS OF 3B 4KV BUS, page 8, rev 10/16/01~~
3. ~~EOP-ECA-0.0, LOSS OF ALL AC POWER, page 26, rev 02/22/02~~

DISTRACTORS:

A Incorrect. Normal Stop is the correct method of EDG shutdown. Reason stated is a potential low speed operation concern on Unit 3 but not on Unit 4 which has electric auxiliary cooling fans. ~~See B. Although fuel dilution into lube oil can occur, it will not occur as a result of the aforementioned operation.~~

B Correct per ONOP-004.1 and 4-OP-023. ~~After 4.5 cumulative hours of operation at synchronous speed (900 rpm) at loads between 0 and 20% (0-500KW), the engine shall be run at a minimum of 50% load (1250KW) for at least 30 minutes to clean out the oil residual that accumulates in the exhaust stack as an exhaust fire could result when the engine is suddenly loaded, raising exhaust temperatures quickly.~~

C Incorrect. Normal Stop is the correct method of EDG shutdown. Reason stated is a potential low speed operation concern on Unit 3 but not on Unit 4 which has electric auxiliary cooling fans. ~~See B. When running for extended periods at idle speed, the air box drain should be open 25% to effectively drain the air box.~~

D Incorrect. Normal Stop is the correct method of EDG shutdown. ~~The Diesel should have been run at a minumum of 50% load (1250KW) vice 500KW. The Diesel must be run an additional 30 minutes at a minimum of 50% load to meet the requirements of Precaution 4.4 of ref 1.~~

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Diesel Generator (EDG) System; Ability to (a) predict the impacts of the following malfunctions or operations on the EDG system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Operating unloaded, lightly loaded, and highly loaded time limit.

Basis for Changes: Incorrect references used to support the question's correct answer. ONOP-004.3 and ECA-0.0 do not apply for the conditions stated. 4-OP-023 is the correct reference but 3-OP-023 was used for this Unit 4 question. Operators would have implemented 4-ONOP-004.1 and at step 9 used 4-OP-023 to perform a normal shutdown of the EDG.

Answer: B

C

Done

12. 065AA2.06 1

Units 3 and 4 are both operating at 100% power when the following occurs on both units ~~Unis~~:

- The Lag CM starts followed shortly by startup of both CD air compressors.
- Annunciator I-6/1, INSTR AIR HI TEMP/LO PRESS actuates
- Annunciator G-1/2, CHARGING PUMP HIGH SPEED, actuates
- Annunciators C-1/1 – 1/3, SG A,B,C, LO/LO-LO LEVEL ALARMS, actuate
- SG levels are 25% and decreasing
- Unit 3 instrument air pressure is 56 ~~60~~ psig
- Unit 4 instrument air pressure is 64 ~~65~~ psig

No change

Which ONE of the following describes the correct operator response?

A. Trip Unit 3 IAW ONOP-013, LOSS OF INSTRUMENT AIR, and perform a Fast Load Reduction on Unit 4 IAW ONOP-100, FAST LOAD REDUCTION.

B. Trip both Units and enter EOP-E-0, REACTOR TRIP OR SAFETY INJECTION on both Units.

C. Perform a Fast Load Reduction on both Units IAW ONOP-100, FAST LOAD REDUCTION and establish AFW flow.

D. Allow both Units to trip and enter EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

REFERENCES:
1. ONOP-013, LOSS OF INSTRUMENT AIR, page 6 & Foldout, rev 12/23/02

DISTRACTORS:

A Incorrect Both Units are below the trip criterion of 65 psig ~~Correct. IAW ref 1.~~

B Correct Both Units are below the trip criterion of 65 psig ~~Incorrect. IA to trip the Unit when its instrument air pressure drops below 65 psig.~~

C Incorrect. IA to trip the Unit when its instrument air pressure drops to ~~below~~ 65 psig.

D Incorrect. IA to trip the Unit when its instrument air pressure drops to ~~below~~ 65 psig.

does not say "to", says "L"

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Instrument Air; Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing.

Basis for changes: Per 0-ONOP-013, FO Page and Step 13, the applicable unit must be tripped if it's IA pressure cannot be maintained above 65 psig. Note that original distractor analysis indicated that "A" was correct but answer was given as "B".

Answer: B

C

13. 076GG2.4.49 1

Unit 3 is operating at 100% power when the following Annunciators actuate:

- ~~I-4/2, ICWP A/B/C TRIP~~
- I 4/1, ICWP A/B/C MOTOR OVERLOAD
- H-8/5, CCW HX OUTLET HI TEMP
- I-5/4, TPCW HI TEMP/LO PRESS
- E-9/4, GEN EXCITER AIR HI TEMP
- TE-3414 (det #31) cold air (fan discharge) as read on R-347 Pts 5 and 6 is 60°C
- TE-3416 (det #33) hot air (exciter armature outlet) as read on R-347 Pts 7 and 8 is 87°C
- Reactive generator load = 150 ~~> 0~~ MVAR in the Lag.

Which ONE of the following describes the correct ~~immediate~~ operator responses?

A. Start standby ICW pump then stop affected ICW pump, reduce reactive load on the generator.

B. Start standby ICW pump then stop affected ICW pump, initiate FAST LOAD REDUCTION.

C. Stop affected ICW pump then start standby ICW pump, initiate FAST LOAD REDUCTION.

D. Stop affected ICW pump then start standby ICW pump, reduce reactive load on the generator.

REFERENCES:

1. ONOP-019, INTAKE COOLING WATER MALFUNCTION, page 5, page 12 rev 10/24/02C ~~01/09/01C~~
2. ~~ARP-097.CR, CONTROL ROOM ANNUNCIATOR RESPONSE, pages 288,445,472,480~~
3. ~~ONOP-030, COMPONENT COOLING WATER MALFUNCTION, pages 6,7, rev 05/24/02~~

DISTRACTORS:

A Correct. ~~Immediate~~ actions IAW ref 1 ~~and 2~~.

B Incorrect. Reduce reactive load vice FAST LOAD REDUCTION.

C Incorrect. Start standby ICW pump THEN stop affected ICW pump. Reduce reactive load vice FAST LOAD REDUCTION.

D Incorrect. Start standby ICW pump THEN stop affected ICW pump.

K/A CATALOGUE QUESTION DESCRIPTION:

- Service Water System (SWS); Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Basis for Change: Changed annunciators. I 4/2 implies the ICW pump has already tripped negating the need to manually stop it. Question enhancement to provide typical value of MVAR load. Line through "immediate" because second half of answer is not an immediate step. Revised references.

Answer: A

C

14. G2.1.14 1

Unit 3 has been in Mode 3 for three days to facilitate the performance of maintenance. Prior to this, Unit 3 had been operating at 100% power for an extended period of time. Following completion of the work the unit will be returned to full power operation.

Which ONE of the following identifies two ~~the~~ plant personnel that are required to be notified to review the requirements of 0-ADM-529, "Unit Restart Readiness" prior to entering Mode 2 and again prior to entering Mode 1?

~~MODE 2~~	~~MODE 1~~
A. Site Vice President	Plant General Manager
B. Reactor Engineering	Work Control Center Supervisor
C. Chemistry Department	Security Department
D. ~~Nuclear Plant Supervisor~~ Health Physics	~~Operations~~ Shift Manager

REFERENCES:

1. GOP-301, HOT STANDBY TO POWER OPERATION, pages 16,19,62, rev 04/22/04

DISTRACTORS:

A Correct. IAW GOP-301, sections 3.1.19 & 3.2.17 the following personnel are to be notified prior to entry into Modes 1 & 2: Site VP, Plant GM, and Operations Shift Manager.

B Incorrect. See A.

C Incorrect. See A.

D Incorrect. See A.

K/A CATALOGUE QUESTION DESCRIPTION:

- Knowledge of system status criteria which require the notification of plant personnel.
-

Basis for Change: Removed Mode1/Mode 2 columns to eliminate confusion since same personnel are notified in each case.. Added reason for notification. Change distractor D to eliminate non-existent position title.

Answer: A

C

15. G2.1.33 1

Plant conditions for Unit 4 are as follows:

- Mode 2
- MTC = $+5.5 \times 10^{-5}$ delta k/k/^{o}F
- BOL
- Low Power Physics testing is in progress
- All shutdown bank rods are fully withdrawn

Which ONE of the following will require Unit 4 to enter a Technical Specification Action Condition?

A. $K_{eff} = .99$ and MTC = -3.0×10^{-4} delta k/k/^{o}F Lowest $T_{avg} = 611^{o}$F

B. $K_{eff} = 1$ and MTC = -3.0×10^{-4} delta k/k/^{o}F Highest $T_{avg} = 610^{o}$F

C. $K_{eff} = .99$ and MTC = $+5.5 \times 10^{-5}$ delta k/k/^{o}F Lowest $T_{avg} = 531^{o}$F

D. $K_{eff} = 1$ and MTC = $+5.5 \times 10^{-5}$ delta k/k/^{o}F Highest $T_{avg} = 530^{o}$F

REFERENCES:

1. Tech Spec 3.1.1.3, MODERATOR TEMPERATURE COEFFICIENT, pages 1-5 & 1-6, Amendment Nos. 137 & 132
2. Tech Spec 3.10.3, PHYSICS TESTS, page 10-3, Amendment Nos. 137 & 132
3. Tech Spec 2.1.2, SAFETY LIMITS – REACTOR CORE, pages 2-1 & 2-2, Amendment Nos. 137 & 132

DISTRACTORS:

A Incorrect. Applicable at EOL, not BOL.

B Incorrect. Applicable at EOL, not BOL.

C Incorrect. Applicable only with K_{eff} greater than or equal to 1.

D Correct. IAW TS 3.10.3.c given the other plant conditions, action step “b” applies.

K/A CATALOGUE QUESTION DESCRIPTION:

- Conduct of Operations; Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

Basis for Changes: Added + signs for clarity. Used Plant specific terminology for physics testing.

Answer: D

(may rvs) C

16. G2.2.20 1

Unit 4 is at 100% power. Mechanical Maintenance is planning to erect a scaffold over redundant safety related equipment to perform trouble shooting activities.

Which ONE of the following identifies the highest level of approval required for the erection of this scaffolding?

A. Operations Manager

B. Unit Supervisor~~ANPS~~

C. Shift Manager ~~NPS~~

D. Assistant Operations Manager ~~Operations Supervisor~~

REFERENCES:

1. ADM-012, step 3.3.2

DISTRACTORS:

A Incorrect.

B Incorrect.

C Incorrect.

D Correct.

K/A CATALOGUE QUESTION DESCRIPTION:

- Equipment Control; Knowledge of the process for managing troubleshooting activities.

Basis for Changes: Changed titles to be consistent with current PTN titles.

Answer: D

C

17. G2.2.32 1

Reactor engineering had designed a core loading pattern that will be performed during the next refueling outage. The CHANGE will result in placing the "twice-burned" fuel assemblies more toward the periphery and the new fuel assemblies more toward the center of the core. Based on engineering calculations, it has been determined that Kexcess will be the same at the beginning of both fuel cycles.

Based on the above information, which ONE of the following describe the affect the new loading pattern will have on the unit?

A. The expected full power loop delta-T value should be significantly LOWER for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

B. The expected full power loop delta-T value should be significantly HIGHER for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

C. If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly BELOW actual power when the 1st calorimetric is performed after the refueling outage.

D. If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly ABOVE actual power when the 1st calorimetric is performed after the refueling outage.

DISTRACTORS:

A. Incorrect.

B. Incorrect.

C. Correct.

D. Incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:
- Equipment Control; Knowledge of the effects of alterations on core configuration.

Basis for Change: Added missing word "delta".

Answer: C

C

18. G2.3.6 1

The following is a time-line of activities associated with Waste Monitoring:

- 14~~00~~30: Waste Monitoring Tank (WMT) "A" is placed on mini-flow recirc for sampling
- 1620: Chemistry completes sampling of WMT "A." Result = 5.5 x 10^{-5} micro Ci/ml.
- 1730: Chemistry submits a Radiological Liquid Waste Discharge Permit for WMT "A"
- 1735: The Shift Manager ~~Nuclear Plant Supervisor~~ authorizes the release of the Radiologica Liquid Waste Permit without the approval of the Radiochemist or Health Physics Supervisor
- 1745: Operators align WMT "A" for discharge and start the release
- 1746: R-18, Waste Disposal System Liquid Effluent Monitor, fails low. The release is terminated and WMT "A" is restored to a normal lineup
- 1930: The R-18 monitor is repaired and restored to service
- 1935: The Shift Manager re-authorizes the release of WMT "A" on the same Radiological Liquid Waste Permit
- 1940: Operators re-align WMT "A" for discharge and start the release

Based on the above information, which ONE of the following represents the problem associated with these actions?

A. The sample taken for the Radiological Liquid Waste Permit may not be ~~is NOT~~ representative o the ~~current~~ contents of WMT "A" now being released.

B. A Radiological Liquid Waste Permit approved for one shift may NOT be used for initiation of a release on the next shift.

C. The discharge required the approval of the Health Physics Supervisor in addition to the Shift Manager ~~Nuclear Plant Supervisor~~.

D. The contents of WMT "A" must first be transferred to the Waste Holdup Tanks for further processing prior to release.

REFERENCES:

1. SD 049/SYS.061A, pages 27-29, rev 02/01/02
2. OP-061.11, CONTROLLED LIQUIED RELEASE TO THE CIRCULATING WATER
3. NCOAP-003, PREPARATION OF LIQUID RELEASE PERMIT

DISTRACTORS:

A Correct. IAW NCOAP–003, PREPARATION OF LIQUID RELEASE PERMIT,Attachments 1 and 6.

B Incorrect. Not required.

C Incorrect. As long as the specific activity of the tank contents is less ~~greate~~r than or equal to 1 x 10^{-4} Ci/ml, only the SM ~~NPS's~~ approval is required.

D Incorrect. Not required.

K/A CATALOGUE QUESTION DESCRIPTION:

- Radiation Control; Knowledge of the requirements for reviewing and approving release permits.

Basis for Changes: Available references did not support answer given. Changed recirc/sample time to < 2 hours in support of requirement to recirc. at least 2 hours when on mini-recirc. Added mini-recirc. Added missing "micro" to curies Changed title to be consistent with plant terminology. Improved answer wording. Identified correct references.

Answer: A

C

19. G2.4.34 1

Turkey Point has experienced a fire in the North/South Breezeway.

- The Fire Suppression System in the N-S Breezeway has activated
- Operators ~~Both crews~~ are in the process of carrying out the actions of ONOP-105, CONTROL ROOM EVACUATION

IAW ONOP-105, which ONE of the following correctly describes an action a particular operator is required to take given existing plant conditions?

A. From the Unit 3 480 Volt Load Center Room the Unit 3 RO will verify LC 3D Supply to LC 3H Breaker, 30402 – CLOSED; and verify 3B Load Center Supply Breaker, 30210 - CLOSED.

B. From the Unit 4 480 Volt Load Center Room the Unit 4 RO will trip LC 4D Supply to LC 4H Breaker, 40402; and verify 4B Load Center Supply Breaker, 40210 - CLOSED.

C. From the Unit 3 480 Volt Load Center Room the Third RO will verify LC 3D Supply to LC 3H Breaker, 30402 – CLOSED; and verify 3B Load Center Supply Breaker, 30210 - CLOSED.

D.✓ From the Unit 4 480 Volt Load Center Room the Third RO will trip LC 4D Supply to LC 4H Breaker, 40402; and verify 4B Load Center Supply Breaker, 40210 - CLOSED.

REFERENCES:

1. ONOP-105, CONTROL ROOM EVACUATION, pages ~~92,~~93,94, rev 10/15 ~~05/04~~/04

DISTRACTORS:

A Incorrect. The Third RO would take this action, NOT the Unit RO and it would be taken only if the fire was NOT in the N-S Breezeway.

B Incorrect. This is the action taken by the Third RO, NOT the Unit RO.

C Incorrect. This is the action the Third RO would take if there was NO fire in the N-S Breezeway.

D Correct. IAW ref 1, this action is to be taken by the Third RO in the event of a fire in the N-S Breezeway.

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Procedures/Plan; Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.
-

Basis for changes: Only one crew. Updated reference

20. G2.4.6 1

Given the following information:

- An event has occurred in the plant that has resulted in a radioactive release in the containment
- The Safety Parameter Display System (SPDS) indicates Critical Safety Function Status Tree display of YELLOW priority for Containment
- FR-Z.3, "Response to High Containment Radiation Level" ~~RESPONSE TO VOIDS IN REACTOR VESSEL,~~ has been entered

Which ONE of the following indicates the mitigation strategy for operator actions directed by FR-Z.3?

A. Allow a controlled release through the containment filtration system prior to exceeding design pressure limits on the containment.

B. Verify containment ventilation isolation and attempt to reduce activity by containment filtration.

C. Reduce containment activity levels with dilution flow using the ~~Main (Preaccess)~~ Containment Purge System.

D. Verify containment isolation Phase "A" and place all containment coolers in slow speed.

REFERENCES:
1. WOG, FR-Z.3, page 2, rev 1C
2. EOP-F-O, CRITICAL SAFETY FUNCTION STATUS TREES, page 11, rev 04/15/99C

DISTRACTORS:

A	Incorrect.	See B.
B	Correct.	IAW WOG.
C	Incorrect.	See B.
D	Incorrect.	See B.

K/A CATALOGUE QUESTION DESCRIPTION:
- Emergency Procedures/Plan; Knowledge of symptom based EOP mitigation strategies.

Basis for Changes: Provided correct procedure title. Provided correct name for containment purge system.

Answer: B

2

21. WE01EA2.2 1

Following a plant event, operators entered the EOP network. ~~complicated reactor trip~~, The control room crew transitioned to, and completed, the FINAL ~~final~~ step in ES-0.0, REDIAGNOSIS.

Which ONE of the following describes the plant conditions that resulted in reaching the end ~~this~~ point in ES-0.0 ~~the plant procedures~~?

A. A ruptured S/G had to have been identified and SI is required.

B. A ruptured S/G had to have been identified and SI is not required.

C. A faulted S/G had to have been identified and SI is required.

D. A faulted S/G had to have been identified and SI is not required.

REFERENCES:
1. EOP-ES-0.0, REDIAGNOSIS, pages 3,5, rev 12/14/02
2. BD-EOP-ES-0.0, REDIAGNOSIS BASIS, pages 8,12, rev 12/14/02

DISTRACTORS:

A Correct. ES-0.0, REDIAGNOSIS, should only be used if SI is in service or in required. To reach step 4, the final step, in ES-0.0, a ruptured S/G had to have been identified in the previous step.

B Incorrect. SI should either be in service or is required.

C Incorrect. A ruptured S/G had to have been identified.

D Incorrect. A ruptured S/G had to have been identified and SI should either be in service or is required.

K/A CATALOGUE QUESTION DESCRIPTION:
- Rediagnosis; Ability to determine and interpret the following as they apply to the (Reactor Trip or Safety Injection Rediagnosis): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Basis for Changes: Improved wording in stem.

Answer: A

C

22. WE03EA2.2 1

Following a LOCA with a concurrent loss of offsite power, Unit 3 entered E-1, LOSS OF REACTOR OR SECONDARY ~~SCEONDARY~~ COOLANT. Currently, the following plant conditions exist:

- T_{avg} is 345°F
- RCS pressure is 350 psig
- Containment temperature is 178°F
- RWST level equals ~~less than~~ ~~1~~255,000 gallons
- A mechanical failure of one train of SI has just occurred

Which ONE of the following describes the required operator actions in accordance with E-1 ~~and the LCO that applies to this condition~~?

A. Transition to ES-1.~~2~~1, SI TERMINATION ~~POST LOCA COOLDOWN AND DEPRESSURIZATION~~, Stop the running HHSI and ~~3A~~ RHR pumps and place in standby. ~~LCO 3.5.4, REFUELING WATER STORAGE TANK, maintain the plant in MODE 4.~~

B. Transition to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, stop the ~~3A~~ running RHR pump and place in standby. ~~LCO 3.5.3, ECCS SUBSYSTEMS, maintain the plant in MODE 4.~~

C. Transition to ES-1.3, TRANSITION TO COLD LEG RECIRCULATION, align RHR suction to containment recirc sump. ~~LCO 3.5.4, REFUELING WATER STORAGE TANK, plant in MODE 5 in 30 hours.~~

D. Transition to ECA-1.1, ~~ES-1.3,~~ LOSS OF EMERGENCY COOLANT RECIRCULATION. Establish makeup to the Unit 3 RWST.~~TRANSITION TO COLD LEG RECIRCULATION~~, ~~align RHR suction to containment recirc sump.~~ ~~LCO 3.5.3, ECCS SUBSYSTEMS, plant in MODE 5 in 30 hours.~~

REFERENCES:

1. E-1, LOSS OF REACTOR OR SECONDARY COOLANT, pages 18,21, rev 04/03/02
2. BD-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, page 35, rev 04/03/02
3. TECHNICAL SPECIFICATION, 3.5.3, page 3/4 5-9, Amendment Nos. 138 & 133
4. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, page 72, rev 05/01/03
5. ES-1.2, POST LOCA COOLDOWN & DEPRESSURIZATION, pgs 3,6,12, rev 04/03/02
6. BD ES-1.2, POST LOCA COOLDOWN AND DEPRESS, page 8, rev 04/03/02
7. TECHNICAL SPECIFICATION, 3.5.4, page 3/4 5-10, Amendment Nos. 138 & 133

DISTRACTORS:

- A Incorrect. The transition is not correct. The stated action would be correct if the transition were made to ES-1.1. ~~The LCO is also correct, however, the plant must be in MODE 5 in 30 hours~~

- B Correct. IAW E-1, Step 19, if RCS pressure is > 250 psig, go to ES-1.2, step 1 which directs stopping RHR pumps. ~~LCO 3.5.3. allows one train of RHR when < 350°F.~~

- C Incorrect. While this is a valid transition from E-1, it occurs when RWST drops below 155,000 gallons ~~is step 23 (after step 19)~~ so this transition would be incorrect.

- D Incorrect. While this is a valid transition from E-1, it occurs if <u>neither</u> RHR pump is available to support cold leg recirculation ~~is step 23 (after step 19)~~ so this transition would be incorrect. ~~The LCO is correct, however, the plant can be maintained in MODE 4.~~

K/A CATALOGUE QUESTION DESCRIPTION:

- LOCA Cooldown and Depressurization; Ability to determine and interpret the following as they apply to the (LOCA Cooldown and Depressurization): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Basis for Changes: As written the correct answer was to transition to ES-1.3. Revised stem to force transition to ES-1.2 per the KA. Added containment temperature non-adverse numbers to force transition to ES-1.2. Increased RWST level to 255K to prevent transition to ES-1.3. Removed references to Tech Specs. Tech Spec compliance is implicit in the WOG EOPs. ES-1.2 directs operators to initiate a rapid cooldown (complying with TS limit of 100°F/hr cooldown rate) to Cold Shutdown that will result in Mode 5 being reached long before the TS time frames are reached. Revised responses to give plausible procedure destinations and plausible activities in those procedures taking into account the mechanical failure of one train of SI.

Answer: B

23. WE05EA2.2 1

Operators are performing EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, and have successfully initiated Bleed and Feed. The BOP subsequently announces secondary heat sink is restored using "A" Standby SG Feed Pump.

Which ONE of the following describes the correct operator response?

A. Return to procedure and step in effect when feed flow is verified to be > 345 ~~354~~ gpm.

B. Continue performing FR-H.1 to completion.

C. Return to procedure and step in effect when narrow range level in any S/G is > 6%[32%].

D. Return to procedure and step in effect only when narrow range levels in all S/Gs are > 6%[32%].

REFERENCES:

1. FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, steps 7,27, rev 04/30/02

DISTRACTORS:

A Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because 345 gpm is a normal indicator of adequate heat sink.

B Correct. After Bleed and Feed is established, FR-H.1 must be completed to ensure SI reduction/termination and PORV closure are completed. Restoration of secondary heat sink is not enough to transition from FR-H. beyond Step 12 of the procedure.

C Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because 6% in any S/G is a normal indicator of adequate heat sink.

D Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because >6% in all S/Gs is a goal of the procedure.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of Secondary Heat Sink; Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Basis for change: Corrected typo.

2

24. WE08EG2.4.4 1

Which ONE of the following conditions would require entering FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, on an orange or red path?

A. Cooldown of cold leg GREATER THAN 100°F in 60 minutes, cold leg = 325°F, RCS pressure = 450 psig

B. Cooldown of cold leg GREATER THAN 100°F in 60 minutes, cold leg = 280°F, RCS pressure = 460 psig

C. Cooldown of cold leg LESS THAN 100°F in 60 minutes, Tavg = 280°F, RCS pressure = 460 psig

D. Cooldown of cold leg LESS THAN 100°F in 60 minutes, Tavg = 270°F, RCS pressure = 450 psig

REFERENCES:

1. EOP-F-0, CRITICAL SAFETY SYSTEM STATUS TREES, page 10, rev 08/03/01

DISTRACTORS:

A Incorrect. Cold leg temperature is NOT less than 320°F, but is less than 350°F, returning a Yellow path. Pressure is of no consequence.

B Correct. Cold leg temperature is less than 290°F, returning a red path.

C Incorrect. Tavg is NOT less than 275°F, returning a green path.

D Incorrect. Tavg is NOT less than 275°F and pressure is NOT greater than 460 psig.

K/A CATALOGUE QUESTION DESCRIPTION:

- Pressurized Thermal Shock; Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Answer: B

C

25. WE16EA2.1 1

Which ONE of the following describes an entry criteria for FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL?

A. Greater than 2.6 E 3 on R-11/12.

B. Greater than 6.1 E 5 on R-11/12.

C. Greater than 1.3 E 4 on CHRRMS.

D. Greater than 1.3 E 3 on CHRRMS.

REFERENCES:
1. FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL, page 3, rev 04/15/99
2. EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES, page 11, rev 04/15/99C
3. ARP-097.CR, CONTROL ROOM ANNUNCIATOR RESPONSE, page 424, rev 07/23/02

DISTRACTORS:

A	Incorrect.	FR-Z.3 is not entered based on this reading.
B	Incorrect.	FR-Z.3 is not entered based on this reading.
C	Correct.	This is the value that will initiate entry into FR-Z.3.
D	Incorrect.	FR-Z.3 is not entered based on this reading.

K/A CATALOGUE QUESTION DESCRIPTION:
- High Containment Radiation; Ability to determine and interpret the following as they apply to the (High Containment Radiation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Answer: C

QUESTIONS REPORT
for Draft TP05-301-SRO

1. 003AG2.2.25 001//T1G2/DROPPED ROD/M(2.5/3.7)/N/TP05301/S/MC

Unit 4 had been operating at 100% power when a Bank "D" rod dropped. The crew entered ONOP-0028.3, DROPPED RCC. The rod was declared inoperable and power was reduced to < 50%.

(is this bases)

Which ONE of the following describes the ~~reason~~ for the ~~LCO in~~ Technical Specifications ~~that~~ requires reducing reactor power to < 75% within one hour?

ABCD which

A. Ensures minimum DNBR in the core remains greater than or equal to the applicable design limit for continued operation and in short-term transients.

B. Provides adequate protection against $F_Q(Z)$,Heat Flux Hot Channel Factor, in the event of a subsequent loss-of-coolant accident.

C. Provides assurance that the effects of residual xenon redistribution impact from past operation near EOL is minimal.

D. ✓ Ensures that design margins to core limits will be maintained under both steady-state and anticipated transient conditions.

Feedback

REFERENCES:
1. Technical Specifications, 3.1.3.1, pages 3/4 1-17 & 1-18, Amendment Nos. 186 & 216
2. ADM-536, TECHNICAL SPECIFICATION BASES CONTROL PROGRAM, pages 32-36,41, rev 05/01/03

DISTRACTORS:

A Incorrect. This is the basis for Tech Spec 3.2, POWER DISTRIBUTION LIMITS

B Incorrect. This is part of the basis for POWER DISTRIBUTION LIMITS but has nothing to do with a LOCA.

C Incorrect. This is the basis for Tech Spec 3.2.1, AXIAL FLUX DIFFERENCE, but had nothing to do with EOL.

D Correct. IAW ref 2, page 33, paragraph 3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Dropped Control Rod; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

QUESTIONS REPORT
for Draft TP05-301-SRO

1. 003AG2.2.25 001//T1G2/DROPPED ROD/M(2.5/3.7)/N/TP05301/S/MC

Unit 4 had been operating at 100% power when a Bank "D" rod dropped. The crew entered ONOP-0028.3, DROPPED RCC. and ~~reduced~~ power. to < 50%. ~~within~~ one hour
The rod was declared inoperable was reduced

Which ONE of the following describes the reason for the LCO in Technical Specifications that requires reducing reactor power to < 75% within one hour?

A. Ensures minimum DNBR in the core remains greater than or equal to the applicable design limit for continued operation and in short-term transients.

B. Provides adequate protection against $F_Q(Z)$,Heat Flux Hot Channel Factor, in the event of a subsequent loss-of-coolant accident.

C. Provides assurance that the effects of residual xenon redistribution impact from past operation near EOL is minimal.

D.✓ Ensures that design margins to core limits will be maintained under both steady-state and anticipated transient conditions.

Feedback

REFERENCES:

1. Technical Specifications, 3.1.3.1, pages 3/4 1-17 & 1-18, Amendment Nos. 186 & 216
2. ADM-536, TECHNICAL SPECIFICATION BASES CONTROL PROGRAM, pages 32-36,41, rev 05/01/03

DISTRACTORS:

A Incorrect. This is the basis for Tech Spec 3.2, POWER DISTRIBUTION LIMITS

B Incorrect. This is part of the basis for POWER DISTRIBUTION LIMITS but has nothing to do with a LOCA.

C Incorrect. This is the basis for Tech Spec 3.2.1, AXIAL FLUX DIFFERENCE, but had nothing to do with EOL.

D Correct. IAW ref 2, page 33, paragraph 3.

K/A CATALOGUE QUESTION DESCRIPTION:

- Dropped Control Rod; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within the Allowed Rod Misalignment between the Analog Rod Position Indication ** and the group step counter demand position within one hour after rod motion. The Allowed Rod Misalignment shall be defined as:

a. for THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 18 steps, and

b. for THERMAL POWER greater than 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 12 steps.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps and THERMAL POWER greater than 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER and confirm that all indicated rod positions are within the Allowed Rod Misalignment, or

   3. Be in HOT STANDBY within the following 6 hours.

c. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 18 steps and THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Be in HOT STANDBY within the following 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3.

**During Unit 4 Cycle 20, the position of Rod C-9 Shutdown Bank A will be determined every 8 hours by verifying gripper coil parameters of the Control Rod Drive Mechanism to determine it has not changed, until the repair of the indication system for this rod is completed.

d. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than the Allowed Rod Misalignment of Specification 3.1.3.1, POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the Allowed Rod Misalignment of Specification 3.1.3.1, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may ~~then continue~~ provided that:

a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the power range neutron flux high trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.d.3.c and 3.1.3.1.d.3.d below are demonstrated, and

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position * of each full length rod shall be determined to be within the Allowed Rod Misalignment of the group step counter demand position at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time invervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours. **

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 92 days.

* During Unit 4 Cycle 20, the position of Rod C-9 Shutdown Bank A will be determined every 8 hours by verifying gripper coil parameters of the Control Rod Drive Mechanism to determine it has not changed state, until the repair of the indication system for this rod is completed.

** During Unit 4 Cycle 20, the position of rod C-9, Shutdown Bank A, may be monitored by verifying gripper coil parameters of the Control Rod Drive Mechanism to determine it has not changed state and it will not provide an input into the Rod Position Deviation Monitor. The use of the alternate method for rod C-9 does not require the 4 hour comparison of demanded versus actual position per 4.1.3.1.1.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

ATTACHMENT 1
(Page 22 of 102)
TECHNICAL SPECIFICATION BASES

3/4.1 REACTIVITY CONTROL SYSTEMS (Continued)

3/4.1.2 BORATION BORATION SYSTEMS (Continued)

The boron concentration of the RWST in conjunction with manual addition of borax ensures that the solution recirculated within containment after a LOCA will be basic. The basic solution minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The temperature requirements for the RWST are based on the containment integrity and large break LOCA analysis assumptions.

The OPERABILITY of one Boron Injection flowpath during REFUELING ensures that this system is available for reactivity control while in MODE 6. Components within the flowpath, e.g., boric acid transfer pumps or charging pumps, must be capable of being powered by an OPERABLE emergency power source, even if the equipment is not required to operate.

The OPERABILITY requirement of 55°F and corresponding surveillance intervals associated with the boric acid tank system ensures that the solubility of the boron solution will be maintained. The temperature limit of 55°F includes a 5°F margin over the 50°F solubility limit of 3.5 wt.% boric acid. Portable instrumentation may be used to measure the temperature of the rooms containing boric acid sources and flow paths.

3.1.2.6 HEAT TRACING

This specification is no longer applicable. Boric acid concentration has been diluted to less than or equal to 3.5 weight percent (wt%). Refer to PC/M 90-440.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the demand counter position. For the Shutdown Banks and Control Banks A and B, the Position Indication requirement is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 200 steps withdrawn and All Rods Out (ARO) inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these bank positions in these two areas satisfies all accident analysis assumptions concerning their position. For Control Banks C and D, the Position Indication requirement is defined as the group demand counter indicated position between 0 steps withdrawn and All Rods Out (ARO) inclusive.

ATTACHMENT 1
(Page 23 of 102)

TECHNICAL SPECIFICATION BASES

3/4.1 REACTIVITY CONTROL SYSTEMS (Continued)

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The increase in the Allowable Rod Misalignment below 90% or Rated Thermal Power **is as a result of the increase in the peaking factor limits as reactor power is reduced.**

Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits.

Rod position indication is provided by two methods: a digital count of actuating pulses which shows demand position of the banks and a linear position indicator Linear Variable Differential Transformer which indicates the actual rod position. The relative accuracy of the linear position indicator Linear Variable Differential Transformer is such that, with the most adverse error, an alarm will be actuated if any two rods within a bank deviate by more than 24 steps for rods in motion and 12 steps for rods at rest. **Complete rod misalignment** (12 feet out of alignment with its bank) **does not result in exceeding core limits in steady-state operation at** RATED THERMAL POWER. **If the condition cannot be readily corrected, the specified reduction in power to** 75% will insure that design margins to core limits will be distortion **maintained under both steady-state and anticipated transient conditions.** The 8-hour permissible limit on rod misalignment is short with respect to the probability of an independent accident.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. **These restrictions provide assurance of fuel rod integrity during continued operation.** In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with Tavg greater than or equal to 500°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.2 POWER DISTRIBUTION LIMITS

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) **maintaining the minimum DNBR in the core greater than or equal to the applicable design limit during normal operation and in short-term transients**, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{XY}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ limit defined in the CORE OPERATING LIMITS REPORT times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

ATTACHMENT 1
(Page 25 of 102)

TECHNICAL SPECIFICATION BASES

3/4.2 POWER DISTRIBUTION LIMITS (Continued)

3/4.2.1 AXIAL FLUX DIFFERENCE (Continued)

At power level below P_T, the limits on AFD are specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation. These limits were calculated in a manner **such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits.** However, in the event that such a deviation occurs, a 15 minute period of time allowed outside of the AFD limits at reduced power levels **will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently** to prevent operation in the vicinity of the power level.

With P_T greater than 100%, two modes are permissible: 1) RAOC with fixed AFD limits as a function of reactor power level and 2) Base Load operation which is defined as the maintenance of the AFD within a band about a target value. Both the fixed AFD limits for RAOC operation and the target band for Base Load operation are defined in the COLR and the Peaking Factor Limit Report, respectively. However, it is possible during extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(Z)$ less than its limiting value. Therefore, P_T is calculated to be less than 100%. To allow operation at the maximum permissible value above P_T Base Load operation restricts the indicated AFD to a relative small target band and power swings. For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed (15 minutes) will not result in significant xenon redistribution such that the envelope of peaking factors will change sufficiently to prohibit continued operation in the power region defined above. **To assure that there is no residual xenon redistribution impact from past operation on the Base Load operation,** a 24-hour waiting period within a defined range of P_T and AFD allowed by RAOC is necessary. During this period, load changes and rod motion are restricted to that allowed by the Base Load requirement. After the waiting period, extended Base Load operation is permissible.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitoring Alarm. The computer monitors the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are: 1) outside the acceptable AFD (for RAOC operation), or 2) outside the acceptable AFD target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) P_T (Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short time period during which operation outside of the target band is allowed.

ATTACHMENT 1
(Page 26 of 102)

TECHNICAL SPECIFICATION BASES

3/4.2 <u>POWER DISTRIBUTION LIMITS</u> (Continued)

3/4.2.2 and 3/4.2.3 <u>HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR</u>

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit. The LOCA peak fuel clad temperature limit may be sensitive to the number of steam generator tubes plugged.

$F_Q(Z)$, <u>Heat Flux Hot Channel Factor</u>, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux.

$F^N_{\Delta H}$ <u>Nuclear Enthalpy Rise Hot Channel Factor</u>, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.2 POWER DISTRIBUTION LIMITS (Continued)

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q(Z)$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or incore thermocouple map are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR above the applicable design limits throughout each analyzed transient. The indicated Tavg value of 581.2°F and the indicated pressurizer pressure value of 2200 psig correspond to analytical limits of 583.2°F and 2175 psig respectively, with allowance for measurement uncertainty.

The measured RCS flow value of 264,000 gpm corresponds to an analytical limit of 255,000 gpm which is assumed to have a 3.5% calorimetric measurement uncertainty.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. Figure 3.1-1 shows the SHUTDOWN MARGIN equivalent to 1.77% $\Delta k/k$ at the end-of-core-life with respect to an uncontrolled cooldown. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With Tavg less than 200°F, the reactivity transients resulting from an inadvertent cooldown of the RCS or an inadvertent dilution of RCS boron are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

The boron rate requirement of 16 gpm of 3.0 wt% (5245 ppm) boron or equivalent ensures the capability to restore the shutdown margin with one OPERABLE charging pump.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -3.5×10^{-4} $\Delta k/k/°F$. The MTC value of -3.0×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -3.5×10^{-4} $\Delta k/k/°F$.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.1 REACTIVITY CONTROL SYSTEMS (Continued)

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

[redacted] **ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F.** This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, and (4) boric acid transfer pumps.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. One flow path from the charging pump discharge is acceptable since the flow path components subject to an active failure are upstream of the charging pumps.

The boration flow path specification allows the RWST and the boric acid storage tank to be the boron sources. Due to the lower boron concentration in the RWST, borating the RCS from this source is less effective than borating from the boric acid tank and additional time may be required to achieve the desired SHUTDOWN MARGIN required by ACTION statement restrictions. ACTION times allow for an orderly sequential shutdown of both units when the inoperability of a component(s) affects both units with equal severity. When a single unit is affected, the time to be in HOT STANDBY is 6 hours. When an ACTION statement requires a dual unit shutdown, the time to be in HOT STANDBY is 12 hours.

FOLDOUT FOR PROCEDURE ONOP-028.3

Following is a list of applicable Tech. Spec. LCOs and procedure steps that verify compliance. These need to be reviewed by the NPS to ensure compliance.

1. **T.S. 3.1.1.1 - SHUTDOWN MARGIN**

    - Covered in step 8 and 12.
    - 1 hour to verify AND repeat every 12 hours.

2. **T.S. 3.1.3.1 - GROUP ROD HEIGHT**

    - Covered in steps 6, 7, ~~9, 10, 19, 20~~
    - 1 hour - to restore **OR** declare inoperable AND be less than 75% power,
      72 hours - to ~~perform Flux Map, and~~
      5 days – to perform re-analysis of accidents.
    - If 2 Control Rods, 7 hours - to be in Hot Standby.
    - If 2 or more Shutdown Rods are not fully withdrawn (230 steps), apply Tech Spec 3.0.3.
    - If Rod Deviation Monitor is inoperable, compare RPIs to Step Counters every 4 hours.

3. **T.S. 3.1.3.5 - SHUTDOWN ROD INSERTION LIMIT**

    - Covered in step 7.
    - 1 hour - to restore **OR** declare inoperable.
    - If 2 or more Shutdown Rods are not fully withdrawn (230 steps), apply Tech Spec 3.0.3.
    - If Rod Deviation Monitor is inoperable, compare RPIs to Step Counters every 4 hours.

4. **T.S. 3.1.3.6 - CONTROL ROD INSERTION LIMIT**

    - Covered in step 9 by power reduction.
    - 2 hours - to restore **OR** to reduce power to within guidelines of COLR.
    - If Rod Deviation Monitor is inoperable, compare RPIs to Step Counters every 4 hours.

5. **T.S. 3.2.1 - AXIAL FLUX DIFFERENCE**

    - Covered in step 4 and 15.
    - 30 minutes to get power less than 50% AND the next 4 hours to set the NIS trip setpoint to 55%.

6. **T.S. 3.2.4 - QUADRANT POWER TILT RATIO**

    - Covered in step 5.
    - 1 hour to perform AND required every hour thereafter while QPTR exceeds 2%.
    - If QPTR is >9%, reduce power 3% for every 1% QPTR exceeds 1 within 30 minutes.
    - If QPTR is >2%, reduce power 3% for every 1% QPTR exceeds 1 within 2 hours.
    - Reduce the NIS trip setpoints the same amount within the next 4 hours.

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 Technical Specifications

1. Section 3/4.1.1, Reactivity Control Systems - Boration Control
2. Section 3/4.1.1.1, Shutdown Margin - Tavg Greater Than 200°F
3. Section 3/4.1.3.1, Reactivity Control Systems – Movable Control Assemblies - Group Height
4. Section 3/4.1.3.5, Reactivity Control Systems – Shutdown Rod Insertion Limit
5. Section 3/4.1.3.6, Reactivity Control Systems – Control Rod Insertion Limits
6. Section 3/4.2.1, Power Distribution Limits – Axial Flux Difference
7. Section 3/4.2.3, Power Distribution Limits – Nuclear Enthalpy Rise Hot Channel Factor
8. Section 3/4.2.4, Power Distribution Limits – Quadrant Power Tilt Ratio

3.1.2 Final Safety Analysis Report

1. Section 14.1.4, Rod Control Cluster Assembly (RCCS) Drop

3.1.3 Plant Procedures

1. 3-GOP-103, Power Operation To Hot Standby
2. 3-ONOP-059.9, Excessive Quadrant Power Tilt Ratio
3. 3-ONOP-059.4, Excessive Axial Flux Difference
4. 3-OSP-040.5, Nuclear Design Verification
5. 3-OSP-059.10, Determination of Quadrant Power Tilt Ratio
6. 0-OP-028.2, Shutdown Margin Calculation

3.1.4 Regulatory Guidelines

1. SOER 84-2 (1,2 and 8) Control Rod Mispositioning
2. SOER 84-2, Control Rod Mispositioning - Addendum

3.1.5 Miscellaneous Documents (i.e., PC/M, Correspondence)

1. PC/M 93-005, Elimination of Turbine Runback from Dropped Rod
2. Unit 3 Plant Curve Book

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- *STEP 1 is an IMMEDIATE ACTION STEP*
- *Foldout page is required to be monitored throughout this procedure.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check Number Of RCCs - DROPPED** • Rod Bottom Lights - LESS THAN 3 ON • Rod Position Indicators – LESS THAN 3 AT ZERO • RCCs misaligned from step counters - LESS THAN 3 • Dropped rods - ALL IN SAME ROD BANK GROUP	Perform the following: a. Manually trip the Reactor and Turbine. b. Go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.
2	**Place Rod Control Selector Switch To MANUAL**	

CAUTIONS

- ***Do NOT dilute the RCS during the performance of this procedure until SHUTDOWN MARGIN calculation has been performed using 0-OP-028.2, SHUTDOWN MARGIN CALCULATION.***
- ***Do NOT increase reactor power during the performance of this procedure.***
- ***Do NOT use control rods for power or temperature adjustments until the cause of the dropped rod is identified and determined not to affect any other rods.***

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Verify Automatic Controls Are Functioning To Stabilize The Unit <u>AND</u> No Transient In Progress**	
	a. Tavg/Tref within 3°F	a. Reduce turbine load to control temperature
	b. PZR level/pressure trending to program	b. Manually control systems to stabilize the unit
	c. S/G level trending to program	c. Manually control systems to stabilize the unit

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Check AFD Within RAOC** • G 5/1, AXIAL FLUX T.S. LIMIT EXCEEDED - OFF • At least 3 channels of AFD Indicating within the RAOC limit, defined in the Plant Curve Book Section 5, Fig. 1	Within 30 minutes, reduce reactor power to less than 50% using 3-ONOP-100, FAST LOAD REDUCTION, while continuing with this procedure.
5	**Initiate Hourly QPTR Determination Using 3-OSP-059.10, DETERMINATION OF QUADRANT POWER TILT RATIO Until Either QPTR Results Are Satisfactory <u>OR</u> Reactor Power Is Less Than 50%**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	**Declare The Dropped RCC(s) Inoperable**	
7	**Check Only 1 RCC DROPPED**	
	a. Verify less than 2 Control Group RCCs misaligned from step counters by checking all indicators • Control Group Rod Bottom Lights - LESS THAN 2 0N • Control Group Rod Position Indicators – LESS THAN 2 RCCs MISALIGNED	a. Perform the following: 1) Within 60 minutes, reduce reactor power to less than 50% using 3-ONOP-100, FAST LOAD REDUCTION 2) Comply with the actions of the following: * Tech Spec 3.1.3.1.b.3 or 3.1.3.1.c.2 **AND** be in Mode 3, Hot Standby, within 7 hours of entering the action statement. * Tech Spec 3.1.1 – To verify adequate shutdown margin using 0-OP-028.2, SHUTDOWN MARGIN CALCULATI ON. 3) Reduce load **AND** go to Mode 3, Hot Standby, as directed by the NPS to comply with the appropriate Tech Spec using 3-ONOP-100, FAST LOAD REDUCTION. 4) Go to appropriate plant procedure as determined by the Nuclear Plant Supervisor.
	b. Verify less than 2 Shutdown Bank RCCs misaligned from step counters **OR** not fully withdrawn by checking all indicators • Shutdown Bank Rod Bottom Lights - LESS THAN 2 ON • Shutdown Bank Rod Position Indicators - LESS THAN 2 MISALIGNED **OR NOT** FULLY WITHDRAWN (230 STEPS)	b. Perform the following: 1) Commence a fast load reduction using 3-ONOP-100, FAST LOAD REDUCTION, **AND** be in Hot Standby within the next 6 hours to comply with Tech Spec 3.1.3.5 and 3.0.3. 2) Go to appropriate plant procedure as determined by the Nuclear Plant Supervisor.
	c. Check Rod Position Deviation Monitor operable – ANNUNCIATOR B 9/3 **NOT** LOCKED IN	c. Compare RPIs to group step counters every 4 hours to comply with Tech Spec Surveillances 4.1.3.1.1 and 4.1.3.6.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

The SHUTDOWN MARGIN is required to be verified within 1 hour of a DROPPED RCC and every 12 hours thereafter.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	**Verify Shutdown Margin Adequate**	
	a. Check RCS boron concentration * GREATER THAN OR EQUAL TO PRE-EVENT VALUE **<u>OR</u>** * GREATER THAN the Minimum Shutdown Boron Versus RCS Temperature as a Function of Burnup Requirements in the Plant Curve Book Section 3, Fig. 5	a. Notify reactor engineering to evaluate SHUTDOWN MARGIN using 0-OP-028.2, SHUTDOWN MARGIN CALCULATION, **<u>AND</u>** go to Step 9.
	b. Log ~~Shutdown Margin Satisfied~~ in the narrative log	
9	**Check Reactor Power Being Reduced To Less Than 50%**	Within one hour, reduce reactor power to less than 50% using 3-ONOP-100, FAST LOAD REDUCTION, while continuing with this procedure.
10	**Notify Reactor Engineering Of Dropped RCC**	

NOTE

PWO will <u>NOT</u> be required; setpoint will be adjusted using 3-OSP-059.4, POWER RANGE NUCLEAR INSTRUMENTATION ANALOG CHANNEL OPERATIONAL TEST.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	**Notify I&C Of Dropped RCC <u>AND</u> Potential Need To Reset POWER RANGE HI FLUX TRIP SETPOINT**	
12	**Calculate Shutdown Margin**	
	a. Perform 0-OP-028.2, SHUTDOWN MARGIN CALCULATION, while continuing with this procedure.	

1. 005AA1.01 001/T1G1/T1G1/3.6/3.4/C/A/NEW/VG01301/BOTH/LM05

Maintaining rod insertion limits within technical specifications, ensures which of the following is maintained?

A. Entropy rise Hot channel factor within limits.

B.✓ Core can be subcritical by shutdown margin with one RCCA stuck.

C. Dropped rod or misaligned bank will not result in peak power density that exceeds the center line melting criteria.

D. Uncontrolled RCCA withdrawal will not result in exceeding DNBR.

B
ref:LO-LP-27101-20

1. 003AG2.1.32 001//T1G1/3.4/3.8/MEMORY/NEW/TP00301/SRO/SDR/TP

A dropped rod at 100% power has caused the Quadrant Power Tilt Ratio (QPTR) to be 1.03.

Which one of the following describes the reason for allowing up to two hours before QPTR or reactor power must be reduced?

To allow time:

A.✓ to retrieve the dropped rod.

B. to perform a calorimetric calculation.

C. for the NPS to notify the NRC.

D. for the NIS High flux trip setpoints to be reduced.

A

REFERENCE:
Tech. Spec. 3.2.4 Basis

2. G2.2.25 001/3//SAFETY LIMIT BASES/M 2.5/3.7/BANK/SM04301/R/GWL

Which ONE of the following is the Technical Specification bases for the pressurizer water level reactor trip?

A. Protects against loss of pressure control due to the spray nozzle being submerged.

B.✓ Protects the pressurizer safety valves against water relief.

C. Prevents exceeding containment design pressure in the event of a LOCA.

D. Prevents solid plant operation while the reactor is critical.

Bank Question from farley exam bank.
Objective IC-9-38.

A. Incorrect, this is not the bases for the trip.

B. Correct this is the bases for the High Pressurizer level trip.

C. Incorrect, mass in the secondary side (S/G) is limited to prevent this.

D. Incorrect, the RCS could be solid with the reactor critical if power was less than P-10.

QUESTIONS REPORT
for Westinghouse 3 Loop Questions

1. 024G2.2.25 077//////HR04301/S/

Which of the following describes a condition in Technical Specifications and its bases which would require Emergency Boration in accordance with AOP-002, "Emergency Boration"?

A.
- During the recovery from a Main Feedwater Pump trip, Control Rods are determined to be below the rod insertion limit
- Control the reactivity transient associated with a steam line break

B.
- During the recovery from a Main Feedwater Pump trip, Control Rods are determined to be below the rod insertion limit
- Control the reactivity transient associated with an inadvertent dilution

C.✓
- During a reactor startup, the Reactor achieves criticality with Bank C rods at 105 steps
- Control the reactivity transient associated with a steam line break

D.
- During a reactor startup, the Reactor achieves criticality with Bank C rods at 105 steps
- Control the reactivity transient associated with an inadvertent dilution

QUESTIONS REPORT

for Draft TP05-301-SRO

1. 006G2.1.28 001//T2G1/ECCS/M(3.2/3.3)/N/TP05301/S/MC

The crew on Unit 4 entered FR-C.2, RESPONSE TO DEGRADED CORE COOLING, after an ORANGE condition was identified. IAW FR-C.2 the crew closes all Accumulator Discharge MOVs.

Which ONE of the following describes why power is not subsequently removed from the Accumulator Discharge MOVs once they are shut?

A. Check valves are installed to prevent back leakage from the RCS into the accumulators.

B✓ Leaving the MOVs energized ensures the operator does not delay actions to restore core cooling by de-energizing the MOVs.

C. The circuit breakers supplying the MOVs are normally open and "racked out" and are returned to this condition during the course of completing FR-C.2.

D. The Accumulators are designed to prevent the introduction of nitrogen into the RCS upon equalizing with RCS pressure.

Feedback

REFERENCES:

1. BD-EOP-FR-C.2, RESPONSE TO DEGRADED CORE COOLING, page 35, rev 12/14/02
2. EOP-FR-C.2, RESPONSE TO DEGRADED CORE COOLING, page 12, rev 12/14/02
3. SD 021/SYS.050,062,064, page 19, rev 04/22/04

DISTRACTORS:

A Incorrect. See B.

B Correct. IAW Basis.

C Incorrect. See B.

D Incorrect. See B.

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Core Cooling System (ECCS); Knowledge of the purpose and function of major system components and controls.

2. 006G2.1.28 001//T2G1/ECCS/M(3.2/3.3)/N/TP05301/S/MC

Unit 4 experienced an ORANGE condition on the Critical Safety Function Status Tree, F-0.2, CORE COOLING. Plant conditions are a follows:

- The crew has entered FR-C.2, RESPONSE TO DEGRADED CORE COOLING
- The procedure directs the crew to close all Accumulator Discharge MOVs

Which ONE of the following describes why power is not subsequently removed from the Accumulator Discharge MOVs once they are shut?

A. Check valves are installed to prevent back leakage from the RCS into the accumulators.

B.✓ Leaving the MOVs energized ensures the operator does not delay actions to restore core cooling by de-energizing the MOVs.

C. The circuit breakers supplying the MOVs are normally open and "racked out" and are returned to this condition during the course of completing FR-C.2.

D. The Accumulators are designed to prevent the introduction of nitrogen into the RCS upon equalizing with RCS pressure.

Feedback

REFERENCES:

1. BD-EOP-FR-C.2, RESPONSE TO DEGRADED CORE COOLING, page 35, rev 12/14/02
2. EOP-FR-C.2, RESPONSE TO DEGRADED CORE COOLING, page 12, rev 12/14/02
3. SD 021/SYS.050,062,064, page 19, rev 04/22/04

DISTRACTORS:

A Incorrect. See B.

B Correct. IAW Basis.

C Incorrect. See B.

D Incorrect. See B.

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Core Cooling System (ECCS); Knowledge of the purpose and function of major system components and controls.

BASIS DOCUMENT

WOG Procedure Step 16 **PTN Procedure Step** 19

Close All Accumulator Discharge MOVs

BASIS:

If not previously isolated, the accumulators should be isolated after SI flow has been established and verified in order to minimize nitrogen injection into the RCS. If it is necessary to vent the nitrogen, the operator should open the vent lines and then continue with this procedure.

Other steps within the EOPs which close accumulator isolation valves have been modified on a plant specific basis to remove power from the MOVs after shutting the valves. In the case of degraded core cooling, it is desirable to follow the WOG practice of leaving the MOVs energized. This will ensure the operator does not delay actions to restore core cooling by de-energizing the MOVs.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

8 The words "SI Accumulator Isolation Valves" were changed to "Accumulator Discharge MOVs" to conform with plant specific terminology.

9 A list of appropriate component designators was added to aid the operator.

9 To ensure proper venting of unisolated accumulators, the RNO was modified to direct the operator to the appropriate plant specific procedure.

PLANT SPECIFIC SETPOINTS:

N/A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18	**Verify SI Flow** * High-head SI pump flow indicator - CHECK FOR FLOW **<u>OR</u>** * RHR pump flow indicator – CHECK FOR FLOW	Perform the following: a. Continue efforts to establish SI flow. b. Try to establish any other high pressure injection as follows: 1) Reset SI. 2) Reset containment isolation phase A. 3) **<u>IF</u>** offsite power is **<u>NOT</u>** available, **<u>THEN</u>** check diesel capacity adequate to run three charging pumps. **<u>IF</u>** adequate diesel capacity is **<u>NOT</u>** available, **<u>THEN</u>** shed non-essential loads. Refer to ATTACHMENT 1 for component KW load rating. 4) Start charging pumps to deliver maximum flow. c. Return to Step 17.
19	**Close All Accumulator Discharge MOVs** • MOV-3-865A • MOV-3-865B • MOV-3-865C	Direct operator to vent any unisolated accumulator using 3-OP-064, SAFETY INJECTION ACCUMULATORS.
20	**Stop All RCPs**	
21	**Check Core Cooling** • RVLMS (QSPDS) plenum indication - GREATER THAN 0% • At least two RCS hot leg temperatures - LESS THAN 350°F	Return to Step 17.
22	**Go To 3-EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 18**	

END OF TEXT

positioned via a potentiometer on VPB and is designed to function as a relief valve if header pressure reaches 900 PSIG.

Two redundant trains of level and pressure sensors are installed on each tank and provide indication on VPB and alarms on annunciator panel H, windows 2/1 through 2/6. A detailed listing of instrumentation and alarms is contained in the tables in the final section.

A motor operated isolation valve and two check valves are installed in each of the 10" lines connecting the accumulators to the RCS cold legs. The check valves prevent back leakage from the RCS into the accumulators. The motor operated isolation valves (865A, B, & C) are the actual RCS pressure boundary. Provisions are made for check valve back leakage testing after repressurizing the RCS above 1000 PSIG. Acceptable leakage criteria is 2cc/hr/inch nominal pipe diameter, with a 1550 PSI differential pressure across the disc. At a leak rate of 30 cc/hr/inch (15 times the acceptable leak rate), accumulator water level would need readjusting about every 5 to 6 months. The motor operated isolation valves are normally open with their circuit breakers "racked out" and are only closed during normal plant cooldown and depressurization evolutions or emergencies. In the event the valves were closed and a safety injection signal was initiated, they would receive automatic open signals.

Miscellaneous accumulator data is contained in the last section.

RESIDUAL HEAT REMOVAL SYSTEM

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 temperature 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 head of about 150 PSID equals the system design pressure of 600 PSIG.

The design heat transfer rate for the RHR System is 58.8 x 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

1.0 **PURPOSE**

1.1 This procedure provides actions to restore adequate core cooling.

1.2 This procedure is applicable when directed by EOP entry conditions.

2.0 **SYMPTOMS OR ENTRY CONDITIONS**

2.1 This procedure is entered from F-0.2, CORE COOLING Critical Safety Function Status Tree, on an ORANGE condition.

3.0 **REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS**

3.1 References

3.1.1 Technical Specifications for Turkey Point Unit 3 and Unit 4

3.1.2 Turkey Point Unit 3 and Unit 4 Final Safety Analysis Report

3.1.3 As-built plant drawings

3.1.4 Procedures

1. 3-OP-064, SAFETY INJECTION ACCUMULATORS

3.1.5 Plant Change/Modifications

1. PC/M 87-025, Replacement of Normal Containment Coolers
2. PC/M 87-194, Containment Spray Pump Restricting Orifice
3. PC/M 87-213, RCS Vent Solenoid Valve Tag Number Change (SV-6320 A&B)
4. PC/M 87-264, EDG 3B/4B, EDG 3A/4A, And New EDG Building Tie-Ins
5. PC/M 90-440, Boric Acid Concentration Reduction
6. PC/M 90-524, SI Accumulator Water Level Instrument Band Increase

1. 008G2.1.28 016//////HR04301/R/

Following a Reactor Trip and Safety Injection due to a RCS leak, the Critical Safety Function Status Trees (CSFST) are being monitored.

When monitoring the CSFST for RCS Inventory, if PRZ level is indicating greater than 92%, why is a check of RVLIS then performed?

A.✓ Determine if the cause of the high PRZ level is excessive RCS inventory or voiding in the Reactor Vessel head

B. Determine if SI termination criteria is met to allow reducing the excessive RCS inventory

C. Determine if Adverse Containment conditions have caused erroneous indications of the PRZ level instruments

D. Determine if the cause of the high PRZ level is excessive RCS inventory or expansion due to an RCS heatup

3. 014A2.07 001//T2G2/RPIS/C/A(2.6/3.0)/M/TP05301/S/MC

Unit 4 is in Mode 3 with all Shutdown Rods withdrawn, Rod Motion Selector Switch in AUTO, and MCC 4B tagged out for maintenance. Power is subsequently lost to 4D01.

For the current plant conditions, which ONE of the following describes the impact on the plant and the actions necessary to mitigate the impact of these conditions?

A. Power has been lost to Demand Position Indication. Open the reactor trip breakers IAW Tech Spec 3.1.3.3, REACTIVITY CONTROL SYSTEMS.

B. Power has been lost to Demand Position Indication. Commence a boration of the RCS to ensure adequate shutdown margin IAW ONOP-028.1, RCC MISALLIGNMENT.

C✓ Power has been lost to Digital Rod Position Indication. Place the Rod Motion Selector Switch to the MAN position and notify the I&C Supervisor IAW ONOP-028.2, RCC POSITION INDICATION MALFUNCTION.

D. Power has been lost to Digital Rod Position Indication. Place the Rod Selector Switch to the MAN position and verify that all Shutdown Bank rods are fully withdrawn by performing an incore flux trace IAW ONOP-028.2, RCC POSITION INDICATION MALFUNCTION.

Feedback

REFERENCES:

1. ONOP-028.2, RCC POSITION INDICATION MALFUNCTION, pages 5,6, rev 04/12/02
2. SD-006/SYS.028B, ROD POSITION INDICATION SYSTEM, pages 9,10, fig 5, rev 04/13/04
3. ONOP-028.1, RCC MISALIGNMENT, pages 6,7, rev 04/12/02
4. Tech Spec 3.1.3.3, REACTIVITY CONTROL SYSTEMS, page 1-23, Amendment Nos. 149 & 144

DISTRACTORS:

A Incorrect. Power has not been lost to Demand Position Indication.

B Incorrect. Power has not been lost to Demand Position Indication and the purpose of boration is to maintain Tavg/Tref within 3 degrees.

C Correct. IAW ONOP-028.2.

D Incorrect. This is the response for a loss of Demand Position Indication.

K/A CATALOGUE QUESTION DESCRIPTION:

- Rod Position Indication System (RPIS); Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of LVDT.

3. 014A2.07 001//T2G2/RPIS/C/A(2.6/3.0)/M/TP05301/S/MC

Unit 4 is in Mode 3 with all Shutdown Rods withdrawn.

- MCC 4B is tagged out for maintenance
- 4D01 has just lost power

Which ONE of the following describes the impact to the plant and the actions necessary to mitigate this condition?

A. Power has been lost to Demand Position Indication. Open the reactor trip breakers IAW Tech Spec 3.1.3.3, REACTIVITY CONTROL SYSTEMS.

B. Power has been lost to Demand Position Indication. Commence a boration of the RCS to ensure adequate shutdown margin IAW ONOP-028.1, RCC MISALLIGNMENT.

C✓ Power has been lost to Digital Rod Position Indication. Place the Rod Motion Selector Switch to the MAN position and notify the I&C Supervisor ONOP-028.2, RCC POSITION INDICATION MALFUNCTION.

D. Power has been lost to Digital Rod Position Indication. Place the Rod Selector Switch to the MAN position and verify that all Shutdown Bank rods are fully withdrawn by performing an incore flux trace IAW ONOP-028.2, RCC POSITION INDICATION MALFUNCTION.

Feedback

REFERENCES:

1. ONOP-028.2, RCC POSITION INDICATION MALFUNCTION, pages 5,6, rev 04/12/02
2. SD-006/SYS.028B, ROD POSITION INDICATION SYSTEM, pages 9,10, fig 5, rev 04/13/04
3. ONOP-028.1, RCC MISALIGNMENT, pages 6,7, rev 04/12/02
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A Incorrect. Power has not been lost to Demand Position Indication.

B Incorrect. Power has not been lost to Demand Position Indication and the purpose of boration is to maintain Tavg/Tref within 3 degrees.

C Correct. IAW ONOP-028.2.

D Incorrect. This is the response for a loss of Demand Position Indication.

K/A CATALOGUE QUESTION DESCRIPTION:

- Rod Position Indication System (RPIS); Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of LVDT.

5.0 SUBSEQUENT ACTIONS

NOTE

As T_{hot} is decreased (i.e., power reduction), RPI signals tend to read higher than the step counters due to increased magnetic coupling from the primary to secondary coils and resultant higher current.

5.1. **IF** the faulty RCC position indicator is associated with a RCC in a control bank, not fully inserted or fully withdrawn, **THEN** place the Rod Motion Control Selector Switch to the MAN position.

5.2 Notify the I&C Supervisor to investigate the RCC position indication malfunction.

NOTE

When RPI power is transferred from RPI inverter to the CVT, the RPI indication may change.

5.3 **IF** the RCC position indication malfunction is caused by an RPI power supply transfer, **THEN** perform the following:

5.3.1 Initiate a PWO to repair the failed inverter.

5.3.2 Initiate a PWO to calibrate the RPI system to compensate for any changes in power supply voltage.

5.4 **IF** a maximum of one individual RCC Position Indicator (RPI) per bank is inoperable in Modes 1 or 2, **THEN**, either:

5.4.1 Determine the position of the non-indicating RCC indirectly by an incore flux trace at least once per 8 hours **AND** within one hour after motion of the non-indicating RCC which exceeds 24 steps in one direction since the last determination of the RCC position,

OR

5.4.2 Reduce thermal power to less than 75 percent within eight hours.

5.5 **IF** two or more RPIs per bank are inoperable in Modes 1 or 2, **THEN** within 1 hour, commence power reduction to Mode 3, Hot Standby, using 3-GOP-103, POWER OPERATION TO HOT STANDBY, **AND** be in Hot Standby within the next 6 hours.

5.6 **IF** a maximum of one Group Step Demand Position Indicator per bank is inoperable, **THEN**, either;

5.6.1 Verify that all individual RCC Position Indicators (RPI) for the affected bank are operable **AND** the highest to lowest rods in the bank are within the Allowed Rod Misalignment of each other at least once per 8 hours,

OR

5.6.2 Reduce power to less than 75 percent within 8 hours.

5.7 **IF** both Group Step Demand Position Indicators per bank are inoperable in Modes 1 or 2, **THEN** within 1 hour, commence power reduction to Mode 3, Hot Standby, using 3-GOP-103, POWER OPERATION TO HOT STANDBY, **AND** be in Hot Standby within the next 6 hours.

5.8 **IF** one or more Group Demand Position Indicator is inoperable in Modes 3, 4 or 5, **THEN** open the reactor trip breakers.

5.9 **IF** either annunciators B 6/4, POWER RANGE CHANNEL DEVIATION, OR B 9/3, SHUTDOWN ROD OFF TOP DEVIATION, are inoperable, **THEN** verify the Demand Position Indication System **AND** the Analog Rod Position Indication System agree within 12 steps (18 steps with power less than 90 percent) at least once per four hours.

5.10 **IF** unit shutdown is required, **THEN** assume that the non-indicating RCC is positioned in the fully withdrawn position, unless prior incore flux traces indicated the RCC was in the inserted position.

5.11 Perform an incore flux trace before **AND** after performance of RCC exercise tests for bank containing non-indicating rod(s) to verify the RCC has returned to the associated bank position.

5.12 **WHEN** the malfunction has been corrected, **THEN** place Rod Control Selector Switch to MANUAL or AUTOMATIC position.

ROD POSITION INDICATION SYSTEM

DETAILED DESCRIPTION

BANK DEMAND POSITION INDICATION

The bank demand position indication system is more commonly known as the "step counters". The step counters count pulses from the rod control system that are used to generate rod motion. They will indicate where the rod groups should be, since each rod in a group receives the same signal to move. Each rod should be at the position indicated by the step counter for its group, however, if the rod does not move when demanded, or is dropped, the step counter will not indicate its true position. For this reason the step counters are not considered very reliable, even though the system accuracy is very high ($\pm$1 step). See Figure 1.

There are twelve group step counters on the console. They are 3-digit electronic counters, and will indicate the number of 5/8" (.625") steps demanded of that group. The counters receive pulses from the logic cabinet in the rod control system. The counters may be electrically reset to 000 by depressing the Rod Control Startup Pushbutton. They may also be manually set to any number using pushbuttons on each counter.

INDIVIDUAL ROD POSITION INDICATION

The individual rod position indication system provides continuous rod position for each of the full length RCCAs. See Figures 2, 3, & 4. The system consists of 45 rod position detectors, rod position channels, and console mounted indicators.

The rod position detectors are mounted on the CRDM rod travel housing. The position of the top of the rod drive shaft within the rod position detector produces an AC signal directly related to its position. This AC signal is transmitted to the rod position channels for processing.

The AC signal is converted into a DC signal, by the rod position channel. The signal is then displayed on console mounted meters, fed to the plant computer, rod deviation monitor, and used to generate alarms and control functions.

ROD POSITION INDICATION SYSTEM

The rod position indicating system receives power from a dedicated 5 KVA inverter which is powered from vital DC bus 3D01 (4D01), and alternate power from the 3C(4B) MCC via a constant voltage transformer. To improve reliability of the RPI system power supply, a static transfer switch, integrally mounted with the inverter, transfers load to the CVT in the event of an inverter failure. (In accordance with ONOP-028.2, when power is transferred to the CVT, the RPI indication may be different due to the voltage of the CVT.)

Problems with the RPI power supply are indicated by actuation of annunciator F-4/6, RPI Power Trouble. Separate indicating fuses protect the primary circuits of each position detector channel. These fuses are located in fuse panels on the top portion of each rack section QR 69, 70, 71, and 72.

As previously stated the RPI AC supply is rectified to DC for RPI module use. The AC is converted and arranged such that there are four internal DC power supplies in an auctioneered arrangement to furnish the positive and negative voltages required by the modules (Rod Position Channels). These internal power supplies (located in QR-70) are identified as PS-1, PS-2, PS-3 and PS-4. Failure of any of the power supplies actuates a ROD POSITION DC AUXILIARY POWER ON alarm on annunciator panel B window 3/3.

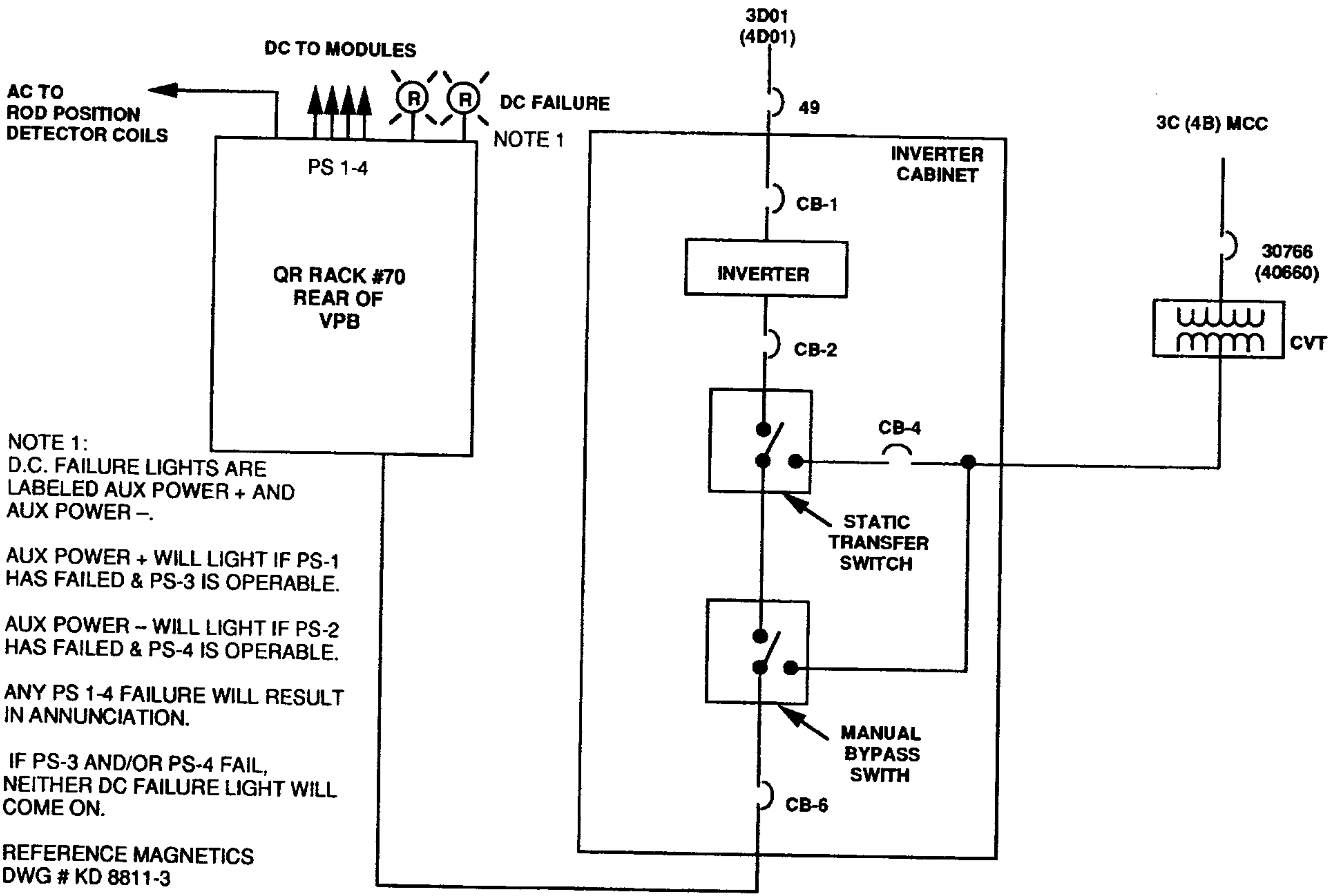
The fuse panel of RPI Rack #2 (QR-70) has two indicator lamps (AUX PWR POS and AUX PWR NEG) that can be used to help determine the power supply failure. If PS-1 or PS-2 fails, the appropriate lamp (POS or NEG) will illuminate. If PS-3 or PS-4 has a failure the lamps will not illuminate.

ROD POSITION DETECTOR

The rod position detector is a linear variable transformer consisting of 72 primary and secondary coils alternately stacked on a stainless steel support tube. The rod drive shaft serves as the transformer core. The primary and secondary coil stacks are series connected.

A stainless steel bottom plate is welded to the inner support tube with a bolt on top plate. This provides a completely self contained and self-supporting rod position detector. To provide maximum cooling the outer winding of the coil stacks are exposed to ambient cooling air. Also the inner surface of the support tube is dimpled to allow minimum contact with the rod travel housing maintaining centering.

RPI POWER SUPPLY



NOTE 1:
D.C. FAILURE LIGHTS ARE LABELED AUX POWER + AND AUX POWER –.

AUX POWER + WILL LIGHT IF PS-1 HAS FAILED & PS-3 IS OPERABLE.

AUX POWER – WILL LIGHT IF PS-2 HAS FAILED & PS-4 IS OPERABLE.

ANY PS 1-4 FAILURE WILL RESULT IN ANNUNCIATION.

IF PS-3 AND/OR PS-4 FAIL, NEITHER DC FAILURE LIGHT WILL COME ON.

REFERENCE MAGNETICS DWG # KD 8811-3

FIGURE 5
Rev.2:3/1/96

5.0 SUBSEQUENT ACTIONS

5.1 Place the Rod Motion Control Selector to the MAN position.

5.2 Proceed as follows:

5.2.1 **IF** reactor power is greater than 75 percent **OR** more than one RCC is misaligned, **THEN**:

1. Borate/dilute AND/OR change Turbine load to maintain Tavg within 3°F of Tref.

2. Use RCC motion only to control axial flux within the target band.

NOTE

If increasing Reactor power to approximately 3 percent (stay in Mode 2) for a flux map, rod motion may be used.

3. Do **NOT** withdraw control rods to increase power until the RCC(s) have been aligned, except as noted above.

5.2.2 **IF** reactor power is less than 75 percent, **AND** one RCC is misaligned, **THEN** at the discretion of the Reactor Engineering Supervisor use rod motion **OR** boration/dilution to limit reactor power to less than 75 percent.

5.3 Notify the Reactor Engineering Supervisor or designee, **AND** provide the following information:

5.3.1 Amount of time the RCC(s) has (have) been misaligned.

5.3.2 Degree of misalignment.

5.3.3 Current reactor status (i.e., rods in manual, reactor power, and RCC position).

NOTE

RPI gain adjustments should be delayed until 0-OSP-059.14 Rod Position Indication (RPI) verification is performed.

5.4 Notify the I&C Supervisor to verify RPI indication **AND** to investigate CRDM system for possible failure.

CAUTION

When more than one RCCA in the same bank is suspected of being misaligned from its step demand counter, flux traces are required to be performed separately for each RCCA. Upon declaring the RCCA not misaligned, the associated RPI must be declared operable prior to performing the next flux trace, to prevent an unnecessary entry into Tech Spec action 3.0.3.

5.5 Have the Reactor Engineering Supervisor perform one or more of the following, at the discretion of the Reactor Engineering Supervisor, to confirm RCC misalignment or RCC position indication malfunction:

5.5.1 Perform 3-OSP-059.10, DETERMINATION OF QUADRANT POWER TILT RATIO.

5.5.2 Check Δ Flux meters.

5.5.3 Check core exit thermocouples.

5.5.4 Two Thimble Flux Map for symmetry check, **OR** to estimate RCC position

1. Visual
2. Computer check of digital data.

5.5.5 Full Core Flux Map for verification of core power distribution.

5.5.6 Stepping of RCCs that are nearly or fully withdrawn.

5.5.7 Rod Deviation/Axial Flux Panel

5.6 **IF** the RCC(s) is determined to be aligned with the associated bank, **AND** an RCC position indicator(s) is failed, **THEN** go to 3-ONOP-028.2, RCC Position Indication Malfunction.

5.7 **IF** more than one RCC is inoperable or misaligned from the group demand step counter position by more than the Allowed Rod Misalignment of Technical Specification 3.1.3.1, **THEN** comply with the actions of Technical Specification 3.1.3.1.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group step counter demand position indicator shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown and control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **

ACTION:

With less than the above required group step counter demand position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each of the above required group step counter demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days.

4.1.3.3.2 OPERABILITY of the group step counter demand position indicator shall be verified in accordance with Table 4.1-1.

* With the Reactor Trip System breakers in the closed position.
** See Special Test Exceptions Specification 3.10.5.

1. Given the following conditions:

- The plant is in Mode 3 with all Shutdown Rods withdrawn.
- The Digital Rod Position Indication emergency power supply is under clearance.
- The normal Digital Rod Position Indication power supply has just tripped.

Which of the following actions is to be taken?

A.✓ Due to the loss of all Digital Rod Position Indication, open the Reactor Trip Breakers in accordance with Technical Specification 3.1.3.3, "Position Indication System – Shutdown"

B. Due to the loss of all Digital Rod Position Indication and Demand Position Indication, verify that all Shutdown Bank Rods are fully withdrawn using the movable incore detectors in accordance with AOP-001, "Malfunction of Rod Control"

C. Due to the loss of all Digital Rod Position Indication and Demand Position Indication, commence a boration of the RCS to ensure adequate Shutdown Margin in accordance with AOP-002, "Emergency Boration"

D. Due to the loss of all Digital Rod Position Indication, verify that all Shutdown Bank Rods are fully withdrawn using Demand Position Indication in accordance with AOP-001, "Malfunction of Rod Control"

QUESTIONS REPORT
for Draft TP05-301-SRO

4. 015AG2.4.49 001//T1G1/RCP/C/A(4.0/4.0)/M/TP05301/S/MC

Unit 3 has been operating at 100% power for three days when the following conditions occur:

- Annunciator A-1/5, RCP SEAL LEAK-OFF HI FLOW, has lit
- Annunciator G-2/2, RCP B STANDPIPE HI LEVEL, has lit
- RCP B seal injection flow is 7.8 gpm
- RCP B seal leak-off flow is 6 gpm
- Seal return temperature is 150^{o}F and rising steadily

Based on the above indications, the crew enters ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL. IAW ONOP-041.1, the SRO should direct the operating crew to perform which ONE of the following?

A. RCP B operation may continue for up to 24 hours due to number two seal sticking.

B.✓ Manually trip the reactor, then stop RCP B and close its seal leakoff valve after the pump stops.

C. Commence unit shutdown using ONOP-100, FAST LOAD REDUCTION, when the turbine is tripped, then trip the reactor, when the reactor is tripped, then stop RCP B.

D. Begin preparations to shutdown and stop RCP B using GOP-103, POWER OPERATION TO HOT STANDBY and contact plant management for further guidance.

Feedback

REFERENCES:

1. ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL, pages 7,12,13,23,30, rev 06/14/99C1
2. ARP-097CR, CONTROL ROOM ANNUNCIATOR RESPONSE, pages 25,352, rev 06/09/03

DISTRACTORS:

A Incorrect. This would be true on initial startup (or for 8 hours during normal operation).

B Correct. IAW ref 1.

C Incorrect. This would be true if seal flow was less than 6 gpm but greater than 5.5 gpm.

D Incorrect. This would be true if seal flow was less than 5.5 gpm.

K/A CATALOGUE QUESTION DESCRIPTION:

- Reactor Coolant Pump Malfunction; Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

4. 015AG2.4.49 001//T1G1/RCP/C/A(4.0/4.0)/M/TP05301/S/MC

Unit 3 is operating at 100% power for three days.

- Annunciator A-1/5, RCP SEAL LEAK-OFF HI FLOW, has lit
- Annunciator G-2/2, RCP B STANDPIPE HI LEVEL, has lit
- RCP B seal injection flow is 7.8 gpm
- RCP B seal leak-off flow is 6 gpm
- Seal return temperature is 150^{O}F and rising steadily

Based on the above indications, you should direct the operating crew to perform which ONE of the following?

A. RCP B operation may continue for up to 24 hours due to number two seal sticking.

B.✓ Manually trip the reactor, then stop RCP B and close its seal leakoff valve after the pump stops.

C. Commence unit shutdown using ONOP-100, FAST LOAD REDUCTION, when the turbine is tripped, then trip the reactor, when the reactor is tripped, then stop RCP B.

D. Begin preparations to shutdown and stop RCP B using GOP-103, POWER OPERATION TO HOT STANDBY and contact plant management for further guidance.

Feedback

REFERENCES:

1. ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL, pages 7,12,13,23,30, rev 06/14/99C1
2. ARP-097CR, CONTROL ROOM ANNUNCIATOR RESPONSE, pages 25,352, rev 06/09/03

DISTRACTORS:

A Incorrect. This would be true on initial startup (or for 8 hours during normal operation).

B Correct. IAW ref 1.

C Incorrect. This would be true if seal flow was less than 6 gpm but greater than 5.5 gpm.

D Incorrect. This would be true if seal flow was less than 5.5 gpm.

K/A CATALOGUE QUESTION DESCRIPTION:

- Reactor Coolant Pump Malfunction; Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Containment entries shall NOT be performed when there are indications of an RCP seal package failure until the reactor is shutdown and RCS pressure/temperature is reduced to minimize leakage.

NOTE

- *Foldout Page is required to be monitored throughout this procedure.*
- *A time delay exists on TR-320 from when an RCP parameter exceeds its setpoint to when the recorder provides indication and alarm. The RCP mimic display on ERDADS is a useful backup to TR-320 for monitoring affected RCP parameters.*
- *RCP Motor Lower Guide Bearing High Temperature Alarm Setpoints may have been reset to a lower value than the normal 185°F. 3-ARP-097.CR, CONTROL ROOM ANNUNCIATOR RESPONSE, provides references for these changes.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check For Proper Seal Injection Flow** • RCP 3A Thermal Barrier ΔP, PI-3-131A - GREATER THAN ZERO INCHES • RCP 3B Thermal Barrier ΔP, PI-3-128A - GREATER THAN ZERO INCHES • RCP 3C Thermal Barrier ΔP, PI-3-125A - GREATER THAN ZERO INCHES • Local Seal Injection Flow Indication - GREATER THAN <u>OR</u> EQUAL TO 6 GPM ON ALL RCPs ✓YES • ERDADS Seal Injection Flow Indication - GREATER THAN <u>OR</u> EQUAL TO 6 GPM ON ALL RCPs	Go to Step 14
2	**Check Number One Seal Leakoff Flow Within Limits Of Enclosure 1** NO →	Observe NOTE prior to Step 16 **AND** go to Step 16.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

When seal injection is lost, increases in the number one seal leakoff and lower pump guide bearing temperatures may be indicative of a loss of CCW cooling to the thermal barrier.

15 **Maintain Thermal Barrier Cooling**

a. Monitor number one seal leakoff **AND** lower pump guide bearing temperatures

b. Maintain the following:

- RCP Thermal Barrier Return CCW Flow FI-3-626 – GREATER THAN 75 GPM
- CCW to RCP Oil Coolers **AND** Thermal Barrier temperature, TI-3-607B - LESS THAN 105°F

RESPONSE NOT OBTAINED: b. Perform 3-ONOP-030, COMPONENT COOLING WATER MALFUNCTION, to restore cooling to thermal barrier.

c. Return to Step 2

NOTE

An RCP STANDPIPE HI LEVEL alarm is indication of 0.5 gpm flow past the number two seal.

16 **Check If Any RCP Number One Seal Leak-off Flow(s), FR-3-154A - GREATER THAN UPPER LIMIT OF ENCLOSURE 1** YES

RESPONSE NOT OBTAINED: Go to Step 21

17 **Check RCP Seal Bypass Valve CV-3-307 - CLOSED** YES

RESPONSE NOT OBTAINED: Perform the following:

a. Manually close CV-3-307

b. Check for corresponding decrease in thermal barrier ΔP

c. Perform cross check of all RCP parameters to determine cause of high leakoff flow

d. Request diagnostic assistance from the System Engineer **AND** Operations Supervision

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18	**Check All RCP Number One Seal Leak-Off Flows On FR-3-154A – LESS THAN 6 GPM** NO →	correct Perform the following: a. 1 Manually trip the reactor **AND** perform 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure. b. 2 **WHEN** the reactor verified tripped, **THEN** stop the affected RCP(s) c. 3 Close affected RCP Seal Leakoff valve(s) after the pump has stopped: * CV-3-303A for RCP A * CV-3-303B for RCP B * CV-3-303C for RCP C d. Monitor RCDT level for indication of number two seal failure. e. DO **NOT** restart the affected RCP until the cause of the seal malfunction has been determined **AND** corrected. f. Return to Step 3.
19	**Check All RCP Number One Seal Leak-Off Flows On FR-3-154A** a. RCP number one seal leak-off flow - LESS THAN **OR** EQUAL TO 5.5 GPM distractor b. Begin preparations to shutdown **AND** stop affected RCP using 3-GOP-103, POWER OPERATION TO HOT STANDBY c. Contact Plant Management for further guidance	distractor a. Perform the following: 1) Commence unit shutdown using 3-ONOP-100, FAST LOAD REDUCTION. 2) **WHEN** turbine tripped, **THEN** trip the reactor. 3) **WHEN** the reactor is tripped, **THEN** stop affected RCP(s). 4) Go to Step 19c.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

distractor

CAUTION

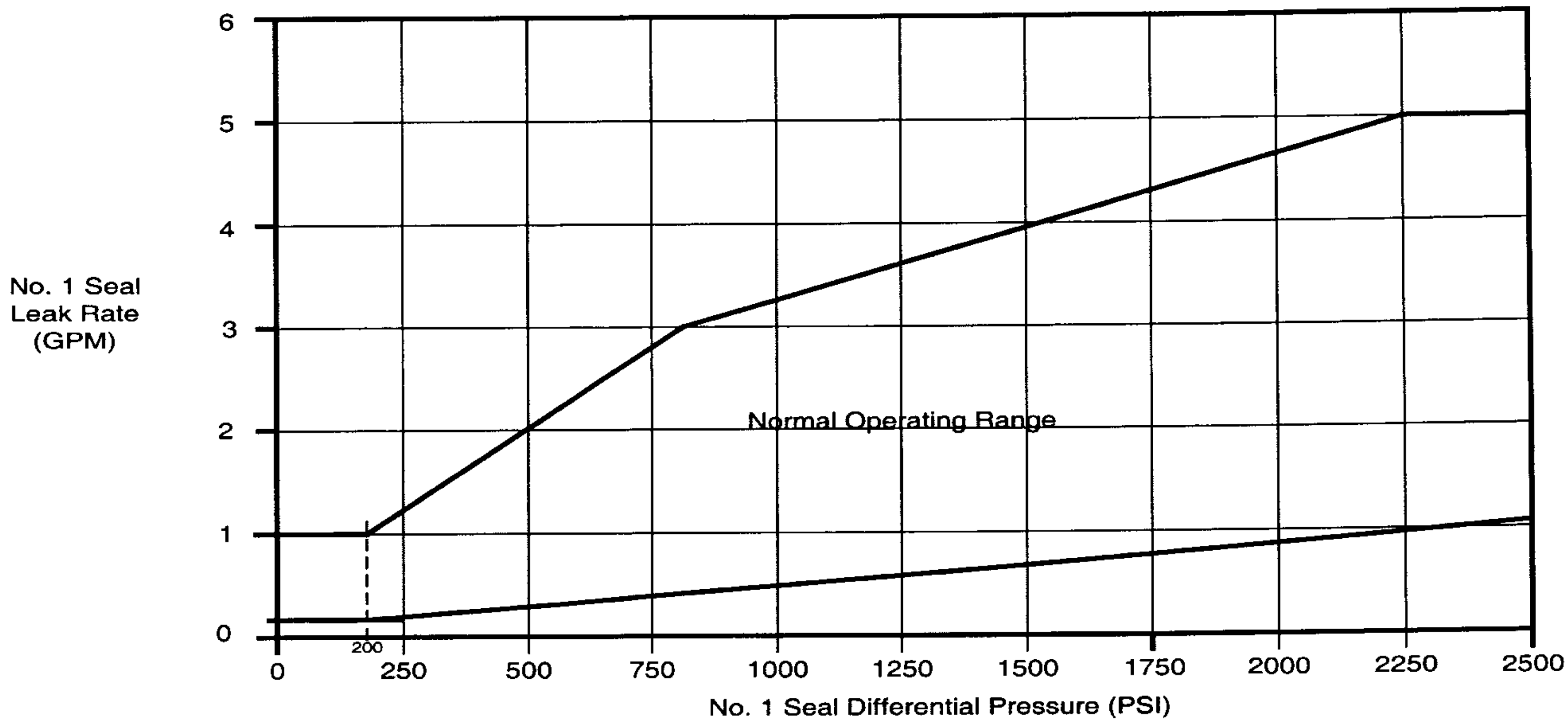
If number two seal high flow is due to number two seal sticking, RCP operation may continue. RCP operation for short period (up to 24 hours on the initial RCP startup or 8 hours during normal operation) may be required before number two seal seats properly.

NOTE

An increase in number two seal leakage of 0.5 gpm or greater is sufficient to give a high standpipe alarm. This should be coincident with a decrease in number one seal leakoff flow of 0.5 gpm or greater.

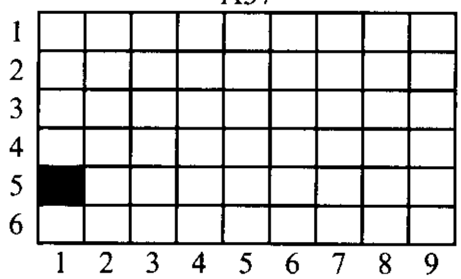
STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
37	**Monitor RCDT At The Waste Disposal/Boron Recovery Panel To Verify RCDT Is Capable Of Handling The Increased Leakage**	Perform the following: a. Manually trip reactor **AND** perform 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure. b. **WHEN** the reactor has been verified tripped, **THEN** stop the affected RCP(s).
38	**Check RCS Leakage Rate Using 3-OSP-041.1, REACTOR COOLANT SYSTEM LEAK RATE CALCULATION - LESS THAN 30 GPM**	Shutdown unit using 3-GOP-103, POWER OPERATION TO HOT STANDBY, **AND** 3-GOP-305, HOT STANDBY TO COLD SHUTDOWN.

ENCLOSURE 1
(Page 1 of 1)
NUMBER ONE SEAL LEAKOFF



BLUE	INVESTMENT PROTECTION	A 1/5

A37



ATTACHMENT 1
Page 5 of 54
Panel A

RCP
SEAL LEAK-OFF
HI FLOW

DEVICES:

FC-3-156A (RCP A)
FC-3-155A (RCP B)
FC-3-154A (RCP C)

SETPOINTS:

5.0 GPM

OPERATOR ACTIONS:

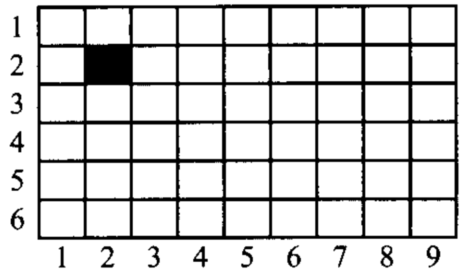
1. Verify alarm by checking the following:
   a. Check seal leak-off greater than 5 GPM as indicated on FR-3-154A (VPA) or ERDADS RCP display.
   b. Check charging flow / seal injection flow normal.
   c. Check VCT temperature, TE-3-116 normal.
2. Corrective actions:
   a. Refer to 3-ONOP-041.1, Reactor Coolant Pump Off-Normal.

CAUSES:
1. Damaged #1 seal.
2. Insufficient / no seal injection
3. High VCT temperature.

REFERENCES:
1. FPL Dwg 5613-M-3047, Sh 3, CVCS – Seal Water Injection to RCP
2. Tech Spec 3/4.4.1.1, 3/4.4.1.2, 3/4.4.1.3

WHITE	STATUS / INFORMATION	G 2/2

G11



ATTACHMENT 7
Page 8 of 54
Panel G

RCP B
STANDPIPE
HI LEVEL

DEVICES:

LC-3-407A

SETPOINTS:

1' above normal

OPERATOR ACTIONS:

1. Verify alarm by checking the following:
   a. RCP temperature recorder TR-320.
   b. RCP seal leakoff flows.
   c. Position of makeup valves CV-3-519A and CV-3-522B.

2. Corrective actions:
   a. Verify standpipe makeup valves CV-3-519A and CV-3-522B are closed.
   b. Monitor RCP parameters **AND** go to 3-ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL

CAUSES:
1. #2 RCP seal damage.
2. CV-3-519A and CV-3-522B leaking by.

REFERENCES:
1. FPL Dwg 5613-M-3041, Sh 3

QUESTIONS REPORT

for Westinghouse 3 Loop Questions

1. 015G2.1.32 001//T2G1/RCP/C/A 3.4/3.8/BANK/NA02301/S/RA/LSM

attachment 3

Given the following conditions:

- Unit 1 is at 100% power.
- C-G2, "RCP 1B STANDPIPE HI LEVEL" has lit. step 1 >6 → step 2 → step 16 → step 21
- "B" seal injection flow is 7.8 GPM. step 2 <6 No → 16 >6 → step 21 → step 3
- "B" seal leak-off flow is 0.2 GPM. 6gpm
- Seal return temperature is 150°F and rising steadily. <190 ✓ step 37 ← step 6
- "B" RCP temperatures are as follows:
  - Motor stator - 242°F
  - Motor upper bearing - steady at 115°F
  - Motor lower bearing - steady at 129°F
  - Pump radial bearing - rising slowly at 129°F

Based on the above indications, you should direct the operating crew to perform which one of the following?

A.✓ Trip the unit, secure "B" RCP and close its no. 1 seal leakoff valve after the pump stops.

B. Trip the "B" RCP and close its no. 1 seal leakoff valve after the pump stops.

C. Close the "B" RCP's no. 1 seal leakoff valve within 5 minutes and shutdown the unit within the next 30 minutes then secure the "B" RCP.

D. Trip the unit, secure "B" RCP and contact OMOC.

Reference required - 1-AP-33.1

BANK 1996 VISION OBJ. 11101
1996/01/29

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

Created: Thursday, May 27, 2004
Modified: Thursday, May 27, 2004
Revised:

RO Tier:		SRO Tier:	T2G1
K/A Value:	RCP	Cog. Level:	C/A 3.4/3.8
Source:	BANK	Exam:	NA02301
Test:	S	Misc:	RA/LSM

QUESTIONS REPORT

for Westinghouse 3 Loop Questions

1. 038EK3.08 002/T1G2/T1G2/RCP TRIP CRITERIA/C/A(4.1/4.2)/M/SM02301/C/GWL

The following conditions exist:

-A SGTR is in progress on the 'B' S/G.
-The Crew has implemented EOP-4.0 "Steam Generator Tube Rupture."
-FI-943 indicates 200 gpm.
-The crew is at the step for determining the required core exit thermocouples for cool down.
-The Reactor Operator reports that RCS pressure has reached 1340 psig.

Which ONE of the following describes what action should be taken next and why?

A. Trip all RCP's. RCP's should be tripped anytime during EOP-4.0 if the trip criteria is met.

B. Do not trip RCP's. Trip criteria does not apply and a controlled cooldown is imminent.

C.✓ Trip all RCP's. the trip criteria has been met and injection flow has been verified.

D. Do not trip RCP's. RCP trip criteria only applies prior to isolation of the steam generator.

Modified Bank Question # 292 open reference bank.
Lesson Plan EOP-4.0 objective # 1919.

A. Incorrect, RCP's should be tripped but the trip criteria only applies prior to a operator controlled cooldown.

B. Incorrect, the trip criteria is met, and a cooldown has not be commenced.

C. Correct, the trip criteria is met and Injection flow has been verified, and an operator controlled cooldown has not been started.

D. Incorrect, the RCP trip criteria applies up until the point that a controlled cooldown has be started.

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

Created: Thursday, May 27, 2004
Modified: Friday, October 15, 2004
Revised:

1. 012A2.02 001/T2G2/T2G2/LOSS OF RCP/M 3.6/3.9/B/SR02301/S/GWL

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

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

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

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

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

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

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

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

Created: Thursday, May 27, 2004
Modified: Wednesday, November 10, 2004
Revised:

RO Tier:	T2G2	SRO Tier:	T2G2
K/A Value:	LOSS OF RCP	Cog. Level:	M 3.6/3.9
Source:	B	Exam:	SR02301
Test:	S	Misc:	GWL

Test Name	Test Date	rpb	p(Diff)	Time	Equ	User Values	
<Cumulative>		0.000	1.000	0	N	1: 0	2: 0
SRO FINAL 2002 Written Examination	03/12/2002	0.000	1.000	0	N	3: 0	4: 0
SRO FINAL 2002 Written Examination	03/12/2002	0.000	1.000	0	N		

	--- A ---			--- B ---			--- C ---			--- D ---					
	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg
<Cumulative> Total: 12 100 Omits: 0 0	0	0	0.00	0	0	0.00	0	0	0.00	12	100	87.00			
SRO FINAL 2002 Written Examination Total: 6 100 Omits: 0 0	0	0	0.00	0	0	0.00	0	0	0.00	6	100	86.67			
SRO FINAL 2002 Written Examination Total: 6 100 Omits: 0 0	0	0	0.00	0	0	0.00	0	0	0.00	6	100	87.33			

QUESTIONS REPORT
for Draft TP05-301-SRO

5. 027AG2.2.25 001//T1G1/PZR PRESS/C/A(2.5/3.7)/M/TP05301/S/MC

Unit 4 is operating at 100% power on February 25, 2005, at 0930, when pressurizer pressure instrument PT-455 fails low. At 1045 the crew verifies that all procedural and Technical Specifications (TS) requirements associated with the failure of PT-455 have been completed. At 1145 the crew determines that pressurizer pressure instrument PT-456 fails to 2300 psig.

Based on the above plant conditions, which ONE of the following describes the actions(s) that must be performed to satisfy Technical Specifications and why?

A. Be in at least Mode 3 within 6 hours; permits the time limits of the ACTION requirements to be reset to the point in time where the plant entered the new Mode to allow completion of remedial measures and a return to POWER.

B.✓ Be in at least Mode 3 within 7 hours; permits shutdown in a controlled and orderly manner.

C. Be in at least Mode 4 within 12 hours; permits shutdown in a controlled and orderly manner.

D. Be in at least Mode 4 within 13 hours; permits the time limits of the ACTION requirements to be reset to the point in time where the plant entered the new Mode to allow completion of remedial measures and a return to POWER.

Feedback

REFERENCES:

1. Tech Specs 3/4.3.1, REACTOR TRIP SYSTEM INSTRUMENTATION, Table 3.3-1, Amendment NOS. 140 & 135
2. Tech Specs 3/4.0, APPLICABILITY, 3.0.3, Amendment NOS. 137 & 132
3. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, pages 22,23, rev 05/01/03

DISTRACTORS:

A Incorrect. Wrong time requirement and reason.

B Correct. IAW 3.0.3, within 1 hour action shall be initiated to place the unit in: (1) at least HOT STANDBY within the next 6 hours (for a total of 7), (2) at least HOT SHUTDOWN within the following 6 hours (for a total of 13), (3) at least COLD SHUTDOWN within the subsequent 24 hours (for a total of 37)..

C Incorrect. Wrong time requirement.

D Incorrect. Wrong reason.

K/A CATALOGUE QUESTION DESCRIPTION:

- Pressurizer Pressure Control System Malfunction; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

5. 027AG2.2.25 001//T1G1/PZR PRESS/C/A(2.5/3.7)/M/TP05301/S/MC

Unit 4 is operating at 100% power with pressurizer pressure instrument PT-455 failed low. All required actions have been completed with PT-455 channel in trip.

- Pressurizer pressure instrument PT-456 fails to 2300 psig

From the time PT-456 fails, which ONE of the following describes the actions(s) that must be performed to satisfy Technical Specifications and why?

A. Be in at least Mode 3 within 6 hours; permits the time limits of the ACTION requirements to be reset to the point in time where the plant entered the new Mode to allow completion of remedial measures and a return to POWER.

B.✓ Be in at least Mode 3 within 7 hours; permits shutdown in a controlled and orderly manner.

C. Be in at least Mode 4 within 12 hours; permits shutdown in a controlled and orderly manner.

D. Be in at least Mode 4 within 13 hours; permits the time limits of the ACTION requirements to be reset to the point in time where the plant entered the new Mode to allow completion of remedial measures and a return to POWER.

Feedback

REFERENCES:

1. Tech Specs 3/4.3.1, REACTOR TRIP SYSTEM INSTRUMENTATION, Table 3.3-1, Amendment NOS. 140 & 135
2. Tech Specs 3/4.0, APPLICABILITY, 3.0.3, Amendment NOS. 137 & 132
3. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, pages 22,23, rev 05/01/03

DISTRACTORS:

A Incorrect. Wrong time requirement and reason.

B Correct. IAW 3.0.3, within 1 hour action shall be initiated to place the unit in: (1) at least HOT STANDBY within the next 6 hours (for a total of 7), (2) at least HOT SHUTDOWN within the following 6 hours (for a total of 13), (3) at least COLD SHUTDOWN within the subsequent 24 hours (for a total of 37)..

C Incorrect. Wrong time requirement.

D Incorrect. Wrong reason.

K/A CATALOGUE QUESTION DESCRIPTION:

- Pressurizer Pressure Control System Malfunction; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

ATTACHMENT 1
(Page 12 of 102)

TECHNICAL SPECIFICATION BASES

3/4.0 APPLICABILITY (Continued)

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

ATTACHMENT 1
(Page 13 of 102)

TECHNICAL SPECIFICATION BASES

3/4.0 <u>APPLICABILITY</u> (Continued)

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

ATTACHMENT 1
(Page 14 of 102)

TECHNICAL SPECIFICATION BASES

3/4.0 APPLICABILITY (Continued)

Specification 3.0.4 establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

Specification 3.0.5 delineates the applicability of each specification to Unit 3 and Unit 4 operation.

Specification 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirement specified in Table 4.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	13
6. Overpower ΔT	3	2	2	1, 2	13
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	13
10. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6

TABLE 3.3-1 (Continued)

TABLE NOTATION

* When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

** When the Reactor Trip System breakers are in the open position, one or both of the backup NIS instrumentation channels may be used to satisfy this requirement. For backup NIS testing requirements, see Specification 3/4.3.3.3, ACCIDENT MONITORING.

*** Reactor Coolant Pump breaker A is tripped by underfrequency sensor UF-3A1(UF-4A1) or UF-3B1(UF-4B1). Reactor Coolant Pump breakers B and C are tripped by underfrequency sensor UF-3A2(UF-4A2) or UF-3B2(UF-4B2).

Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

possible alternate distractor

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 10- With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable, except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ACTUATION LOGIC TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 13 - With the number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

QUESTIONS REPORT

for Westinghouse 3 Loop Questions

1. 027AG2.2.22 001/1/1//C/A 3.4/4.1/N/FA03301/S/SDR

Unit 1 is operating at full power with Pressurizer Pressure Instrument PT-455 failed low. All required actions have been completed with PT-455 channel in trip. Pressurizer Pressure instrument, PT-456 fails to 2300 psig.

Which ONE of the following describes the action(s) that must be performed to satisfy Technical Specifications?

REFERENCE PROVIDED

A. Place PT-456 channel in trip within the next 6 hours.

B. Shutdown the plant to Mode 3 within 12 hours.

C. Reduce THERMAL POWER to < P-7 within 12 hours.

D.✓ Shutdown the plant to Mode 3 within 7 hours; Mode 4 within 13 hours; and Mode 5 within 37 hours.

Reference: TECH SPEC 3.3.1/3.3.2

A - Incorrect; This is the Action 'E.1' of TS 3.3.1, for one channel inoperable, and is the correct action if this were the only pressurizer pressure channel failed however, with two channels failed placing this channel in trip will cause a reactor trip.

B - Incorrect; This is the alternative Action for 'E.1', Action 'E.2' of TS 3.3.1, and is an action for one channel inoperable.

C - Incorrect; This is the alternative Action for 'M.1' , Action 'M.2' of TS 3.3.1, Tripping the bistable action of 'M.1' was already performed for the PT-455 failure. This is an action for one channel being inoperable.

D - Correct; There is no condition stated in LCO 3.3.1 for these concurrent failures, therefore LCO 3.0.3 must be entered and the plant shutdown.

This question meets the requirements of 10CFR55.43(b)(2) for an SRO only question.

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

Created: Thursday, May 27, 2004
Modified: Thursday, May 27, 2004
Revised:

6. 033GG2.4.30 001//T2G2/SFPCS/C/A(2.2/3.6)/N/TP05301/S/MC

Unit 4 is in Mode 6. On January 29th at 0900 the water level in the Spent Fuel Pool was lowered to 56' to perform maintenance. All movement of fuel assemblies and crane operation in the fuel storage area had been suspended at 0700 the same day. Maintenance was scheduled for completion with level restored by February 5th at 0900.

- Maintenance was completed and level restored on February 6th at 1000

Which ONE of the following correctly completes the statement:

This event is ___(1)___ to the NRC because ___(2)___.

A. Reportable; SFP level remained below the minimum for more than 24 hours beyond originally scheduled.

B.✓ Reportable; SFP level remained below the minimum for more than 7 days.

C. Not Reportable; SFP level was not lowered below the minimum level required by Tech Specs.

D. Not Reportable; the amount of time the SFP level remained below the minimum did not exceed the maximum time allowed by Tech Specs.

Feedback

REFERENCES:

1. Tech Specs, 3.9.11, REFUELING OPERATIONS – WATER LEVEL – STORAGE POOL, page 9-12, Amendment Nos. 224 & 219

DISTRACTORS:

A Incorrect. The time exceeded 7 days.

B Correct. SFP level shall be maintained greater than or equal to elevation 56' – 10". If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

C Incorrect. Level was lowered below the minimum required by Tech Specs.

D Incorrect. The amount of time the SFP level remained below the minimum exceeded the maximum time allowed by Tech Specs.

K/A CATALOGUE QUESTION DESCRIPTION:

- Spent Fuel Pool Cooling System (SFPCS); Knowledge of which events related to system operations/status should be reported to outside agencies.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The water level shall be maintained greater than or equal to elevation 56' - 10' the spent fuel storage pool.*

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

*The requirements of this specification may be suspended for more than 4 hours hours to perform maintenance provided a 10 CFR 50.59 evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

..008.2.4.30 7/17/2002

Braidwood 1 **Exam Level** S

Mark Question | Print Recor | New Search | Exit

Question

Unit 1 is in Mode 3
RCS pressure control was lost resulting in RCS pressure peaking at 2500 psig.
Both Pzr PORVs and 1 Pzr Safety valve opened, then closed.
Operators have subsequently stabilized RCS pressure at 2235 psig.

This event is _____(1)_____ because _____(2)_____.

_____(1)_____ _____(2)_____

Answer:	Reportable	The Pzr PORVs and Safeties were challenged
Distracter 1	Reportable	Only the Pzr Safety Valve was challenged
Distracter 2	Not Reportable	RCS pressure did not exceed the safety limit
Distracter 3	Not Reportable	The PORVs and Safety closed after opening

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

QUESTIONS REPORT
for Westinghouse 3 Loop Questions

1. Which ONE of the following identifies an event that is required to be reported to the NRC within 1 hour of discovery?
(Reference provided)

A. An inadvertant Safety Injection due to an instrument surveillance error.

B. The Shift Supervisor authorizes the individual insertion of control rods into the core without bank overlap to shutdown the reactor in an emergency.

C. A hypochlorite spill outside the Polishing Building of which the EPA has been notified.

D. A radioactive release such that if an individual had been present for 24 hours, they could have received an intake in excess of one occupational annual limit on intake.

7. 039GG2.1.32 001//T2G1/MRSS/M(3.4/3.8)/N/TP05301/S/MC

Unit 4 is operating at 100% power when "A" MSIV is found to be inoperable in the OPEN position at 1400, 02-21-05.

Which ONE of the following describes the actions to be taken and why?

A. Be in MODE 3 by 0200, 02-22-05. Minimizes the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown in the event of a steam line rupture.

B. Be in MODE 3 by 1400, 02-22-05. Minimizes the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown in the event of a steam line rupture.

C✓ Be in MODE 3 by 2000, 02-22-05. Limits the pressure rise within containment in the event a steam line rupture occurs within containment.

D. Be in MODE 3 by 2000, 02-21-05. Limits the pressure rise within containment in the event a steam line rupture occurs within containment.

Feedback

REFERENCES:

1. Tech Specs, 3.7.1.5, MAIN STEAM ISOLATION VALVES, page 7-10, Amendment Nos. 137 & 132
2. 0-ADM-536, TECHNICAL SPECIFICATION BASES CONTROL PROGRAM, page 85, rev 05/01/03

DISTRACTORS:

A Incorrect. Action is incorrect. Basis is correct.

B Incorrect. Action is incorrect. Basis is correct.

C Correct. "With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. The OPERABILITY of the MSIVs ensures that no more than one steam generator will blow down in the event of a steam line rupture." "This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment."

D Incorrect. Action is incorrect. Basis is correct.

K/A CATALOGUE QUESTION DESCRIPTION:

- Main and Reheat Steam; Ability to explain and apply all system limits and precautions.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

TECHNICAL SPECIFICATION BASES

3/4.7 PLANT SYSTEMS (Continued)

3/4.7.1.4 SPECIFIC ACTIVITY

The limit on secondary coolant specific activity is based on a postulated release of secondary coolant equivalent to the contents of three steam generators to the atmosphere due to a net load rejection. The limiting dose for this case would result from radioactive iodine in the secondary coolant. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary making allowance for plate-out and retention in water droplets. The inhalation thyroid dose at the site boundary is then;

Dose (Rem) = C * V * B * DCF * X/Q * 0.1

Where: C = secondary coolant dose equivalent I-131 specific activity

= 0.2 curies/ m^3 (*μCi/cc) or 0.1 Ci/m^3 , each unit

V = equivalent secondary coolant volume released = 214 m^3

B = breathing rate = 3.47×10^{-4} m^3/sec.

X/Q = atmospheric dispersion parameter = 1.54×10^{-4} sec/m^3

0.1 = equivalent fraction of activity released

DCF = dose conversion factor, Rem/Ci

The resultant thyroid dose is less than 1.5 Rem.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses. The 24–hour action time provides a reasonable amount of time to troubleshoot and repair the backup air and/or nitrogen system.

QUESTIONS REPORT

for Westinghouse 3 Loop Questions

1. 003AG2.1.32 001//T1G1/3.4/3.8/MEMORY/NEW/TP00301/SRO/SDR/TP

A dropped rod at 100% power has caused the Quadrant Power Tilt Ratio (QPTR) to be 1.03.

Which one of the following describes the reason for allowing up to two hours before QPTR or reactor power must be reduced?

To allow time:

A.✓ to retrieve the dropped rod.

B. to perform a calorimetric calculation.

C. for the NPS to notify the NRC.

D. for the NIS High flux trip setpoints to be reduced.

A

REFERENCE:

Tech. Spec. 3.2.4 Basis

1. 040G2.1.32 090//////HR04301/S/

Given the following conditions:

- A steam break has occurred on SG 'A'.
- Following a Reactor Trip and Safety Injection, a transition has been made to EPP-015, "Uncontrolled Depressurization of All Steam Generators."

Why should the Unit-SCO direct the operators to attempt to close the Main Steam Isolation Valves before taking any other actions to isolate the SGs?

A.✓ To minimize the positive reactivity effects associated with the cooldown of the RCS due to more than one (1) SG blowing down **AND** to minimize the pressure rise inside Containment in the event the steam break is inside Containment

B. To minimize the positive reactivity effects associated with the cooldown of the RCS due to more than one (1) SG blowing down **AND** to ensure an adequate supply of AFW in the Condensate Storage Tank in the event of a subsequent loss of all AC power

C. To ensure an adequate supply of AFW in the Condensate Storage Tank in the event of a subsequent loss of all AC power **AND** to limit the likelihood of a radiological release to the environment in the event of a subsequent SGTR

D. To minimize the pressure rise inside Containment in the event the steam break is inside Containment **AND** to limit the likelihood of a radiological release to the environment in the event of a subsequent SGTR

1. G2.1.32 001/T3/T3/3.4/3.8/C/A/BANK/FA01301/COM (66)/SDR

In EEP-2, "FAULTED STEAM GENERATOR ISOLATION," the operator is cautioned that any faulted steam generator should remain isolated during subsequent recovery actions unless needed as a heat sink for RCS cooldown.

Which ONE of the following is the reason for this caution?

A. AFW pumps could reach run-out flow and cavitate causing damage to the pumps and possibly rendering them inoperable.

B. Additional steaming from the S/G will increase the likelihood of damaging other equipment, power supplies, or instrumentation in the vicinity of the break.

C.✓ Un-isolating a faulted steam generator could result in an RCS cooldown causing a severe transient that challenges the primary-secondary barrier.

D. Re-establishing feed flow to the faulted S/G would cause SI to re-actuate on high steam flow and interfere with the RCS cooldown to Mode 5, Cold Shutdown.

A - Incorrect, The AFW system is designed to prevent the conditions of run-out. Procedures limit AFW flow to specific value designed to ensure run-out conditions are not created.
B - Incorrect, This is true but is not the reason for maintaining the S/G isolated.
C - Correct
D - Incorrect, This could occur but is not the reason for maintaining the S/G isolated.

1. WE06G2.1.32 091//////HR04301/S/

The unit has tripped due to a LOCA and ESF equipment has failed to start. As a result, FRP-C.2, "Response to Degraded Core Cooling," has been entered.

A depressurization of the Steam Generators (SGs) to 80 psig is being performed, in accordance with FRP-C.2, when the STA reports that a Red Path condition for Integrity has occurred.

Which of the following actions should be taken?

A. Immediately transition to FRP-P.1, "Response to Imminent Pressurized Thermal Shock Conditions"

B. Stop the S/G depressurization and, if the red path does not clear, transition to FRP-P.1, "Response to Imminent Pressurized Thermal Shock Conditions"

C.✓ Complete FRP-C.2 and then transition to FRP-P.1, "Response to Imminent Pressurized Thermal Shock Conditions," if the red path still exists

D. Complete the S/G depressurization and then transition to FRP-P.1, "Response to Imminent Pressurized Thermal Shock Conditions," if the red path still exists

1. WE09G2.1.32 095//////HR04301/S/

A reactor trip occurred due to a loss of offsite power. The plant is being cooled down on RHR per EPP-006, Natural Circulation Cooldown with Steam Void in Vessel with RVLIS.

- RCS cold leg temperatures are 190°F.
- Steam generator pressures are 50 psig.
- RVLIS upper range indicates greater than 100%.
- Three CRDM fans have been running during the entire cooldown.

Steam should be dumped from all SGs to ensure ...

A. boron concentration is equalized throughout the RCS prior to taking a sample to verify cold shutdown boron conditions.

B.✓ all inactive portions of the RCS are below 200°F prior to complete RCS depressurization.

C. RCS and SG temperatures are equalized prior to any subsequent RCP restart.

D. RCS temperatures do not increase during the required 29 hour vessel soak period.

QUESTIONS REPORT

for Westinghouse 3 Loop Questions

1. 039K4.08 001/ROT2G2/SROT2G2/3.3/3.4/C/A/BANK/SM00301/BOTH/MM52

The unit was operating at 80% power when a spurious low steamline pressure SI signal was received. The cause of the spurious signal was corrected and the SI signal has been reset. During preparations to return the unit to power operations, the MSIVs fail to open when their control switches are taken to OPEN. The unit is currently in MODE 3 with steam pressure downstream of the MSIVs at 1080 psig and Tavg at 557°F.

Which one of the following is the cause for the MSIVs' failure to open?

A. The differential pressure across the MSIVs is excessive.
B. The MSIV bypass valves are CLOSED.
C. The MSIV motor control center breaker is OPEN.
D.✓ The MSIV isolation signal has not been reset.
REF: Lesson Plan TB-2
SOURCE: VCS TB-2 Exam Bank # 1876

8. 055EA2.03 001//T1G1/STA BLACKOUT/M(3.9/4.7)/B/TP05301/S/MC

A loss of all AC power has occurred. The STA reports the status of the CSFs are as follows:

- Subcriticality – RED
- Core Cooling – RED
- Heat Sink – RED
- Integrity – GREEN
- Containment – GREEN
- Inventory - YELLOW

Which ONE of the following procedures should be used to mitigate these conditions?

A. EOP-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS

B.✓ EOP-ECA-0.0, LOSS OF ALL AC POWER

C. EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK

D. EOP-FR-C.1, RESPONSE TO INADEQUATE CORE COOLING

Feedback

REFERENCES:

1. EOP-ECA-0.0, LOSS OF ALL AC POWER, page 6, rev 02/22/02

DISTRACTORS:

A Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

B Correct. Note that precedes Step 1 of EOP-ECA-0.0 states that "CSF Status Trees are required to be monitored for information only. FRPs shall NOT be implemented."

C Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

D Incorrect. FRP's shall NOT be implemented while in ECA-0.0.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of Offsite and Onsite Power (Station Blackout); Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- *Steps 1 and 2 are IMMEDIATE ACTION steps.*
- *CSF Status Trees are required to be monitored for information only. FRPs shall NOT be implemented.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Verify Reactor Trip** • Rod bottom lights - ON • Reactor trip and bypass breakers - OPEN • Rod position indicators – AT ZERO • Neutron flux - DECREASING	Manually trip reactor.

1.0 **PURPOSE**

1.1 This procedure provides actions to respond to a loss of all AC power.

1.2 This procedure is applicable for Modes 1, 2, and 3 (greater than 1000 psig).

2.0 **SYMPTOMS OR ENTRY CONDITIONS**

2.1 The symptom of a loss of all AC power is the indication that the A and B 4KV buses are both de-energized.

2.2 This procedure is entered from E-0, REACTOR TRIP OR SAFETY INJECTION, Step 3, on the indication that the A and B 4KV buses are de-energized.

3.0 **REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS**

3.1 References

3.1.1 Technical Specifications for Turkey Point Unit 3 and Unit 4

3.1.2 Turkey Point Unit 3 and Unit 4 Final Safety Analysis Report

3.1.3 As-built plant drawings

3.1.4 Procedures

1. 3-ONOP-004.2, LOSS OF 3A 4KV BUS
2. 3-ONOP-004.3, LOSS OF 3B 4KV BUS
3. 3-ONOP-004.4, LOSS OF 3C 4KV BUS
4. 3-ONOP-023.2, EMERGENCY DIESEL GENERATOR FAILURE
5. 0-ONOP-025.3, DC EQUIPMENT AND INVERTER ROOM SUPPLEMENTAL COOLING
6. 3-ONOP-033.1, SPENT FUEL PIT COOLING SYSTEM MALFUNCTION
7. 3-ONOP-075, AUXILIARY FEEDWATER SYSTEM MALFUNCTION
8. 3-OP-023, EMERGENCY DIESEL GENERATOR
9. 4-OP-030, COMPONENT COOLING WATER SYSTEM

1. 055EA2.03 001/1/1/BLACKOUT SBO/MEM 3.9/4.7/B/SR04301/S/MAB/SDR

The following conditions exist:
- A loss of all AC power has occurred.
- The STA reports the status of the CSFs are as follows:
  - Subcriticality - RED
  - Core Cooling - RED
  - Heat Sink - RED
  - Integrity - GREEN
  - Containment - GREEN
  - Inventory - YELLOW

Which ONE of the following procedures should be used to mitigate these conditions?

A. 1-FR-S.1, Response to Nuclear Power Generation / ATWS

B.✓ 1-ECA-0.0, Loss of All AC Power

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

D. 1-FR-C.1, Response to Inadequate Core Cooling

Surry

References:
1-ECA-0.0, Loss of All AC Power, Rev. 21

Distractor Analysis:
A. Incorrect because FR's should not be implemented while in ECA-0.0. (see NOTE prior to step 1 of ECA-0.0)
B. Correct because this is the correct procedure to mitigate the loss of ac power.
C. Incorrect because FR's should not be implemented while in ECA-0.0.
D. Incorrect because FR's should not be implemented while in ECA-0.0.

Surry ILT Exam Bank Question #899

055 Station Blackout
EA2.03: Ability to determine or interpret the following as they apply to Station Blackout: Actions necessary to restore power.

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

Created: Thursday, May 27, 2004
Modified: Thursday, October 21, 2004
Revised:

9. 056AA2.51 001//T1G1/OFFSITE PWR/C/A(3.3/3.4)/B/TP05301/S/MC

Unit 3 was operating at 100% power when a loss of offsite power occurred.

- Subsequently, a loss of CCW occurs and all RCPs are tripped

Which ONE of the following describes the response of the reactor CORE delta T from the time the RCPs are tripped until one hour later later in the event?

Delta T ________ as natural circulation is being established, then ________.

A. Lowers; remains constant as heat removal is established with the atmospheric steam dumps.

B. Lowers; rises as decay heat load diminishes and heat removal is controlled by the atmospheric steam dumps.

C. Rises; lowers as decay heat load diminishes and heat removal is controlled by the atmospheric steam dumps.

D.✓ Rises; remains constant as heat removal is established with the atmospheric steam dumps.

Feedback

REFERENCES:
1. FSAR

DISTRACTORS:

A Incorrect. Delta T has to become higher to establish a driving head for natural circulation.

B Incorrect. Delta T has to become higher to establish a driving head for natural circulation. Does not take decay heat into account. Distractor provides opposite of actual effect.

C Incorrect. Does not take decay heat into account.

D Correct.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Offsite Power; Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Delta T, (core, heat exchanger, etc.).

INPO EXAM BANK

..056.AA2.51 03/10/2003

Indian Point 2 (Unit) **Exam Level** R

Mark Question | Print Recor | New Searc | Exit

Question

Given the following conditions:

oThe plant was at 100% power, BOL
oA loss of off-site power has occurred
oSubsequently, a loss of CCW required a reactor trip and a trip of all RCPs

Which ONE (1) of the following describes the response of the reactor core ?T from the time the RCPs are tripped until one hour later in the event?

Core ?T

Answer: Rises as natural circulation is being established, then lowers as decay heat load diminishes and heat removal is controlled by the atmospheric steam dumps

Distracter 1 Rises as natural circulation is being established, then remains constant as heat removal is established with the atmospheric steam dumps

Distracter 2 Lowers as natural circulation is being established, then remains constant as heat removal is established with the atmospheric steam dumps

Distracter 3 Lowers as natural circulation is being established, the rises as decay heat load diminishes and heat removal is controlled by the atmospheric steam dumps

Distracter Analysis:

Answer:

A.Incorrect. Does not take decay heat into account
B.Correct
C.Incorrect. Delta T has to become higher to establish a driving head for natural circulation
D.Incorrect. Delta T has to become higher to establish a driving head for natural circulation. Distractor provides opposite of actual effect

Distracter 1:

A.Incorrect. Does not take decay heat into account
B.Correct
C.Incorrect. Delta T has to become higher to establish a driving head for natural circulation
D.Incorrect. Delta T has to become higher to establish a driving head for natural circulation. Distractor provides opposite of actual effect

Distracter 2:

A.Incorrect. Does not take decay heat into account
B.Correct
C.Incorrect. Delta T has to become higher to establish a driving head for natural circulation
D.Incorrect. Delta T has to become higher to establish a driving head for natural circulation. Distractor provides opposite of actual effect

Distracter 3:

A.Incorrect. Does not take decay heat into account
B.Correct
C.Incorrect. Delta T has to become higher to establish a driving head for natural circulation
D.Incorrect. Delta T has to become higher to establish a driving head for natural circulation. Distractor provides opposite of actual effect

QUESTIONS REPORT

for Draft TP05-301-SRO

12. 058AG2.2.25 001//T1G1/DC PWR TS/C/A(2.5/3.7)/N/TP05301/S/MC

Unit 3 is operating at 100% power and Unit 4 is in COLD SHUTDOWN. For the past two hours, Unit 3 operators have been unable to raise float charge voltage above 124 volts on battery bank 3A. Checks of individual cell voltages have confirmed this value.

Based on the above plant conditions, which of the following describes the Technical Specification (TS) requirements and the reason for the requirements?

A.✓ Have Unit 3 in at least HOT STANDBY within 34 hours. This time allows the orderly shutdown of one unit at a time so as not to jeopardize the stability of the electrical grid by imposing a dual unit shutdown.

B. Have Unit 3 in at least HOT STANDBY within 12 hours. This time allows restoring the battery to within limits and not jeopardizing the stability of the electrical grid by imposing a dual unit shutdown.

C. Have Unit 3 in at least HOT STANDBY within 34 hours. This time allows the orderly shutdown of one unit at a time in order to avoid dual unit natural circulation cooldown in the event of a loss of both startup transformers.

D. Have Unit 3 in at least HOT STANDBY within 12 hours. This time allows restoring the battery to within limits and avoiding dual unit natural circulation cooldown in the event of a loss of both startup transformers.

Feedback

REFERENCES:

1. Tech Specs, 3/4.8.2, DC SOURCES, pages 8-13 – 8-16, Amendment Nos. 138 & 133
2. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, pages 92-95, rev 05/01/03

DISTRACTORS:

A Correct. IAW note (3) of Table 4.8-2, the "allowable value" for float voltage for each connected cell must be = 2.07 or the battery is considered INOPERABLE. IAW ACTION step "b," operators have 2 hours* *(which can be extended to 24 hours with Unit 4 in Mode 5) to correct the problem or have both Units in Mode 3 within the next 12 hours.

B Incorrect. 34 hours would be correct.

C Incorrect. The time is correct but the reason is incorrect.

D Incorrect. The time and reason are incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of DC Power; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

10. 058AG2.2.25 001//T1G1/DC PWR TS/C/A(2.5/3.7)/N/TP05301/S/MC

Unit 3 is operating at 100% power and Unit 4 is in COLD SHUTDOWN. For the past two hours, Unit 3 operators have been unable to raise float charge voltage above 124 volts on battery bank 3A. Checks of individual cell voltages have confirmed this value. Which of the following describes the requirements IAW ~~Chapter~~ 3.8.2 of Technical Specifications and the reason for the requirements?
Section ??

A. ✓ Have Unit 3 in at least HOT STANDBY within 34 hours. This time allows the orderly shutdown of one unit at a time so as not to jeopardize the stability of the electrical grid by imposing a dual unit shutdown.

B. Have Unit 3 in at least HOT STANDBY within 12 hours. This time allows restoring the battery to within limits and not jeopardizing the stability of the electrical grid by imposing a dual unit shutdown.

C. Have Unit 3 in at least HOT STANDBY within 34 hours. This time allows the orderly shutdown of one unit at a time in order to avoid dual unit natural circulation cooldown in the event of a loss of both startup transformers.

D. Have Unit 3 in at least HOT STANDBY within 12 hours. This time allows restoring the battery to within limits and avoiding dual unit natural circulation cooldown in the event of a loss of both startup transformers.

Feedback

REFERENCES:

1. Tech Specs, 3/4.8.2, DC SOURCES, pages 8-13 – 8-16, Amendment Nos. 138 & 133
2. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, pages 92-95, rev 05/01/03

DISTRACTORS:

A Correct. IAW note (3) of Table 4.8-2, the "allowable value" for float voltage for each connected cell must be = 2.07 or the battery is considered INOPERABLE. IAW ACTION step "b," operators have 2 hours* *(which can be extended to 24 hours with Unit 4 in Mode 5) to correct the problem or have both Units in Mode 3 within the next 12 hours.

B Incorrect. 34 hours would be correct.

C Incorrect. The time is correct but the reason is incorrect.

D Incorrect. The time and reason are incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of DC Power; Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following D.C. electrical sources shall be OPERABLE:*#

a. 125-volt D.C. Battery Bank 3A or spare battery bank D-52 and associated full capacity charger(s)
   1) 3A1 powered by motor control center (MCC) 3C with EDG 3A OPERABLE, or
   2) 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
   3) 3A1 powered by MCC 3C with EDG 3A OPERABLE and 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

b. 125-volt D.C. Battery Bank 3B or spare battery bank D-52 and associated full capacity charger(s)
   1) 3B1 powered by MCC 3B with EDG 3B OPERABLE, or
   2) 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
   3) 3B1 powered by MCC 3B with EDG 3B OPERABLE and 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

c. 125-volt D.C. Battery Bank 4A or spare battery bank D-52 and associated full capacity charger(s)
   1) 4A1 powered by MCC 4C with EDG 4A OPERABLE, or
   2) 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or
   3) 4A1 powered by MCC 4C with EDG 4A OPERABLE and 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE,

d. 125-volt D.C. Battery Bank 4B or spare battery bank D-52 and associated full capacity charger(s)
   1) 4B1 powered by MCC 4B with EDG 4B OPERABLE, or
   2) 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or
   3) 4B1 powered by MCC 4B with EDG 4B OPERABLE and 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one or more of the required battery chargers OPERABLE but not capable of being powered from its associated OPERABLE diesel generator(s), restore the capability within 72 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

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* All battery chargers required to satisfy the LCO shall be powered from separate MCCs.

Inoperability of the required EDG's specified in the LCO requirements below does not constitute inoperability of the associated battery chargers or battery banks.

D.C. SOURCES

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. With one of the required battery banks inoperable, or with none of the full-capacity chargers associated with a battery bank OPERABLE, restore all battery banks to OPERABLE status and at least one charger associated with each battery bank to OPERABLE status within two hours* or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.2.1 Each 125-volt battery bank and its associated full capacity charger(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 129 volts on float charge and the battery charger(s) output voltage is ≥ 129 volts, and

3) If two battery chargers are connected to the battery bank, verify each battery charger is supplying a minimum of 10 amperes, or demonstrate that the battery charger supplying less than 10 amperes will accept and supply the D.C. bus load independent of its associated battery charger.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts (108.6 volts for spare battery D-52), or battery overcharge with battery terminal voltage above 143 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) There is no visible corrosion at either terminals or connectors, or the connection resistance is less than 150×10^{-6} ohm, and

3) The average electrolyte temperature of every sixth cell is above 60°F.

c. At least once per 18 months by verifying that:

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, and

*Can be extended to 24 hours if the opposite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

SURVEILLANCE REQUIREMENTS (Continued)

4) Each 400 amp battery charger (associated with Battery Banks 3A and 4B) will supply at least 400 amperes at ≥ 129 volts for at least 8 hours, and each 300 amp battery charger (associated with Battery Banks 3B and 4A) will supply at least 300 amperes at ≥ 129 volts for at least 8 hours.

d. At least once per 18 months, during shutdown**, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 18 months, during shutdown**, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

f. At least once per 60 months, during shutdown**, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1.d.

** Except that the spare battery bank D-52, and any other battery out of service when spare battery bank D-52 is in service may be tested with simulated loads during operation.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(6)	≥ 2.07 volts
Specific Gravity (4)	≥ 1.200(5)	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all Connected cells > 1.205	Average of all connected cells ≥ 1.195(5)

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, three 125 volt battery banks, each with at least one associated full capacity charger capable of being powered by an OPERABLE diesel generator, shall be OPERABLE.

APPLICABILITY: MODES 5* and 6*.

ACTION:

With one or more of the required 125 volt battery banks or required associated full-capacity chargers inoperable or not capable of being powered from an OPERABLE diesel generator, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required battery banks and associated full-capacity chargers to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2.2 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125 volt battery banks and associated full-capacity chargers shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

*CAUTION - If the opposite unit is in MODES 1, 2, 3 or 4, see the corresponding Limiting Condition for Operation 3.8.2.1.

ATTACHMENT 1
(Page 82 of 102)

TECHNICAL SPECIFICATION BASES

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility.

The loss of an associated diesel generator for system(s), subsystem(s), train(s), component(s) or device(s) does not result in the system(s), subsystem(s), train(s), component(s) or device(s) being considered inoperable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation for the affected unit provided (1) its corresponding normal power source is OPERABLE; and (2) its redundant system(s), subsystem(s), train(s), component(s), and device(s) that depend on the remaining OPERABLE diesel generators as a source emergency power to meet all applicable LCO's are OPERABLE. This allows operation to be governed by the time limits of the ACTION statement associated with the inoperable diesel generator, not the individual ACTION statements for each system, subsystem, train, component or device. However, due to the existence of shared systems, there are certain conditions that require special provisions. These provisions are stipulated in the appropriate LCO's as needed.

More specifically, LCO's 3.5.2 and [redacted]require that associated EDG's be OPERABLE in addition to requiring that Safety Injection pumps, battery chargers, and battery banks, respectively also be OPERABLE. This EDG requirement was placed in these particular LCO's due to the shared nature of these systems to ensure adequate EDG availability for the required components. A situation could arise where a unit in MODES 1,2,3, or 4 could be in full compliance with LCO 3.8.1.1, yet be using shared equipment that could be impacted by taking an EDG out-of-service on the opposite unit. In this situation, diesel generator ACTION 3.8.1.1.d which verifies redundant train OPERABILITY, may not be applicable to one of the units. Thus, specific requirements for EDG **OPERABILITY have been added to the appropriate LCO's of the shared systems (3.5.2 and 3.8.2.1). It is important to note that in these particular LCO's, the inoperability of a required EDG does not constitute inoperability of the other components required to be OPERABLE in the LCO. Specific ACTION statements are included in 3.5.2 and 3.8.2.1 for those situations where the required components are OPERABLE (by the definition of OPERABILITY) but not capable of being powered by an OPERABLE EDG.**

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.8 ELECTRICAL POWER SYSTEMS (Continued)

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The ACTION requirements specified for the levels of degradation of the power sources provide restrictions upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources is consistent with the initial condition assumptions of the safety analysis and is based upon maintaining adequate onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of one onsite A.C. source. Two physically independent A.C. circuits exist between the offsite transmission network and the onsite Class 1E Distribution System by utilizing the following:

(1) a total of eight transmission lines which lead to five separate transmission substations tie the Turkey Point Switchyard to the offsite power grid;
(2) two dual-winding startup transformers each provide 100% of the A and B train 4160 volt power from the switchyard to its associated unit.

In addition, each startup transformer has the capability to supply backup power of approximately 2500 kw to the opposite unit's A-train 4160 volt bus. Two emergency diesel generators (EDG) provide onsite emergency A.C. power for each unit. EDG's 3A and 3B provide Unit 3 A-train, and B-train emergency power, respectively. EDG's 4A and 4B provide Unit 4 A-train and B-train emergency power, respectively.

Due to the shared nature of numerous electrical components between Turkey Point Units 3&4, the inoperability of a component on an associated unit will often affect the operation of the opposite unit. These shared electrical components consist primarily of both startup transformers, three out of four 4160 volt busses, and associated 480 volt motor control centers, all four 125 volt D.C. busses, all eight 120 volt vital A.C. panels and eight out of twelve vital A.C. inverters, **four out of eight battery chargers, and all four battery banks**. Depending on the component(s) which is (are) determined inoperable, the resulting ACTION can range from the eventual shutdown of the opposite unit long after the associated unit has been shutdown (30 days) to an immediate shutdown of both units. Therefore, **ACTION times allow for an orderly sequential shutdown of both units when the inoperability of a component(s) affects both units with equal severity. When a single unit is affected, the time to be in HOT STANDBY is** 6 hours. When an ACTION statement requires a dual unit shutdown, the time to be in HOT STANDBY is 12 hours. This is to allow the orderly shutdown of one unit at a time and not jeopardize the stability of the electrical grid by imposing a dual unit shutdown.

TECHNICAL SPECIFICATION BASES

3/4.8 ELECTRICAL POWER SYSTEMS (Continued)

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

As each startup transformer only provides the limited equivalent power of approximately one EDG to the opposite Units A-train 4160 volt bus, the allowable out-of-service time of 30 days has been applied before the opposite unit is required to be shutdown. Within 24 hours, a unit with an inoperable startup transformer must reduce THERMAL POWER to less than or equal to 30% RATED THERMAL POWER. The 30% RATED THERMAL POWER limit was chosen because at this power level the decay heat and fission product production has been reduced and the operators are still able to maintain automatic control of the feedwater trains and other unit equipment. At lower power levels the operators must use manual control with the feedwater bypass lines. By not requiring a complete unit shutdown, the plant avoids a condition requiring natural circulation and avoids intentionally relying on engineered safety features for non-accident conditions.

With one startup transformer and one of the three required EDGs inoperable, the unit with the inoperable transformer must reduce THERMAL POWER to less than or equal to 30% RATED THERMAL POWER within 24 hours, based on the loss of its associated startup transformer, whereas operation of the unit with the OPERABLE transformer is controlled by the limits for inoperability of the EDG. The notification of a loss of startup transformer(s) to the NRC (ACTION STATEMENT 3.8.1.1.c) is not a 10 CFR 50.72/50.73 requirement and as such will be made for information purposes only to the NRC Operations Center via commercial lines.

With an EDG out of service, ACTION statement 3.8.1.1.b and Surveillance Requirement (SR) 4.8.1.1.1.a are provided to demonstrate operability of the required startup transformers and their associated circuits within 1 hour and at least once per 8 hours thereafter. For a planned EDG inoperability, SR 4.8.1.1.1.a may be performed up to 1 hour prior to rendering the EDG inoperable. The frequency of SR 4.8.1.1.1.a after it has been performed once, is at least once per 8 hours until the EDG is made operable again. When one diesel generator is inoperable, there is also an additional ACTION requirement to verify that required system(s), subsystem(s), train(s), component(s), and device(s) that depend on the remaining required OPERABLE diesel generators as a source of emergency power to meet all applicable LCO's, are OPERABLE. This requirement is intended **to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable.** This requirement allows continued operation to be governed by the time limits of the ACTION statement associated with the inoperable diesel generator. The loss of a diesel generator does not result in the associated system(s), subsystem(s), train(s), component(s), or device(s) being considered inoperable provided: (1) its corresponding normal power source is OPERABLE: and (2) its redundant system(s), subsystem(s), train(s), component(s), and device(s) that depend on the remaining required OPERABLE diesel generators as a source of emergency power to meet all applicable LCO's, are OPERABLE.

ATTACHMENT 1
(Page 85 of 102)

TECHNICAL SPECIFICATION BASES

3/4.8 ELECTRICAL POWER SYSTEMS (Continued)

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

All diesel generator inoperabilities must be investigated for common cause failures regardless of how long the diesel generator inoperability persists. When one diesel generator is inoperable, TS 3.8.1.1 ACTION statements b and c provide an allowance to avoid unnecessary testing of other required diesel generators. If it can be determined that the cause of the inoperable diesel generator does not exist on the remaining required diesel generators, then SR 4.8.1.1.2a.4 does not have to be performed. Twenty-four (24) hours (or eight (8) hours if both a startup transformer and diesel generator are inoperable) is reasonable to confirm that the remaining required diesel generators are not affected by the same problem as the inoperable diesel generator. If it cannot otherwise be determined that the cause of the initial inoperable diesel generator does not exist on the remaining required diesel generators, then satisfactory performance of SR 4.8.1.1.2a.4 suffices to provide assurance of continued OPERABILITY of the remaining required diesel generators. If the cause of the initial inoperability exists on one or more of the remaining required diesel generators, those diesel generators affected would also be declared inoperable upon discovery, and TS 3.8.1.1 ACTION statement f or TS 3.0.3, as appropriate, would apply.

When in Modes 1, 2, 3 or 4, a unit depends on one EDG and its associated train of busses from the opposite unit in order to satisfy the single active failure criterion for safety injection (SI) pumps and other shared equipment required during a loss-of-coolant accident with a loss-of-offsite power. Therefore, one EDG from the opposite unit is required to be OPERABLE along with the two EDG's associated with the applicable unit.

For single unit operation (one unit in Modes 1-4 and one unit in Modes 5-6 or defueled) TS 3.8.1.1 ACTION d. refers to one of the three required emergency diesel generators. For dual unit operation (both units in Modes 1-4), TS 3.8.1.1 ACTION d. refers to one of the four required emergency diesel generators. This conclusion is based on the portion of ACTION d. that states "... in addition to ACTION b. or c. ...." Since ACTIONs b. and c. both refer to "one of the required diesel generators," this implies that ACTION d. also refers to one of the required diesel generators. ACTION d. says "in addition to ACTION b. or c. above, ..." therefore ACTION d. is merely providing additional requirements applicable to the conditions that required satisfaction of ACTIONs b. or c.

With both startup transformers inoperable, the unit(s) are required to be shutdown consecutively, after 24 hours. **[redacted] shutdown is used because a unit without its associated transformer must perform a natural circulation cooldown.** By placing one unit in COLD SHUTDOWN before starting shutdown of the second unit, a dual unit natural circulation cooldown is avoided.

The term verify means to administratively check by examining logs or other information to determine if required components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

ATTACHMENT 1
(Page 88 of 102)

TECHNICAL SPECIFICATION BASES

3/4.8 ELECTRICAL POWER SYSTEMS (Continued)

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

Surveillance Requirement 4.8.1.1.2.g.7) demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal surveillances, and achieve the required voltage and frequency within 15 seconds. **The 15 second time is derived from the requirements of the accident analysis to respond to a design large break Loss of Coolant Accident (LOCA).** By performing this SR after 24 hours (or after two hours, in accordance with the proposed revised footnote), the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the EDG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY. The requirement that the diesel has operated for at least two hours at full load is based on NRC staff guidance for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test.

In accordance with Technical Specification Amendments 215/209, the EDGs will be inspected in accordance with a licensee controlled maintenance program referenced in the UFSAR. The maintenance program will require inspections in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service. Changes to the maintenance program will be controlled under 10 CFR 50.59.

The fuel supply specified for the Unit 3 EDG's is based on the original criteria and design bases used to license the plant. The specified fuel supply (diesel oil storage tank or temporary storage system) will ensure sufficient fuel for either EDG associated with Unit 3 for at least a week. The fuel supply specified for the Unit 4 EDG's is based on the criteria provided in ANSI N195-1976 as endorsed by Regulatory Guide 1.137. The specified fuel supply will ensure sufficient fuel for each EDG associated with Unit 4 for at least a week.

Surveillance Requirement 4.8.1.1.2.g.7, verifying that the diesel generator operates for at least 24 hours, may be performed during POWER OPERATION (Mode 1) per Licensing Amendment # 221/215.

DIESEL FUEL OIL TESTING PROGRAM

In accordance with TS 6.8.4, a diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. For the intent of this specification, new fuel oil shall represent diesel fuel oil that has not been added to the Diesel Fuel Oil Storage Tanks. Once the fuel oil is added to the Diesel Fuel Oil Storage Tanks, the diesel fuel oil is considered stored fuel oil, and shall meet the Technical Specification requirements for stored diesel fuel oil.

ATTACHMENT 1
(Page 93 of 102)

TECHNICAL SPECIFICATION BASES

3/4.8 ELECTRICAL POWER SYSTEMS (Continued)

DIESEL FUEL OIL TESTING PROGRAM (Continued)

The ACTION requirements specified for the inoperability of certain Motor Control Centers (MCCs), Load Centers (LCs) and the 4160-Volt Busses provide restrictions upon continued facility operation commensurate with the level of degradation on each unit and the amount of time one could reasonably diagnose and correct a minor problem. The level of degradation is based upon the types of equipment powered and the out-of-service limit imposed on that equipment by the associated ACTION statement. If this degradation affects the associated unit only, then no restriction is placed on the opposite unit and an out-of-service limit of 8 hours (except for MCC's 3A, 3K, 4J and 4K) is applied to the associated unit. Since MCC's 3A, 3K, 4J and 4K are used to power EDG auxiliaries, an out-of-service limit of 72 hours is applied as required by 3.8.1.1. If the degradation impacts both units (i.e., required shared systems or cross-unit loads), then an out-of-service limit of 8 hours is applied to the associated unit and an out-of-service limit based on the most restrictive ACTION requirement for the applicable shared or cross-unit load is applied to the opposite unit.

For example, if being used to satisfy **3.8.2.1,** the Battery Chargers 3A2, 3B2, 4A2, and 4B2 are cross-unit loads and have out-of-service limits of 2 hours. This is the most restrictive limit of the applicable equipment powered from MCC 3D and 4D. Therefore, an out-of-service limit of 2 hours is applied if the battery charger is required to be OPERABLE.

The ACTION requirements specified when an A.C. vital panel is not energized from an inverter connected to its associated D.C. bus provides for two phases of restoration. Expedient restoration of an A.C. panel is required due to the degradation of the Reactor Protection System and vital instrumentation. The first phase requires reenergization of the A.C. vital panel within two hours. During this phase the panel may be powered by a Class 1E constant voltage transformer (CVT) fed from a vital MCC. However, the condition is permissible for only 24 hours as the second phase of the ACTION requires reenergization of the A.C. vital panel from an inverter connected to its associated D.C. bus within 24 hours. Failure to satisfy these ACTIONS results in a dual unit shutdown.

Chapter 8 of the UFSAR provides the description of the A.C. electrical distribution system. The 480 Volt Load Center busses are arranged in an identical manner for Units 3 and 4. For each unit there are five safety related 480v load center busses, four of which are energized from different 4.16 kv busses (Load Centers A and C are fed from Train A and Load Centers B and D are fed from Train B). This arrangement ensures the availability of equipment associated with a particular function in the event of loss of one 4.16 kV bus.

The fifth safety related 480V load center in each unit is a swing load center, which can swing between Load Center C and D of its associated unit. These load centers are labeled as 3H for Unit 3 and 4H for Unit 4. When the 480V swing load center is connected to either 480V supply bus, it is considered to be an extension of that 480V supply bus.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.8 ELECTRICAL POWER SYSTEMS (Continued)

DIESEL FUEL OIL TESTING PROGRAM (Continued)

Technical Specification 3/4.8.3.1 states that, ... the electrical busses shall be energized in the specified manner...

Footnote 3.8.3.1*** states in part, Electrical bus can be energized from either train of its unit....

These statements establish that the load center is an extension of the train it is supplied from, and the associated bus is energized in the specified manner when it is supplying the load center.

The second half of the footnote pertains to the swing capability of the LC, and reads, ...and swing function to opposite train must be OPERABLE for the Unit(s) in MODES 1, 2, 3, and 4.

Although the swing load center swing function may be inoperable, the associated bus and swing loads are clearly OPERABLE, because the associated train was established by the first half of the footnote. The swing bus is capable of being powered from the opposite train, and the swing function is only applicable to the opposite train. If the swing LC cannot be powered from, or swing to, the opposite train, then the opposite train is incapable of being fully energized and is INOPERABLE.

Therefore, the correct interpretation of the footnote for the swing LCs and MCCs is as follows:

> Electrical bus can be energized from either train of its unit (establishes the associated bus) and swing function to opposite train must be OPERABLE for the Unit(s) in MODES 1, 2, 3, and 4 (or the opposite train is INOPERABLE).

The swing load centers are used to supply shared system and cross-unit loads, and other Technical Specification ACTION statements may be invoked for loss of swing capability. As discussed above, the Unit 3 DC battery chargers 3A2 and 3B2 are powered from Unit 4 via swing MCC 4D, and the Unit 4 DC battery chargers 4A2 and 4B2 are powered from Unit 3 via swing MCC 3D. Inoperability of the swing capability could impact both units if any of the swing battery chargers is credited for satisfying Technical Specification **3.8.2.1**. Both EDGs are required to be OPERABLE for a swing battery charger. An inoperable swing function prevents one EDG from supporting that battery charger, and a dual-unit 72 hour ACTION statement applies in accordance with TS **3.8.2.1 A**CTION statement a.

With a unit shutdown one 4160-volt bus on the associated unit can be deenergized for periodic refueling outage maintenance. The associated 480-volt Load Centers can then be cross-tied upon issuance of an engineering evaluation.

QUESTIONS REPORT

for Westinghouse 3 Loop Questions

1. 058AG2.1.12 001/T1G2/T1G2/LOSS OF DC/C/A (3.7/4.1)/B/TP02301/S/SDR

Both Units are at 100% power with all systems operable except the '4A' EDG which is Out Of Service on a clearance. The '3B1' battery charger fails and all DC bus loads are automatically shifted to the '3B2' battery charger.

Which ONE of the following describes the required operator response?

Restore the '3B1' battery charger to service within:

A.✓ 72 hours or shutdown both Units within the next 12 hours.

B. 2 hours or shutdown both Units within the next 12 hours.

C. 72 hours or shutdown only Unit 4 within the next 12 hours.

D. 2 hours or shutdown only Unit 4 within the next 12 hours.

Question source: Turkey Point requal question bank 69025280301-ORQ; ORQ#600

Distractor analysis:
A: Correct, 3B1 has failed TS 3.8.2.1.b.1) not satisfied. 3B2 does not have 4A EDG operable TS 3.8.2.1.b.2) not satisfied. The conditions for 125-volt DC battery bank 3B and associated full capacity chargers can not be satisfied. TS 3.8.2.1 ACTION a. is applicable since 3B2 is available but not capable of being powered from its associated EDG
B: Incorrect, Actions required for TS 3.8.2.1 ACTION b. Applicable if required battery banks are inoperable or no chargers operable.
C: Incorrect, Correct action but TS 3.8.2.1 ACTION a. is applicable to both units simultaneously.
D: Incorrect, Actions required for TS 3.8.2.1 ACTION b. Applicable if required battery banks are inoperable or no chargers operable. TS 3.8.2.1 ACTION b. is applicable to both units simultaneously.

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

Created: Thursday, May 27, 2004
Modified: Friday, November 05, 2004
Revised:

RO Tier:	T1G2	SRO Tier:	T1G2
K/A Value:	LOSS OF DC	Cog. Level:	C/A (3.7/4.1)
Source:	B	Exam:	TP02301
Test:	S	Misc:	SDR

QUESTIONS REPORT
for Westinghouse 3 Loop Questions

1. 024G2.2.25 077//////HR04301/S/

Which of the following describes a condition in Technical Specifications and its bases which would require Emergency Boration in accordance with AOP-002, "Emergency Boration"?

A. • During the recovery from a Main Feedwater Pump trip, Control Rods are determined to be below the rod insertion limit
   • Control the reactivity transient associated with a steam line break

B. • During the recovery from a Main Feedwater Pump trip, Control Rods are determined to be below the rod insertion limit
   • Control the reactivity transient associated with an inadvertent dilution

C.✓ • During a reactor startup, the Reactor achieves criticality with Bank C rods at 105 steps
   • Control the reactivity transient associated with a steam line break

D. • During a reactor startup, the Reactor achieves criticality with Bank C rods at 105 steps
   • Control the reactivity transient associated with an inadvertent dilution

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

Created: Thursday, May 27, 2004
Modified: Thursday, May 27, 2004
Revised:
RO Tier: SRO Tier:
K/A Value: Cog. Level:
Source: Exam: HR04301
Test: S Misc:

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

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

QUESTIONS REPORT

for Draft TP05-301-SRO

1. 064A2.06 001//T2G1/EDG/C/A(2.9/3.3)/N/TP05301/S/MC

You are the SRO on Unit 4. Following recovery from a loss of offsite power, the 4A Diesel Generator continued to run unloaded for 4.5 hours at 900 rpm. The BOP discovered this and placed a 500KW load on the diesel for 30 minutes with the EDG Air Box drain fully open and has requested to secure the EDG.

Considering the conditions given above, which ONE of the following describes your response and why?

A. Have him run the diesel an additional hour with a minimum of 2500KW load with the air box drain shut to remove the fuel that accumulated in the crankcase to prevent the possibility of a crankcase explosion if the diesel loads suddenly on its next start.

B.✓ Have him run the diesel an additional half hour with a minimum of 1250KW load with the air box drain shut to clean out the oil that accumulated in the exhaust stack to prevent the possibility of an exhaust fire if the diesel loads suddenly on its next start.

C. He should not have run the diesel partially loaded. Have him run the diesel an additional half hour with no load and with the air box drain open 25 percent to effectively drain the air box and to ensure the diesel is sufficiently cooled down prior to securing.

D. His actions were correct. The diesel may be secured with no adverse impact.

Feedback

REFERENCES:

1. OP-023, EMERGENCY DIESEL GENERATOR, pages 14,50,120,126, rev 08/29/02
2. ONOP-004.3, LOSS OF 3B 4KV BUS, page 8, rev 10/16/01
3. EOP-ECA-0.0, LOSS OF ALL AC POWER, page 26, rev 02/22/02

DISTRACTORS:

A Incorrect. See B. Although fuel dilution into lube oil can occur, it will not occur as a result of the aforementioned operation.

B Correct. After 4.5 cumulative hours of operation at synchronous speed (900 rpm) at loads between 0 and 20% (0-500KW), the engine shall be run at a minimum of 50% load (1250KW) for at least 30 minutes to clean out the oil residual that accumulates in the exhaust stack as an exhaust fire could result when the engine is suddenly loaded, raising exhaust temperatures quickly.

C Incorrect. See B. When running for extended periods at idle speed, the air box drain should be open 25% to effectively drain the air box.

D Incorrect. The Diesel should have been run at a minumum of 50% load (1250KW) vice 500KW. The Diesel must be run an additional 30 minutes at a minimum of 50% load to meet the requirements of Precaution 4.4 of ref 1.

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Diesel Generator (EDG) System; Ability to (a) predict the impacts of the following malfunctions or operations on the EDG system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Operating unloaded, lightly loaded, and highly loaded time limit.

11. 064A2.06 001//T2G1/EDG/C/A(2.9/3.3)/N/TP05301/S/MC

You are the SRO on Unit 4. Following recovery from a loss of offsite power, the 4A Diesel Generator continued to run unloaded for 4.5 hours at 900 rpm. The BOP discovered this and placed a 500KW load on the diesel for 30 minutes with the EDG Air Box drain open and has requested to secure the EDG.

Considering the conditions given above, which ONE of the following describes your response and why?

A. Have him run the diesel an additional hour with a minimum of 2500KW load with the air box drain shut to remove the fuel that accumulated in the crankcase to prevent the possibility of a crankcase explosion if the diesel loads suddenly on its next start.

B.✓ Have him run the diesel an additional half hour with a minimum of 1250KW load with the air box drain shut to clean out the oil that accumulated in the exhaust stack to prevent the possibility of an exhaust fire if the diesel loads suddenly on its next start.

C. He should not have run the diesel partially loaded. Have him run the diesel an additional half hour with no load and with the air box drain open 25 percent to effectively drain the air box and to ensure the diesel is sufficiently cooled down prior to securing.

D. His actions were correct. The diesel may be secured with no adverse impact.

Feedback

REFERENCES:

1. OP-023, EMERGENCY DIESEL GENERATOR, pages 14,50,120,126, rev 08/29/02
2. ONOP-004.3, LOSS OF 3B 4KV BUS, page 8, rev 10/16/01
3. EOP-ECA-0.0, LOSS OF ALL AC POWER, page 26, rev 02/22/02

DISTRACTORS:

A Incorrect. See B. Although fuel dilution into lube oil can occur, it will not occur as a result of the aforementioned operation.

B Correct. After 4.5 cumulative hours of operation at synchronous speed (900 rpm) at loads between 0 and 20% (0-500KW), the engine shall be run at a minimum of 50% load (1250KW) for at least 30 minutes to clean out the oil residual that accumulates in the exhaust stack as an exhaust fire could result when the engine is suddenly loaded, raising exhaust temperatures quickly.

C Incorrect. See B. When running for extended periods at idle speed, the air box drain should be open 25% to effectively drain the air box.

D Incorrect. See B.

K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Diesel Generator (EDG) System; Ability to (a) predict the impacts of the following malfunctions or operations on the EDG system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Operating unloaded, lightly loaded, and highly loaded time limit.

4.0 PRECAUTIONS/LIMITATIONS

4.1 For test purposes the generator load shall not exceed 2750 KW and generator current shall not exceed 477 amps. (Basic Overload rating).

4.2 An electric motor driven soakback pump shall be in operation at all times when the engine is not running. This is necessary to provide turbocharger prestart lubrication and post shutdown bearing cooling.

4.3 Do not allow the engine to run unloaded at 900 rpm for periods in excess of 4.5 hours. The Diesel Generators should not be operated at less than 25 percent load due to the accumulation of lube oil in the exhaust during light load operation (souping). Depending on the amount of souping that has taken place, an exhaust fire could result when the engine is suddenly loaded, raising exhaust temperatures quickly.

4.4 After 4.5 cumulative hours of operation at synchronous speed at loads between 0 and 20 percent (0-500KW), the engine shall be run at a minimum of 50 percent load for at least 30 minutes to clean out the oil residual that accumulates in the exhaust stack.

4.5 After 8 hours of continuous operation at idle speed the engine shall be run at a minimum of 50 percent load for at least 30 minutes to clean out the oil residual that accumulates in the exhaust stack.

4.6 During testing, only one of the Unit 3 Emergency Diesel Generators shall be paralleled with the off-site transmission network at a time.

4.7 Caution must be used when paralleling Unit 3 Emergency Diesel Generators with the system grid since there are no synch check relays.

4.8 During testing, only one of the Emergency Diesel Generators for each unit shall have its Master Control Switch in the LOCAL position with the Rapid Start/Auto Start Bypass Keylock switch positioned to Auto Start Bypass. When the switches are in this configuration the Emergency Diesel Generator will not automatically start upon loss of bus voltage or a Safety Injection signal.

4.9 When the Emergency Diesel Generators are in Standby Mode, Governor Control switches and Voltage Control switches at the Local and Control Room panels shall not be operated. Actuation of these switches will alter the preset speed or voltage settings.

4.10 All alarms shall be investigated promptly and corrective action taken to assure availability of the Emergency Diesel Generator.

4.11 Technical Specification requirements shall be observed and any deviation from these requirements shall be reported immediately to the Nuclear Plant Supervisor. Technical Specifications should be consulted for any change in system status.

4.12 Hearing protection shall be worn in the EDG Rooms when operating the Emergency Diesel Generator.

4.13 Any problems or abnormalities encountered when performing this procedure should be reported immediately to the Nuclear Plant Supervisor.

INITIALS
CK'D VERIF

5.3.2 (Cont'd)

NOTE

A lack of oil on the rags indicates a failure of the lubricators for the air start motors. Foreign particles (rust, dirt, etc.) may indicate impending air start motor failure. The Nuclear Plant Supervisor should be notified of any malfunction.

____ 23. Remove and inspect the rags placed over the air start motor exhausts for evidence of oil and foreign particles.

CAUTION

Do not operate the diesel generator unloaded and at rated speed (900 rpm) for extended periods of time (over 4.5 hours). A minimum load of 25 percent should be applied to the generator in a timely manner to reduce the possibility of souping which can result in an exhaust system fire.

24. **WHEN** 3A Diesel Generator reaches rated speed (900 rpm), **THEN** perform the following:

CAUTION

Loss of crankcase vacuum in conjunction with a decrease of one or more cylinder exhaust pyrometer reading(s) could be indicative of fuel oil line or fitting failure in the crankcase area and fuel oil intrusion of the lube oil. When fuel oil intrusion is substantiated by the smell of fuel oil at the lube oil dipstick opening the EDG should be shut down and not restarted until lube oil quality (absence of fuel oil) is determined to be acceptable.

NOTE

Guidelines for determining EDG crankcase vacuum/pressure are provided in Enclosure 2.

____ a. Slowly Open 3A EDG Crankcase Vacuum Gauge PI-3-6679A Isol, 3-70-283A.

INITIALS
CK'D VERIF

5.7.2 (Cont'd)

CAUTIONS

- ***Extended periods of operation at idle speed can cause oil to accumulate in the EDG Air Box. During idle operation, the EDG Air Box drain should be open approximately 25 percent to effectively drain the air box.***
- ***The EDG is required to be loaded to at least 50 percent of rated load following extended idle periods (no more than 8 hours) to preclude the possibility of an exhaust system fire due to the accumulation of lube oil in the exhaust system at light load operation.***

____ 28. Operate 3A EDG at idle speed as required per the direction of the Nuclear Plant Supervisor, AND repeat Substeps 5.7.2.18 through 5.7.2.26 as required for repetitive idle starts.

a. If the EDG is to be shut down, proceed to Substep 5.7.2.42.

NOTE

When the 3A EDG Idle Release is actuated, the diesel generator will accelerate from idle speed to full speed, 900 RPM.

____ 29. WHEN all checks (i.e., maintenance) requiring 3A EDG idle speed operation have been successfully completed, THEN position the Idle Release/Start control switch to RELEASE.

a. Verify the following as the 3A Diesel Generator accelerates to full speed:

____ (1) Observe Gen Voltage and Gen Frequency indicators to verify the generator field flashes when speed reaches approximately 800 rpm.

____ 30. IF the overspeed test was performed and it is desired to restart the EDG, THEN return to Substep 5.7.2.18.

____ 31. IF the EDG is not going to be restarted using this section, THEN go to Substep 5.7.2.43.

INITIALS
CK'D VERIF 5.7.2 (Cont'd)

NOTE

A change in day tank level should occur during diesel operation. A failure of the day tank indicated level to change may be indicative of an isolated or malfunctioning level indicator.

37. Verify skid tank and day tank levels are being maintained as follows:

____ a. Day Tank, LG-3-1428A

(1) Minimum level: 4 feet 9 inches

(2) Maximum level: 6 feet 2 inches

____ b. Skid Tank, LI-3-3402A

(1) Minimum level: 40 to 150 gal (EDG operating)
150 gal (EDG in standby)

(2) Maximum level: 210 gal (EDG in standby)

____ 38. Verify 3A Diesel Generator operating parameters are within normal ranges per Enclosure 1 (except lube oil level).

____ 39. **IF** operating at a load other than 2300 KW to 2500 KW, **THEN** verify 3A Diesel Generator operating parameters as specified by System Engineer or applicable procedure.

CAUTION

The Emergency Diesel Generator should be operated for a minimum of 1 hour at greater than 50 percent (1250KW) load any time the diesel generator has been subjected to extended idle periods or low load operating conditions.

____ 40. Operate the 3A Diesel Generator for a minimum of 1 hour, **OR** as directed by the Nuclear Plant Supervisor.

____ a. **IF** Diesel Generator is to be run greater than 1 hour, **THEN** perform Attachments 11 and 12.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Emergency diesel generators should NOT be run unloaded for more than 4.5 hours.

2	**Check 3B 4KV Bus Lockout Relay - RESET**	Perform the following: a. IF the 3A and 3B 4KV buses are both deenergized, THEN reset 3B 4KV bus lockout relay. b. IF 3A 4KV bus is energized, THEN perform the following: 1) Determine and correct cause of 3B 4KV bus lockout relay actuation. 2) WHEN cause of 3B 4KV bus lockout relay actuation has been determined and corrected, THEN reset lockout relay. c. WHEN 3B 4KV bus lockout relay has been reset, THEN observe CAUTION prior to Step 3 AND go to Step 3.

CAUTION

If an SI signal exists or an SI signal is actuated during the performance of this procedure, it is required to be reset to enable closure of the Emergency Diesel Breaker or Startup Transformer Breaker, or to permit manual loading of equipment on the 4KV bus.

3	**Verify SI Reset.**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Steady state loading on each Unit 3 emergency diesel generator shall not exceed 2500 KW. Load transients up to 2750 KW are acceptable when starting additional equipment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
35	**Verify The Following Equipment Loaded On Energized 4KV Buses**	
	a. 480 volt load centers	a. Manually close load control center breakers to energize 480 volt load centers.
	b. Battery chargers	
	c. Instrumentation and control	
	d. Communications	

	e. HVAC Equipment
	• Computer Room Chiller
	• Battery Room Air Conditioners -
	  * E16E (30609)
	  * E16F (40625)	

	f. One Auxiliary Building Exhaust Fan	
	g. Spent Fuel Pit Exhaust Fan	
	h. Spent Fuel Pit Cooling Water Pump	
	i. Radiation Monitors • Unit 3 SFP SPING • Plant Vent SPING • SJAE SPING	

17. 065AA2.06 001//T1G1/INST AIR/C/A(3.6/4.2)/M/TP05301/S/MC

Units 3 and 4 are both operating at 100% power when the following occurs on both Unis:

- The Lag CM starts followed shortly by startup of both CD air compressors.
- Annunciator I-6/1, INSTR AIR HI TEMP/LO PRESS actuates
- Annunciator G-1/2, CHARGING PUMP HIGH SPEED, actuates
- Annunciators C-1/1 – 1/3, SG A,B,C, LO/LO-LO LEVEL ALARMS, actuate
- SG levels are 25% and decreasing
- Unit 3 instrument air pressure is 60 psig
- Unit 4 instrument air pressure is 65 psig

Which ONE of the following describes the correct operator response?

A.✓ Trip Unit 3 IAW ONOP-013, LOSS OF INSTRUMENT AIR, and perform a Fast Load Reduction on Unit 4 IAW ONOP-100, FAST LOAD REDUCTION.

B. Perform a Fast Load Reduction on both Units IAW ONOP-100, FAST LOAD REDUCTION and establish AFW flow.

C. Trip both Units and enter EOP-E-0, REACTOR TRIP OR SAFETY INJECTION on both Units.

D. Allow both Units to trip and enter EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

Feedback

REFERENCES:
1. ONOP-013, LOSS OF INSTRUMENT AIR, page 6 & Foldout, rev 12/23/02

DISTRACTORS:

A Correct. IAW ref 1.

B Incorrect. IA to trip the Unit when its instrument air pressure drops below 65 psig.

C Incorrect. IA to trip the Unit when its instrument air pressure drops below 65 psig.

D Incorrect. IA to trip the Unit when its instrument air pressure drops below 65 psig.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Instrument Air; Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing.

12. 065AA2.06 001//T1G1/INST AIR/C/A(3.6/4.2)/M/TP05301/S/MC

Units 3 and 4 are both operating at 100% power when the following occurs on both Unis:

- The Lag CM starts followed shortly by startup of both CD air compressors.
- Annunciator I-6/1, INSTR AIR HI TEMP/LO PRESS actuates
- Annunciator G-1/2, CHARGING PUMP HIGH SPEED, actuates
- Annunciators C-1/1 – 1/3, SG A,B,C, LO/LO-LO LEVEL ALARMS, actuate
- SG levels are 25% and decreasing
- Unit 3 instrument air pressure is 60 psig
- Unit 4 instrument air pressure is 65 psig

Which ONE of the following describes the correct operator response?

A.✓ Trip Unit 3 IAW ONOP-013, LOSS OF INSTRUMENT AIR, and perform a Fast Load Reduction on Unit 4 IAW ONOP-100, FAST LOAD REDUCTION.

B. Perform a Fast Load Reduction on both Units IAW ONOP-100, FAST LOAD REDUCTION and establish AFW flow.

C. Trip both Units and enter EOP-E-0, REACTOR TRIP OR SAFETY INJECTION on both Units.

D. Allow both Units to trip and enter EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

Feedback

REFERENCES:

1. ONOP-013, LOSS OF INSTRUMENT AIR, page 6 & Foldout, rev 12/23/02

DISTRACTORS:

A Correct. IAW ref 1.

B Incorrect. IA to trip the Unit when its instrument air pressure drops below 65 psig.

C Incorrect. IA to trip the Unit when its instrument air pressure drops below 65 psig.

D Incorrect. IA to trip the Unit when its instrument air pressure drops below 65 psig.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of Instrument Air; Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Random failures of equipment and components may be misleading. It is important to determine affected unit(s), the cause and to correct the problem as rapidly as possible to minimize the impact of an Instrument Air Malfunction.

NOTES

- *Foldout Pages shall be monitored throughout this procedure.*
- *This procedure shall be followed until the loss of instrument air is corrected and the system returned to a normal configuration per 3/4-OP-013, Instrument Air System, to facilitate recovery actions following the event.*
- *The first two steps are immediate actions steps.*
- *Steps utilizing an asterisk (*) are referring to the affected unit(s).*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Determine The Actual Instrument Air Pressure On PI-3-1444 AND PI-4-1444 (VPA For Each Unit)**	
2	**Maintain Instrument Air Pressure Greater Than 65 PSIG On PI-*-1444 (VPA)**	IF pressure is less than 65 psig **AND** the available Instrument Air Compressor(s) is unable to restore pressure, **THEN** trip the affected unit(s) and enter 3/4-EOP-E-0 while continuing with this procedure.
3	**Instrument Air Pressure Less Than 95 PSIG On PI-*-1444 (VPA)**	* Dispatch an Operator to check operation of Instrument Air Compressor After Coolers. IF the cooler is not operating properly, **THEN** place the other unit's Instrument Air Compressor(s) in service. * IF either unit is experiencing symptoms of a loss of Instrument Air **AND** system pressure is greater than 95 psig, **THEN** go to Step 8. * IF neither unit has symptoms of a loss of Instrument Air, **THEN** go to Step 27.

FOLDOUT PAGE

UNIT TRIP CRITERIA

IF Instrument Air System cannot be maintained greater than 65 psig, requires isolation of the Auxiliary Building, Containment Building or Turbine Building Air Headers, **OR** results in the inability to maintain S/G levels on either unit, **AND** Instrument Air Compressors are unable to restore pressure, **THEN** the affected unit(s) shall be tripped **AND** the appropriate EOP network entered while continuing to restore Instrument Air to the unit(s).

MAJOR COMPONENT IMPACTS

1. A single unit loss of Instrument Air (less than 90 psig) may result in the loss of function to the following valves if the Nitrogen Backup Systems cannot be maintained to the components:
   - Pressurizer PORVs
   - Unit 3 MSIVs (Unit 3 Only)
   - Steam Dumps to Atmosphere (Manual operation ONLY if N_2 Backup from dewar is available)
   - Train 2 of AFW (same unit), operate FCV's in MANUAL to conserve N_2.
   - Train 1 of AFW (opposite unit), operate FCV's in MANUAL to conserve N_2.
   - Unit 3 EDG Fuel Oil Transfer capability (Unit 3 Only)

2. A single unit loss of instrument air (less than 65 psig) results in the loss or partial loss of function depending on the spring bench setting of the following valve(s):
   - Letdown Isolation Valves
   - Feedwater Reg Valves
   - Feedwater Bypass Valves
   - Unit 3A and B Emergency Diesel Generator Day Tank Level Control Valve

3. A dual unit loss of instrument air (less than 60 psig) results in the loss of the additional functions:
   - Charging pump speed control (fails to high speed)
   - Primary Water Makeup capability
   - Loss of AUTOMATIC transfer of the Charging Pump Suction to the RWST
   - Unit 3 ECC valves fail open after a delay of 20 or more minutes.
   - Unit 4 ECC valves fail open.
   - Inability to control RCS cooldown using HCV-*-758 and FCV-*-605 and may require stopping the RHR Pump to stop a cooldown, if in progress when air was lost.

0-ONOP-105, Control Room Evacuation, provides useful guidance in determining components required for a plant cooldown. This procedure should be referenced if a cooldown must be performed without the availability of Instrument Air. Plant Management Staff and Engineering should be involved in any decision to attempt a cooldown without Instrument Air.

Question #: 1.1.44.45.1.2,M

71021450102; Annunciator I-6/1, INSTR AIR HI TEMP/LO PRESS" has alarmed on both Unit 3 & 4. Both Units are at 100% power. The following conditions have been identified: Unit 3 Instrument Air Pressure (PI-3-1444) < 65 psig and decreasing. Unit 4 Instrument Air Pressure (PI-4-1444) at 85 psig and beginning to slowly increase. What immediate action(s) must be performed by the RCO?

Distractor:
Direct an NPO to start additional Instrument Air Compressor.

Distractor:
Rapidly reduce load on Unit 3 IAW 3-ONOP-100.

Answer:
Trip Unit 3 and enter 3-EOP-E-0, while continuing 0-ONOP-013.

Distractor:
Trip both Unit 3 and 4 and enter EOP-E-0 on both Units, while continuing 0-ONOP-013.

REFERENCES:
; :REFERENCE: 0-ONOP-013

KEYWORDS/KSAs:
; RCO; 065 AA2.05

TP05301

Test Name	Test Date	rpb	p(Diff)	Time	Equ	User Values	
<Cumulative>		0.065	0.875	0	N	1: 0.000000	2: 0.000000
Farley 2001-301 SRO Test	07/17/2001	0.065	0.875	0	N	3: 0.000000	4: 0.000000

	--- A ---			--- B ---			--- C ---			--- D ---					
	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg
<Cumulative>						Total:	8	100				Omits:	0	0	
	7	88	84.71	1	13	84.00	0	0	0.00	0	0	0.00			
Farley 2001-301 SRO Test						Total:	8	100				Omits:	0	0	
	7	88	84.71	1	13	84.00	0	0	0.00	0	0	0.00			

52. 065AA2.06 001/1/1//C/A 3.6/4.2/B/FA03301/R/SDR

The reactor is at 30% power. For an unknown reason, instrument air pressure is falling. All available air compressors have been started and pressure continues to fall. The main feed regulating valve operation has started to become erratic. Feed flow is decreasing to the Steam Generators, levels are at 60% and slowly decreasing.

Which ONE of the following describes the action(s) the operator should take?

A.✓ Trip the reactor and go to EEP-0.

B. Trip the turbine and ramp the reactor to less than 2% power and establish AFW flow.

C. Ramp the turbine and reactor to below 5% and establish AFW flow.

D. Dispatch operators to manually jack open the main feed regulating valves to control SG level.

TP05301

Source: Farley Exam Bank Question #AOP-6.0-52520F08

A - Correct; AOP-6.0 Step 1, WHEN reactor critical AND control of critical AOVs erratic, THEN trip the reactor and go to EEP-0 REACTOR TRIP OR SAFETY INJECTION. FRV's are "critical valves.

B - Incorrect; turbine trip is not a priority in AOP-6 these are the actions of AOP-13.0, loss of feedwater.

C - Incorrect; ramping down is not an option.

D - Incorrect; not proceduralized, non-conservative, defeats FW Isolation signal.

Test Name	Test Date	rpb	p(Diff)	Time	Equ	User Values	
<Cumulative>		0.000	1.000	0	N	1: 0	2: 0
2002LOCTCY7Wk4Form2-2RO	05/17/2002	0.000	1.000	0	N	3: 0	4: 0
2002LOCTCY7Wk4Frm2-2SRO	05/17/2002	0.000	1.000	0	N		

	--- A ---			--- B ---			--- C ---			--- D ---					
	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg
<Cumulative>						Total:	18	100				Omits:	0	0	
	18	100	91.94	0	0	0.00	0	0	0.00	0	0	0.00			
2002LOCTCY7Wk4Form2-2RO						Total:	4	100				Omits:	0	0	
	4	100	93.75	0	0	0.00	0	0	0.00	0	0	0.00			
2002LOCTCY7Wk4Frm2-2SRO						Total:	14	100				Omits:	0	0	
	14	100	91.43	0	0	0.00	0	0	0.00	0	0	0.00			

13. 076GG2.4.49 001//T2G1/SWS/M(4.0/4.0)/N/TP05301/S/MC

Unit 3 is operating at 100% power when the following Annunciators actuate:

- I-4/2, ICWP A/B/C TRIP
- H-8/5, CCW HX OUTLET HI TEMP
- I-5/4, TPCW HI TEMP/LO PRESS
- E-9/4, GEN EXCITER AIR HI TEMP
- TE-3414 (det #31) cold air (fan discharge) as read on R-347 Pts 5 and 6 is 60^{o}C
- TE-3416 (det #33) hot air (exciter armature outlet) as read on R-347 Pts 7 and 8 is 87^{o}C
- Reactive generator load > 0 MVAR
- Operators verified all other associated plant parameters and conditions consistent with the above alarming conditions

Which ONE of the following describes the correct immediate operator responses?

A.✓ Start standby ICW pump then stop affected ICW pump, reduce reactive load on the generator.

B. Start standby ICW pump then stop affected ICW pump, initiate FAST LOAD REDUCTION.

C. Stop affected ICW pump then start standby ICW pump, initiate FAST LOAD REDUCTION.

D. Stop affected ICW pump then start standby ICW pump, reduce reactive load on the generator.

Feedback

REFERENCES:

1. ONOP-019, INTAKE COOLING WATER MALFUNCTION, page 5, rev 01/09/01C
2. ARP-097.CR, CONTROL ROOM ANNUNCIATOR RESPONSE, pages 288,445,472,480
3. ONOP-030, COMPONENT COOLING WATER MALFUNCTION, pages 6,7, rev 05/24/02

DISTRACTORS:

A Correct. Immediate actions IAW ref 1 and 2.

B Incorrect. Reduce reactive load vice FAST LOAD REDUCTION.

C Incorrect. Start standby ICW pump THEN stop affected ICW pump. Reduce reactive load vice FAST LOAD REDUCTION.

D Incorrect. Start standby ICW pump THEN stop affected ICW pump.

K/A CATALOGUE QUESTION DESCRIPTION:

- Service Water System (SWS); Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- *If the cause of the Intake Cooling Water Malfunction is determined to be due to high differential pressure on the traveling screens, then 3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION, should be used.*
- *If an Intake Cooling Water Pump is stopped in this procedure and the reason for stopping the pump has not been corrected, that pump is not available for starting in subsequent procedure steps.*

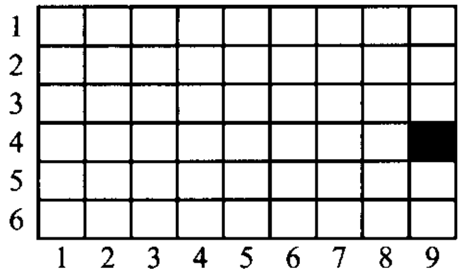
NOTE

Steps 1 and 2 are IMMEDIATE ACTION steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check Traveling Screens - CLEAN:** • Alarm I 3/3, Traveling Screen HI ΔP - OFF • Traveling Screen DP - LESS THAN 7.5 INCHES OF WATER	Go to 3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION
2	**Verify All Intake Cooling Water Pump Alarms - OFF** • I 4/1, ICWP A/B/C MOTOR OVERLOAD • I 4/2, ICWP A/B/C TRIP • I 4/3, ICWP A/B/C MOTOR BRG HI TEMP	Perform the following: 1. Have operator check pump(s) locally 2. Determine affected intake cooling water pump. 3. Start standby intake cooling water pump. 4. Stop affected intake cooling water pump.

BLUE	INVESTMENT PROTECTION	E 9/4

E36



ATTACHMENT 5
Page 52 of 54
Panel E

**GEN
EXCITER AIR
HI TEMP**

NOTES

- *CR Recorder R-3-347 Point #7 (hot air side Unit 3 north exciter cooler from TE-3-3416) was recording 6 deg C higher than actual which caused Ann E-9/4 spurious alarms. Point #7 was re-enabled with a 6 deg C lower offset to maintain Ann E-9/4 alarm clear.*
- *(OPS Action Item OP-03-06-004 will track this OTSC and verify that Point #7 is returned to normal value after repair scheduled by PWO 33008216).*

DEVICES:

PT	DESCRIPTION	SETPOINTS:
5	TE-3414 (det #31) cold air (fan discharge)	59 degrees C as read on R-347 Pts 5 and 6 for cold air
6	TE-3415 (det #32) cold air (diode wheel inlet)	
7	TE-3416 (det #33) hot air (exciter armature outlet)	87 degrees C as read on R-347 Pts 7 and 8 for hot air
8	TE-3417 (det #34) hot air (diode wheel outlet)	

OPERATOR ACTIONS:

1. Verify alarm by checking the following:
   a. Recorder R-347 pts 5 or 6 at 59 degrees C (VPA) **OR**
   b. Recorder R-347 pts 7 or 8 at 87 degrees C (VPA)
   c. Verify temperatures locally by using temperature probes at the test fixtures on the exciter house.
2. Corrective actions:

CAUTION

Switchyard voltage should be maintained at or above 232 Kv. Actions may be necessary to maintain Switchyard voltage greater than or equal to 232 Kv.

NOTES

- *An increasing trend in point 5 or 6 without a corresponding increase in point 7 or 8 (respectively) is indicative of a possible instrument problem.*
- *Hot air temperatures need to be maintained less than 90 degrees C, and cold air temperature limitations are required to be followed in order to preclude equipment damage.*
- *All voltage changes should be coordinated through FPL System Load Dispatcher.*

   a. **IF** the TPCW Supplemental Cooling System is expected to be in service, **THEN** verify that the system is still operational **WHILE** continuing with the rest of the corrective actions below.
   b. Monitor temperatures closely at intervals determined by the Shift Manager.
   c. **IF** either hot air temperature is ≥ 87 degrees C, **THEN** monitor grid voltage to ensure ≥ 232 kV is maintained **AND** reduce reactive load on the generator.
   d. **IF** either hot air temperature is ≥ 87 degrees C with reactive load at 0 MVAR, **THEN** monitor grid voltage to ensure ≥ 232 KV is maintained **AND** reduce generator load using 3-ONOP-100, FAST LOAD REDUCTION, until hot air temperatures stabilize below 87 degrees C.
   e. **IF** either hot air temperature is ≥ 87 degrees C and increasing due to an unknown cause, **THEN** consider tripping the Reactor and Turbine.
   f. **IF** either hot air temperature exceeds 90 degrees C, and there is no expectation of fast recovery, **THEN** trip the Reactor and Turbine and enter 3–EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.
   g. **IF** either cold air temperature is ≥ 59 degrees C, **THEN** dispatch an operator to increase flow or lower temperature of TPCW to the exciter air coolers.

CAUSES:

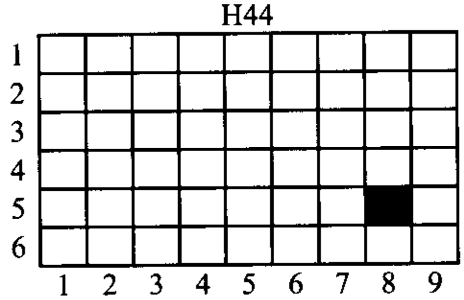
1. High exciter air temp
2. Instrument failure

REFERENCES:

1. W Dwg 687J759 TAB 7
2. W Field Action Report (FAR) Number 2-JB-3401-510
3. PTN-BFSI 94-006 Engineering Setpoint Calculation
4. CR 98-1130
5. CR 00-0935
6. PC/M 03-086 – U3 S/U Trans Volt. Tap Change

BLUE	INVESTMENT PROTECTION	H 8/5

H44



ATTACHMENT 8
Page 47 of 54
Panel H

CCW HX
OUTLET
HI TEMP

DEVICES:
TE-3-607A on A header
down stream from CCW HXs

TE-3-607A on B header
down stream from CCW HX

SETPOINTS:
120°F

OPERATOR ACTIONS:

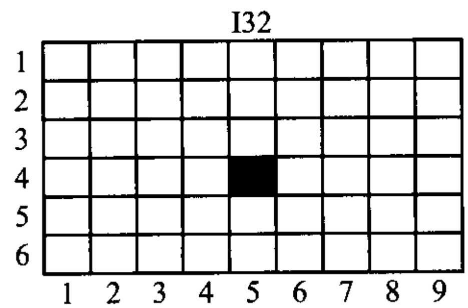
1. Verify alarm by checking the following:
   a. Flow/temperature of RHR HX, if in operation (VPB).
   b. CCW header flows (VPB).
   c. ICW discharge pressure (VPA).

2. Corrective actions:
   a. Dispatch Operator to check ΔP across ICW/CCW basket strainer.
   b. **IF** ΔP greater than 1.0 psid, **THEN** backwash ICW/CCW basket strainer.
   c. Verify all CCW HX in operation.
   d. **IF** RHR in cooldown alignment, **THEN** reduce rate of RCS cooldown.
   e. **IF** required, **THEN** start another CCW/ICW using 3-OP-019, INTAKE COOLING WATER SYSTEM, or 3-OP-030, COMPONENT COOLING WATER SYSTEM.
   f. Refer to 3-ONOP-030, Component Cooling Water Malfunction.
   g. Refer to 3-ONOP-019, Intake Cooling Water Malfunction.
   h. Refer to TS 3.7.2, 3.7.3, 3.4.1.3, 3.4.1.4.1, and 3.4.1.4.2.

CAUSES:
1. High CCW pump suction temperature due to:
   a. High RCS to CCW flow in RHR HX
   b. Dirty CCW HX tubes
   c. Dirty ICW/CCW basket strainer
   d. Low CCW System flow
   e. Clogged CCW HX tube sheets
   f. High RCS cooldown rate

REFERENCES:
1. FPL Dwg 5613-M-3030
2. Tech Spec Sections 3.7.2, 3.7.3, 3.4.1.3, 3.4.1.4.1, and 3.4.1.4.2

BLUE	INVESTMENT PROTECTION	I 5/4



ATTACHMENT 9
Page 28 of 54
Panel I

**TPCW
HI TEMP/
LO PRESS**

DEVICES:

PS-3-1621
TS-3-2106

SETPOINTS:

15 PSIG (Equivalent to 75 PSIG Header Pressure)
110°F (Tolerance between 108°F - 112°F)

OPERATOR ACTIONS:

1. Verify alarm by checking the following:
   a. Check TPCW header pressure equal to or less than 75 PSIG.

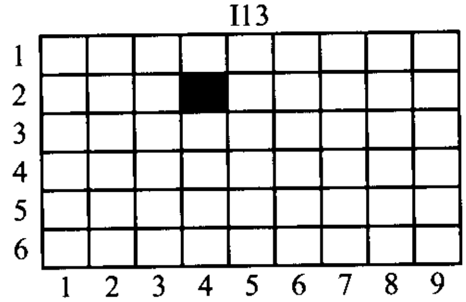
2. Corrective actions:
   a. In ERDADS, check TE-3-1472 to be between 108°F and 112°F.
   b. <u>**IF**</u> TPCW header low pressure condition exists <u>**THEN**</u> perform the following:
      (1) Start standby TPCW pump using 3-OP-008, TURBINE PLANT COOLING WATER MALFUNCTION.
      (2) Monitor pump amp indication on 3C04.
      (3) Locally check for system leakage.
   c. <u>**IF**</u> TPCW header high temperature condition exists <u>**THEN**</u> perform the following:
      (1) Verify ICW pumps running.
      (2) Verify at least one TPCW Heat Exchanger in service.
   d. Refer to 3-ONOP-008, Turbine Plant Cooling Water Malfunction.

CAUSES:
1. TPCW system leakage.
2. Inadequate ICW flow through TPCW HXs.
3. Inadequate TPCW system flow.
4. Plugged ICW/TPCW basket strainer.

REFERENCES:
1. 3-ONOP-008, Turbine Plant Cooling Water Malfunction
2. FPL Dwg 5613-M-3008
3. FPL EWD 5610-E-27, Sh 26, Mechanical Auxiliaries

YELLOW	POWER PRODUCTION AVAILABILITY	I 4/2

I13



ATTACHMENT 9
Page 20 of 54
Panel I

ICWP A/B/C
TRIP

DEVICES:

DEVICES	SETPOINTS
3AA19-174-RELAY (3A ICWP) TDO	105 AMPS
3AB17-174-RELAY (3B ICWP) TDO	105 AMPS
3AD05-174 RELAY (3C ICWP) TDO	105 AMPS

OPERATOR ACTIONS:

1. Verify alarm by checking the following:
   a. Check for any ICW pump mismatched breaker indication:
      (1) green indicating light(s) lit.
      AND
      (2) red control switch flag present.

2. Corrective actions:
   a. Refer to 3-ONOP-019, ICW Malfunction.
   b. Refer to Tech Spec 3.7.3, ICW System.

CAUSES:
1. Motor failure.
2. Pump failure.
3. Excessive ΔP across traveling screens.

REFERENCES:
1. Tech Spec 3/4.7.3
2. 3-ONOP-019, Intake Cooling Water Malfunction
3. FPL EWD 5613-E-27, Sh 2A and 2A1, ICW Pump 3A/3AA19
4. FPL EWD 5613-E-27, Sh 2B and 2B1, ICW Pump 3B/3AB17
5. FPL EWD 5613-E-27, Sh 2C and 2C1, ICW Pump 3C/3AD05

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If any RCP bearing temperature annunciator alarm actuates AND its associated motor bearing temperature is greater than 195°F, THEN trip the reactor and stop the affected RCP(s).

NOTES

- *Steps 1 and 2 are IMMEDIATE ACTION steps.*
- *Foldout page should be monitored throughout this procedure.*
- *A time delay exists on TR-320 from when an RCP parameter exceeds its setpoint to when the recorder provides indication and alarm. Use the RCP mimic display on ERDADS as a backup to TR-320 to monitor affected RCP parameters.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Verify Flow In Both Component Cooling Water Headers - NORMAL** • FT-3-613A for header A • FT-3-613B for header B	Perform the following: a. **IF** starting an idle CCW pump will **NOT** overload an EDG, **THEN** start CCW pumps as necessary to establish flow in both headers. b. **IF** CCW flow to RCPs can **NOT** be established, **THEN** manually trip the reactor **AND** verify reactor trip using the EOP Network, **THEN** stop all RCPs. c. Isolate Letdown and Excess Letdown. d. **IF** any charging pump is running, **THEN** operate at maximum speed until Attachment 1 is completed. e. Dispatch an operator to establish emergency cooling water to desired charging pump using Attachment 1.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *The top of the component cooling water surge tank divider plate is located at approximately 25% indicated level.*
- *If a cross tie valve between the units is leaking or open, the surge tank on the opposite unit may be experiencing level control problems.*
- *If in Modes 1 through 3, and CCW System level is NOT maintained within the CCW Head Tank, restore CCW System level to be within the CCW Head Tank within 24 hours.*
- *LI-3-613A and LI-3-614A are NOT overlapping (i.e., LI-3-614A will go off scale low before LI-3-613A comes off its high peg with decreasing level).*

2 **Verify Component Cooling Water Surge Tank Level Being Maintained**

a. Component Cooling Water Surge Tank Level, LI-3-613A -

- GREATER THAN 25%

AND

- STABLE OR INCREASING

Response Not Obtained:

Perform the following:

1. Open Component Cooling Water Surge Tank Makeup, MOV-3-832 as necessary to add makeup.
2. **IF** Component Cooling Water Surge Tank Level can **NOT** be maintained, **THEN** perform the following:
   a) Trip the reactor.
   b) Stop all RCPs.
   c) Perform 3-EOP-E-O, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.
3. OBSERVE NOTES PRIOR TO STEP 6 and Go To Step 6.

1.0 PURPOSE

1.1 This procedure provides instructions to be followed in the event of a malfunction or failure in the Intake Cooling Water (ICW) System.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 Visible evidence of excessive system leakage.

2.2 Catastrophic failure due to passage of heavy loads across system piping or electrical ducts.

2.3 Annunciators:

2.3.1 H 8/5, CCW HX OUTLET HI TEMP

2.3.2 I 4/1, ICWP A/B/C MOTOR OVERLOAD

2.3.3 I 4/2, ICWP A/B/C TRIP

2.3.4 I 4/3, ICWP A/B/C MOTOR BRG HI TEMP

2.3.5 I 4/4, ICW HEADER A/B LO PRESS

2.3.6 I 5/4, TPCW HI TEMP/LO PRESS

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 Technical Specifications

1. Section 3.7.3, Intake Cooling Water System

3.1.2 FSAR

1. Section 9, Auxiliary and Emergency Systems

3.1.3 Plant Drawing

1. 5613-M-3019, Sht 1 and Sht 2, Intake Cooling Water System

FOLDOUT PAGE FOR 3-ONOP-019

1. **TRIP CRITERIA**

   - Component Cooling Water temperature as read on TI-3-607A and TI-3-607B cannot be maintained less than 120°F.
   - Turbine or Generator bearing temperatures cannot be maintained less than 180°F.

2. **MINIMUM FLOW REQUIREMENTS FOR CCW HXs**

   While isolating a CCW/ICW strainer, ICW flow less than minimum required through the CCW HXs can be tolerated without entry into Technical Specification Action 3.0.3, provided flow is restored to the minimum allowable, as determined by Enclosure 1 of 3-OP-019, Intake Cooling Water System, in less than 5 minutes by reopening the strainer isolation valves. If flow is below the minimum allowable value for greater than 5 minutes, then entry into Technical Specification Action 3.0.3 is started at the point where flow first fell below the minimum value. [Reference 3.1.4]

QUESTIONS REPORT
for Draft TP05-301-SRO

14. G2.1.14 001//T3/OPS/M(2.5/3.3)/N/TP05301/S/MC

Unit 3 has been in Mode 3 for three days to facilitate the performance of maintenance. Prior to this, Unit 3 had been operating at 100% power for an extended period of time. Following completion of the work the unit will be returned to full power operation.

Which ONE of the following identifies the plant personnel that are required to be notified prior to entering Mode 2 and again prior to entering Mode 1?

	MODE 2	MODE 1
A.✓	Site Vice President	Plant General Manager
B.	Reactor Engineering	Work Control Center Supervisor
C.	Chemistry Department	Security Department
D.	Nuclear Plant Supervisor	Operations Shift Manager

Feedback

REFERENCES:

1. GOP-301, HOT STANDBY TO POWER OPERATION, pages 16,19,62, rev 04/22/04

DISTRACTORS:

A Correct. IAW GOP-301, sections 3.1.19 & 3.2.17 the following personnel are to be notified prior to entry into Modes 1 & 2: Site VP, Plant GM, and Operations Shift Manager.

B Incorrect. See A.

C Incorrect. See A.

D Incorrect. See A.

K/A CATALOGUE QUESTION DESCRIPTION:

- Knowledge of system status criteria which require the notification of plant personnel.

14. G2.1.14 001//T3/OPS/M(2.5/3.3)/N/TP05301/S/MC

Unit 3 had been operating at 100% power prior to shutting down to Mode 3 for three days for maintenance.

Based on the above information and IAW GOP-301, HOT STANDBY TO POWER OPERATION, which ONE of the following indicates the plant personnel that are required to be notified prior to entry into Mode 2 and again prior to entry into Mode 1?

A.✓	Site Vice President	Plant General Manager
B.	Reactor Engineering	Work Control Center Supervisor
C.	Chemistry Department	Security Department
D.	Nuclear Plant Supervisor	Operations Shift Manager

Feedback

REFERENCES:

1. GOP-301, HOT STANDBY TO POWER OPERATION, pages 16,19,62, rev 04/22/04

DISTRACTORS:

A Correct. IAW GOP-301, sections 3.1.19 & 3.2.17 the following personnel are to be notified prior to entry into Modes 1 & 2: Site VP, Plant GM, and Operations Shift Manager.

B Incorrect. See A.

C Incorrect. See A.

D Incorrect. See A.

K/A CATALOGUE QUESTION DESCRIPTION:

- Knowledge of system status criteria which require the notification of plant personnel.

INITIALS

____ 3.1.16 Verify 0-ADM-009, Containment Entry When Integrity is Established, Attachment 4, Entry and Exit Requirements Data Sheet, Subsection 5.5, Exiting the Containment, has been satisfactorily completed. (N/A if no containment entry has been made.)

3.1.17 Verify the following documents have been reviewed to ensure no Technical Specification related equipment required for Mode 2 is inoperable:

____ 1. EOOS records [Commitment – Step 2.3.5]

____ 2. Equipment Clearance Orders [Commitment – Step 2.3.5]

____ 3. Locked Valve Deviation Log

____ 4. Caution Tag Indexes

____ 5. Operator Log Readings

____ 3.1.18 Verify 3-OP-038.9, Refueling Activities Checkoff List, Attachment 1 has been completed. (N/A if not returning from refueling.)

3.1.19 Notify the following personnel to review the requirements of 0-ADM-529, Unit Restart Readiness, prior to entry into Mode 2:

____ 1. Site Vice President

____ 2. Plant General Manager

____ 3. Operations Shift Manager

3.2 Verify that the following steps have been completed prior synchronizing the generator:

NOTE

The following system alignment is required to be completed when returning from a refueling outage. For cold SNOs, the Operations Manager or Designee may waive any or all of the alignment requirements by initialing next to the item to be waived, and putting N/A in the completed by column.

CMPLTD BY

3.2.1 Perform the following system alignment unless waived by the Operations Manager or Designee:

____ 1. 3-OP-093.1, ATWS MITIGATING SYSTEM ACTUATION CIRCUITRY (AMSAC)

INITIALS
CMPLTD
BY

3.2.17 Notify the following personnel to review the requirements of 0-ADM-529, Unit Restart Readiness, prior to entry into Mode 1:

____ 1. Site Vice President

____ 2. Plant General Manager

____ 3. Operations Shift Manager

____ 3.2.18 Verify that all required surveillances for entering Mode 1, are current using 0-ADM-215, PLANT SURVEILLANCE TRACKING PROGRAM, Mode Change Report, required for entry into Mode 1, Power Operation. (If the surveillance computer is out of service, the manual log for 0-ADM-215, Plant Surveillance Tracking Program, may be utilized.)

NOTES

- *The Operations Manager or designee shall initial next to all steps to be waived, and N/A the completed by column prior to beginning Subsection 3.3. All required steps shall be initialed by the operator when completed.*
- *This is a work list of alignments that, in addition to the minimum required alignments for refueling outages listed in Subsection 3.2, may be required to be completed prior to placing the system in service or entry into Mode 1, as designated by Operations Manager or designee.*
- *Any step not completed shall be listed in the Remarks Section of this step. The exceptions shall list the specifics of each exception as shown in the documentation example prior to the Remarks Section.*

3.3 Verify the following systems have the applicable alignments completed as indicated by the Operations Manager or designee:

____ 3.3.1 0-OP-001.1, Plant Page System

____ 3.3.2 0-OP-003.1, 125V Vital DC System

INIT

5.98 **IF** the Reactor has not operated at full power since the last refueling outage, **THEN** notify Reactor Engineering that the following items will be needed:

____ 5.98.1 Perform an NIS calibration using 0-OSP-059.15, NUCLEAR INSTRUMENTATION CHANNEL CHECK AND CALIBRATION

____ 5.98.2 Perform a power coefficient test, power defect test, or Critical Boron Concentration Test as determined by the Reactor Supervisor.

____ 5.98.3 Perform 3-OSP-040.15, CALORIMETRIC VERIFICATION OF REACTOR COOLANT SYSTEM flow.

____ 5.98.4 Perform 3-OSP-059.5, POWER RANGE NUCLEAR INSTRUMENTATION SHIFT CHECKS AND DAILY CALIBRATION, at greater than 99 percent power and equilibrium reactor conditions **AND** verify the extrapolated full power Delta T values on Data Sheet, Full Power Delta T Extrapolation and Measurement, are still acceptable.

____ 5.98.5 Complete Intermediate Range Setpoint Check and Calibration using 0-OSP-059.15, NUCLEAR INSTRUMENTATION CHANNEL CHECK AND CALIBRATION.

____ 5.99 Request Reactor Engineering to align MIMS for Mode 1, Steady State Operations, using 3–OP-099, METAL IMPACT MONITORING SYSTEM.

5.99.1 **IF** Reactor Engineering is not available, **THEN** place MIMS in service using 3-OP-099, METAL IMPACT MONITORING SYSTEM.

____ 5.100 Verify that the Gamma Metric Wide Range Percent Power Meter reads within 1.5 percent of the Westinghouse Power Range Instrumentation when reactor power is at 98.5 to 100 percent.

5.100.1 **IF** a Gamma Metrics Channel is not reading within 1.5 percent of the Westinghouse Power Range Instrumentation, **THEN** have I&C perform 3-PMI-059.2, GAMMA METRICS WIDE RANGE PERCENT OF POWER METER CALIBRATION.

5.101 **WHEN** the Steam Jet Air Ejector is in service **AND** the Hogging Ejector is secured, **THEN** perform the following:

____ 5.101.1 Close or verify closed, SJAE Main Stm Sply CV-3700 Byp Angle Vlv, 3-30-026.

____ 5.101.2 Close or verify closed, SJAE Main Stm Sply CV-3700 Byp Throt, 3-30-027.

____ 5.101.3 Close or verify closed, SJAE Main Stm Sply RO-1454 Inlet Isol, 3-30-029.

____ 5.101.4 Close or verify closed, SJAE Main Stm Sply RO-1455 Inlet Isol, 3-30-031.

ENCLOSURE 3
(Page 1 of 1)
PLANT MANAGEMENT NOTIFICATIONS

NOTES

- *These events may require the generation of a Condition Report. Refer to 0-ADM-518, Condition Reports for further guidance.*
- *Ensure the ENS worksheet (Form similar to Attachment 2) is attached to the Condition Report, IF NRC reportability is required.*

The following notifications are informational in nature. The Duty Call Supervisor shall confirm with the Plant General Manager that the NRC Resident Inspector does not need to be notified for the events listed below:

1. An unplanned unit load reduction of greater than 50 MW for greater than 1 hour.
2. Any event listed in Nuclear Policy 600 (NP-600), Events Warranting Public Dissemination via Corporate Communications, that is not listed in Enclosure 1, or included in Enclosure 2.
3. Significant equipment/component problems that could jeopardize continued plant operation.
4. Fires requiring Fire Brigade activation or any unplanned major impairment of the Fire Protection System.
5. Events requiring Special Reports to the NRC, e.g., lifting a PORV. |
6. Acts or events where there may be indications of Tampering, Vandalism, or Malicious Mischief.

INITIALS
CMPLTD
BY

3.1.1 (Cont'd)

____ 9. 3-OSP-046.3, CVCS - Boration Systems Flowpath Verification, Attachments 1 through 3

____ 10. 3-OP-047, CVCS - Charging and Letdown, Attachment 1

____ 11. 3-OP-050, Residual Heat Removal System, Attachments 2 through 5

____ 12. 3-OP-055, Emergency Containment Cooling and Filtering System, Attachments 1 through 3

____ 13. 3-OP-061.3, Reactor Coolant Drain Tank, Attachments 1 and 2

____ 14. 3-OP-062, Safety Injection, Attachments 1 through 5

____ 15. 0-OP-065.3, Nitrogen Gas Supply System, Attachment 4

____ 16. 0-OP-065.4, Steam Dump to Atmosphere Controller Backup N_2 Gas Supply System, Attachment 1

____ 17. 3-OP-067, Process Radiation Monitoring System, Attachments 1 and 2

____ 18. 3-OP-068, Containment Spray System, Attachments 1 and 2

____ 19. 3-OP-071, Steam Generator Blowdown Recovery System, Attachment 1. [Commitment - Step 2.3.14]

____ 20. 3-OP-072, Main Steam System, Attachment 1

____ 3.1.2 **WHEN** recovering from a refueling outage, **THEN** the Refueling Coordinator Verification Point in 3-OP-038.9, REFUELING ACTIVITIES CHECKOFF LIST, shall be signed off prior to beginning RCS System heatup. (N/A if not returning from refueling)

NOTE

Steam Generator samples should be obtained between 180°F to 200°F.

____ 3.1.3 Verify with the Chemistry Department that the Steam Generator Chemistry meets the requirements for entering Mode 4.

____ 3.1.4 To ensure revitalization of required areas per 0-ADM-404, Devitalization and Revitalization of Vital Areas, notify the Security Department approximately 24 hours prior to changing the status of a unit from Mode 5 to Mode 4. This notification may be made by telephone and shall be documented in the Unit 3 RCO Logbook and Security Logs.

QUESTIONS REPORT
for Westinghouse 4-Loop Questions

2. G2.1.14 001

Which ONE of the following is required to be notified when a Temporary Alteration installation has been completed or returned to normal?

A. Unit Operator

B. Operations Superintendent

C.✓ Shift Manager

D. Unit Manager

Feedback

Ref: WB procedure SPP-9.5, Temporary Alterations

Distractor analysis:
SPP-9.5 states Shift Manager or "designee", which is the Unit Supervisor.

Notes

Categories

RO Tier:	T3	SRO Tier:	T3
K/A Value:	NOTIFICATION	Cog. Level:	M 2.5/3.3
Source:	N	Exam:	WB020301
Test:	S	Misc:	RLM

Reactor Operator
Unit Supervisor
Shift Manager

Site VP, Plant GM, ~~Ops Shift Mngr~~
~~Site VP, Plant GM~~,
Plant GM, ~~Ops Shift Mngr~~, Work Control Cen ~~Cp~~
Ops Shift Mngr, Reactor Engineers
Ops Shift Mngr, Security Dept

Work Control Center Supervisor (WCCS)

NAP-402

GOP-103 pg 11

GOP-301 pg 16

ADM-503 pg 20

Shift Eng
Shift Manager

Ops manager
Site VP
Plant GM
Ops shift mngr
System Engineer
Work Control Center Sup.
ERDADS System Eng
— NPS
Nuclear Plant Supervisor
Reactor Engineering
Security Dept

TP05301

Test Name	Test Date	rpb	p(Diff)	Time	Equ	User Values	
<Cumulative>		0.000	0.000	0	N	1: 0.000000	2: 0.000000
						3: 0.000000	4: 0.000000

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

61. G2.1.14 002/T3/T3/NOTIFICATION/M 2.5/3.3/N/VG02301/S/CKFOFV

Given the following Conditions/events:

- Unit 1 is at 100% power
- Steam Generator Tube Rupture has occurred
- Reactor is manually tripped and SI is actuated
- Emergency Director declares an ALERT

Which ONE of the following is the correct notification required for plant personnel according to Procedure 00004-C, "Plant Communications"?

A. A warble tone would be sounded for 15 seconds, then make page announcement warning personnel of the tube rupture and its location.

B. A siren tone would be sounded for 15 seconds, then make page announcement warning personnel of the tube rupture and its location

C. Page announcement warning personnel of the tube rupture and its location there is no alarm required.

D.✓ Page announcement warning personnel of the tube rupture and its location, then sound warble tone for 15 seconds.

Ref: VG 00004-c

Distractor analysis

per procedure step 5.1.2

15. G2.1.33 001//T3/OPS/C/A(3.4/4.0)/N/TP05301/S/MC

Plant conditions for Unit 4 are as follows:

- Mode 2
- MTC = 5.5×10^{-5} delta k/k/^oF
- BOL
- Physics testing is in progress
- All shutdown bank rods are fully withdrawn

Which ONE of the following will require Unit 4 to enter a Technical Specification Action Condition?

A. K_{eff} = .99 and MTC = -3.0×10^{-4} delta k/k/^oF Lowest T_{avg} = 611^oF

B. K_{eff} = 1 and MTC = -3.0×10^{-4} delta k/k/^oF Highest T_{avg} = 610^oF

C. K_{eff} = .99 and MTC = 5.5×10^{-5} delta k/k/^oF Lowest T_{avg} = 531^oF

D.✓ K_{eff} = 1 and MTC = 5.5×10^{-5} delta k/k/^oF Highest T_{avg} = 530^oF

Feedback

REFERENCES:

1. Tech Spec 3.1.1.3, MODERATOR TEMPERATURE COEFFICIENT, pages 1-5 & 1-6, Amendment Nos. 137 & 132
2. Tech Spec 3.10.3, PHYSICS TESTS, page 10-3, Amendment Nos. 137 & 132
3. Tech Spec 2.1.2, SAFETY LIMITS – REACTOR CORE, pages 2-1 & 2-2, Amendment Nos. 137 & 132

DISTRACTORS:

A Incorrect. Applicable at EOL, not BOL.

B Incorrect. Applicable at EOL, not BOL.

C Incorrect. Applicable only with K_{eff} greater than or equal to 1.

D Correct. IAW TS 3.10.3.c given the other plant conditions, action step "b" applies.

K/A CATALOGUE QUESTION DESCRIPTION:

- Conduct of Operations; Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than or equal to 5.0×10^{-5} $\Delta k/k/°F$ for all rods withdrawn, beginning of the cycle life (BOL), hot zero THERMAL POWER (HZP) conditions; and

b. Less positive than or equal to 5.0×10^{-5} $\Delta k/k/°F$ from HZP to 70% RATED THERMAL POWER condition; and

c. Less positive than or equal to 5.0×10^{-5} $\Delta k/k/°F$ from 70% RATED THERMAL POWER decreasing linearly to less positive than or equal to 0 $\Delta k/k/°F$ at 100% RATED THERMAL POWER conditions; and

d. Less negative than -3.5×10^{-4} $\Delta k/k/°F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a, b and c. - MODES 1 and 2* only**.
Specification 3.1.1.3d. - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a, b or c above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive or equal to limits described in 3.1.1.3a, b and c above within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

* With K_{eff} greater than or equal to 1.

** See Special Test Exceptions Specification 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. With the MTC more negative than the limit of Specification 3.1.1.3d. above, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.0×10^{-4} $\Delta k/k/°F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.0×10^{-4} $\Delta k/k/°F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3d., at least once per 14 EFPD during the remainder of the fuel cycle.

c. Perform design calculation to verify conformance to Specifications 3.1.1.3b and c.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F at least once per 30 minutes during PHYSICS TESTS.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and **the highest operating** loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1, for 3 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

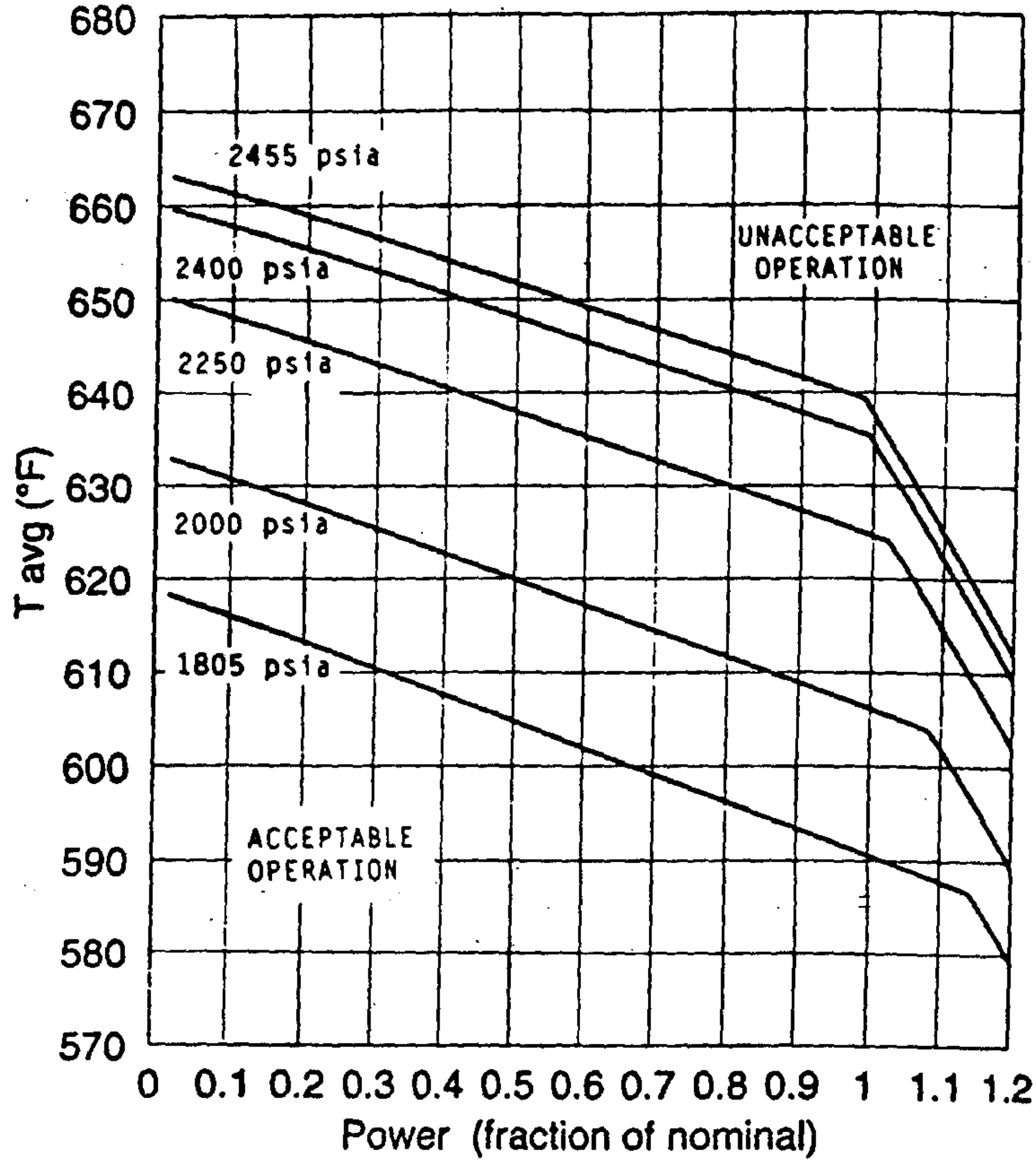


Figure 2.1-1 Reactor Core Safety Limit – Three Loops in Operation

16. G2.2.20 001//T3/EQUIP CONT/M(2.2/3.3)/B/TP05301/S/MC

Unit 4 is at 100% power. Mechanical Maintenance is planning to erect a scaffold over Redundant Safety Related Equipment to perform trouble shooting activities.

Which ONE of the following identifies the highest level of approval required for the erection of this scaffolding?

A. Operations Manager

B. ANPS

C. NPS

D✓ Operations Supervisor

Feedback

REFERENCES:

1. ADM-012, step 3.3.2

DISTRACTORS:

A Incorrect.

B Incorrect.

C Incorrect.

D Correct.

K/A CATALOGUE QUESTION DESCRIPTION:

– Equipment Control; Knowledge of the process for managing troubleshooting activities.

Question #: 1.1.23.35.2.9,M

69020350209; Unit 3 is at 100% power. Mechanical Maintenance is planning to erect a scaffold over Redundant Safety Related Equipment to perform trouble shooting activities. Which ONE of the following identifies the highest level of approval required for the erection of this scaffolding?

Distractor:
ANPS

Distractor:
NPS

Answer:
Operations Supervisor

Distractor:
Operations Manager

REFERENCES:
; :Reference: ADM-012 step 3.3.2, PTN NRC SRO Q, RCO 18

KEYWORDS/KSAs:
; NRC Exam Question; 2.2.20

3.7 PWO Originator

3.7.1 Provide all required information for originating a PWO.

3.7.2 Place a Deficiency Tag on deficient power block (excluding containment) equipment/components when originating a PWO.

3.7.3 Notify the NPS/ANPS of all Trouble and Breakdown PWOs originated relating to permanently installed plant equipment necessary for plant operation.

3.7.4 Notify the appropriate department of all Non-Trouble and Breakdown PWOs originated and provide supporting documents (PC/M, MSP, CR, letters, etc.).

3.8 Nuclear Plant Supervisor/Assistant Nuclear Plant Supervisor

3.8.1 Review and approve all Trouble and Breakdown PWOs in accordance with this procedure.

3.8.2 Authorize the start of work and acknowledge the completion of work on power block related PWOs in accordance with this procedure.

3.8.3 Shall designate a SRO as the liaison between the NPS and the troubleshooting ~~team~~ for troubleshooting sensitive or load threatening equipment. The troubleshooting plan shall be reviewed by the SRO prior to the implementation of the plan. Completed troubleshooting activities shall have an independent assessment by the SRO and applicable maintenance supervision to ensure prop development of subsequent troubleshooting steps. All changes to th troubleshooting plan, for sensitive or load threatening equipment, shall be reviewed by the SRO prior to implementation. The SRO shall keep the NPS informed of the troubleshooting plan of action, progress and subsequent changes to the plan.

3.9 Planning Supervisor/Fin Team Leader

3.9.1 Review and sign all Safety Related (SR) and Quality Related (QR) work control documents generated to implement a PC/M (MSP) and any changes thereto prior to implementation. The review shall include proper parts Procurement Classification (PC) levels, as well as, proper planning from design documents.

3.9.2 Review all SR and QR work orders for content and quality to ensure that the correct part/material PC levels are being installed in SR and QR systems in accordance with 0-ADM-047, Identification and Control of Safety Related and Quality Related Parts, Materials and Components, as follows:

1. Safety Related (SR) systems shall only have PC-1 or PC-2 level material installed.

   a. PC-2 material shall have a Dedication Package for that application.

   b. PC-3 material may only be installed in a SR Host Component with an approved Procurement Engineering (PE) evaluation.

2. Quality Related (QR) systems shall have only PC-3 level material (or better) installed.

ATTACHMENT E
REACTIVITY MANAGEMENT CONTROLS
(Page 14 of 27)

3.6 Fuel Handling SRO

1. Stops fuel handling when any discrepancy arises that adversely affects nuclear safety, specifically core reactivity.

2. Supervises fuel handling activities, including core loading, unloading, and fuel transfers within the fuel pool.

3. Ensures fuel-handling activities are conducted in accordance with approved procedures.

4. Ensures fuel-handling personnel demonstrate proper respect for the reactivity potential of the reactor by communicating the importance of procedural adherence, attention to detail, professional conduct, and conservative response to abnormal events.

5. Determines when fuel handling ends and troubleshooting and maintenance activities begin

3.7 Fueling Handling personnel

1. Perform fuel handling activities, including core loading and unloading, fuel transfers within the fuel pool, fuel receipt and inspection, fuel pool gate moves, testing of fuel handling interlocks.

2. Perform fuel-handling activities in accordance with approved procedures.

3. Demonstrate proper respect for the reactivity potential of the reactor through procedural adherence, attention to detail, professional conduct, and conservative response to abnormal events.

ATTACHMENT I
OPERATION OF PLANT EQUIPMENT
(Page 2 of 2)

4.0 MAIN BODY

4.1 Control room personnel shall ensure reactor control manipulations are performed by licensed Reactor Operators or Senior Reactor Operators only, with the exception of trainees enrolled and participating in an approved license training program under the direct supervision of a licensed operator.

4.2 Devices in the control room and the plant may need to be manipulated as part of troubleshooting, surveillance or routine test procedures. Normally, the individual performing the test should request that the appropriate control room operator or equipment operator manipulate the controls as required by the applicable procedure. Shift Management may allow the manipulation of controls by other than the duty control room operator(s) or equipment operators under the following conditions:

1. The required manipulations are part of an approved procedure or work package (i.e. surveillance/routine, troubleshooting control procedure or special test).
2. The proposed manipulations have been discussed with the on-duty control room team in a pre-job brief.
3. The devices to be manipulated are not "reactor controls" or impair the monitoring of reactor parameters.
4. Plant system and status is such that the manipulation of the control(s)/device(s) will not change plant conditions (i.e. within the isolation boundary of an active clearance order).
5. The responsible operator must be notified of changes to equipment status.

END OF ATTACHMENT I

3.10 Maintenance Planners

3.10.1 Research, plan and produce ready to work PWO packages in accordance with appropriate procedures, instructions, and drawings.

3.10.2 Interface with and assist, as required, other departments to provide information, clarification, and PWO requirements coordination.

3.11 General Maintenance Leader (Leader)/Supervisor

3.11.1 Ensure prerequisite conditions and required notifications to appropriate groups or individuals are complete prior to starting work.

3.11.2 Ensure that the correct part/material Procurement Classification (PC) Levels are being installed in SR and QR systems in accordance with 0-ADM-047, Identification and Control of Safety Related and Quality Related Parts, Materials and Components, as follows:

1. Safety Related (SR) systems shall only have PC-1 or PC-2 level material installed.

   a. PC-2 material shall have a Dedication Package for that application.

   b. PC-3 material may only be installed in a SR Host Component with an approved Procurement Engineering (PE) evaluation.

2. Quality Related (QR) systems shall have only PC-3 level material (or better) installed.

3.11.3 Ensure that proper work methods are being employed, all equipment is properly protected, and housekeeping requirements are maintained at all times.

3.11.4 Ensure post maintenance testing is adequate to check work performed and the Journeyman's Work Report is complete.

1. In specific cases where PMT is complex, assistance may be obtained from Engineering to determine/verify the PMT.

3.11.5 Ensure that Hazard Assessment Tailboard Briefings are held prior to the performance of any job.

3.11.6 Ensure that only qualified and capable personnel are assigned to perform Safety Related/Quality Related work independently.

3.11.7 Ensure that non-qualified personnel work under the direction of a qualified individual.

3.11.8 When troubleshooting sensitive or load threatening equipment, ensure that a SRO is designated as the liaison between the NPS and the troubleshooting team. The troubleshooting plan shall be reviewed by the SRO prior to the implementation of the plan. Completed troubleshooting activities shall have an independent assessment by the SRO and applicable maintenance supervision to ensure proper development of subsequent troubleshooting steps. All changes to the troubleshooting plan, for sensitive or load threatening equipment, shall be reviewed by the SRO prior to implementation. The SRO shall keep the NPS informed of the troubleshooting plan of action, progress and subsequent changes to the plan.

ATTACHMENT B

ROLES AND RESPONSIBILITIES OF OPERATIONS GROUP PERSONNEL

(Page 8 of 22)

4.2 Unit Supervisor (continued)

9. Maintaining close involvement with load-threatening activities.

10. Has the authority to direct the shut down of their assigned unit if conditions warrant this action.

11. Has the authority to initiate the Emergency Plan and, in the absence of/or unavailability of the Shift Manager, is the Emergency Coordinator when the Emergency Plan is in effect.

12. During emergencies or transients, the Unit Supervisor is in charge of the control room and shall remain in the control room unless properly relieved of duty.

13. Monitors and controls access to and activities in the control room to ensure proper decorum and professionalism are maintained.

14. Authorizes testing, surveillances, outages, and maintenance on all equipment and systems affecting plant safety. Similarly, ensures equipment is properly restored following completion of these activities.

15. Ensures activities listed on the site work schedule impacting the control room are completed as scheduled, and shall ensure appropriate personnel are notified if planned work is delayed.

16. Ensures abnormal conditions are investigated, including initial troubleshooting, verifying proper information is gathered and verifying appropriate corrective actions are established.

17. Report all significant plant changes, unsafe trends, unsafe conditions, or Operator Work Arounds to the Shift Manager.

18. Factors making up this professional atmosphere include knowledge of all aspects of plant status by licensed control room operators, maintaining an orderly and clean working environment, aggressiveness of the operating staff to prevent operational problems, and correcting observed deficiencies.

QUESTIONS REPORT
for Draft TP05-301-SRO

17. G2.2.32 001//T3/EQUIP CONT/C/A(2.3/3.3)/B/TP05301/S/MC

Reactor engineering had designed a core loading pattern that will be performed during the next refueling outage. The CHANGE will result in placing the "twice-burned" fuel assemblies more toward the periphery and the new fuel assemblies more toward the center of the core. Based on engineering calculations, it has been determined that Kexcess will be the same at the beginning of both fuel cycles.

Based on the above information, which ONE of the following describe the affect the new loading pattern will have on the unit?

A. The expected full power loop T value should be significantly LOWER for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

B. The expected full power loop T value should be significantly HIGHER for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

C.✓ If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly BELOW actual power when the 1st calorimetric is performed after the refueling outage.

D. If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly ABOVE actual power when the 1st calorimetric is performed after the refueling outage.

Feedback

DISTRACTORS:

A Incorrect.

B Incorrect.

C Correct.

D Incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:

- Equipment Control; Knowledge of the effects of alterations on core configuration.

17. G2.2.32 001//T3/EQUIP CONT/C/A(2.3/3.3)/B/TP05301/S/MC

If the core loading pattern will be CHANGED during the next refueling outage to place the new fuel assemblies more toward the center of the core and the "twice-burned" assemblies more toward the periphery, THEN what affect would this loading pattern have on the unit?

Assume K_{excess} will be the same at the beginning of both fuel cycles.

A. The expected full power loop T value should be significantly LOWER for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

B. The expected full power loop T value should be significantly HIGHER for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

C.✓ If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly BELOW actual power when the 1st calorimetric is performed after the refueling outage.

D. If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly ABOVE actual power when the 1st calorimetric is performed after the refueling outage.

Feedback

DISTRACTORS:

A Incorrect.

B Incorrect.

C Correct.

D Incorrect.

K/A CATALOGUE QUESTION DESCRIPTION:
- Equipment Control; Knowledge of the effects of alterations on core configuration.

..2.2.32

9/1/2003

Prairie Island 1

Exam Level S

Mark Question | Print Recor | New Searc | Exit

Question

IF the core loading pattern will be CHANGED during the next refueling outage to place the new fuel assemblies more toward the center of the core and the "twice-burned" assemblies more toward the periphery, THEN what affect would this loading pattern have on the unit?

Assume Kexcess will be the same at the beginning of both fuel cycles.

Answer: IF PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly below actual power when the 1st calorimetric is performed after the refueling outage.

Distracter 1 The expected full power loop T value should be significantly lower for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

Distracter 2 The expected full power loop T value should be significantly higher for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

Distracter 3 IF PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly above actual power when the 1st calorimetric is performed after the refueling outage.

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

..2.2.32

10/1/2002

Diablo Canyon 1

Exam Level S

Mark Question | Print Recor | New Searc | Exit

Question

While refueling, two fuel assemblies of different enrichments were inadvertently loaded into the wrong core position. The error was not caught during reload verification.

Which one of the following would detect the error first during startup testing?

Answer: Incore flux detectors.

Distracter 1 Core physics data at 10-8 amps.

Distracter 2 Power range QPTR.

Distracter 3 Incore thermocouple data

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

K/A	TIER	KEYWORD	COGNITIVE LEVEL	SOURCE	EXAM
G2.2.32	ST3	EQUIP CONT	C/A(2.3/3.3)	B	TP05301

If the core loading pattern will be CHANGED during the next refueling outage to place the new fuel assemblies more toward the center of the core and the "twice-burned" assemblies more toward the periphery, THEN what affect would this loading pattern have on the unit?

Assume K_{excess} will be the same at the beginning of both fuel cycles.

A The expected full power loop T value should be significantly LOWER for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

B The expected full power loop T value should be significantly HIGHER for this fuel cycle when compared to the value of full power loop T for the previous fuel cycle.

C If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly BELOW actual power when the 1st calorimetric is performed after the refueling outage.

D If PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly ABOVE actual power when the 1st calorimetric is performed after the refueling outage.

REFERENCES:

1. reference, page ##, rev ##/##/##

DISTRACTORS:

A Incorrect. Why.

B Incorrect. Why.

C Correct. Why.

D Incorrect. Why.

K/A CATALOGUE QUESTION DESCRIPTION:

- Equipment Control; Knowledge of the effects of alterations on core configuration.

QUESTIONS REPORT
for Draft TP05-301-SRO

18. G2.3.6 001//T3/RAD CONTROL/C/A(2.1/3.1)/M/TP05301/S/MC

The following is a time-line of activities associated with Waste Monitoring:

- 1400: Waste Monitoring Tank (WMT) "A" is placed on recirc for sampling
- 1620: Chemistry completes sampling of WMT "A." Result = 5.5×10^{-5} Ci/ml
- 1730: Chemistry submits a Radiological Liquid Waste Discharge Permit for WMT "A"
- 1735: The Nuclear Plant Supervisor authorizes the release of the Radiological Liquid Waste Permit without the approval of the Radiochemist or Health Physics Supervisor
- 1745: Operators align WMT "A" for discharge and start the release
- 1746: R-18, Waste Disposal System Liquid Effluent Monitor, fails low. The release is terminated and WMT "A" is restored to a normal lineup
- 1930: The R-18 monitor is repaired and restored to service
- 1935: The Shift Manager re-authorizes the release of WMT "A" on the same Radiological Liquid Waste Permit
- 1940: Operators re-align WMT "A" for discharge and start the release

Based on the above information, which ONE of the following represents the problem associated with these actions?

A.✓ The sample taken for the Radiological Liquid Waste Permit is NOT representative of the current contents of WMT "A" now being released.

B. A Radiological Liquid Waste Permit approved for one shift may NOT be used for initiation of a release on the next shift.

C. The discharge required the approval of the Health Physics Supervisor in addition to the Nuclear Plant Supervisor.

D. The contents of WMT "A" must first be transferred to the Waste Holdup Tanks for further processing prior to release.

Feedback

REFERENCES:

1. SD 049/SYS.061A, pages 27-29, rev 02/01/02
2. OP-061.11, CONTROLLED LIQUIED RELEASE TO THE CIRCULATING WATER
3. NCAP-003, PREPARATION OF LIQUID RELEASE PERMIT

DISTRACTORS:

A Correct. IAW NCAP-003, PREPARATION OF LIQUID RELEASE PERMIT.

B Incorrect. Not required.

C Incorrect. As long as the specific activity of the tank contents is greater than or equal to 1 x 10^{-4} Ci/ml, only the NPS's approval is required.

D Incorrect. Not required.

K/A CATALOGUE QUESTION DESCRIPTION:

– Radiation Control; Knowledge of the requirements for reviewing and approving release permits.

18. G2.3.6 001//T3/RAD CONTROL/C/A(2.1/3.1)/M/TP05301/S/MC

The following is a time-line of activities associated with Waste Monitoring:

- 1400, Waste Monitoring Tank (WMT) "A" is placed on recirc for sampling
- 1620, Chemistry completes sampling of WMT "A" with a result of 5.5 x 10^{-5} Ci/ml
- 1730, Chemistry submits a Radiological Liquid Waste Discharge Permit #05-XX for WMT "A"
- 1735, the Nuclear Plant Supervisor authorizes the release of Radiological Liquid Waste Permit #05-XX without the approval of the Radiochemist or Health Physics Supervisor
- 1745, Operators align WMT "A" for discharge and start the release
- 1746, R-18, Waste Disposal System Liquid Effluent Monitor, fails low. The release is terminated and WMT "A" is restored to a normal lineup
- 1930, the R-18 monitor is repaired and restored to service
- 1935, the Shift Manager re-authorizes the release of WMT "A" on the Radiological Liquid Waste Permit #05-XX
- 1940, Operators re-align WMT "A" for discharge and start the release

Which ONE of the following represents the problem associated with these actions?

A.✓ The sample taken for the Radiological Liquid Waste Permit is NOT representative of the current contents of WMT "A" now being released.

B. A Radiological Liquid Waste Permit approved for one shift may NOT be used for initiation of a release on the next shift.

C. The discharge required the approval of the Health Physics Supervisor in addition to the Nuclear Plant Supervisor.

D. The contents of WMT "A" must first be transferred to the Waste Holdup Tanks for further processing prior to release.

Feedback

REFERENCES:

1. SD 049/SYS.061A, pages 27-29, rev 02/01/02
2. OP-061.11, CONTROLLED LIQUIED RELEASE TO THE CIRCULATING WATER
3. NCAP-003, PREPARATION OF LIQUID RELEASE PERMIT

DISTRACTORS:

A Correct. IAW NCAP-003, PREPARATION OF LIQUID RELEASE PERMIT.

B Incorrect. Not required.

C Incorrect. As long as the specific activity of the tank contents is greater than or equal to 1 x 10^{-4} Ci/ml, only the NPS's approval is required.

D Incorrect. Not required.

K/A CATALOGUE QUESTION DESCRIPTION:

- Radiation Control; Knowledge of the requirements for reviewing and approving release permits.

LIQUID WASTE DISPOSAL

Waste Holdup Tank Pump-Back System

During a loss-of-coolant accident, conditions such as loss of system integrity, equipment malfunction, or valve failures can result in drainage of highly contaminated liquid into the No. 1 WHT. This situation could cause high airborne activity and high radiation levels in the surrounding auxiliary building areas. A corresponding reduction in personnel stay times would adversely effect the ability to control a given accident. The WHT pump-back system has been provided to pump the tank contents back into containment. This procedure is performed only when the unit is in the recirculation mode of safety injection following a loss-of-coolant accident. This procedure is given in detail in OP-061.12, Waste Disposal System - Waste Monitor Tanks and Demineralizer Operation.

Accidental Release of Radioactive Liquid

Accidents in the auxiliary building or rad waste facility which would result in the release of radioactive liquids are those which involve the rupture or leakage from storage tanks or piping systems. Any leakage from these components will be collected in the respective sumps and is pumped back into the liquid waste system. The gross rupture of the tanks is not considered credible but is accommodated in the plant design. All spilled liquid is contained within the tank cubicles and no uncontrolled release will occur. Flooded cubicles are drained to the sumps and then pumped to another unaffected waste handling tank. Piping external to containment and running to the auxiliary building is in concrete pipe chases to contain leakage.

Uncontrolled release to the circulating water is prevented by the waste release procedures and administrative controls. In the liquid waste release header, double valve isolation by locked closed valve 4749 and RCV-18 and continuous monitoring by RE-18 present positive controls during the release procedure to ensure accidental release do not occur. Detailed analysis of this accident is discussed in the FSAR section 14.2.2.

CONTROLLED LIQUID RELEASE TO THE CIRCULATING WATER

To perform a controlled radioactive liquid release certain prerequisites must be met. First, the monitor tank to be released must be sufficiently recirculated to obtain a representative sample and

LIQUID WASTE DISPOSAL

then a sample is drawn by the chemist for radiochemical analysis. These sample results are necessary to prepare a liquid release permit, which is required. Nuclear chemistry procedure 0-NCAP-003, Preparation of Liquid Release Permit, provides a step by step procedure for the preparation of a permit. From the waste liquid activities, the chemist will calculate the maximum release flow rate to the circulating water. This flow rate is based on the resultant activities in the circulating water remaining below 10CFR20 limits. The release permit must be approved by the Nuclear Plant Supervisor. Approvals of the radiochemist and Health Physics Supervisor are necessary only if the specific activity of the tank contents is equal to or greater than 1 x 10^{-4} Ci/ml.

The number of operating circulating water pumps required for dilution flow is determined by the chemist's/prerelease calculations. This is based on the specific activity of the tank sample and the required dilution water flow to remain below Technical Specification liquid waste release limits. Additionally, the process radiation monitor must be functional, set to alarm and trip the liquid waste discharge valve RCV-18.

The setpoints are calculated in accordance with the Technical Specifications for liquid waste release to unrestricted areas. The flow indication in the liquid waste release header shall also be operable.

After the liquid release permit is approved, the discharge lineup is performed in accordance with OP-061.11, Controlled Liquid Release To The Circulating Water. The first step is to check the operation of the liquid waste monitor alarm and control functions. The manual waste discharge valve, 4749, is verified closed to prevent inadvertent waste releases. RCV-18 is then opened from the waste/boron panel. A channel R-18 alarm is initiated from the process radiation monitoring cabinet in the control room to trip the valve closed. Once proper operation of the channel alarm and trip functions have been verified, the channel is reset. RCV-18 and manual valve 4749 remain closed. The tank to be released to the circulating water is selected and flow is established from that system to the liquid waste release header. In the case of the auxiliary building monitor tanks, flow is established to valve 1296, RCV-18 is opened, and valve 4749 is opened. Valve 1296 is then slowly opened and throttled to regulate the predetermined release flow rate as indicated on FI-1064. When discharging from the rad waste facility monitor tanks, valve 1804 is opened to the waste release header, RCV-18 is opened, and then valve 4749 is opened and throttled to obtain the proper liquid waste release flow rate as indicated on FI-1064. R-18 count rates should be continuously monitored during the release. An expected count rate for R-18 is calculated by the chemists. During the release, R-18 response is verified by comparing actual count rate to the expected count rate. Start and stop times, tank levels, and the R-18 count rate are recorded on the release permit. Should R-18 exceed its alarm setpoint during the discharge, the monitor tank pump is stopped and RCV-18 and valve 4749 are closed. The Nuclear Plant Supervisor is then notified. When the tank being released reaches its low level alarm setpoint, the transfer pump automatically stops and the lineup is then return to normal. RCV-18 is closed, valve 4749 is closed and locked, and valve 1296 or 1804 is closed.

BASIC LIQUID WASTE DISPOSAL SYSTEM FLOW PATHS

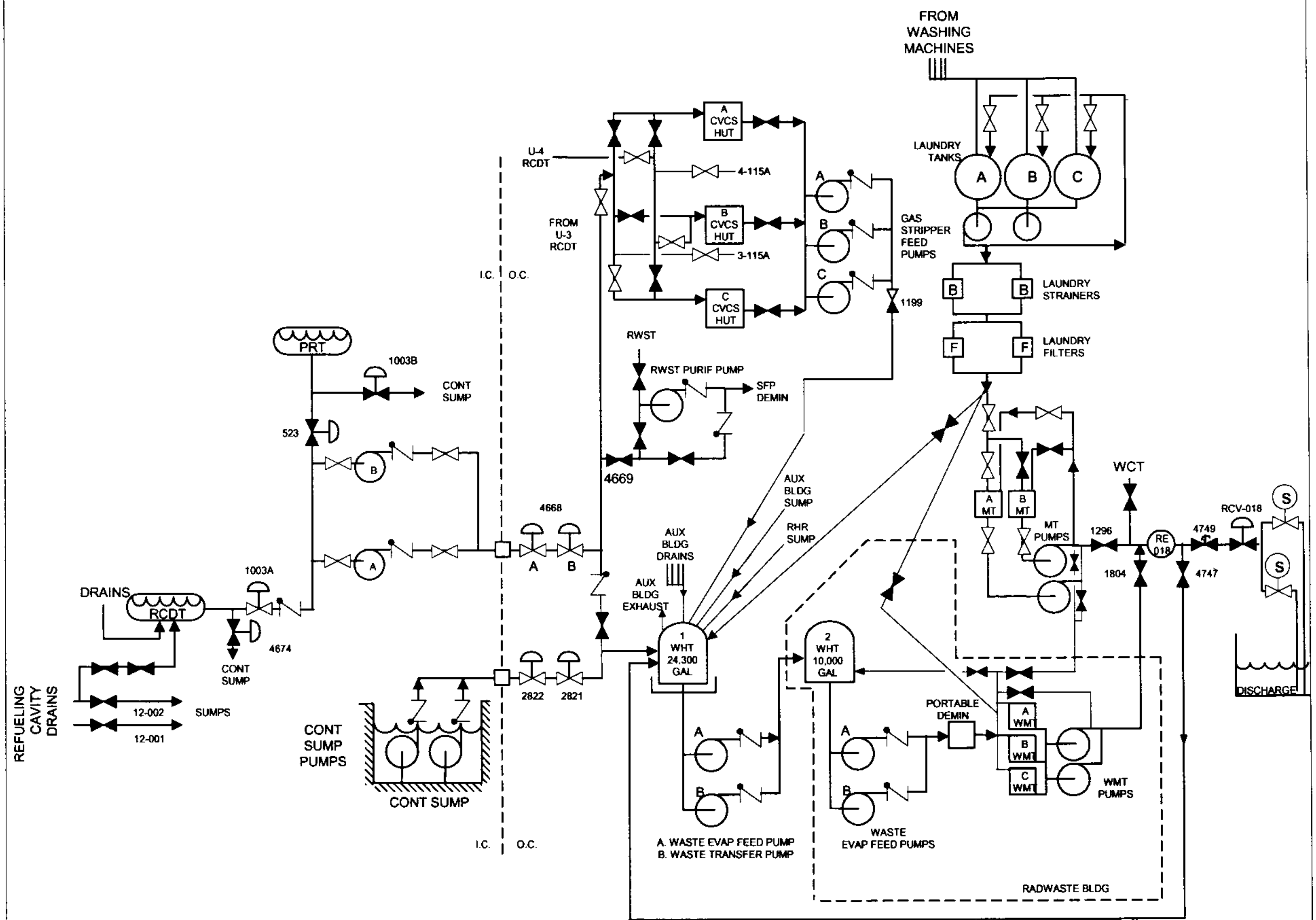


FIGURE 1
1/16/01

LIQUID WASTE IN CONTAINMENT

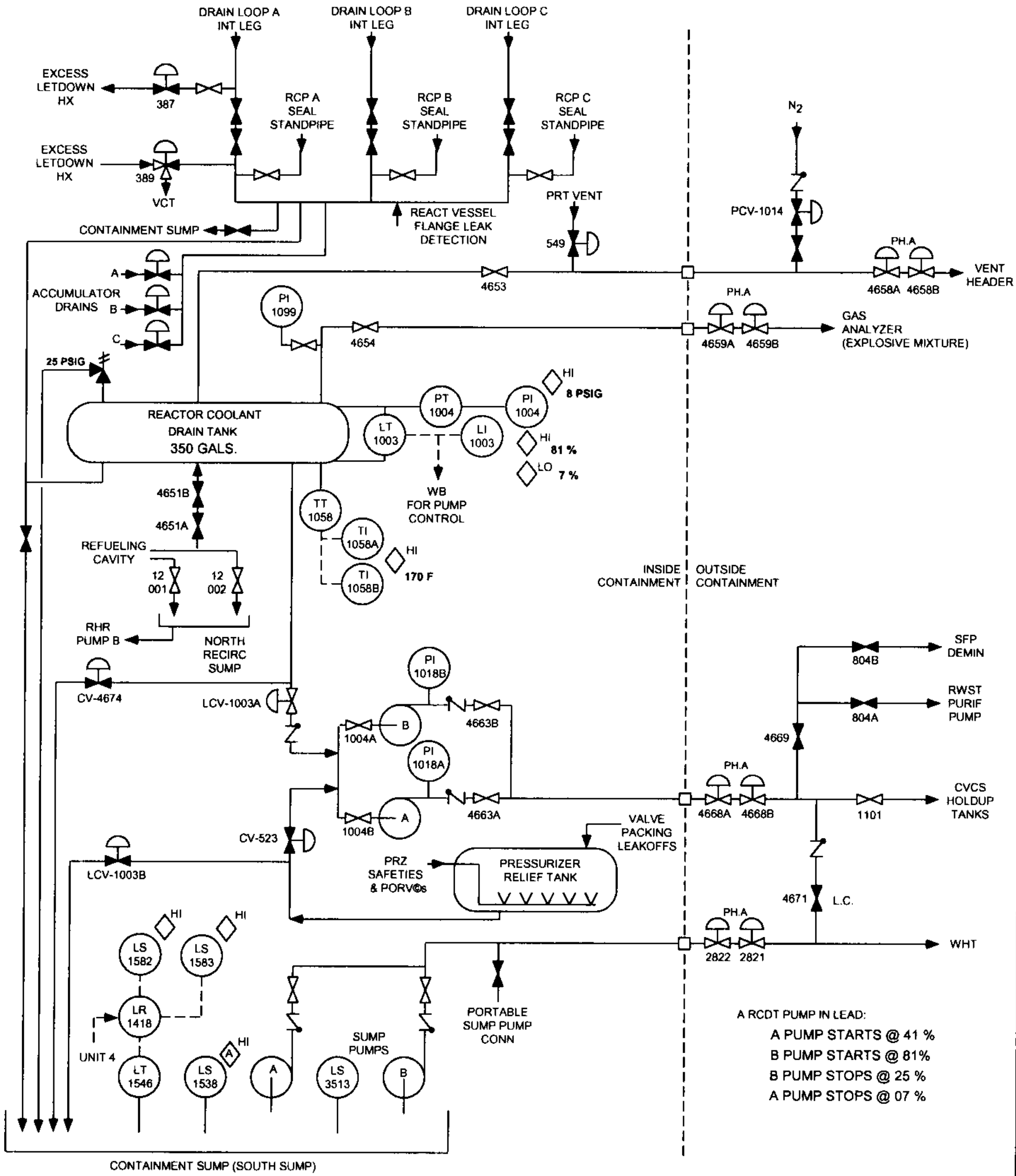


FIGURE 2
01/16/01

OUTLINE OF INSTRUCTION	INSTRUCTIONAL ACTIVITIES & NOTES
	Question #3: What information is recorded before making a liquid release?
	Answer #3: 1. Start (Stop) times 2. Tank levels 3. R-18 count rate
	Ask students if there are any questions on the material.
	Ask questions.
	Resolve questions.

OUTLINE OF INSTRUCTION	INSTRUCTIONAL ACTIVITIES & NOTES
3. WHT No. 1	
4. Demineralizer and Waste Monitor Tank	
5. WHT No. 2	
B. Infrequent Operation	
1. Draining Cold legs to RCDT	
2. Draining Refueling Cavity to RCDT	
3. RCP seal leakoff to RCDT	
C. Abnormal Operation	
1. Containment Isolation, Phase A	
2. Excessive RCS Leakage	
3. WHT pumpback to Containment	
4. Accidental Release	
D. Controlled Release	
1. Prerequisites	
2. R-18/RCV-18 Operability Test	
3. Flow established from release tank to liquid release header	

OUTLINE OF INSTRUCTION	INSTRUCTIONAL ACTIVITIES & NOTES
4. Release header flow rate established	
a. Throttling valve 1296 when releasing monitor tanks A & B	
b. Throttling valve 4749 when releasing waste monitor tanks	
5. Count rate monitored	
a. Expected count rate calculated by chemist	
b. Expected compared to actual count rate to verify R-18 response	
6. Information recorded in release	
a. Start and stop times	
b. Tank levels	
c. R-18 count rate	
7. R-18 setpoint exceeded	
a. Monitor tank pump stopped	
b. RCV-18 and 4749 closed	
c. Nuclear plant supervisor is notified	

OUTLINE OF INSTRUCTION	INSTRUCTIONAL ACTIVITIES & NOTES
8. When monitor tank (waste monitor tank) reaches low level alarm	
a. Pump stops	
b. Release lineup returned to normal	
(1) RCV-18 closed	
(2) 4749 locked closed	
(3) 1296 or 1804 closed	
VI. TECH SPECS	Briefly explain that Tech. Specs. Address these items and that a detailed review of Tech. Specs. Will be provided in a separate lesson.
A. Instantaneous Release Limits	EO 7, ETP
B. Individual Exposure Guidelines	
C. Noble Gas Concentration Limits	
VII. RECOGNIZE MAINTENANCE RULE APPLICATIONS FOR THE LIQUID WASTE DISPOSAL SYSTEM	Direct the student to Maintenance Rule Function Summary for the LWDS
	EO 8
A. Maintenance Rule	Use the Function Summary to explain which components and functions make the Rule applicable to this system.

..2.3.6 | 9/6/2002

Kewaunee 1 | **Exam Level** S

Mark Question | Print Recor | New Searc | Exit

Question

The following is a time-line of activities associated with Waste Condensate Tank 'A':

- 1400,Waste Condensate Tanks 'A' and 'B' are placed on recirculation for sampling.
- 1620,Chemistry completes sampling of Waste Condensate Storage Tanks.
- 1730, Chemistry submits a Radiological Liquid Waste Discharge Permit # 02-XX for the Waste Condensate Storage Tanks.
- 1735, Shift Manager authorizes the release on Radiological Liquid Waste Permit # 02-XX
- 1745, NAO aligns the Waste Condensate Storage Tanks for discharge to the Aux. Bldg Standpipe and starts the release.
- 1746, R-18, Waste Disposal Liquid Monitor, fails low. The release is terminated and the Waste Condensate Storage Tanks are restored to a normal lineup.
- 1930, R-18 monitor is repaired and restored to service.
- 1935, Shift Manager reauthorizes the release of the Waste Condensate Tanks on the Radiological Liquid Waste Permit # 02-XX.
- 1940, NAO realigns the Waste Condensate Storage Tanks for discharge to the Aux. Bldg Standpipe.

What is the problem associated with these actions?

Answer: The sample taken for the Radiological Liquid Waste Permit is NOT representative of the current contents of the Waste Condensate Tanks now being released.

Distracter 1 A Radiological Liquid Waste Permit approved for one shift may NOT be used for initiation of a release on the next shift.

Distracter 2 The release rate calculations and the discharge line valving require independent verification prior to initiating the second release.

Distracter 3 The Waste Condensate Tanks must first be transferred to the Waste Holdup Tank for further processing prior to release.

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

Question #: 1.1.24.49.3.4,M

69021490304; Which ONE of the following describes the actions that must be taken to release a Waste Monitor Tank with PRMS R-18 out of service?

Distractor:
Two separate tank samples must be analyzed and the Health Physics Supervisor must approve the release.

Distractor:
Tank specific activity must be less than 1E-4 (Ci/ml and the Radiochemistry Supervisor must approve the release.

Answer:
Two separate tank samples must be analyzed and two qualified members of plant staff must independently verify the release rate calculations and discharge valving.

Distractor:
Tank specific activity must be less than 1E-4 (Ci/ml and two qualified members of plant staff must independently verify the discharge valving.

REFERENCES:

KEYWORDS/KSAs:
; 2.3.6

..2.3.6 10/29/2001

Braidwood 1 **Exam Level** S

Mark Question | Print Recor | New Searc | Exit

Question

Which ONE of the following can provide final authorization for a Liquid Rad Waste release?

Answer: Shift Manager.

Distracter 1 Plant Manager.

Distracter 2 Rad Protection Supervisor.

Distracter 3 Chemistry Supervisor.

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

TP05301

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

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

51. G2.3.06 001/3///MEM 2.1/3.1/B/FA03301/S/SDR

Unit 1 is at 100% steady-state reactor power with the following plant conditions:

- 1A Steam Generator has a confirmed tube leak of 20 gpd.
- 1B Steam Generator has a confirmed tube leak of 5 gpd .
- The Turbine Building water sump is full and needs to be discharged.

Which ONE of the following, if any, describes the release permit(s) you would expect to review (be in affect) to authorize the release?

A.✓ A batch release permit.

B. A continuous release permit.

C. Both a batch and continuous release permit.

D. No permit is required.

Source: Farley 2001 NRC Exam

A - Correct, Batch release permit is required if there is evidence of a SGTL creating the possibility that the sump contents may be contaminated.
B - Incorrect, Continuous release permit is required if there is no evidence of a SGTL.
C - Incorrect, Both types of release permits would not be in effect with the evidence of a SGTL it is inappropriate to have a continuous release permit.
D - Incorrect, A release permit is required.

19. G2.4.34 001//T3/E PRO/PLAN/M(3.8/3.6)/N/TP05301/S/MC

Turkey Point has experienced a fire in the North/South Breezeway.

- The Fire Suppression System in the N-S Breezeway has activated
- Both crews are in the process of carrying out the actions of ONOP-105, CONTROL ROOM EVACUATION

IAW ONOP-105, which ONE of the following correctly describes an action a particular operator is required to take given existing plant conditions?

A. From the Unit 3 480 Volt Load Center Room the Unit 3 RO will verify LC 3D Supply to LC 3H Breaker, 30402 – CLOSED; and verify 3B Load Center Supply Breaker, 30210 - CLOSED.

B. From the Unit 4 480 Volt Load Center Room the Unit 4 RO will trip LC 4D Supply to LC 4H Breaker, 40402; and verify 4B Load Center Supply Breaker, 40210 - CLOSED.

C. From the Unit 3 480 Volt Load Center Room the Third RO will verify LC 3D Supply to LC 3H Breaker, 30402 – CLOSED; and verify 3B Load Center Supply Breaker, 30210 - CLOSED.

D✓ From the Unit 4 480 Volt Load Center Room the Third RO will trip LC 4D Supply to LC 4H Breaker, 40402; and verify 4B Load Center Supply Breaker, 40210 - CLOSED.

Feedback

REFERENCES:
1. ONOP-105, CONTROL ROOM EVACUATION, pages 92,93,94, rev 05/04/04

DISTRACTORS:

A Incorrect. The Third RO would take this action, NOT the Unit RO and it would be taken only if the fire was NOT in the N-S Breezeway.

B Incorrect. This is the action taken by the Third RO, NOT the Unit RO.

C Incorrect. This is the action the Third RO would take if there was NO fire in the N-S Breezeway.

D Correct. IAW ref 1, this action is to be taken by the Third RO in the event of a fire in the N-S Breezeway.

K/A CATALOGUE QUESTION DESCRIPTION:
- Emergency Procedures/Plan; Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 5
(Page 7 of 14)

THIRD LICENSED REACTOR OPERATOR

NOTE

The breakers listed in Step 18 are physically locked in the OFF position to prevent spurious valve operation. The breakers are unlocked and energized (turned to ON) only when the Unit(s) are in Mode 4.

18 **Place the following breakers in the locked OFF position OR verify the following breakers are in the locked OFF position:**

- 40615 for MOV-4-750
- 40616 for MOV-4-862B

19 **Notify the TSC That Steps 1 Through 18 Have Been Completed**

20 **Proceed To Unit 4 480 Volt Load Center Room**

NOTE

Breakers listed in Steps 21 and 23 are locked open by holding in trip pushbutton and pulling out tab under trip pushbutton.

21 **Perform The Following On Appropriate Load Center:**

a. Position Pressurizer Backup Heater 4B Key Switch on back of 4D load center to EMERGENCY position

b. Verify 4D Load Center Supply Breaker, 40410 – CLOSED

RESPONSE NOT OBTAINED: b. Close breaker.

c. Lock open Containment Spray Pump B Breaker, 40403

d. Place Normal/Isolate switch for LC 4D Supply to LC 4H, XS-40402, to ISOLATE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 5
(Page 8 of 14)

THIRD LICENSED REACTOR OPERATOR

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	**Perform The Following On Appropriate Load Center (Cont'd):**	
	e. Check fire is in North-South Breezeway	e. Perform the following: (1) Verify LC 4D Supply to LC 4H Breaker 40402 – CLOSED. (2) Go to Step 21g.
	f. Trip LC 4D Supply to LC 4H Breaker, 40402	
	g. Verify 4B Load Center Supply Breaker, 40210 - CLOSED	g. Close breaker.
22	**Proceed To Unit 3 ~~480 Volt Load Center~~ Room**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 5
(Page 9 of 14)

THIRD LICENSED REACTOR OPERATOR

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23	**Perform The Following On Appropriate Load Center:**	
	a. Position Pressurizer Backup Heater 3B Key Switch on back of 3D load center to EMERGENCY position	
	b. Verify 3D Load Center Supply Breaker, 30410 – CLOSED	b. Close breaker.
	c. Lock open Containment Spray Pump B Breaker, 30403	
	d. Place Normal/Isolate switch for LC 3D to LC 3H, XS-30402, to ISOLATE	
	e. Check fire is in North-South Breezeway	e. Perform the following: (1) Verify LC 3D Supply to LC 3H Breaker 30402 – CLOSED. (2) Go to Step 23g.
	f. Trip LC 3D Supply to LC 3H Breaker, 30402	
	g. Verify 3B Load Center Supply Breaker, 30210 - CLOSED	g. Close breaker.
24	**Notify The TSC That Steps 1 Though 23 Are Complete**	
25	**Check If Unit 3 MSIV Bypass Valves Energized Prior To Evacuation**	Go to Step 28.
26	**Proceed To Unit 3 Main Steam Platform**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 5
(Page 1 of 14)

THIRD LICENSED REACTOR OPERATOR

NOTES

- *Steps 1 and 2 are IMMEDIATE ACTION steps.*
- *Steps 1 through 23 should be completed within 15 minutes.*

1 Announce Control Room Evacuation:

a. Depress and hold the PAGE BOOST pushbutton

b. Announce over the plant PA system:

INITIATING CONTROL ROOM EVACUATION DUE TO (Specify Cause). ALL OPERATIONS PERSONNEL PERFORM THEIR CONTROL ROOM EVACUATION DUTIES

c. Repeat announcement

d. Release PAGE BOOST pushbutton

e. Actuate fire alarm

f. Obtain Radio

Make announcement immediately after evacuation of the Control Room using one of the following:

* Radios
* Telephones
* Alternate Shutdown Communication System

2 Evacuate Control Room As Follows:

a. Proceed to Turbine Deck Work Station

b. Obtain the following:

- One copy of this procedure
- Third RO key ring

CAUTION

At least one AFW pump should be in operation within 20 minutes following a unit trip.

3 Proceed To AFW Pump Cage

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 5
(Page 2 of 14)

THIRD LICENSED REACTOR OPERATOR

CAUTION

AFW Pump A and Train 1 are NOT Alternate Shutdown protected, therefore should only be operated under close supervision.

NOTES

- *AFW pump B T&T valve controlled from Unit 4 ASP, pump C T&T valve from Unit 3 ASP.*
- *AFW pumps B and C are Alternate Shutdown protected and normally aligned to Train 2. AFW pump A is normally aligned to Train 1.*
- *Only Train 2 auxiliary feedwater flow is controllable from the alternate shutdown panels, therefore A AFW pumps is tripped to ensure zero Train 1 auxiliary feedwater flow.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Mechanically Trip A AFW Pump**	
5	**Check If AFW Required:** * Unit 3 RHR System - NOT IN SERVICE PRIOR TO CONTROL ROOM EVACUATION **OR** * Unit 4 RHR System - NOT IN SERVICE PRIOR TO CONTROL ROOM EVACUATION	**Perform the following:** a. Verify B and C AFW pumps - STOPPED. b. Go to Step 8.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 5
(Page 3 of 14)

THIRD LICENSED REACTOR OPERATOR

NOTE

When the unit ROs place their respective T&T valve transfer switches to local, a trip of the related AFW pump may occur and require a restart of the tripped pump(s).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	**Verify B And C AFW Pumps - NOT TRIPPED**	Locally reset pump(s) as follows: a. Place Keylock Switch on applicable AFW pump local control panel to Local. b. Mechanically trip the applicable AFW pump. c. **WHEN** the pump stops rotating, **THEN** turn manual speed fully counterclockwise. d. Restore manual speed controller to fully clockwise position. e. Reset mechanical trip by moving trip linkage towards MOV.
7	**Verify B And C AFW Pumps - BOTH RUNNING**	Start AFW pump(s) as follows: a. Place Keylock Switch on applicable AFW pump local control panel to LOCAL. b. Depress the open T&T pushbutton for applicable AFW pump. c. **IF** applicable AFW pump does **NOT** start, **THEN** manually open Stm Gen 3A (4A) Supply to AFW pumps, MOV-3 (4)-1403.

1. GEN2.4.34 001//T3/3.8/3.6/MEMORY/NEW/VG01301/SRO/LRM14

The operations personnel are being evacuated because of fire in the control room and dispatched to their assigned locations per procedure 18038-1,"Operation from the Remote Shutdown Panels".

How do the Reactor Operator and BOP Operator know which instruments are fire event qualified instrumentation and where are fire event qualified instrumentation located?

A. Fire event qualified instrumentation is located on shutdown panel A and has a red bar at the base of the instrument.

B.✓ Fire event qualified instrumentation is located on shutdown panel B and has a red bar at the base of the instrument.

C. Fire event qualified instrumentation is located on shutdown panel A and has an orange dot at the top of the instrument.

D. Fire event qualified instrumentation is located on shutdown panel B and has an orange dot at the top of the instrument.

Feedback

B

Notes

Ref: 18038-1 "Operation from the Remote Shutdown Panels", LO-LP-60327-06-C, LO-LP-60328-09-C
CFR 43.5

20. G2.4.6 001//T3/E PRO/PLAN/M(3.1/4.0)/B/TP05301/S/MC

Given the following information:

- An event has occurred in the plant that has resulted in a radioactive release in the containment
- The Safety Parameter Display System (SPDS) indicates Critical Safety Function Status Tree display of YELLOW priority for Containment
- FR-Z.3, RESPONSE TO VOIDS IN REACTOR VESSEL, has been entered

Which ONE of the following indicates the mitigation strategy for operator actions directed by FR-Z.3?

A. Allow a controlled release through the containment filtration system prior to exceeding design pressure limits on the containment.

B.✓ Verify containment ventilation isolation and attempt to reduce activity by containment filtration.

C. Reduce containment activity levels with dilution flow using the Main (Preaccess) Purge System.

D. Verify containment isolation Phase "A" and place all containment coolers in slow speed.

Feedback

REFERENCES:

1. WOG, FR-Z.3, page 2, rev 1C
2. EOP-F-O, CRITICAL SAFETY FUNCTION STATUS TREES, page 11, rev 04/15/99C

DISTRACTORS:

A	Incorrect.	See B.
B	Correct.	IAW WOG.
C	Incorrect.	See B.
D	Incorrect.	See B.

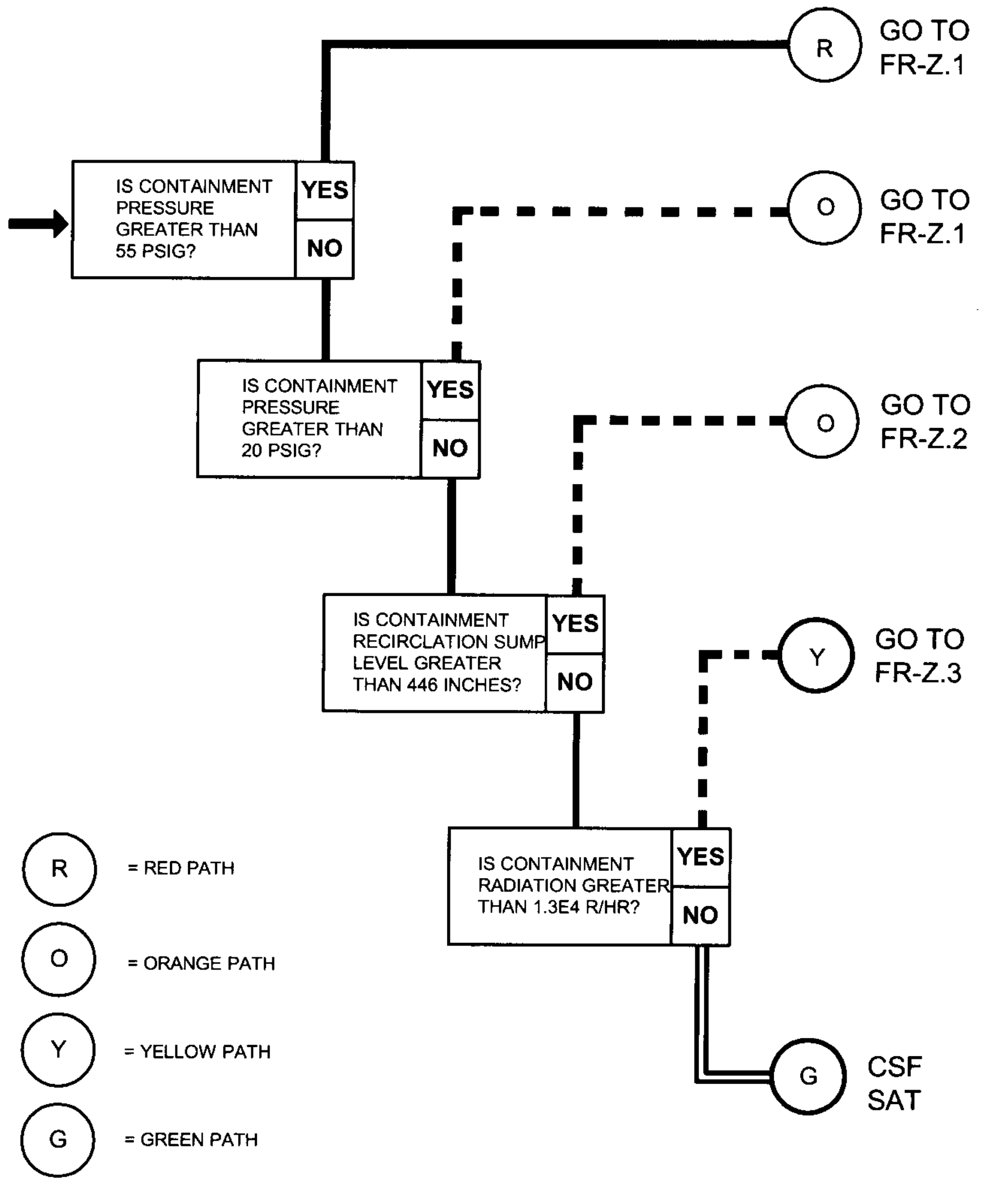
K/A CATALOGUE QUESTION DESCRIPTION:

- Emergency Procedures/Plan; Knowledge of symptom based EOP mitigation strategies.

ENCLOSURE 5

(Page 1 of 1)

CSF F-0.5 CONTAINMENT



2. DESCRIPTION

Guideline FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL, provides actions to respond if high radiation is present inside containment. This guideline is entered on a YELLOW priority on the Containment Status Tree based on operator judgement when the containment radiation level is above a value corresponding to the radiation alarm setpoint for the post accident containment radiation monitor. The main purpose of the post accident containment radiation monitor is to follow the radiation level in the containment during and after an accident that releases a significant quantity of radioactivity into the containment. The radiation alarm setpoint for the post accident radiation monitor is plant specific, however, a typical setpoint value would be 2 R/hr to 3 R/hr. The radiation alarm setpoint would be reached due to any significant RCS leakage into containment or after a steamline break inside containment assuming technical specification leakage from the steam generators.

Additionally, it should be pointed out that the radiation level for the post accident containment radiation monitor, which is used in FR-Z.3, is higher than the setpoint that actuates containment ventilation isolation. The setpoint for containment ventilation isolation corresponds to a radiation level which is slightly above background radiation during normal plant operations.

The actions of this guideline verify containment ventilation isolation and attempt to reduce activity by containment filtration. The plant engineering staff is then notified of the existence of the excessively high radiation level.

W26

1. W/E16GEN2.4.6 001//T1G2/3.1/4.0/MEMORY/NEW/VG01301/SRO/LRM20

Given the following information:

- An event has occurred in the plant that has resulted in a radioactive release in the containment.
- The Safety Parameter Display System (SPDS) indicates Critical Safety Function Status Tree display of YELLOW priority for Containment.
- Procedure 19253-C "FR-2.3 Response to High Containment Radiation Level" has been entered.

What is the mitigation strategy of the operator actions directed by Procedure 19253-C "FR-2.3 Response to High Containment Radiation Level?"

A. Allow a controlled release through containment filtration system prior to exceeding design pressure limits on the containment.

B.✔ Verify containment ventilation isolation and attempt to reduce activity by containment filtration.

C. Reduce containment activity levels with dilution flow using the Main (Preaccess) Purge System.

D. Verify containment isolation Phase "A" and place all containment coolers in slow speed.

Feedback

B

Notes

Ref: Procedure 19253-C "FR-2.3 Response to High Containment Radiation Level", Background information for WOG ERG FR-Z.3 "Response to High Containment Radiation"
CFR 43.5

..g2.4.6 6/29/2000

Byron 1 **Exam Level** R

Mark Question | Print Recor | New Searc | Exit

Question The primary basis for depressurizing all intact steam generators to atmospheric pressure in FR-C.1, "RESPONSE TO INADEQUATE CORE COOLING," is to:

Answer: reduce RCS pressure for establishing low-head safety injection.

Distracter 1 insure core exit thermocouple temperatures are reduced to less than 700 F.

Distracter 2 reduce S/G pressure to increase feedwater flow.

Distracter 3 enhance natural circulation cooling of the reactor core.

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

58. G2.4.6 001/T3/T3/QUENCH SPRAY/C/A 3.1/4.0/BANK/NA02301/C/RA/LSM

Given a spurious CDA, the operating crew uses E-1, Attachment #3, "Termination of Quench Spray," as a guide. Which one of the following represents the correct sequence for securing quench spray?

A.✓ Reset CDA, stop the QS pumps, close the QS pump discharge valves, then close the CAT discharge valves.

B. Reset CDA, stop the QS pumps, close the CAT discharge valves, then close the QS pump discharge valves.

C. Stop the QS pumps, close the CAT discharge valves, then close the QS pump discharge valves.

D. Stop the QS pumps, close the QS pump discharge valves, then reset CDA.

Source - Bank ID: 4074

Associated objective(s):

13593

Explain the guidelines for using the following types of procedures (OPAP-0002; SER-1999-2).

- Operating procedures
- Emergency operating procedures
- Abnormal procedures
- Annunciator response procedures

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

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

59. G2.4.6 002/T3//3.1/4.0/MEMORY/NEW/TP00301/RO/SDR/LSM/TP

Select the EOP(s) that can be entered directly:

A. E-0 only

B. E-0 and FR-S.1

C.✓ E-0 and ECA-0.0

D. E-0, ECA-0.0 and FR-S.1

C

Entry conditions

Test Name	Test Date	rpb	p(Diff)	Time	Equ	User Values
<Cumulative>		0.000	0.000	0	N	1: 0.000000 2: 0.000000 3: 0.000000 4: 0.000000

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

60. 059AA1.03 001/T1G2/T1G1/3.0/2.9/MEMORY/NEW/VG01301/BOTH/GTH09

Unit 2 is performing a liquid effluent release of waste monitor tank 9. Which of the following would require immediate termination of the release in progress ?

A. High dilution flow rate indication.

B.✓ Blowdown sump dilution flow of 12,000 gpm.

C. High radiation indication on 2-RE-0016.

D. 2-RE-0018 reads less than expected.

B

Ref: procedure 13216-2 (taken from bank NRC-14 #65)

21. WE01EA2.2 001//T1G2/REDIAGNOSIS/M(3.3/3.9)/N/TP05301/S/MC

Following a complicated reactor trip, the control room transitioned to, and completed, the final step in ES-0.0, REDIAGNOSIS.

Which ONE of the following describes the plant conditions that resulted in reaching this point in the plant procedures?

A.✓ A ruptured S/G had to have been identified and SI is required.

B. A ruptured S/G had to have been identified and SI is not required.

C. A faulted S/G had to have been identified and SI is required.

D. A faulted S/G had to have been identified and SI is not required.

Feedback

REFERENCES:

1. EOP-ES-0.0, REDIAGNOSIS, pages 3,5, rev 12/14/02
2. BD-EOP-ES-0.0, REDIAGNOSIS BASIS, pages 8,12, rev 12/14/02

DISTRACTORS:

A Correct. ES-0.0, REDIAGNOSIS, should only be used if SI is in service or in required. To reach step 4, the final step, in ES-0.0, a ruptured S/G had to have been identified in the previous step.

B Incorrect. SI should either be in service or is required.

C Incorrect. A ruptured S/G had to have been identified.

D Incorrect. A ruptured S/G had to have been identified and SI should either be in service or is required.

K/A CATALOGUE QUESTION DESCRIPTION:

- Rediagnosis; Ability to determine and interpret the following as they apply to the (Reactor Trip or Safety Injection Rediagnosis): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

1.0 PURPOSE

1.1 This procedure provides a mechanism to allow the operator to determine or confirm the most appropriate post accident recovery procedure.

1.2 This procedure is applicable when directed by EOP entry conditions.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 This procedure may be entered based on operator judgment when all of the following criteria have been satisfied:

2.1.1 **Safety Injection is in service or is required.**

2.1.2 E-0, REACTOR TRIP OR SAFETY INJECTION, has been executed and a transition has been made to another Optimal Recovery Procedure.

2.1.3 A Functional Restoration Procedure is not being performed.

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 Technical Specifications for Turkey Point Unit 3 and Unit 4

3.1.2 Turkey Point Unit 3 and Unit 4 Final Safety Analysis Report

3.1.3 As-built plant drawings

3.1.4 Procedures

1. 3-OP-018.1, CONDENSATE STORAGE TANK
2. 3-OP-023, EMERGENCY DIESEL GENERATOR
3. 4-OP-023, EMERGENCY DIESEL GENERATOR

3.1.5 Plant Change/Modifications

1. None

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTES • *Foldout page is required to be monitored throughout this procedure.* • *This procedure should only be used if SI is in service or is required.*	
1	**Check If Any S/G Is <u>NOT</u> Faulted** • Check pressures in all S/Gs - ANY STABLE **<u>OR</u>** INCREASING	**<u>IF</u>** a controlled cooldown is in progress, **<u>THEN</u>** go to Step 2. **<u>IF</u>** controlled cooldown **<u>NOT</u>** in progress, **<u>THEN</u>** the following applies: * **<u>IF</u>** main steamlines **<u>NOT</u>** isolated, **<u>THEN</u>** you should be in 3-EOP-E-2, FAULTED STEAM GENERATOR ISOLATION. **<u>OR</u>** * **<u>IF</u>** main steamlines isolated, **<u>THEN</u>** you should be in 3-EOP-ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.
2	**Check If All S/Gs Are <u>NOT</u> Faulted** a. Check pressures in all S/Gs - • NO S/G PRESSURE DECREASING IN AN UNCONTROLLED MANNER • NO S/G COMPLETELY DEPRESSURIZED	**<u>IF</u>** any faulted S/G **<u>NOT</u>** previously isolated **<u>AND</u>** that faulted S/G is **<u>NOT</u>** needed for RCS cooldown, **<u>THEN</u>** you should be in 3-EOP-E-2, FAULTED STEAM GENERATOR ISOLATION.
3	**Check If S/G Tubes Are Ruptured** * ANY S/G LEVEL INCREASING IN AN UNCONTROLLED MANNER **<u>OR</u>** * ANY S/G WITH HIGH RADIATION	You should be in an E-1 or ECA-1 series procedure.
4	**You Should Be In An E-3 Or ECA-3 Series Procedure**	

END OF TEXT

BASIS DOCUMENT

WOG Procedure Step 1 – NOTE 2 **PTN Procedure Step** 1 – NOTE 2

This procedure should only be used if SI is in service or is required.

BASIS:

The particular sequence of steps in this procedure was based on the assumption that SI is in service or should be in service. Therefore, this procedure should be used only if SI is in service or is required and E-0, REACTOR TRIP OR SAFETY INJECTION, has been completed.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

N/A

PLANT SPECIFIC SETPOINTS:

N/A

BASIS DOCUMENT

WOG Procedure Step 4 **PTN Procedure Step** 4

You Should Be In An E-3 Or ECA-3 Series Procedure

BASIS:

In order to reach this step, a ruptured S/G had to have been identified in the previous step. Thus, if any S/G is ruptured, the appropriate procedure is an E-3 or ECA-3 series procedure.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

N/A

PLANT SPECIFIC SETPOINTS:

N/A

3. WE01G2.4.5 001/T1G2/T1G1//M 2.9/3.6/B/SR02301/R/GWL
Which one of the following correctly describes the conditions that allow implementation of ES-0.0, "Re-Diagnosis?"

A. Entry is based solely on operator judgement and ES-0.0 may be entered at any time.

B. ES-0.0, may be entered anytime that an SI has been actuated, and an ERG is in effect.

C.✓ Entry is based on an SI being in service and E-0, "Reactor Trip/Safety Injection" has been completed.

D. ES-0.0 may be entered anytime that the EOP's have been entered.

Surry Exam Bank Question # 903.
Surry Lesson Plan ND-95.3-LP-33 objectives A and B.

A. Incorrect, ES-0.0 entry requires an SI to be in service and E-0 to be completed.

B. Incorrect, E-0 must be completed.

C. Correct, an SI must be in service, and E-0 has been completed.

D. Incorrect, an SI must be present, and E-0 completed.

2. WE01EK3.2 001//SROT1G1/3.0/3.9/MEMORY/NEW/SM00301/SRO/MM24

Following a complicated reactor trip, the control room is considering transitioning to EOP-1.5, "Rediagnosis." Which one of the following represents valid entry conditions for EOP-1.5?

A.✓ A safety injection actuation is required and EOP-1.0, "Reactor Trip/Safety Injection Actuation" is complete.

B. A safety injection actuation has occurred and EOP-1.0, "Reactor Trip/Safety Injection Actuation" is being completed.

C. Reactor coolant system pressure is trending toward the safety injection actuation setpoint and EOP-1, "Reactor Trip/Safety Injection Actuation" is complete.

D. Reactor coolant system pressure is trending toward the safety injection actuation setpoint and EOP-1, "Reactor Trip/Safety Injection Actuation" is being completed.

REF: EOP-1.5, rev 2

SOURCE: NEW (MSM)

QUESTIONS REPORT

for Westinghouse 3 Loop Questions

1. WE02EA2.2 001//T1G1/TS/C/A 3.5/4.0/NEW/NA02301/S/RA/LSM

Unit one is in post-LOCA cooldown. The current temperature is 345°F. A mechanical failure of one train of SI has just occurred. The other train is available.

Which one of the following describes the ITS LCO that applies to this condition?

A. LCO 3.5.2, ECCS - Operating.

B. LCO 3.4.7, RCS Loops - Mode 5, Loops Filled.

C. LCO 3.9.5, Residual Heat Removal and Coolant Circulation—High Water Level.

D.✓ LCO 3.5.3, ECCS - Shutdown.

One train is required in mode 4 by LCO 3.5.3

North Anna Units 1 and 2 B 3.5.3-2 Rev 0 (Draft 1), 06/05/00
ECCS—Shutdown
B 3.5.3
BASES
LCO

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the three cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot or cold legs.

APPLICABILITY In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350....F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

2 WE03EA2.2 001//T1G2/LOCA C/D/C/A(3.5/4.1)/N/TP05301/S/MC

Following a LOCA with a concurrent loss of offsite power, Unit 3 entered E-1, LOSS OF REACTOR OR SCEONDARY COOLANT. Currently, the following plant conditions exist:

- T_{avg} is 345^{o}F
- RCS pressure is 350 psig
- RWST level less than 155,000 gallons
- A mechanical failure of one train of SI has just occurred

Which ONE of the following describes the required operator actions in accordance with E-1 and the LCO that applies to this condition?

A. Transition to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, stop the 3A RHR pump and place in standby. LCO 3.5.4, REFUELING WATER STORAGE TANK, maintain the plant in MODE 4.

B.✓ Transition to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, stop the 3A RHR pump and place in standby. LCO 3.5.3, ECCS SUBSYSTEMS, maintain the plant in MODE 4.

C. Transition to ES-1.3, TRANSITION TO COLD LEG RECIRCULATION, align RHR suction to containment recirc sump. LCO 3.5.4, REFUELING WATER STORAGE TANK, plant in MODE 5 in 30 hours.

D. Transition to ES-1.3, TRANSITION TO COLD LEG RECIRCULATION, align RHR suction to containment recirc sump. LCO 3.5.3, ECCS SUBSYSTEMS, plant in MODE 5 in 30 hours.

Feedback

REFERENCES:

1. E-1, LOSS OF REACTOR OR SECONDARY COOLANT, pages 18,21, rev 04/03/02
2. BD-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, page 35, rev 04/03/02
3. TECHNICAL SPECIFICATION, 3.5.3, page 3/4 5-9, Amendment Nos. 138 & 133
4. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, page 72, rev 05/01/03
5. ES-1.2, POST LOCA COOLDOWN & DEPRESSURIZATION, pgs 3,6,12, rev 04/03/02
6. BD ES-1.2, POST LOCA COOLDOWN AND DEPRESS, page 8, rev 04/03/02
7. TECHNICAL SPECIFICATION, 3.5.4, page 3/4 5-10, Amendment Nos. 138 & 133

DISTRACTORS:

A Incorrect. The transition is correct. The LCO is also correct, however, the plant must be in MODE 5 in 30 hours

B Correct. IAW E-1, Step 19, if RCS pressure is > 250 psig, go to ES-1.2, step 1 which directs stopping RHR pumps. LCO 3.5.3. allows one train of RHR when < 350^{o}F.

C Incorrect. While this is a valid transition from E-1, it occurs is step 23 (after step 19) so this transition would be incorrect.

D Incorrect. While this is a valid transition from E-1, it occurs is step 23 (after step 19) so this transition would be incorrect. The LCO is correct, however, the plant can be maintained in MODE 4.

K/A CATALOGUE QUESTION DESCRIPTION:

- LOCA Cooldown and Depressurization; Ability to determine and interpret the following as they apply to the (LOCA Cooldown and Depressurization): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

22. WE03EA2.2 001//T1G2/LOCA C/D/C/A(3.5/4.1)/N/TP05301/S/MC

As the result of a LOCA on Unit 3 concurrent with a loss of offsite power, the crew entered E-1, LOSS OF REACTOR OR SCEONDARY COOLANT. Current plant conditions are as follows:

- T_{avg} is 345°F
- RCS pressure is 350 psig
- RWST level less than 155,000 gallons
- A mechanical failure of one train of SI has just occurred

Which ONE of the following describes the required operator actions in accordance with E-1 and the LCO that applies to this condition?

A. Transition to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, stop the 3A RHR pump and place in standby. LCO 3.5.4, REFUELING WATER STORAGE TANK, maintain the plant in MODE 4.

B.✓ Transition to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, stop the 3A RHR pump and place in standby. LCO 3.5.3, ECCS SUBSYSTEMS, maintain the plant in MODE 4.

C. Transition to ES-1.3, TRANSITION TO COLD LEG RECIRCULATION, align RHR suction to containment recirc sump. LCO 3.5.4, REFUELING WATER STORAGE TANK, plant in MODE 5 in 30 hours.

D. Transition to ES-1.3, TRANSITION TO COLD LEG RECIRCULATION, align RHR suction to containment recirc sump. LCO 3.5.3, ECCS SUBSYSTEMS, plant in MODE 5 in 30 hours.

Feedback

REFERENCES:

1. E-1, LOSS OF REACTOR OR SECONDARY COOLANT, pages 18,21, rev 04/03/02
2. BD-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, page 35, rev 04/03/02
3. TECHNICAL SPECIFICATION, 3.5.3, page 3/4 5-9, Amendment Nos. 138 & 133
4. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, page 72, rev 05/01/03
5. ES-1.2, POST LOCA COOLDOWN & DEPRESSURIZATION, pgs 3,6,12, rev 04/03/02
6. BD ES-1.2, POST LOCA COOLDOWN AND DEPRESS, page 8, rev 04/03/02
7. TECHNICAL SPECIFICATION, 3.5.4, page 3/4 5-10, Amendment Nos. 138 & 133

DISTRACTORS:

A Incorrect. The transition is correct. The LCO is also correct, however, the plant must be in MODE 5 in 30 hours

B Correct. IAW E-1, Step 19, if RCS pressure is > 250 psig, go to ES-1.2, step 1 which directs stopping RHR pumps. LCO 3.5.3. allows one train of RHR when < 350^{o}F.

C Incorrect. While this is a valid transition from E-1, it occurs is step 23 (after step 19) so this transition would be incorrect.

D Incorrect. While this is a valid transition from E-1, it occurs is step 23 (after step 19) so this transition would be incorrect. The LCO is correct, however, the plant can be maintained in MODE 4.

K/A CATALOGUE QUESTION DESCRIPTION:

- LOCA Cooldown and Depressurization; Ability to determine and interpret the following as they apply to the (LOCA Cooldown and Depressurization): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19	**Check If RCS Cooldown And Depressurization Is Required**	
	a. RCS pressure - GREATER THAN 250 PSIG[650 PSIG]	a. Perform the following: 1) **IF** RHR pump flow greater than 1500 gpm, **THEN** go to Step 20. 2) **IF** RHR pump flow less than or equal to 1500 gpm, **THEN** go to 3-EOP-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1.
	b. Go to 3-EOP-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22	**Check If Auxiliary Feedwater Pumps Should Be Stopped**	
	a. Verify narrow range level in all S/Gs - GREATER THAN 15%	a. Go to Step 23.
	b. Reset AMSAC	
	c. Verify AFW actuation signal reset – BOTH AFW AUTO START WHITE LIGHTS OUT (3QR50 AND 3QR51)	c. Perform the following: 1) Verify SI – RESET 2) Verify main feedwater pump switch flags and indicating lights – MATCHED 3) <u>**IF**</u> AFW Auto Start Lights (3QR50 AND 3QR51) are OUT, <u>**THEN**</u> go to Step 22d. 4) <u>**IF**</u> either AFW Auto Start Light (3QR50 AND 3QR51) is LIT, <u>**THEN**</u> go to Step 23.
	d. Establish feedwater flow from one of the following * One main feedwater pump using 3-OP-074, STEAM GENERATOR FEEDWATER PUMP <u>**OR**</u> * Standby feedwater using 0-OP-074.1, STANDBY S/G FEEDWATER SYSTEM	d. Go to Step 23.
	e. Stop AFW pumps using 3-OP-075, AUXILIARY FEEDWATER SYSTEM	
23	**Check If Transfer To Cold Leg Recirculation Is Required**	
	a. RWST level - LESS THAN 155,000 GALLONS	a. Return to Step 16.
	b. SI system - ALIGNED FOR INJECTION	b. <u>**IF**</u> SI system has already been aligned for cold leg recirculation, <u>**THEN**</u> go to Step 24.
	c. Go to 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1	

BASIS DOCUMENT

WOG Procedure Step 17 **PTN Procedure Step** 19

Check If RCS Cooldown And Depressurization Is Required

BASIS:

The operator should stay in E-1 only for loss of reactor coolant accidents for which the RCS pressure is less than the RHR shutoff head and flow from the RHR pumps has been verified. The RHR pump flow should be verified even though the RCS pressure is less than 250 psig, which is the shutoff head pressure of the RHR pumps plus allowances for normal channel accuracy. Since the post accident transmitter errors are added on to determine the pressure requirement, the actual plant pressure may be significantly less.

For any break in the RCS for which the RCS pressure remains above the shutoff head pressure of the RHR pumps, the operator should transfer to procedure, ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. If the RCS pressure is less than the RHR pump shutoff head but RHR pump flow into the RCS cannot be verified, the operator should also transfer to ES-1.2. From this point on ES-1.2 would be used for plant recovery.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

8 The words "low-head SI pumps" were changed to "RHR pumps" to conform to plant specific terminology.

9 The WOG RNO relies on the rules of usage to direct the operator to ES-1.2 when RHR pump flow can not be verified. To eliminate confusion, the RNO was modified to clarify that a transition to ES-1.2 is necessary if RHR pump flow is less than or equal to the minimum flow indication.

PLANT SPECIFIC SETPOINTS:

250 psig Shutoff head pressure of the RHR pumps plus normal channel accuracy. (EOP Setpoint B.7)

650 psig Shutoff head pressure of the RHR pumps plus normal channel accuracy and post accident transmitter errors. (EOP Setpoint B.8)

1000 gpm Minimum RHR flow which indicates injection into the RCS. (EOP Setpoint S.3)

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, the following ECCS components and flow path shall be OPERABLE:

a. One OPERABLE RHR heat exchanger,

b. One OPERABLE RHR pump, and

c. An OPERABLE flow path capable of (1) taking suction from the refueling water storage tank upon being manually realigned and (2) transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no OPERABLE ECCS flow path from the refueling water storage tank, restore at least one ECCS flow path to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With either the residual heat removal heat exchanger or RHR pump inoperable, restore the components to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date since January 1, 1990.

SURVEILLANCE REQUIREMENTS

4.5.3 The above ECCS components shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.5 EMERGENCY CORE COOLING SYSTEMS (Continued)

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of ECCS components and flowpaths required in Modes 1, 2 and 3 ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming any single active failure consideration. Two SI pumps and one RHR pump operating in conjunction with two accumulators are capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all pipe break sizes up to and including the maximum hypothetical accident of a circumferential rupture of a reactor coolant loop. In addition, the RHR subsystem provides long–term core cooling capability in the recirculation mode during the accident recovery period.

Motor Operated Valves (MOVs) 862A, 862B, 863A, 863B are required to take suction from the containment sump via the RHR system. PC-600 supplies controlling signals to valves MOVs 862B and 863B, to prevent opening these valves if RHR pump B discharge pressure is above 210 psig. PC-601 provides similar functions to valves MOVs 862A and 863A. Although all four valves are normally locked in position, with power removed, the capability to power up and stroke the valves must be maintained in order to satisfy the requirements for OPERABLE flow paths (capable of taking suction from the containment sump).

When PC-600/-601 are calibrated, a test signal is supplied to each circuit to check operation of the relays and annunciators operated by subject controllers. This test signal will prevent MOVs 862A, 862B, 863A, 863B from opening. Therefore it is appropriate to tag out the MOV breakers, and enter Technical Specification Action Statement 3.5.2.a. and 3.6.2.1 when calibrating PC-600/-601.

With the RCS temperature below 350°F, operation with less than full redundant equipment is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

TS 3.5.2, Action g. provides an allowed outage/action completion time (AOT) of up to 7 days to restore an inoperable RHR pump to OPERABLE status, provided the affected ECCS subsystem is inoperable only because its associated RHR pump is inoperable. This 7 day AOT is based on the results of a deterministic and probabilistic safety assessment, and is referred to as a 'risk-informed" AOT extension. Planned entry into this AOT requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP), which is described in the administrative procedure that implements the maintenance rule pursuant to 10CFR50.56.

1.0 PURPOSE

1.1 This procedure provides actions to cool down and depressurize the RCS to cold shutdown conditions following a loss of reactor coolant inventory.

1.2 This procedure is applicable when directed by EOP entry conditions.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 This procedure is entered from E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 19, when RCS pressure is greater than the shutoff head pressure of the RHR pumps.

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 Technical Specifications for Turkey Point Unit 3 and Unit 4

3.1.2 Turkey Point Unit 3 and Unit 4 Final Safety Analysis Report

3.1.3 As-built plant drawings

3.1.4 Procedures

1. 3-ONOP-004.1, SYSTEM RESTORATION FOLLOWING LOSS OF OFFSITE POWER
2. 3-ONOP-030, COMPONENT COOLING WATER MALFUNCTION
3. 3-OP-018.1, CONDENSATE STORAGE TANK
4. 3-OP-023, EMERGENCY DIESEL GENERATOR
5. 4-OP-023, EMERGENCY DIESEL GENERATOR
6. 3-OP-030, COMPONENT COOLING WATER SYSTEM
7. 3-OP-033, SPENT FUEL PIT COOLING SYSTEM
8. 3-OP-064, SAFETY INJECTION ACCUMULATORS
9. 3-OP-072.1, MOISTURE SEPARATOR REHEATERS

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***If RWST level decreases to less than 155,000 gallons, the SI System is required to be aligned for cold leg recirculation using 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.***
- ***RCS pressure is required to be monitored. If RCS pressure decreases in an uncontrolled manner to less than 250 psig[650 psig], the RHR pumps must be manually restarted to supply water to the RCS.***
- ***CCW System load requirements of 3-OP-030, COMPONENT COOLING WATER SYSTEM, SHALL NOT be exceeded.***

NOTE

Foldout page is required to be monitored throughout this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check If RHR Pumps Should Be Stopped**	
	a. Check RHR pumps - ANY RUNNING	a. Go to Step 2.
	b. Check RCS pressure	b. Go to Step 2.
	1) Pressure - GREATER THAN 250 PSIG[650 PSIG]	
	2) Pressure - STABLE OR INCREASING	
	c. Stop RHR pumps **AND** place in Standby	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

DO NOT CONTINUE until RHR pump suction is isolated from the RWST.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13	**Align RHR Suction To Containment Recirc Sump** • Open MOV-3-860A • Open MOV-3-860B • Open MOV-3-861A • Open MOV-3-861B	**IF** at least one path from a containment recirc sump to an operable RHR pump can **NOT** be established, **THEN** go to 3-EOP-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.
14	**Verify Adequate Recirculation Sump Level**	
	a. Verify RWST level – GREATER THAN 60,000 GALLONS	a. **IF** containment recirc sump level less than 427 inches, **THEN** go to 3-EOP-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 31.
	b. Verify containment recirculation sump level - GREATER THAN 427 INCHES	b. Return to Step 14a.

BASIS DOCUMENT

WOG Procedure Step 2 - CAUTION **PTN Procedure Step** 1-CAUTION 2

RCS pressure is required to be monitored. If RCS pressure decreases in an uncontrolled manner to less than 250 psig[650 psig], the RHR pumps must be manually restarted to supply water to the RCS.

BASIS:

Except for relatively large LOCAs, the RCS pressure should remain higher than the shutoff head of the RHR pumps until later in the recovery following a controlled cooldown and depressurization. To avoid damage to the RHR pumps, instructions are provided to stop these pumps early in the recovery if RCS pressure is greater than their shutoff head. However, if RCS pressure decreases to less than the shutoff head, then the pumps will have to be restarted manually since no automatic signal is available. During the controlled depressurization in ES-1.2, injection from the RHR pumps is not desirable unless the closed-loop RHR system has been placed in service. The RHR pumps may have to be restarted upon transfer to cold leg recirculation (ES-1.3)

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

9 This caution was moved to keep it in front of the applicable step. See Step deviation section for WOG step 2.

8 The WOG guidelines do not provide distinct definitions for the terms should and shall. The word "should" was changed to "is required to" to denote a requirement.

8 The words "low-head SI pumps" were changed to "RHR pump" to conform to plant specific terminology.

PLANT SPECIFIC SETPOINTS:

650 psig Shutoff head pressure of the RHR pumps plus normal channel accuracy and post-accident transmitter errors. (EOP Setpoint B.7)

250 psig Shutoff head pressure of the RHR pumps plus normal channel accuracy. (EOP Setpoint B.7)

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 For single Unit operation, one refueling water storage tank (RWST) shall be OPERABLE or for dual Unit operation two RWSTs shall be OPERABLE with:

a. A minimum indicated borated water volume of 320,000 gallons per RWST,

b. A minimum boron concentration of 1950 ppm of boron,

c. A minimum solution temperature of 39°F, and

d. A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With less than the required number of RWST(s) OPERABLE, restore the tank(s) to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The required RWST(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the indicated borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for consecutive 23 hours, and

2) At least once per 24 hours while the outside temperature exceeds its limit.

1. WE02EA2.2 001//T1G1/TS/C/A 3.5/4.0/NEW/NA02301/S/RA/LSM

Unit one is in post-LOCA cooldown. The current temperature is 345°F. A mechanical failure of one train of SI has just occurred. The other train is available.

Which one of the following describes the ITS LCO that applies to this condition?

A. LCO 3.5.2, ECCS - Operating.

B. LCO 3.4.7, RCS Loops - Mode 5, Loops Filled.

C. LCO 3.9.5, Residual Heat Removal and Coolant Circulation—High Water Level.

D.✓ LCO 3.5.3, ECCS - Shutdown.

One train is required in mode 4 by LCO 3.5.3

North Anna Units 1 and 2 B 3.5.3-2 Rev 0 (Draft 1), 06/05/00
ECCS—Shutdown
B 3.5.3
BASES
LCO

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the three cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot or cold legs.

APPLICABILITY In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350....F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

QUESTIONS REPORT
for Westinghouse 3 Loop Questions

1. WE14EA2.2 087//////HR04301/S/

A LOCA occurred several hours ago. Only one (1) Containment Spray Pump is running due to actions taken in EPP-012, "Loss of Emergency Coolant Recirculation."

A transition has just been made to FRP-J.1, "Response to High Containment Pressure." Containment Pressure is 14 psig.

Which of the following actions should be taken?

A. Start the second Containment Spray Pump if Containment pressure does **NOT** decrease below 10 psig before exiting FRP-J.1.

B. Start the second Containment Spray Pump per FRP-J.1since pressure is above 10 psig.

C.✓ Continue operation with one Containment Spray Pump per EPP-012 unless Containment pressure exceeds design, then start the second pump.

D. Continue operation with one Containment Spray Pump per EPP-012 unless Containment pressure begins increasing, then start the second pump.

1. WE03G2.4.4 002/T1G2/T1G2//C/A 4.0/4.3/B/SR02301/R/GWL

-The team has transitioned to ES-1.1, "SI Termination" from E-1 "Loss of Reactor or Secondary Coolant."
-At step 4 of ES-1.1, the team secures all but one charging pump as directed.

Which one of the following describes the required operator actions in accordance with ES1.1 if RCS Pressure begins to decrease after the pump is secured?

A. Restart the last charging pump secured and transition back to E-1, "Loss of Reactor or Secondary Coolant."

B. Start another Charging pump to stabilize RCS pressure and continue with ES-1.1, "SI Termination."

C. Manually re-initiate SI and Transition to E-0, "Reactor Trip or Safety Injection" step 4.

D.✓ Monitor RCS Pressure, if it continues to fall, then transition to ES-1.2, "Post LOCA Cooldown and Depressurization."

Surry Exam Bank Question # 1005, slightly modified.
Surry Lesson Plan ND-95.3-LP-8 and 9. Objective A.

A. Incorrect, the procedure step directs a transition to ES-1.2 if RCS pressure continues to fall.

B. Incorrect, the procedure does not direct another charging pump to be started in ES-1.1.

C. Incorrect, SI re-initiation is not directed and the team would not transition to step 4 of E-0.

D. Correct, If RCS pressure continues to fall the team is directed to transition to ES-1.2.

..WE03G2.4.6.1 03/14/2003

Surry 1 **Exam Level** S

Mark Question | Print Recor | New Searc | Exit

Question

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

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

Answer: Two RCPs should be stopped to minimize RCS heat input, and still produce effective heat transfer and RCS pressure control.

Distracter 1 One RCP should be stopped to produce effective heat transfer, provide boron mixing for RHR operations, and provide RCS pressure control.

Distracter 2 All RCPs should be stopped to minimize RCS inventory loss when the break uncovers.

Distracter 3 Three RCPs should be left running to ensure symetric heat transfer to the S/Gs, to aid in RCS pressure control, and prevent steam voiding in the Reactor vessel head.

Distracter Analysis:

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

Distracter 1: A. Incorrect, Two RCPs should be secured. Mixing for placing RHR on service is not a reason for running RCPs.

Distracter 2: B. Incorrect, The procedure directs the operator to leave an RCP operation if possible.

Distracter 3: D. Incorrect, The procedure directs only one RCP to be left running.

..E03.EA2.2

08/16/2002

Prairie Island 2

Exam Level R

Question

The following conditions exist on Unit 1:

Unit 1 has tripped from 100% power due to a small break LOCA
SI has actuated and all equipment is operating properly
SI flow is 300 gpm
Containment pressure is 3 psig
Containment radiation is 100 R/hour
RCS pressure is at 1600 psig
RCS temperature is at 530 F and stable
ES-1.1, POST LOCA COOLDOWN AND DEPRESSURIZATION has just been entered, and step 2 to stop the RHR pumps was just completed

Which of the following by itself would require tripping of both RCPs?

Answer: RCS pressure decreases to 1100 psig

Distracter 1 PRZR level decreases to 7%

Distracter 2 RCS subcooling decreases to 50 F

Distracter 3 Containment pressure increases to 23 psig

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

..E03.EA2.2

09/29/2003

Point Beach 1

Exam Level R

Mark Question | Print Recor | New Searc | Exit

Question

A small break LOCA has occurred on Unit 1. The crew is performing EOP 1.2, Small Break LOCA Cooldown and Depressurization. Which of the following describes the Reactor Coolant System cooldown rate that is called for in EOP 1.2?

Answer: Less than 100 F / hour in order to preclude violating thermal shock limits.

Distracter 1 Less than 100 F / hour in order to minimize outsurge from the Pressurizer.

Distracter 2 As rapid as possible in order to conserve RWST inventory.

Distracter 3 As rapid as possible in order to shorten the time until Residual Heat Removal is placed in service.

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

23. WE05EA2.2 001//T1G1/HEAT SINK/C/A(3.7/4.3)/B/TP05301/S/MC

Operators are performing EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, and have successfully initiated Bleed and Feed. The BOP subsequently announces secondary heat sink is restored using "A" Standby SG Feed Pump.

Which ONE of the following describes the correct operator response?

A. Return to procedure and step in effect when feed flow is verified to be > 354 gpm.

B.✓ Continue performing FR-H.1 to completion.

C. Return to procedure and step in effect when narrow range level in any S/G is > 6%[32%].

D. Return to procedure and step in effect only when narrow range levels in all S/Gs are > 6%[32%].

Feedback

REFERENCES:

1. FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, steps 7,27, rev 04/30/02

DISTRACTORS:

A Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because 345 gpm is a normal indicator of adequate heat sink.

B Correct. After Bleed and Feed is established, FR-H.1 must be completed to ensure SI reduction/termination and PORV closure are completed. Restoration of secondary heat sink is not enough to transition from FR-H. beyond Step 12 of the procedure.

C Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because 6% in any S/G is a normal indicator of adequate heat sink.

D Incorrect. After Step 12, restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because >6% in all S/Gs is a goal of the procedure.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of Secondary Heat Sink; Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	**Check S/G Levels**	
	a. Narrow range level in at least one S/G - GREATER THAN 6%[32%]	a. Perform the following: 1) Verify standby feedwater flow to at least one S/G. **IF** flow can **NOT** be verified, **THEN** go to Step 8. 2) Maintain flow to restore narrow range level to greater than 6%[32%] in at least one S/G.
	b. Return to procedure **AND** step in effect	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***If RWST level decreases to less than 155,000 gallons, the SI System is required to be aligned to cold leg recirculation using 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.***
- ***Feed flow is required to be initiated at a rate NOT to exceed 100 gpm to avoid excessive RCS cooldown and limit thermal stress in the S/Gs with wide range level less than 10% [33%].***
- ***Operation of RHR pumps for greater than 30 minutes on minimum recirculation may cause overheating and failure of the RHR pump.***

27 **Continue Attempts To Establish Secondary Heat Sink In At Least One S/G**

* AFW flow using Step 2
* Main feedwater flow using Step 4
* Standby S/G feedwater pumps using Step 6
* Unit 2 or Unit 4 feedwater flow using Step 8
* Condensate flow using Step 10

54. WE05EA2.2 REPLACEMEN 001/1/1/HEAT SINK/C/A (3.7/4.3)/N/TP03301/S/RFA

Operators are performing EOP-FR-H.1, Response to Loss of Secondary Heat Sink, and have successfully initiated Bleed and Feed.

The BOP subsequently announces secondary heat sink is restored using "A" Standby Steam Generator Feed Pump.

Which ONE of the following describes the correct operator response?

A.✓ Continue performing FR-H.1 to completion.

B. Return to procedure and step in effect when feed flow is verified to be > 345 gpm.

C. Return to procedure and step in effect when narrow range level in any S/G is > 6% [32%].

D. Return to procedure and step in effect only when narrow range levels in all S/Gs are > 6% [32%].

References: FR-H.1, step 7, 27

A. Correct because after Bleed and Feed is estabalished, Fr-H.1 must be completed to ensure SI reduction/termination and PORV closure are completed. Restoration of secondary heat sink is not enough to transition from FR-H.1 beyond step 12 of the procedure.

B. Incorrect because after Step 12 restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because 345 gpm is a normal indicator of adequate heat sink.

C. Incorrect because after Step 12 restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because 6% in any S/G is a normal indicator of adequate heat sink.

D. Incorrect because after Step 12 restoration of secondary heat sink is not enough to transition from FR-H.1. Plausible because > 6% in all S/Gs is a goal of the procedure.

53. WE05EA2.2 001/1/1/H.1 PROCEDURE/M 3.7/4.3/BANK/SM04301/R/GWL

Which ONE of the following describes the sequence of actions required to establish bleed and feed heat removal, in accordance with, EOP-15.0, "Response to Loss of Secondary Heat Sink?"

A. "Feed" is established first by initiating Main Feed to a S/G, then "Bleed" is established by depressurizing a S/G with an atmospheric steam dump.

B. "Bleed is established first by opening Both Pressurizer PORVs, then "Feed" is established by initiating Safety Injection.

C.✓ "Feed" is established first by initiating Safety Injection, then "Bleed" is established by opening All Pressurizer PORVs.

D. "Bleed is established first by depressurizing a S/G with an atmospheric steam dump, then "Feed" is established by initiating Main Feed to a S/G.

Summer Bank Question # 3076.

A. Incorrect, feed is established first, however it is with SI flow, and Bleed is with pressurizer PORVs.

B. Incorrect, bleed is established after Feed is established.

C. Correct, Feed by initiating an SI is established first, and then the Pressurizer PORVs are opened to establish a bleed path.

D. Incorrect, bleed is established after Feed is established, and not from these sources.

Test Name	Test Date	rpb	p(Diff)	Time	Equ	User Values	
<Cumulative>		0.000	1.000	0	N	1: 0	2: 0
Summer RO Test	04/15/2004	0.000	1.000	0	N	3: 0	4: 0

	--- A ---			--- B ---			--- C ---			--- D ---					
	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg	Resp	%	Avg
<Cumulative>						Total:	6	100				Omits:	0	0	
	0	0	0.00	0	0	0.00	6	100	82.89	0	0	0.00			
Summer RO Test						Total:	6	100				Omits:	0	0	
	0	0	0.00	0	0	0.00	6	100	82.89	0	0	0.00			

..W/E05.EA1.2

05/30/2003

Seabrook 1

Exam Level R

Mark Question | Print Record | New Search | Exit

Question

A reactor trip with SI has occurred. The crew transitioned from E-0, AReactor Trip or Safety Injection@, TO FR-H.1, ALoss of Secondary Heat Sink@, based on valid red path condition on the heat sink CSF.

When the crew checked whether heat sink was required the Primary Operator reported that RCS pressure was 700 psig and slowly decreasing. The secondary operator reported that all S/G pressures were approximately 950 psig and stable.

Based on this information, the unit supervisor transitioned to E-1, ALoss of Reactor or Secondary Coolant@, Step 1.

Which of the following summarizes plant conditions?

Answer: A LOCA is in progress. Heat transfer in the RCS during the casualty was such that the S/Gs are currently not functioning as a heat sink and therefore not required, resulting in transition to E-1 to combat LOCA.

Distracter 1 A LOCA is in progress. Heat transfer in the RCS during the casualty was such that the S/Gs are currently a heat sink but are not required resulting in transition to E-1 to combat LOCA.

Distracter 2 A LOCA is in progress. Because S/Gs are the sole heat sink, a transition to E-1 is made to minimize coolant loss and restore S/G levels to normal band.

Distracter 3 A LOCA is in progress. Heat is being transferred from the S/Gs to the RCS. The US incorrectly transitioned to E-1. Remain in FR-H.1 to recover S/Gs.

Distracter Analysis:

Answer: S/Gs are now a heat sourceBheat is being transferred from S/G to RCS.
B, C, and D are incorrect

Distracter 1:

Distracter 2:

Distracter 3:

..W/E05.EA1.2 05/30/2003

Seabrook 1 **Exam Level** S

Mark Question | Print Recor | New Searc | Exit

Question

The following plant conditions exist:

$The crew is responding to a LOCA in accordance with E-1, ALoss of Reactor or Secondary Coolant@.
$A RED path occurs on the Heat Sink Critical Safety Function Status Tree.
$The crew transitions to FR-H.1, AResponse to Loss of Secondary Heat Sink@.
$Total available EFW flow is 350 gpm.
$RCS pressure is 470 psig and STABLE.
$Containment pressure is 17 psig and INCREASING.
$SG <A=, >B=, >C= and >D= pressures are all 950 psig and STABLE.
$SG <A=, >B=, >C= and >D= wide range levels are 59% and DECREASING.

Which of the following actions are required?

Answer: Transition back to E-1, ALoss of Reactor or Secondary Coolant@.

Distracter 1 Establish EFW flow from SUFP.

Distracter 2 Immediately perform FR-H.1, AResponse to Loss of Secondary Heat Sink@, steps 10 B 14, to initiate feed and bleed.

Distracter 3 Attempt to establish EFW flow to at least ONE steam generator.

Distracter Analysis:

Answer: A - correct - RCS pressure is less than the pressure of any non-faulted S/G. This means that the S/Gs cannot be used as a heat sink - FR-H.1 directs the user to the procedure and step in effect.

Distracter 1:
B - incorrect - this is step 3 RNO - never gets there.
C - incorrect - this is action for SG levels less than 26% (50%).
D - incorrect - step 3 has user establish EFW to at least one S/G - user never gets to step 3.

Distracter 2:
B - incorrect - this is step 3 RNO - never gets there.
C - incorrect - this is action for SG levels less than 26% (50%).
D - incorrect - step 3 has user establish EFW to at least one S/G - user never gets to step 3.

Distracter 3:
B - incorrect - this is step 3 RNO - never gets there.
C - incorrect - this is action for SG levels less than 26% (50%).
D - incorrect - step 3 has user establish EFW to at least one S/G - user never gets to step 3.

24. WE08EG2.4.4 001//T1G2/PTS/M(4.0/4.3)/B/TP05301/S/MC

Which ONE of the following conditions would require entering FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, on an orange or red path?

A. Cooldown of cold leg GREATER THAN 100°F in 60 minutes, cold leg = 325°F, RCS pressure = 450 psig

B.✓ Cooldown of cold leg GREATER THAN 100°F in 60 minutes, cold leg = 280°F, RCS pressure = 460 psig

C. Cooldown of cold leg LESS THAN 100°F in 60 minutes, Tavg = 280°F, RCS pressure = 460 psig

D. Cooldown of cold leg LESS THAN 100°F in 60 minutes, Tavg = 270°F, RCS pressure = 450 psig

Feedback

REFERENCES:

1. EOP-F-0, CRITICAL SAFETY SYSTEM STATUS TREES, page 10, rev 08/03/01

DISTRACTORS:

A Incorrect. Cold leg temperature is NOT less than 320°F, but is less than 350°F, returning a Yellow path. Pressure is of no consequence.

B Correct. Cold leg temperature is less than 290°F, returning a red path.

C Incorrect. Tavg is NOT less than 275°F, returning a green path.

D Incorrect. Tavg is NOT less than 275°F and pressure is NOT greater than 460 psig.

K/A CATALOGUE QUESTION DESCRIPTION:

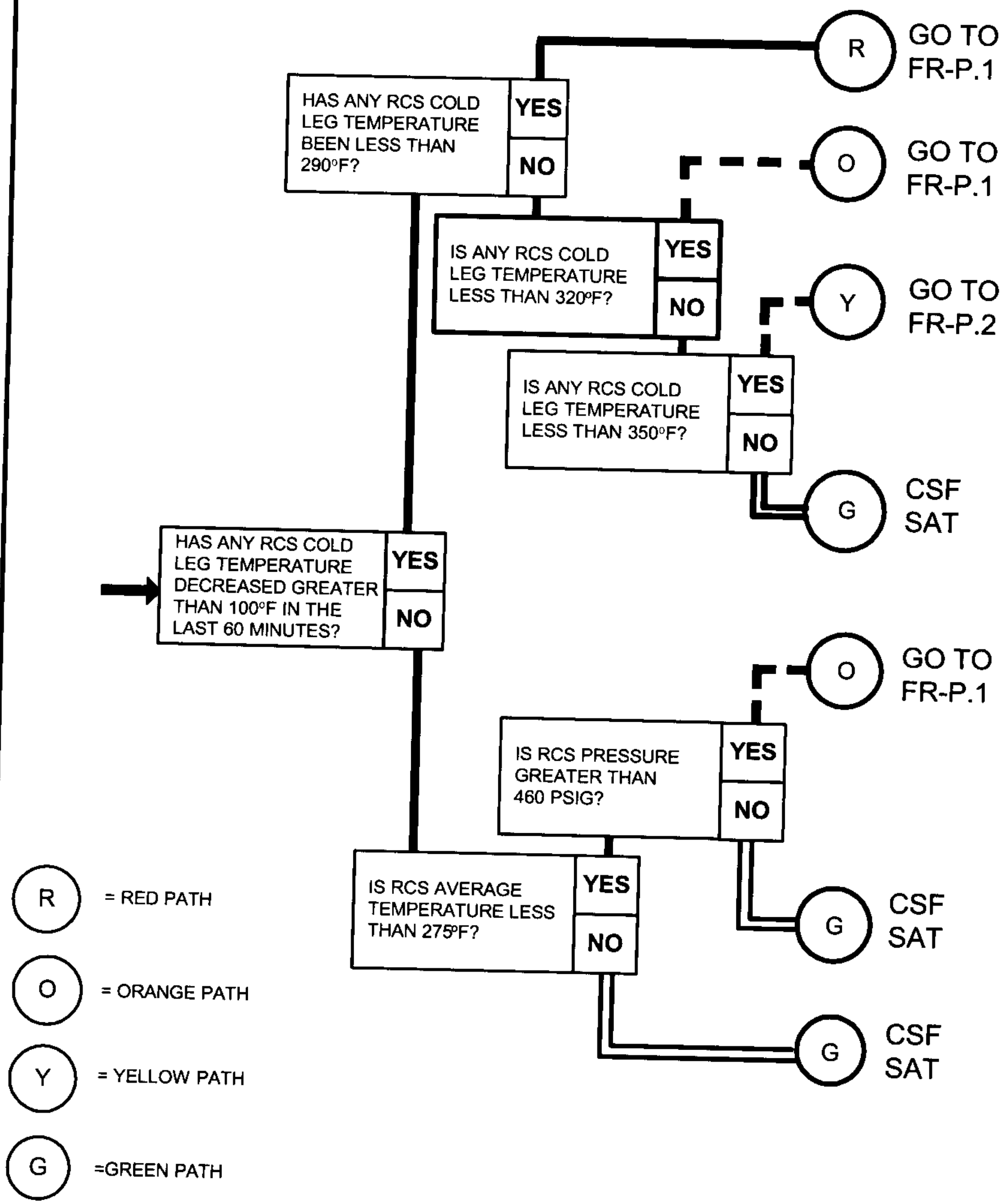
- Pressurized Thermal Shock; Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Procedure No:	Procedure Title:	Page:
3-EOP-F-O	CRITICAL SAFETY SYSTYEM STATUS TREES	10
		Approval Date: 08/03/01

ENCLOSURE 4

(Page 1 of 1)

CSF F-0.4 INTEGRITY



QUESTIONS REPORT
for Westinghouse 3 Loop Questions

1. WE08EA2.1 001/T1G1/T1G1//C/A 3.4/4.2/B/SR02301/S/GWL

Which one of the following conditions would require entering FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition" on an orange or red path?

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

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

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

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

Feedback

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

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

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

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

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

Notes

25. WE16EA2.1 001//T1G2/HIGH RAD/M(2.9/3.3)/B/TP05301/S/MC

Which ONE of the following describes an entry criteria for FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL?

A. Greater than 2.6 E 3 on R-11/12.

B. Greater than 6.1 E 5 on R-11/12.

C✓ Greater than 1.3 E 4 on CHRRMS.

D. Greater than 1.3 E 3 on CHRRMS.

Feedback

REFERENCES:

1. FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL, page 3, rev 04/15/99
2. EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES, page 11, rev 04/15/99C
3. ARP-097.CR, CONTROL ROOM ANNUNCIATOR RESPONSE, page 424, rev 07/23/02

DISTRACTORS:

A	Incorrect.	FR-Z.3 is not entered based on this reading.
B	Incorrect.	FR-Z.3 is not entered based on this reading.
C	Correct.	This is the value that will initiate entry into FR-Z.3.
D	Incorrect.	FR-Z.3 is not entered based on this reading.

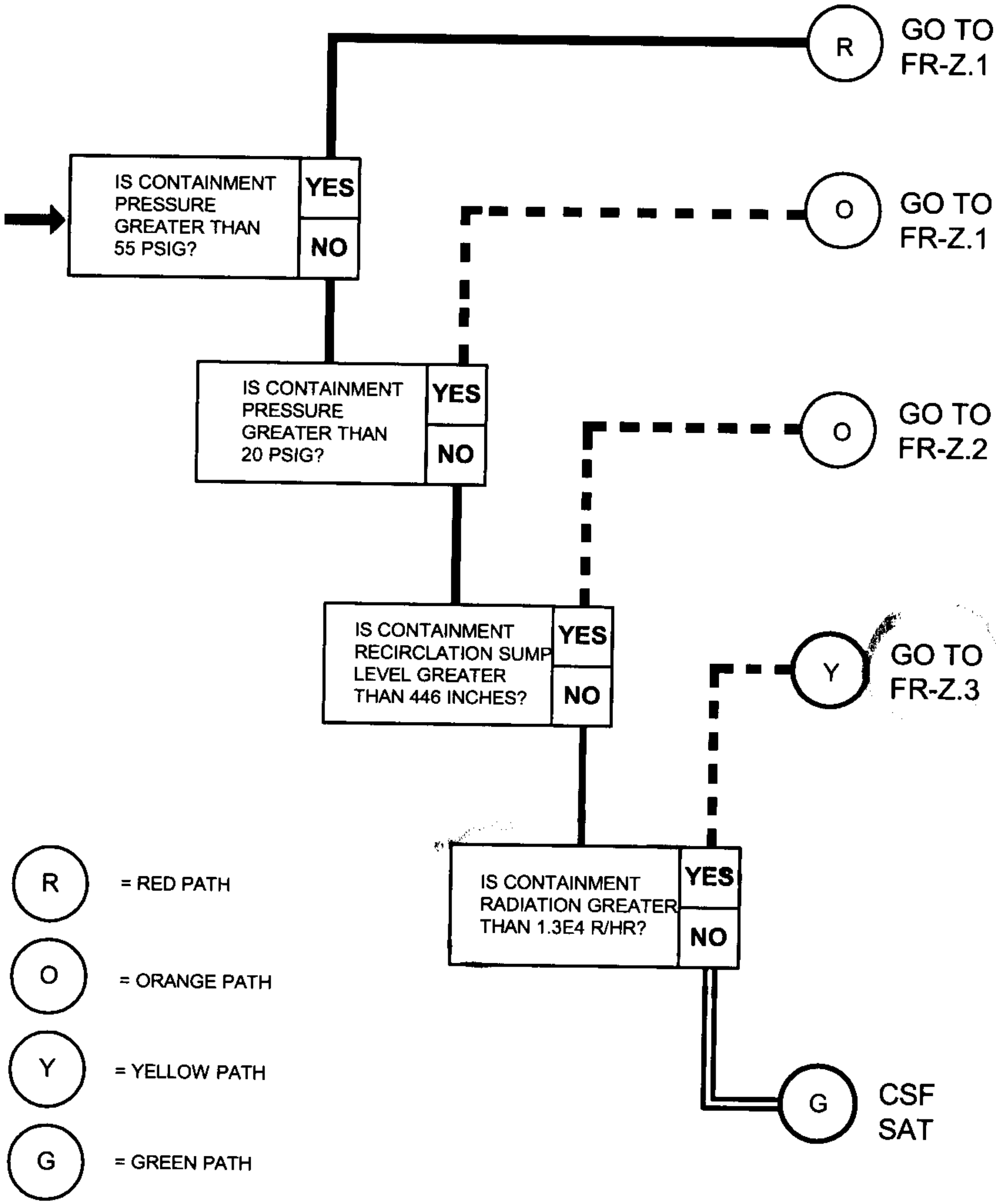
K/A CATALOGUE QUESTION DESCRIPTION:

- High Containment Radiation; Ability to determine and interpret the following as they apply to the (High Containment Radiation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

ENCLOSURE 5

(Page 1 of 1)

CSF F-0.5 CONTAINMENT



1.0 PURPOSE

1.1 This procedure provides actions to respond to high containment radiation level.

1.2 This procedure is applicable when directed by EOP entry conditions.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 This procedure is entered from F-0.5, CONTAINMENT Critical Safety Function Status Tree on a YELLOW condition.

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 Technical Specifications for Turkey Point Unit 3 and Unit 4

3.1.2 Turkey Point Unit 3 and Unit 4 Final Safety Analysis Report

3.1.3 As-built plant drawings

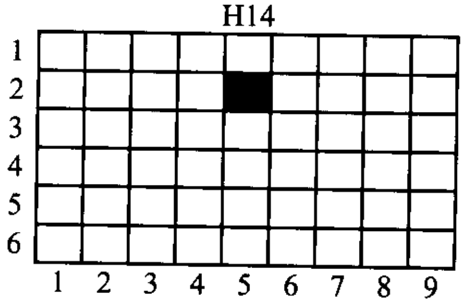
3.1.4 Procedures

1. None

3.1.5 Plant Change/Modifications

1. None

YELLOW	POWER PRODUCTION AVAILABILITY	H 5/2



ATTACHMENT 8
Page 26 of 54
Panel H

CNTMT
ISOLATION
ACTIVATED

DEVICES:
Lockout relays in CNTMT Isolation racks QR50/51 for both trains of Phase A and B CNTMT Isolation and Control Room Ventilation Isolation

SETPOINTS:
R-11 (6.1×10^5 CPM)
R-12 (2.6×10^3 CPM)
2/3 HI (4 psig) **AND** 2/3 HI-HI (20 psig)
Safety Injection
Manual Phase A or B

OPERATOR ACTIONS:

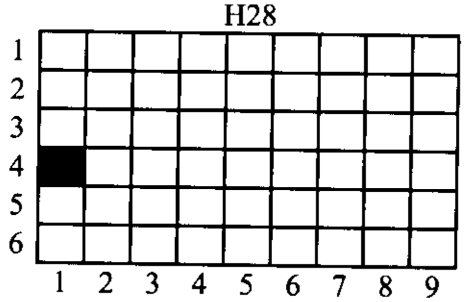
1. Verify alarm by checking the following:
   a. CNTMT Phase A and/or B valve white lights (VPB).
   b. Train A and B control room ventilation system panels in QR-81/82 in recirculation alignment (Unit 4).
2. Verify the following automatic actions have occurred:
   a. For any auto SI - CNTMT Isolation Phase A
   b. For HI and HI-HI CNTMT pressure - CNTMT Isolation Phase B
   c. For any SI or R-11/R-12 HI Alarm trip - CNTMT **AND** Control Room ventilation isolation
3. Corrective actions:
   a. **IF** SI/Rx trip initiating event, **THEN** follow 3-EOP-E.0, REACTOR TRIP OR SAFETY INJECTION.
   b. Verify proper CNTMT Isolation Phase A/B and CNTMT/Control Room ventilation isolation alignment, as appropriate.
   c. Refer to 3-ONOP-067, Radioactive Effluent Release to verify Control Room ventilation recirculation operation **AND** to restore it to normal alignment when required.
   d. **IF** reactor cavity is filled in refueling mode **AND** CNTMT pressure is increasing **THEN** perform the following:
      (1) Close SFP transfer canal gate valve, 3-12-031.
      (2) Terminate service/breathing air activities in CNTMT.
      (3) Isolate service air and breathing air to containment.
   e. Refer to Tech Spec 3/4.3.2, 3/4.6, and 3/4.7.5

CAUSES:
1. CNTMT isolation Phase A/B or CNTMT/Control Room ventilation isolation due to auto/manual SI, HI/HIHI CNTMT pressure, R-11/R-12 HI alarm, or Manual Phase A/B initiation.

REFERENCES:
1. FPL Logic Diag 5610-T-L1, Sh 11
2. Tech Spec Section 3/4.3.2, 3/4.6, 3/4.7.5

YELLOW	POWER PRODUCTION AVAILABILITY	H 1/4

H28



ATTACHMENT 8
Page 4 of 54
Panel H

PRMS
HI RADIATION

DEVICES:

R-11, R-12, R-14,
R-15, R-17A/B,
R-18, R-19, & R-20

SETPOINTS:

Variable with each PRMS channel

OPERATOR ACTIONS:

1. Verify alarm by checking the following:
   a. Countrate meter on each PRMS drawer in rack QR-66.
   b. Alarm indicators on each drawer in rack QR-66.

2. Verify the following automatic actions have occurred:
   a. Refer to 3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE, for alarms on R-11, R-12, R-14, R-17A/B, R-18, **OR** R-20.

   b. Refer to 3-ONOP-071.2, STEAM GENERATOR TUBE LEAKAGE, for an alarm on R-15 **OR** R-19.

3. Corrective actions:
   a. Verify valid alarm on affected PRMS channel (for all channels except R-11/12 and R-20): |
      (1) Check FAIL/TEST light not lit.
      (2) Push FAIL/TEST light (meter reading of 288 or 289K)
      (3) Push SOURCE CHECK light (should get meter increase).
      (4) Push HIGH ALARM light to determine if meter level is above high alarm setpoint.
   b. Verify or manually initiate required automatic actions.
   c. **IF** the alarm is on R-11, R-12, R-14, R-17A/B, R-18 **OR** R-20, **THEN** refer to 3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE.
   d. **IF** the alarm is on R-15 **OR** R-19, **THEN** refer to 3-ONOP-071.2 STEAM GENERATOR TUBE LEAKAGE.
   e. Refer to TS 3.3.3, 3.4.6, and 3.9.13 for additional required actions.

CAUSES:

1. High radiation in on of the systems monitored by PRMS.
2. PRMS system component failure.

REFERENCES:

1. Tech Spec Sections 3.3.3, 3.4.6, and 3.9.13

QUESTIONS REPORT

for Westinghouse 3 Loop Questions

1. WE16EK1.3 002/T1G2/T1G2//M 3.0/3.3/B/SR02301/C/GWL

   Which one of the following describes an <u>entry</u> criteria for FR-Z.3, "Response to Containment High Radiation Level?"

   A. High alarm reading on the Manipulator Crane radiation monitor.

   B. High alarm reading on the Reactor Containment Area radiation monitor.

   C.✓ Greater than 3.0 E 2 on the CHRRMS radiation monitor.

   D. Greater than 50 uCi/cc on the Kaman Radiation High Range High Monitor.

   Surry Exam bank Question # 1032 slightly modified.
   Surry Lesson Plans ND-95.3-LP-50A. ND-93.5-LP-1; and ND-L93.5-LP-3.

   A,B, and D incorrect, FR-Z.3 is not entered based on any of these readings.

   C, Correct, this is the value that will initiate entry in to FR-Z.3.

PRMS Channels R-11 &/or R-12 radiation particulate gaseous release rate

CHRRMS - containment dose rate level indication RRD-6311A & 6311B