QUESTION 1

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	007	EK1.06
Importance Rating	3.7	

Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Relationship of emergency feedwater flow to S/G and decay heat removal following reactor trip

Proposed Question: RO Question # 1

Initial Conditions:

- A reactor trip has occurred on Unit 3 due to a Loss of Offsite Power.

Current Conditions:

- The crew is performing actions of 3-EOP-ES-0.1, Reactor Trip Response.
- Offsite Power has NOT been restored.
- RCS Thot is 565°F and slowly lowering.
- RCS Tcold is 540°F and slowly lowering.
- S/G Steam Dumps to Atmosphere are modulated open.
- Total AFW flow is 450 gpm and stable.
- All SG levels indicate 1% NR and rising slowly.

In accordance with 3-EOP-ES-0.1, which ONE of the following describes the required adjustments in steam flow and total feed flow for this condition?

A. Increase dumping steam and raise total feed flow as necessary until one S/G is greater than 6% narrow range level.

B. Stop dumping steam and if cooldown continues reduce total feed flow to just above 345 gpm until one S/G is greater than the 32% narrow range level.

C. Increase dumping steam and raise total feed flow as necessary until one S/G is greater than 32% narrow range level.

D. Stop dumping steam and if cooldown continues reduce total feed flow to just above 345 gpm until one S/G is greater than the 6% narrow range level.

Proposed Answer: D

Explanation (Optional):

A. Incorrect because heat sink requirements are met with AFW flow at 450 gpm and stable. Plausible because these actions are similar to those taken in 3-EOP-ES-0.1, RNO Step 3, when Tcold is increasing.

B. Incorrect because 32% NR is not required for secondary heat sink. Actions are correct but level is higher than required by the procedure.

C. Incorrect because heat sink requirements are met with AFW flow at 450 gpm and stable. Also, incorrect because 32% NR is not required for secondary heat sink. Plausible because these actions are similar to those taken in 3-EOP-ES-0.1, RNO Step 3, when Tcold is increasing.

D. CORRECT. When Tcold is below 547 degrees F and decreasing, 3-EOP-ES-0.1, RNO Step 3 requires operator to reduce total feed flow to greater than 345 gpm until NR level in at least 1 SG is >6%.

Technical Reference(s): 3-EOP-ES-0.1, Step 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 69-00323 Obj 3 (As available)

Question Source: Bank # WTSI 63679

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Harris

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that the operational implications of "required adjustments for AFW flow", during decay heat removal, are tested.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Maintain RCS Cold Leg Temperature**	
	a. All RCS Cold Leg Temperatures - GREATER THAN OR EQUAL TO 525°F	a. Emergency borate for uncontrolled cooldown using 3-ONOP-046.1, EMERGENCY BORATION while continuing with this procedure.
	b. RCS Cold Leg Temperature * STABLE AT 547°F **OR** * TRENDING TO 547°F **OR** * Stable at post trip value - LESS THAN 547°F	b. Perform the following: 1) **IF** temperature less than 547°F **AND** temperature is decreasing, **THEN** perform the following: a) Stop dumping steam. b) Verify S/G blowdown isolation valves closed. c) **IF** cooldown continues, **THEN** control total feed flow. Maintain total feed flow greater than 345 gpm until narrow range level greater than 6% in at least one S/G. d) **IF** cooldown continues due to excessive steam flow, **THEN** close main steamline isolation and bypass valves. 2) **IF** temperature greater than 547°F **AND** temperature is increasing, **THEN** perform the following: * Dump steam to condenser. **OR** * Dump steam using S/G steam dump to atmosphere valves.

QUESTION 2

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	008	AA1.03
	Importance Rating	2.8	

Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: Turbine bypass in manual control to maintain header pressure

Proposed Question: RO Question # 2

Given the following:

- Unit 3 has tripped and safety injection actuated due to a stuck open Pressurizer PORV.
- The crew is performing actions of 3-EOP-E-0, Reactor Trip or Safety Injection.
- PI-3-1406, Condenser Vacuum, indicates 18" Hg and lowering.
- Tavg is 560°F and RISING slowly.

Which ONE of the following steam dump manipulations result in meeting the RCS temperature requirements in 3-EOP-E-0?

A. Manually adjust the S/G Steam Dump To Atmosphere Controller Setpoints to LOWER Tavg.

B. Manually control the S/G Steam Dump To Atmosphere Controller Output Demands to MAINTAIN RCS Tavg.

C. Manually operate the Steam Dump To Condenser Press. Control Output Demand in the Steam Pressure mode to LOWER Tavg.

D. Manually adjust the Steam Dump To Condenser Press. Control Potentiometer to control the Condenser Steam Dumps to MAINTAIN Tavg.

Proposed Answer: A

Explanation (Optional):

A. CORRECT. According to lesson plan 6902118, Steam Dump, Page 26, with condenser vacuum less than 20" Hg, steam dumps to the condenser are blocked. 3-EOP-E-0, RNO Step 10.b requires dumping steam, by either condenser or atmospheric steam dump valves, if Tcold is greater than 547°F and increasing.

B. Incorrect because maintaining current temperature is incorrect. RCS Tcold should be maintained at 547°F, not 560°F per 3-EOP-E-0, Step 10. Plausible because there are other conditions in EOPs which require RCS temperature to be stabilized where it is at before proceeding. Also plausible because the atmospheric dumps must be used (that part is correct) and control via setpoint adjustment is available.

C. Incorrect because the condenser dumps are not available since vacuum is less than 20" Hg. Plausible because the manual operation of the Steam Dump Controller <u>is</u> available and because the desired response of RCS temperature is correct.

D. Incorrect because maintaining current temperature is incorrect and because the condenser dumps are not available since vacuum is less than 20" Hg. Plausible because there are other conditions in EOPs which require RCS temperature to be stabilized where it is at before proceeding.

Technical Reference(s): 3-EOP-E-0, RNO Step 10.b (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 6918321, Obj. 4 (PowerPoint) (As available)

Question Source: Bank # WTSI 70173

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Comanche Peak

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 7

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that, during the onset of a PRZ vapor space accident (PORV stuck open), manual operation of the atmospheric steam dump valves (to lower RCS temperature) is tested.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	**Maintain RCS Cold Leg Temperature**	Perform the following:
	* STABLE AT **OR** TRENDING TO 547°F IF ANY RCP RUNNING	a. **IF** temperature is decreasing, **THEN** perform the following:
	OR	1) Stop dumping steam.
	* LESS THAN 547°F **AND** STABLE IF NO RCP RUNNING	2) Reduce total feed flow to 345 gpm until narrow range level greater than 6%[32%] in at least one S/G.
		3) **IF** cooldown is due to excessive steam flow, **THEN** close main steamline isolation and bypass valves.
		b. **IF** temperature greater than 547°F **AND** increasing, **THEN** perform the following:
		* Dump steam to condenser.
		OR
		* Dump steam using S/G steam dump to atmosphere valves.

BASIS DOCUMENT

WOG Procedure Step 20 **PTN Procedure Step** 10

Maintain RCS Cold Leg Temperature

BASIS:

RCS temperature stable at or trending to the no-load value indicates that the secondary steam dump system is operating as designed. If the RCS cooldown is excessive, then steam dump should be stopped. Excessive feed to the steam generators can also result in cooling down the RCS and it may be necessary to reduce feed flow to the minimum for decay heat removal until S/G level is in the narrow range. If the cooldown continues, the main steamlines are isolated to stop any steam leakage downstream of the MSIVs, such as a stuck open condenser steam dump valve.

If RCS temperature is greater than no-load and increasing, then steam dump from the secondary must be increased for decay heat removal.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

9 Feedback from operator training and EOP validation has shown that proper operator response is not always obtained if Tavg is stable but not at 547°F (natural circulation). The high level step was reworded to clarify the two acceptable situations for RCS Tavg.

9 The RNO was reindexed to improve readability.

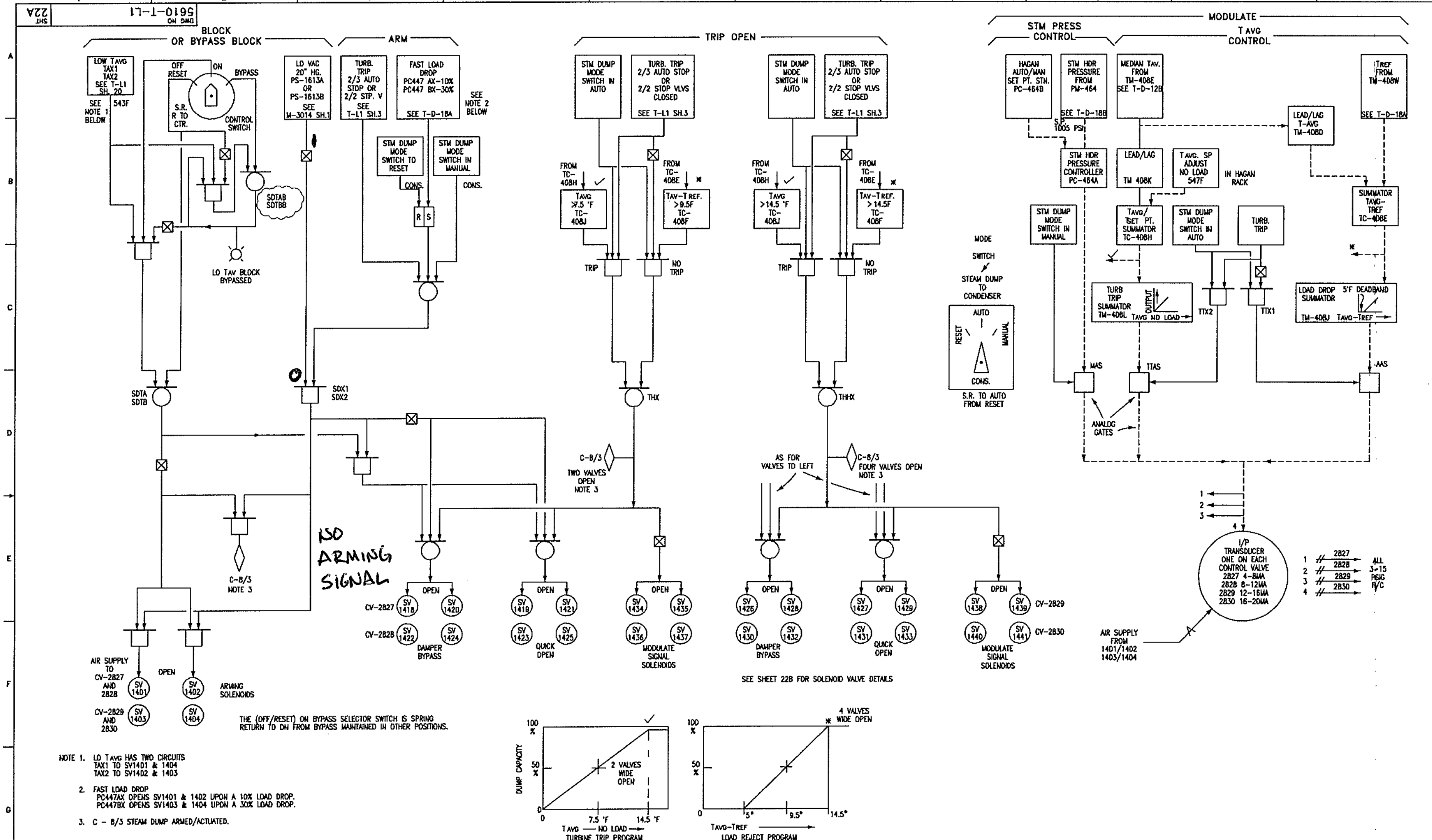
9 The IF, THEN logic statements were modified to improve readability and to conform to the plant specific Writer's Guide.

8 The words "SG PORVs" were changed to "S/G steam dump to atmosphere valves" to conform with plant specific terminology.

9 This step was made to be a continuous action step to provide a control parameter for the operator. As such, the action verb was changes from "Check" to "Maintain".

1 The WOG RNO directs the operator to close MSIVs if the cooldown continues after reducing feedwater flow. The RCS cooldown could be the result of low decay heat level combined with minimum required AFW flow or from a LOCA induced cooldown. To preclude unnecessary MSIV closure, the words "due to excessive steam flow" were added.

(Continued on next page)



NOTE 1. LO TAVG HAS TWO CIRCUITS
TAX1 TO SV1401 & 1404
TAX2 TO SV1402 & 1403

2. FAST LOAD DROP
PC447AX OPENS SV1401 & 1402 UPON A 10% LOAD DROP.
PC447BX OPENS SV1403 & 1404 UPON A 30% LOAD DROP.

3. C - B/3 STEAM DUMP ARMED/ACTUATED.

REV	DATE	REVISION	BY	CH	APP	APP
12	1-20-98	ISSUED AS-BUILT PER CRN-I-3883 (PC/M 97-042).	RV	RH	MA	TS
11	11-9-95	INCORPORATED CRN-I-3253 FOR PC/M 95-115.	RV	RH	BAP	JCW
10	5-18-95	INCORPORATED CRN-I-3171 FOR PC/M 95-042.	RV	RH	JJN	JCW

REV	DATE	REVISION	BY	CH	APP	APP
9	5-11-95	ISSUED AS-BUILT FOR PC/M 94-120	RV	RH	BAW	JCW
8	3-21-95	INCORPORATED CRN-I-3140 FOR PC/M 95-001.	RV	LH	–	JCW
7	12-19-94	ISSUED AS-BUILT FOR PC/M 94-110 AND INCORP. CRN-I-3052.	RV	LH	RAS	JJN
6	9-25-91	AS-BUILT FOR DCR-TPI-91-400	RV	LH	ET	JG
5	12-27-90	AS-BUILT FOR DCR-TPI-90-403	RV	LH	JG	FV

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

ELECTRICAL

STEAM DUMP
TO CONDENSER

POD

DRAWING NUMBER 5610-T-L1

SHEET 22A

SYS

REV 12

CAD FILE: UTE01032.DWG

JPN ADESS

QUESTION 3

Facility: Turkey Point

~~Vendor:~~ ~~WEC~~

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	009	EK3.20
Importance Rating	3.5	

Knowledge of the reasons for the following responses as the apply to the small break LOCA: Tech-Spec leakage limits

Proposed Question: RO Question # 3

Initial Conditions:

- Unit 4 is in Mode 1 at 100% power
- A RCS Auto Makeup is in progress.
- There is identified S/G tube leakage through the 4C S/G at 0.09 gpm.
- Condenser Air Ejector Gas Monitor, R-4-15, is OOS.
- RCS Pressure is at 2235 psig and stable.
- An identified RCS leak inside Containment is calculated at 1.5 gpm.
- The Unit Supervisor has reviewed TS 3.4.6.2 for RCS operational leakage

The following occurs without any operator actions:

- 4B Charging Pump speed maximizes.
- RCS Pressure is at 2210 psig and lowering.
- Steam Generator levels are at setpoint.
- The CNTMT SUMP HI LEVEL (G 9/5) annunciator alarms.
- Pressurizer Level is 46% and decreasing.
- Containment Pressure is rising steadily.

Which ONE of these choices (1) is the reason for this integrated system response and (2) is entry into TS 3.4.6, Reactor Coolant System Operational Leakage, required and why?

A. (1) 4C S/G tubes have ruptured as revealed by the Pressurizer level lowering.
(2) Enter TS 3.4.6.2 and take appropriate action to limit further degradation of the RCS.

B. (1) 4C S/G tubes have ruptured as revealed by S/G levels oscillations.
(2) Do Not Enter TS 3.4.6.2 due to leakage limits not exceeded.

C. (1) The RCS leak has increased as reflected by Charging Pump speed rising.
(2) Do Not Enter TS 3.4.6.2 due to leakage limits not exceeded.

D. (1) The RCS leak has increased as reflected by Pressurizer level lowering.
(2) Enter TS 3.4.6.2 and take appropriate action to limit further degradation of the RCS.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. 4C SG shows no sign of rupture since there is no rise in secondary activity and containment sump level is rising, indicating a different RCS leak. Second half is correct.

B. Incorrect. 4C SG shows no sign of rupture since there is no rise in secondary activity and containment sump level is rising, indicating a different RCS leak. Second half is incorrect, and applicant must determine that leak rate has changed to above TS limits when PZR level is dropping.

C. Incorrect. The first part is a correct statement, but second half is incorrect, as TS LCO has been exceeded for RCS leakage

D. Correct. PZR level lowering indicates an imbalance that in this case is caused by RCS leakage. TS action is required

Technical Reference(s): TS 3.4.6.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902913, Obj. 7 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3PEO)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATING

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator (SG),

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4-1 up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than allowed by 3.4.6.2.e above operation may continue provided:

   1. Within 4 hours verify that at least two valves in each high pressure line having a non-functional valve are in, and remain in that mode corresponding to the isolated condition, i.e., manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized. Follow applicable ACTION statement for the affected system, and

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* Test pressure less than 2235 psig are allowed. Minimum differential test pressure shall not be less than 150 psid. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

ATTACHMENT 3
(Page 1 of 4)

LEAK RATE CALCULATION DATA SHEET (MANUAL METHOD)

QA RECORD PAGE

Date: __________

PEAK SHIFT

	ERDADS	Meter No.	Start	Stop	Start + Stop / 2	*CF	(Stop-Start) xCF
1. TIME	N/A	N/A			////////////	N/A	min
2. VCT Level	L115_A	LI-3-115	%	%	////////////	14.15	gal
3. PZR Level	L462_A	LI-3-459A	%	%	////////////	42.1	gal
4. Primary Water Totalizer	TOTPWCG_V	N/A	gal	gal	////////////	1	gal
5. Boric Acid Totalizer	TOTBACG_V	N/A	gal	gal	////////////	1	gal
6. PRT Level	L470_A	LI-3-470	%	%	////////////	100	gal
7. RCDT Level	3L1003_A	LI-3-1003	%	%	////////////	3.2	gal
8. Cont. Sump Level	L1546_A	LR-3-1418	gal	gal	////////////	1	gal
9. Tavg	AUCT_TAV_A	**** TR-3-408	°F	°F	°F	**	gal

* CF = Conversion Factor

** Obtain CF for Tavg from Enclosure 1

**** **IF** Tavg ≤ 540°F, use TR-3-410, TR-3-413, **AND** average the temperatures $\frac{(T_C + T_H)}{2}$ of the operating loops.

RCS LEAK RATE CALCULATION

10. ________ + ________ - ________ - ________ - ________ = ________ gal
ΔVCT Lvl Line 2; ΔPZR Lvl Line 3; Primary Water Line 4; Boric Acid Line 5; ΔTavg Line 9; ΔTotal gal

11. (-) ________ ÷ ________ = ________ gpm
ΔTotal gal Line 10; ΔTime Line 1; Gross RCS Leak Rate (Note 1)

12. **IF** RCS Gross Leak Rate is greater than 0.1 gpm, **THEN** obtain primary to secondary leak rate from Chemistry and record: ________________ gpm.

REFERENCE LEAKAGE (For Information)

13. (________ + ________) ÷ ________ = ________ gpm
ΔPRT Level Line 6; ΔRCDT Level Line 7; ΔTime Line 1; Identified Leakage

14. ________ – ________ – ________ – ________ = ________ gpm
Gross RCS Leak Rate Line 11; Identified Leakage Line 12; (Note 2); (Note 3); Unidentified Leakage

Note 1- **IF** the RCS gross leak rate is greater than 1 gpm, **THEN** immediately notify the Shift Manager.

Note 2 - This is the combined Charging Pump primary leakage in gpm rounded to two decimal places. This value is determined using Attachment 4. However, the value from 3-OP-047, CVCS-Charging and Letdown, Subsection 7.14, Determination of Charging Pump Primary and Secondary Packing Leakage, may be used if it is performed during the same time frame as the Leak Rate Calculation.

Note 3 - This is the sum of all measured Non-RCPB leakage (other than Charging Pump packing leakage) in gpm rounded to two decimal places. This value is determined using Attachment 6.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.6.2 Operational Leakage

10 CFR 50, Appendix A, GDC (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

Applicable Safety Analyses

The primary-to-secondary leakage safety analysis assumption for individual events varies. The assumption varies depending on whether the primary-to-secondary leakage from a single steam generator (SG) can adversely affect the dose consequences for the event. In which case, the affected SG is assumed to have the maximum allowable leakage (500 gallons per day). Collectively, however, the safety analyses for events resulting in steam discharge to the atmosphere assume that primary-to-secondary leakage from all steam generators (SGs) is 1 gpm total and 500 gallons per day through any one SG accident conditions or increases to these levels as a result of accident conditions. The LCO requirement to limit primary-to-secondary leakage through any one SG to less than or equal to 150 gpd at room temperature is significantly less than the conditions assumed in the safety analysis.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a locked rotor accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a SG tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released to the atmosphere via the atmospheric dump valves and/or main steam safety valves for a limited period of time. Operator action is taken to isolate the affected SG within the time period. The 500 gallons per day primary-to-secondary leakage in each of the two intact SGs at accident conditions in the safety analysis assumption is relatively inconsequential.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.6.2 (Cont'd)

Accidents for which the radiation dose release path is primary-to-secondary leakage, the locked rotor accident is more limiting for site radiation dose releases. The safety analysis for the locked rotor accident assumes that primary-to-secondary leakage from all SGs is 1 gpm total. The dose consequences resulting from the locked rotor accident are well within the limits defined in 10 CFR 100 or the NRC approved licensing basis (i.e., a small fraction of these limits).

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

RCS operational leakage shall be limited to:

a. Pressure Boundary Leakage

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. Unidentified Leakage

One gallon per minute (gpm) of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Identified Leakage

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the RCS Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leak-off (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.6.2 (Cont'd)

d. Primary-to-Secondary Leakage Through Any One SG

The limit of 150 gpd per SG at room temperature is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day. The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of SG tube ruptures.

e. RCS Pressure Isolation Valve Leakage

RCS pressure isolation valve leakage is IDENTIFIED LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The specified leakage limits for the RCS pressure isolation valves are sufficiently low to ensure early detection of possible in-series check valve failure.

Applicability

In MODES 1, 2, 3, and 4, the potential for reactor coolant PRESSURE BOUNDARY LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

ACTIONS

a. If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. The reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

b. UNIDENTIFIED LEAKAGE or IDENTIFIED LEAKAGE in excess of the LCO limits must be reduced to within the limits within 4 hours. This allowable outage time allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This ACTION is necessary to prevent further deterioration of the RCPB.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.6.2 (Cont'd)

c. The leakage from any RCS Pressure Isolation Valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. With one or more RCS Pressure Isolation Valves with leakage greater than that allowed by Specification 3.4.6.2.e, within 4 hours, at least two valves in each high pressure line having a non-functional valve must be closed and remain closed to isolate the affected lines. In addition, the ACTION statement for the affected system must be followed and the leakage from the remaining Pressure Isolation Valves in each high pressure line having a valve not meeting the criteria of Table 3.4-1 shall be recorded daily. If these requirements are not met, the reactor must be brought to at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more RCS Pressure Isolation Valves with leakage greater than 5 gpm, the leakage must be reduced to below 5 gpm within 1 hour or the reactor must be brought to at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

The allowable outage times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillance Requirements

SR 4.4.6.2.1

Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of a Reactor Coolant System water inventory balance.

a.&
b. These SRs demonstrate that the RCS operational leakage is within the LCO limits by monitoring the containment atmosphere gaseous or particulate radioactivity monitor and the containment sump level at least once per 12 hours.

QUESTION 4

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011	EA2.10
	Importance Rating	4.5	

Ability to determine or interpret the following as they apply to a Large Break LOCA: Verification of adequate core cooling

Proposed Question: RO Question # 4

Given the following conditions:

- A LOCA has occurred on Unit 3.
- The crew has entered 3-EOP-E-0, Reactor Trip or Safety Injection.
- Immediate Operator Actions steps are complete.
- RCS Tavg is 535°F and trending down.
- RCS Subcooling on QSPDS is 10°F and lowering.
- Containment Temperature is 240°F and rising slowly.
- Containment Pressure is 15 psig and rising slowly.
- All CSF Status Trees are YELLOW or GREEN.

While reviewing the Foldout Page, the Reactor Operator verifies NO HHSI pumps are running, and NONE can be started.

Which ONE of the following describes the required operation of the RCPs and the reason for the action?

A. ALL RCPs remain in operation to ensure core cooling.

B. ONE RCP must be tripped immediately to save for future use in the Functional Recovery Procedures.

C. ALL BUT ONE RCP must be tripped immediately to prevent core uncovery and an inadequate core cooling condition due to the mass being pumped out of the RCS break.

D. ALL RCPs must be tripped immediately because the two-phase flow is creating an artificially high vessel level indication and core uncovery will eventually occur if RCPs are left running.

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Per 3-EOP-E-0, Step 12 (Check if RCPs Should be Stopped), if RCS Subcooling is LESS THAN 25°F and there is no HHSI flow, the operator transitions to the RNO column, which directs them to Go to Step 13, bypassing the step for stopping RCPs.

B. Incorrect because ALL RCPs must remain running since RCP trip criteria has not been met with no SI pumps available. Plausible because this is an action that may be performed in 3-EOP-FR-C.2.

C. Incorrect because RCP trip criteria has not been met with no SI pumps available. Plausible because there are other steps in 3-EOP-E-0 where various combinations of RCPs may be tripped (such as Step 9, where *affected* RCPs must be tripped - and Step 11, where RCPs are tripped as necessary to stop spray flow)

D. Incorrect because RCP trip criteria has not been met with no SI pumps available. Plausible because ALL RCPs are tripped in other places in 3-EOP-E-0 (for example, Step 6).

Technical Reference(s): 3-EOP-E-0, *Reactor Trip Or Safety Injection* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 69-321, Obj. 6 (As available)

Question Source: Bank # WTSI 19102

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2002 BVPS-2

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that, during a LOCA, it requires the operator to determine that core cooling is not adequate because subcooling indicates 0°F and no HHSI pumps are running.

FOLDOUT FOR PROCEDURE E-0

1. **ADVERSE CONTAINMENT CONDITIONS**

   IF either of the conditions listed below occur, **THEN** use adverse containment setpoints:

   Containment atmosphere temperature ≥ 180°F

   OR

   Containment radiation levels ≥ 1.3×10^5 R/hr

   WHEN containment parameters drop below the above values, **THEN** normal setpoints can again be used **IF** the TSC determines that containment integrated dose rate has not exceeded 10^6 Rads.

2. **RCP TRIP CRITERIA**

   a. **IF** both conditions listed below occur, **THEN** trip all RCPs:
      1) High-head SI pumps - AT LEAST ONE RUNNING **AND** SI FLOWPATH VERIFIED.
      2) RCS subcooling - LESS THAN 25°F[65°F]

   b. **IF** phase B actuated, **THEN** trip all RCPs.

3. **FAULTED S/G ISOLATION CRITERIA**

   IF any S/G pressure decreasing in an uncontrolled manner **OR** any S/G completely depressurized, **THEN** the following may be performed:

   a. Maintain total feedwater flow greater than 345 gpm until narrow range level in at least one S/G is greater than 6%[32%].
   b. Isolate AFW flow to faulted S/G(s).
   c. Stabilize RCS hot leg temperature using steam dumps when faulted S/G has blown down to less than 10% wide range.

4. **RUPTURED S/G ISOLATION CRITERIA**

   IF any S/G level increases in an uncontrolled manner **OR** any S/G has abnormal radiation, **AND** narrow range level in affected S/G(s) is greater than 6%[32%], **THEN** feed flow may be stopped to affected S/G(s).

5. **AFW SYSTEM OPERATION CRITERIA**

   a. **IF** two AFW pumps are operating on a single train, **THEN** one of the pumps shall be shut down within one hour of the initial start signal
   b. **IF** two AFW trains are operating and one of the AFW pumps has been operating at low flow of 60 gpm or less for one hour, **THEN** that AFW pump shall be shut down

6. **CST MAKEUP WATER CRITERIA**

   IF CST level decreases to less than 10%, **THEN** add makeup to CST using 3-NOP-018.01, CONDENSATE STORAGE TANK (CST).

7. **RHR SYSTEM OPERATION CRITERIA**

   IF RHR flow is less than 1000 gpm, **THEN** the RHR pumps shall be shut down within 44 minutes of the initial start signal.

BASIS DOCUMENT

WOG Procedure Step 22 **PTN Procedure Step** 12

Check If RCPs Should Be Stopped

BASIS:

Tripping of the Reactor Coolant Pumps during accident conditions prevents excessive depletion of Reactor Coolant System water inventory through a break in the Reactor Coolant System. See WOG Executive Volume, Generic Issues, for additional information on RCP trip criteria. If the RCPs are to remain running, normal containment coolers must be restarted to cool the RCP motors. Because offsite power must be available if RCPs are running, emergency diesel capacity does not need to be checked prior to starting normal containment coolers.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

9 If RCPs are already off due to loss of offsite power or due to operator action directed by the foldout page, unnecessary delays are created by checking RCP trip criteria. Substep 12a was added to skip the remainder of the step if RCPs are already off. This change was made in response to an EOP validation comment.

8 The words "RCP trip parameter - LESS THAN (12) [(13) FOR ADVERSE CONTAINMENT]" were replaced with the plant specific RCP trip criteria, which is RCS subcooling.

8 Verification of a HHSI flowpath was added to this step to match the criteria for stopping RCPs at PTN. This is consistent with the RCP trip criteria found on the Foldout Page.

PLANT SPECIFIC SETPOINTS:

25°F RCP trip criteria of 0°F subcooling based on Loop RTDs and RCS wide range pressure plus normal channel accuracy. (EOP Setpoint W.4)

65°F RCP trip criteria of 0°F subcooling based on Loop RTDs and RCS wide range pressure plus normal channel accuracy and post accident transmitter errors. (EOP Setpoint W.5)

FOP STEP: 1 WOG STEP: FOP 1

Changed criteria for checking "SI pumps - AT LEAST ONE RUNNING" by adding "AND CAPABLE OF DELIVERING FLOW" based on discussion in Section 2.3, RCP Trip Criteria, of the document RCP TRIP/RESTART in the Generic Issues section of the ERG Executive Volume which states the following:

- It cannot be emphasized too strongly that a fundamental condition which must be satisfied for RCP trip during an emergency condition is that at least one high pressure SI pump be in operation and capable of delivering flow to the RCS. If this fundamental condition is not met, the RCPs should not be tripped regardless of whether or not the plant parameters indicate that a trip setpoint has been reached. Analysis has shown that if the SI system is not in operation, the RCPs can be operated to provide core heat removal. As discussed in WCAP 9753 (Reference 4), for SBLOCAs with the high head safety injection (HHSI) pumps not in operation, the RCPs continue to provide core heat removal via the break and the S/Gs. With the RCPs running, the RCS can safely be depressurized to the point where the accumulators and the low head safety injection (LHSI) pumps can ensure core heat removal before symptoms of Inadequate Core Cooling (ICC) are exhibited.

REFERENCES:

Rev 2 DW-04-007 change to Executive Volume FOLDOUT PAGE section

Plant Specific Setpoints:

- 19°F - RCP trip criteria [Normal Containment] (EOP setpoint W.04)
- 41°F - RCP trip criteria [Adverse Containment] (EOP setpoint W.05)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	**Check PRZ PORVs, Spray Valves And Excess Letdown Isolated**	
	a. PORVs – CLOSED	a. **IF** PRZ pressure less than 2335 psig, **THEN** manually close PORVs. **IF** any PRZ PORV can **NOT** be closed, **THEN** manually close its block valve. **IF** block valve can **NOT** be closed, **THEN** perform the following: 1) Monitor Critical Safety Functions using 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES. 2) Go to 3-EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. Normal PRZ spray valves – CLOSED	b. **IF** PRZ pressure less than 2260 psig, **THEN** manually close valves. **IF** valve(s) can **NOT** be closed, **THEN** stop RCP(s) as necessary to stop spray flow.
	c. Auxiliary Spray Valve, CV-3-311 – CLOSED	c. Manually close auxiliary spray valve. **IF** auxiliary spray valve can **NOT** be closed, **THEN** close Charging Flow to Regen Heat Exchanger, HCV-3-121.
	d. Excess letdown isolation valves – CLOSED • CV-3-387, Excess Letdown Isolation Valve From Cold Leg To Excess Letdown Heat Exchanger • HCV-3-137, Excess Letdown Flow Controller	d. Manually close valve(s).
12	**Check If RCPs Should Be Stopped**	
	a. Check RCPs - ANY RUNNING	a. Go to Step 13.
	b. Check RCS subcooling – LESS THAN 25°F[65°F]	b. Go to Step 13.
	c. High-Head SI Pump – AT LEAST ONE RUNNING **AND** FLOWPATH VERIFIED	c. Go to Step 13.
	d. Stop all RCPs	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	**Check If One RCP Should Be Stopped**	
	a. All RCPs - RUNNING	a. Go to Step 11.
	b. Stop RCP in loop B	
	c. Go to Step 11	
10	**Check Core Cooling**	
	a. Core exit TCs - LESS THAN 700°F	a. Perform one of the following: * **IF** temperature is decreasing, **THEN** observe NOTE prior to Step 1 **AND** return to Step 1. * **IF** temperature is **NOT** decreasing, **THEN** go to Step 11.
	b. Return to procedure **AND** step in effect	
11	**Check SI Accumulator Isolation Valve Status**	
	a. Power to Accumulator Discharge MOVs - AVAILABLE	a. Locally unlock and close the following breakers: • 30532 for MOV-3-865A • 30631 for MOV-3-865B • 30733 for MOV-3-865C
	b. Accumulator Discharge MOVs - OPEN • MOV-3-865A • MOV-3-865B • MOV-3-865C	b. **IF** accumulators are **NOT** discharged, **THEN** open Accumulator Discharge MOVs.

QUESTION 5

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015	AK3.07
	Importance Rating	4.1	

Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) : Ensuring that S/G levels are controlled properly for natural circulation enhancement

Proposed Question: RO Question # 5

Given the following:

- Unit 3 was operating at power when a Loss of Offsite Power occurred.
- The operating crew has implemented 3-EOP-ES-0.1, Reactor Trip Response.
- Stable plant conditions have been established.
- Offsite Power is expected to be restored within 24 hours.

In accordance with 3-EOP-ES-0.1, which one of the following describes (1) the control band used for S/G levels and (2) the reason for that band?

A. (1) BETWEEN 15% and 50%
(2) to preclude AFW re-initiation and establish a heat sink which will enhance natural circulation

B. (1) BETWEEN 32% and 50%
(2) to precluded AFW re-initiation and establish a heat sink which will enhance natural circulation

C. (1) BETWEEN 15% and 50%
(2) to maintain an adequate inventory to start a required natural circulation cooldown

D. (1) BETWEEN 32% and 50%
(2) to maintain an adequate inventory to start a required natural circulation cooldown

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Step 24.c of 3-EOP-ES-0.1 requires SG levels to be maintained 15-50%. BD-EOP-ES-0.1, Page 37, states: "S/G level is maintained in the narrow range with the low end of the control band at the AFW actuation setpoint plus 5% to preclude AFW actuation."

B. Incorrect since the control band is 15-50%, not 32-50%. Plausible because the 2nd part is correct. Also plausible because the control band of 32-50% (adverse value) is the range of operation which prevents RPS actuation.

C. Incorrect since the control band is based on establishing heat sink. Plausible because the 1st part is correct. Also plausible because 3-EOP-ES-0.2 will require a natural circulation cooldown.

D. Incorrect since the control band is based on establishing heat sink, not preventing a main steamline isolation. Also incorrect since the control band is 15-50%, not 32-50%. Plausible because the control band of 32-50% (adverse value) will prevent RPS actuation. Also plausible because 3-EOP-ES-0.2 will require a natural circulation cooldown.

Technical Reference(s): 3-EOP-ES-0.1, Reactor Trip Response; BD- 3-EOP-ES-0.1, Reactor Trip Response (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902323, Obj. 3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests why S/G levels are maintained (establish heat sink to enhance natural circulation) during a loss of RCS flow (RCPs lost due to LOOP)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13	**Maintain S/G Levels**	
	a. Narrow range level – GREATER THAN 6%	a. Maintain total feed flow greater than 345 gpm until narrow range level greater than 6% in at least one S/G.
	b. Control feed flow to maintain narrow range level between 15% and 50%	
	c. Narrow range level - LESS THAN 50%	c. **IF** narrow range level in any S/G continues to increase, **THEN** stop feed to that S/G.
14	**Establish Condenser Steam Dump Control**	**IF** condenser steam dumps are **NOT** available, **THEN** set S/G steam dump to atmosphere valve controllers to maintain desired S/G pressure.
	a. Set condenser steam dump controller to maintain desired S/G pressure	
	b. Place condenser steam dumps in Manual Mode	
	c. Verify condenser - AVAILABLE	
15	**Check RCPs - ALL STOPPED**	Go to Step 20.
16	**Check Plant Conditions For Starting Desired RCP**	Perform the following:
	a. A or B 4KV bus - ENERGIZED FROM STARTUP TRANSFORMER	1. Verify natural circulation using ATTACHMENT 1. **IF** natural circulation can **NOT** be verified, **THEN** increase dumping steam.
	b. Number one seal ΔP – GREATER THAN 200 PSID	2. Go to Step 20.
	c. Thermal barrier ΔP – GREATER THAN 0 INCHES OF WATER	
	d. Verify proper number one seal leak-off flow - GREATER THAN 0.8 GPM	
	e. RCP number one seal leak-off temperature - LESS THAN 225°F	

BASIS DOCUMENT

WOG Procedure Step 6 **PTN Procedure Step** 13

Maintain S/G Levels

BASIS:

The minimum feed flow requirement satisfies the feed flow requirement of the Heat Sink Status Tree until level in at least one S/G is restored into the narrow range. Narrow range level is re-established in all S/Gs to maintain symmetric cooling of the RCS. The control range ensures adequate inventory with level readings on span.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

9 The WOG guideline step description table identifies this as a continuous action step. The generic step uses the words "check S/G levels" which does not remind the operator to return to this step if proper S/G level response is not possible. The high level step was reworded to "Maintain S/G Levels." This will identify the step as a continuous action step.

4 S/G level must be restored in the narrow range to ensure an adequate heat sink. The level should then be raised to preclude AFW actuation. This will allow a shutdown of AFW pumps later in the procedure. PTN Substep b was modified accordingly using the minimum level specified for the E-3 series ERGs which adds 5% to the AFW actuation setpoint.

9 The generic wording of this step is technically correct but relies heavily on operator knowledge of the rules of usage. To prevent operator confusion noted during EOP validation, Substep c was added to provide specific actions for high S/G level. This will clarify the intended actions of the WOG guidelines.

PLANT SPECIFIC SETPOINTS:

6%	S/G level just in the narrow range plus normal channel accuracy. (EOP Setpoint M.2)
345 gpm	Minimum safeguards AFW flow required for heat removal plus normal channel accuracy. This is the capacity of one AFW pump. (EOP Setpoint S.2)
15%	Low-low S/G level AFW actuation setpoint plus 5% operating band. (EOP Setpoint M.10)
50%	S/G level at the middle of the narrow range. (EOP Setpoint M.20)

FOLDOUT FOR PROCEDURE ES-0.1

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs, **THEN** actuate SI, actuate containment isolation phase A, and go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1:
   * RCS subcooling based on core exit TCs - LESS THAN 30°F
     OR
   * PRZ level – CAN **NOT** BE MAINTAINED GREATER THAN 12%

2. **EMERGENCY BORATION CRITERIA**
   IF one of the following conditions exists, **THEN** emergency borate using 3-ONOP-046.1, EMERGENCY BORATION, until termination criteria are met.
   a. Any RCS Cold Leg temperature decreases to less than 525°F
   b. Two or more control rods **NOT** fully inserted.

3. **PRESSURIZER LEVEL CRITERIA**
   IF Pressurizer level is <20%, **THEN** control charging to maintain pressurizer level between 20% and 30%.

4. **S/G LEVEL CRITERIA USING AFW**
   IF AFW was actuated, **THEN** adjust auxiliary feed flow to the S/G to perform the following:
   a. Maintain total feedwater flow greater than 345 gmp until narrow range level in at least one S/G is greater than 6%.
   b. Stop auxiliary feed flow to any S/G with a narrow range level of GREATER THAN 50%.

5. **CST MAKEUP WATER CRITERIA**
   IF CST level decreases to less than 10%, **THEN** add makeup to CST using 3-NOP-018.01, CONDENSATE STORAGE TANK (CST).

6. **RED PATH SUMMARY**
   IF any condition listed below occurs, **THEN** go to 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES, Step 1:
   a. Subcriticality: Nuclear power - GREATER THAN 5%
   b. Core Cooling: Core exit TCs - GREATER THAN 1200°F
   c. Heat Sink: Narrow range level in all S/Gs - LESS THAN 6%
      AND total feedwater flow - LESS THAN 345 GPM
   d. Integrity: Cold leg temperature decrease - GREATER THAN 100°F IN LAST 60 MINUTES **AND** any RCS cold leg temperature - HAS BEEN LESS THAN 290°F
   e. Containment: Containment pressure - GREATER THAN 55 PSIG

ATTACHMENT 1
(Page 1 of 1)

NATURAL CIRCULATION INDICATIONS

The following conditions support or indicate natural circulation flow:

- RCS subcooling based on core exit TCs - GREATER THAN 30°F
- S/G pressures - STABLE OR DECREASING
- RCS hot leg temperatures - STABLE OR DECREASING
- Core exit TCs - STABLE OR DECREASING
- RCS cold leg temperatures - WITHIN 35°F OF SATURATION TEMPERATURE FOR S/G PRESSURE

BASIS DOCUMENT

WOG Procedure Step 16 **PTN Procedure Step** 8

Verify Proper AFW Flow

BASIS:

AFW flow is necessary for secondary heat sink. If S/G level is in the narrow range in at least one S/G, a heat sink is available. Therefore, AFW flow is needed only to maintain level. If adequate feed flow for decay heat removal cannot be established, the transition to the FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, is necessary to establish an alternate source of feed flow or an alternate heat sink.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

9 Due to the addition of plant specific actions, the high level step was reworded to more accurately describe the intent of the step. The WOG wording is "Verify Total AFW Flow".

9 This step was rewritten to first check for level in at least one SG level > 6% instead of AFW flow > 345 gpm. AFW flow may have already been reduced because of a viable heat sink being established a reduction in AFW may have been performed to control plant temperature. If adequate level does not exist, then the operators verify that adequate AFW flow is in service in the RNO step.

4 The words "IF AFW flow" were changed to "IF total feed flow" to prevent an unnecessary transition to FR-H.1 if an alternate source of feed has already been established through the concurrent use of plant off-normal procedures.

8 The WOG guidelines require initiation of Critical Safety Function status tree monitoring whenever exiting E-0. The RNO was modified to provide procedural guidance for performance of this task so that the need to memorize User's Guide requirements is eliminated.

9 Step 8b was added to provide a normal operating control band for the operators once they have verified that they have an acceptable heat sink. This is consistent with operator fundamentals to control the plant precisely and represents that manner in which the operators are trained.

8 The step numbering difference between the WOG procedure and the PTN procedure comes about because of the decision by PTN to incorporate the prompt action verifications attachment, which is allowed by the WOG and discussed in the Step 5 basis. The WOG allowed this step to be moved to the attachment, but operator feedback and the subsequent operator training sessions and the preference for only one operator (the RO or the BOP) adjusting/monitoring AFW so they do not compromise each others actions.

(Continued on next page)

BASIS DOCUMENT

WOG Procedure Step 16 **PTN Procedure Step** 8

(Continued)

PLANT SPECIFIC SETPOINTS:

6%	S/G level just in the narrow range plus normal channel accuracy. (EOP Setpoint M.2)
32%	S/G level just in the narrow range plus normal channel accuracy, post accident transmitter errors, and reference leg process errors. (EOP Setpoint M.3)
15%	S/G level AFW actuation setpoint plus 5%. (EOP Setpoint M.10)
32%	S/G level to prevent AFW auto start including adverse errors. (EOP Setpoint M.11)
50%	S/G narrow range level corresponding to middle of span. (EOP Setpoint M.20)
345 gpm	Minimum safeguards AFW flow required for heat removal plus normal channel accuracy. This is the capacity of one AFW pump. (EOP Setpoint S.2)

QUESTION 6

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022	AK1.01
	Importance Rating	2.8	

Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Consequences of thermal shock to RCP seals

Proposed Question: RO Question # 6

The crew has transitioned from 3-EOP-ECA-0.0, Loss of All AC Power, to 3-EOP-ECA-0.1, Loss of All AC Power Recovery Without SI Required.

The following plant conditions exist:

- Power is restored to all 4160 kV Buses.
- Annunciator A 1/2, RCP THERMAL BARR COOLING WATER HI TEMP, is lit.
- Annunciator A 1/6, RCP #1 SEAL LEAK OFF HI TEMP, is lit.
- Seal Injection Manual Isolation Valves, 3-297A, 3-297B, and 3-297C are closed.
- RCP Thermal Barrier CCW Outlet Valve, MOV-3-626, is closed.
- Common Seal Water Return Temperature to CVCS is 250°F
- No. 1 Seal Water Outlet Temperature for all RCPs is 250°F

Which ONE of the following describes (1) the required action(s) and (2) the result of these action(s) on RCP Seal Cooling in accordance with 3-EOP-ECA-0.1?

A. (1) RCP Seal Injection Manual Valves and Thermal Barrier Cooling Outlet Valves must remain CLOSED.
(2) This isolation prevents thermal shock to RCP seals and potential failure.

B. (1) RCP Seal Injection Manual Valves and Thermal Barrier Cooling Outlet Valves are SLOWLY OPENED in a controlled manner.
(2) This controlled re-initiation of flow prevents thermal shock to RCP seals and potential failure.

C. (1) RCP Seal Injection Manual Valves and Thermal Barrier Cooling Outlet Valves must remain CLOSED.
(2) This isolation ensures RCP Seal integrity during a 100 °F/hr RCS cooldown in

ES- 0.2

accordance with EOP-ECA-0.1.

D. (1) RCP Seal Injection Manual Valves and Thermal Barrier Cooling Outlet Valves are SLOWLY OPENED in a controlled manner.
(2) This controlled re-initiation of flow ensures RCP Seal integrity during a 100 °F/hr RCS cooldown in accordance with EOP-ECA-0.1.

ES - 0.2

Proposed Answer: A

Explanation (Optional):

A. Correct. With charging pumps off and valves already closed, there will be no attempt to restore RCP seal cooling at step 1 of ECA-0.1

B. Incorrect. Because it is already isolated, it will not be unisolated. Plausible because it is logical to believe that a slow introduction of cooling flow would be prudent, and the second part is correct

C. Incorrect. Plausible because first part is correct, but second part is an incorrect statement because seal integrity is not ensured following a loss of AC power, whether cooling down the RCS or not. Logical because cooldown is typically associated with thermal shock

D. Incorrect. Because it is already isolated, it will not be unisolated. Plausible because it is logical to believe that a slow introduction of cooling flow would be prudent, and the second part is logical because cooldown is typically associated with thermal shock

Technical Reference(s): 3-EOP-ECA-0.1 Step 1 and RNO (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 6902349 Obj 3,8 (As available)

Question Source: Bank # WTSI 62337

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2005 Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Turkey Point 2005. Not in last 3 exams

Changed wording of stem, all distractors and changed distractor analysis. Could be considered new

Meets KA because loss of RCS makeup is initiated with loss of AC Power. Operational implication of thermal shock to RCP seals must be evaluated in conjunction with a loss of CCW, and procedure action for this event is an implication.

BASIS DOCUMENT

WOG Procedure Step 8 **PTN Procedure Step** 13

Locally Close Valves To Isolate RCP Seals

BASIS:

This step is aimed at isolating the RCP seals. These actions all require an auxiliary operator, dispatched from the Control Room, to locally close containment isolation valves. Turkey Point plant utilizes motor operated valves for the RCP seal return and RCP thermal barrier CCW return lines and manual valves for the RCP seal injection lines.

Isolating the RCP seal injection lines prepares the plant for recovery while protecting the RCPs from seal and shaft damage that may occur when a charging pump is started as part of the recovery. With the RCP seal injection lines isolated, a charging pump can be started in the normal charging mode without concern for cold seal injection flow thermally shocking the RCPs. Seal injection can subsequently be established to the RCP consistent with the appropriate procedure.

Isolating the RCP thermal barrier CCW return outside containment isolation valve prepares the plant for recovery while protecting the CCW system from steam formation due to RCP thermal barrier heating. Following the loss of all ac power, hot reactor coolant will gradually replace the normally cool seal injection water in the RCP seal area. As the hot reactor coolant leaks up the shaft, the water in the thermal barrier will heat up and potentially form steam in the thermal barrier and in the CCW lines adjacent to the thermal barrier. If abnormal RCP seal leakage had developed in a pump, the abnormally high leakage rate could exceed the cooling capacity of the CCW flow to that pump thermal barrier and tend to generate more steam in the RCP thermal barrier CCW return lines. Isolating these lines prevents the potential introduction of this steam into the main portion of the CCW system upon CCW pump start.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

2 The step was rewritten to only perform the local isolation of the seal cooling lines. The actions to place the appropriate control switches in the closed position are not required at Turkey Point. The control switches for the MOVs are spring return to auto.

PLANT SPECIFIC SETPOINTS:

N/A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If an SI signal is actuated prior to Step 11 of this procedure, it needs to be reset to permit manual loading of equipment on an ac bus.

NOTE

CSF status trees are required to be monitored for information only. FRPs shall NOT be implemented prior to completion of Step 11.

1 **Check RCP Seal Isolation Status**

a. Check RCP seal injection manual isolation valves - CLOSED

- 3-297A
- 3-297B
- 3-297C

a. **IF** valves open **OR** position **NOT** known, **THEN** check charging pump status:

1) **IF** charging pump running, **THEN** go to Step 2.

2) **IF** charging pump **NOT** running, **THEN** locally close the following RCP seal injection isolation valves before starting charging pump:

- 3-297A
- 3-297B
- 3-297C

b. Check RCP Thermal Barrier CCW Outlet, MOV-3-626 - CLOSED

b. **IF** valve open **OR** position **NOT** known, **THEN** check CCW pump status:

1) **IF** CCW pump running, **THEN** go to Step 2.

2) **IF** CCW pump **NOT** running, **THEN** manually or locally close MOV-3-626. **IF** MOV-3-626 can **NOT** be closed, **THEN** locally close CCW return manual isolation valve outside containment, 3-736.

2 **Check Containment Isolation Phase A - NOT ACTUATED**

Reset containment isolation phase A and phase B.

BASIS DOCUMENT

WOG Procedure Step 1 **PTN Procedure Step** 1

Check RCP Seal Isolation Status

BASIS:

The operator should check the status of RCP seal isolation in preparation for loading the charging pump on the 4KV bus. If the RCP has been isolated, the operator proceeds to the next step to verify and load equipment as appropriate.

If the operator determines that the seal injection isolation valves are open or their position is not known, the operator should check charging pump status. If the pumps are running, the operator is instructed to go to the next step to continue plant recovery. In this situation, the charging pumps have automatically started upon ac power restoration and have already initiated seal injection flow. The RCPs have already been subjected to the potential thermal shock. If the charging pumps are not running, the operator should attempt to isolate the seal injection line by remotely closing any other plant specific valves, if available, or by dispatching personnel to locally close the seal injection isolation valves. The operator should not start a charging pump until seal injection is isolated to avoid potential RCP damage.

If the operator determines that the thermal barrier CCW return outside containment isolation valve is open, the operator should check CCW pump status. If a pump is running, the operator is instructed to go to the next step to continue plant recovery. In this situation, the CCW pump has already started upon ac power restoration and thermal barrier flow has been initiated. The operator should be aware of potential CCW operational problems. If the CCW pumps are not running, the operator is instructed to manually close the outside containment isolation valve. If the valve cannot be remotely closed, the operator should close some other valve in the thermal barrier line.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

8 The words "isolation valves outside containment" were changed to "manual isolation valves" to conform with plant specific terminology.

9 The word "pump" was changed to "charging pump" to clarify which pump is being discussed.

(Continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12	**Evaluate Plant Conditions And Continue Plant Operations**	
	a. Direct the Plant Manager to review • 0-ADM-115, NOTIFICATION OF PLANT EVENTS, for applicability • Applicable parts of Technical Specification 3.4.1 for compliance • 0-EPIP-20101, DUTIES OF EMERGENCY COORDINATOR, for applicability	
	b. Check Annunciator A 1/4, RCP SEAL LEAK-OFF LO FLOW - ALARM OFF	b. Go to Step 2.
	c. Continue monitoring pertinent plant parameters affecting any RCP abnormalities addressed in this procedure	
	d. Go to appropriate plant procedure as determined by the Shift Manager	
13	**Check Proper Seal Injection Temperature**	
	a. Check all number one seal leakoff temperatures on DCS - LESS THAN 170°F	a. Restore seal injection flow using ATTACHMENT 1 while continuing with procedure by observing NOTE prior to Step 15 **AND** going to Step 15.

ATTACHMENT 1
(Page 1 of 2)

RESTORATION OF RCP SEAL INJECTION

CAUTION

Do NOT use this attachment if seal return temperature has exceeded 235°F. If seal return temperature has exceeded 235°F, then Step 22 RNO in the body of the procedure should be performed.

1. Establish Seal Injection Flow

a. Locally close **affected** RCP Seal Injection Throttle Valve(s)

* 3-297A for RCP A
* 3-297B for RCP B
* 3-297C for RCP C

b. Verify Charging Flow to Regen Heat Exchanger, HCV-3-121 – FULL OPEN

c. Verify charging pumps - AT LEAST ONE RUNNING

d. Verify operable seal injection filter in service using 3-OP-047, CVCS-CHARGING AND LETDOWN

d. **WHEN** operable seal injection filter in service, **THEN** go to Step 1.e of this attachment. Continue with procedure and step in effect.

e. Locally throttle open **affected** RCP Seal Injection Throttle Valve to establish flow of 2 gpm to affected RCP

* 3-297A, FI-3-130 for RCP A
* 3-297B, FI-3-127 for RCP B
* 3-297C, FI-3-124 for RCP C

f. Maintain Annunciator A 6/6, SEAL WATER INJ FILTER HI ΔP, alarm - CLEAR

f. Go to Step 1.i of this attachment.

g. Maintain affected RCP lower bearing cooldown rate - LESS THAN OR EQUAL TO 1°F/MINUTE

g. Adjust RCP Seal Injection Manual Isolation valve to affected RCP(s) to establish pump bearing cooldown rate less than or equal to 1°F/minute.

ATTACHMENT 1
(Page 2 of 2)

RESTORATION OF RCP SEAL INJECTION

1 (Cont'd). Establish Seal Injection Flow (Cont'd)

h. Go to Step 2 of this attachment

i. **IF** standby seal injection filter available for use, **THEN** place it in service using 3-OP-047, CVCS-CHARGING AND LETDOWN

i. Perform the following:

1) Record time seal injection lost:

____

2) **WHEN** time since seal injection lost equals 18 hours, **THEN** perform 3-GOP-103, POWER OPERATION TO HOT STANDBY, to shut down Unit 3 **AND** stop affected RCP within 24 hours of the above recorded time in this step.

3) **WHEN** seal injection filter available for use, **THEN** return to Step 1.a of this attachment.

4) Return to procedural step in effect.

2. Reestablish Normal Seal Injection Flow

a. Check the affected RCP number one seal leakoff temperature(s) on DCS - LESS THAN 170°F

a. **WHEN** RCP number one seal leakoff temperatures are less than 170°F, **THEN** go to Step 2.b of this attachment. Continue with procedural step in effect.

b. Establish between 6 and 13 gpm seal injection to each RCP

3. Return To Procedural Step In Effect

QUESTION 7

Facility: Turkey Point
Vendor: WEC
Exam Date: 2011
Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	025	AA1.02
Importance Rating	3.8	

Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: RCS inventory

Proposed Question: RO Question # 7

Given the following initial conditions:

- Unit 3 is cooling down using 'A' RHR Train.
- RCS temperature is 310°F.
- RCS pressure is 340 psig.
- PZR level is currently 10%.

When:

- PZR level starts lowering at a greater rate.
- Charging is at maximum flow.
- Letdown is isolated.
- Containment radiation levels are rising.
- The running RHR pump trips.

Which ONE of the following mitigation strategies will be used FIRST to raise PZR level?

A. Actuate Safety Injection

B. Manually align Safety Injection Pumps to RCS Cold Legs

C. Locally unlock and close Unit 3 SI Accumulator breakers for valve operation

D. Manually align the non-operating train of RHR for Injection

Proposed Answer: B

Explanation (Optional):

A. Incorrect because SI actuation at this point could cause overpressurization of the RCS. Plausible because if the plant was in Mode 3 SI would be actuated

B. CORRECT. Per 3-ONOP-041.7, first action is to raise charging and isolate letdown, which has been performed. Second action is to align one train of HPSI to cold legs.

C. Incorrect but plausible because this action is performed in 3-EOP-E-1 for valve operation.

D. Incorrect but plausible because this check is made after aligning SI to cold legs. If not aligned for injection, RHR is realigned.

Technical Reference(s): 3-ONOP-041.7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902210, Obj. 4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

The K/A is matched in that, during implementation of 3-ONOP-041.7, Shutdown LOCA, Mode 3 less than 1000 psig or Mode 4, the operator must determine appropriate actions, given a lowering RCS inventory.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-041.7

Title:

Shutdown LOCA [Mode 3 (Less than 1000 PSIG) or Mode 4]

Safety Related Procedure	
Responsible Department:	Operations
Revision Approval Date:	10/21/04

RTSs *93-1422P, 94-0349P, 94-1102P, 95-0956P, 96-0982P, 96-1180P, 96-1463P, 96-1534P, 00-0485P, 01-0215P, 01-0417P, 02-0016P, 02–0145P, 04-0887P*
PCMs *96-012, 96-081, 00-027*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *Foldout page shall be monitored throughout this procedure.*
- *RCS inventory should be controlled using the level instrument(s) in use for existing plant conditions prior to the event.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Monitor Conditions To Determine If RHR Pumps Should be Stopped:**	
	a. Check the following: * PZR level - LESS THAN 12% [50%] **OR** * RCS subcooling based on core exit TCs - LESS THAN 30°F [210°F]	a. **IF** neither condition satisfied **THEN** Go to STEP 2
	b. Stop RHR pumps and place them in PULL TO LOCK	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	**Isolate RCS letdown**	
	a. Excess letdown isolation valves- CLOSED • CV-3-387, Excess Letdown Isolation Valve From Cold Leg To Excess Letdown Heat Exchanger • HCV-3-137, Excess Letdown Flow Controller	a. Manually close valves.
	b. Normal Letdown isolation valves - CLOSED • CV-3-200A, 45 gpm LTDN Isolation • CV-3-200B, 60 gpm LTDN Isolation • CV-3-200C, 60 gpm LTDN Isolation • LCV-3-460, High Pressure Letdown Isolation From Loop B Cold Leg	b. Manually close valves.
	c. RHR letdown Isolation Valves - CLOSED • HCV-3-142, RHR LTDN to CVCS	c. Manually close valve.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Check If Charging Flow Is Adequate**	Go to Step 4

a. Verify RCS Makeup Control Switch in Stop

b. Adjust charging flow as necessary to maintain PZR level

c. Check PZR level

- GREATER THAN 12% [50%]
- STABLE **OR** INCREASING

d. RCS subcooling based on core exit TCs - GREATER THAN 30°F [210°]

e. Charging flow - ADEQUATE

- FI-3-122A -LESS THAN 140 GPM
- Check PZR level
  1) GREATER THAN 12% [50%]
  2) STABLE **OR** INCREASING

f. Go to appropriate plant procedure as determined by the Shift Manager.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

4 **Dispatch Personnel To Locally Restore Power To Locked Out SI Equipment As Follows:**

a. Verify the following breakers CLOSED

- 30622 for MOV-3-843B
- 30621 for MOV-3-866B
- 30605 for MOV-3-864B
- 30615 for MOV-3-750
- 30616 for MOV-3-862B
- 30626 for MOV-3-863B

b. Verify the following breakers CLOSED

- 30738 for MOV-3-843A
- 30737 for MOV-3-869
- 30712 for MOV-3-864A
- 30720 for MOV-3-862A
- 30726 for MOV-3-863A
- 30731 for MOV-3-751
- 30732 for MOV-3-866A

c. Verify the following breakers RACKED IN:

- 3AA13 for 3A HHSI PUMP
- 3AB12 for 3B HHSI PUMP
- 4AA13 for 4A HHSI PUMP
- 4AB12 for 4B HHSI PUMP

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Evacuate Non-essential Personnel In Containment**	
	a. Announce over the plant PA system • **Attention all personnel inside Unit 3 Containment: Evacuate Unit 3 Containment**	a. Request Shift Manager pass supervisory announcement over MTX-900 radio to order personnel out of containment.
	b. Sound the containment evacuation alarm	b. Notify Health Physics Shift Supervisor **OR** Operations Department personnel inside containment to order all personnel to evacuate the containment building.
	c. Announce over the plant PA system • **Attention all personnel inside Unit 3 Containment: Evacuate Unit 3 Containment**	c. Request Shift Manager pass supervisory announcement over MTX-900 radio to order personnel out of containment.
6	**Actuate Containment Isolation Phase A**	
	a. Manually actuate containment isolation phase A	
	b. Containment isolation phase A valve white lights on VPB – ALL BRIGHT	b. **IF** any containment isolation phase A valve is **NOT** closed, **THEN** manually close valve. **IF** valve(s) can **NOT** be manually closed, **THEN** manually or locally isolate affected containment penetration.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	**Monitor Conditions To Determine If RCPs Must Be Stopped**	
	a. Check RCPs - ANY RUNNING	a. Go to Step 8.
	b. Check the following * Number one seal differential pressure - LESS THAN 200 PSID **<u>OR</u>** * Number one seal leakoff flow - LESS THAN 0.8 GPM	b. **<u>IF</u>** neither condition satisfied, **<u>THEN</u>** go to Step 8.
	c. Stop affected RCP(s)	

CAUTION

If RWST level decreases to less than 155,000 gallons, the SI System should be aligned for Cold Leg Recirculation using Attachment 2.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	**Check If One HHSI Pump Should Be Started**	
	a. Check the following * PZR level - LESS THAN 12% [50%] **<u>OR</u>** * RCS Subcooling based on core exit TCs - LESS THAN 30°F[210°F]	a. **<u>IF</u>** neither condition satisfied, **<u>THEN</u>** go to Step 12
	b. Align One Train Of Safety Injection * SI To Cold Leg Isol Valve, MOV–3–843A - OPEN **<u>OR</u>** * SI To Cold Leg Isol Valve, MOV–3–843B - OPEN	
	c. Start One High-head SI Pump	
	d. Verify HHSI flow on FI-3-943	d. Manually start pumps and align valves

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

If the RHR system has been isolated due to excessive leakage, then observe NOTE prior to Step 11 and go to Step 11.

9 **Check RHR Pump Aligned For Injection**

a. RHR Valves aligned for injection prior to event

- MOV-3-862A, RHR Suction from RWST – OPEN
- MOV-3-862B, RHR Suction from RWST – OPEN

b. RHR pump suction from RCS - CLOSED

- MOV-3-750, Loop 3C RHR Pump Suction Stop Valve
- MOV-3-751, Loop 3C RHR Pump Suction Stop Valve

RESPONSE NOT OBTAINED:

Continue with Step 11 while performing the following:

1. Realign RHR suction to RWST as follows:

   * <u>**IF**</u> RCS hot leg temperature less than 245°F, <u>**THEN**</u> align RHR for Injection as follows:

     a) Stop RHR pumps and place them in standby

     b) Close both Loop 3C RHR Suction Stop Valves
        - MOV-3-750
        - MOV-3-751

     c) Isolate RHR Flow to CVCS
        - HCV-3-142
        - Locally close 3-205B

     d) Isolate RHR Heat Exchanger Bypass Line:
        - Close FCV-3-605, RHR HX BYPASS FLOW VALVE
        - Locally Close 3-757D, RHR HX A Bypass Hdr Isolation
        - Locally Close 3-757C, RHR HX B Bypass Hdr Isolation

     e) Open HCV-3-758, RHR Heat Exchanger Outlet Flow Valve

     f) Open the following valves:
        - MOV-3-862A, RHR SUCTION FROM RWST
        - MOV-3-862B, RHR SUCTION FROM RWST

   <u>**OR**</u>

   * <u>**IF**</u> RCS Hot Leg temperature GREATER THAN 245°F, <u>**THEN**</u> cooldown the RHR suction piping and align RHR for injection using ATTACHMENT 3

2. <u>**WHEN**</u> RHR suction aligned to RWST, <u>**THEN**</u> perform Step 10.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-041.8

Title:

Shutdown LOCA [Mode 5 or 6]

(Continuous Use)

Safety Related Procedure

Responsible Department:	Operations
Revision Number:	1
Issue Date:	9/26/10
Revision Approval Date:	9/23/10

***PCRs** 08-5660, 10-0480*

W97:DH/mrg/cls/cls

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***Changes in RCS pressure may result in inaccuracies in RCS level readings.***
- ***If the refueling Cavity is flooded, then go to 3-ONOP-033.2, REFUELING CAVITY SEAL FAILURE.***
- ***If entering this procedure from 3-ONOP-050, LOSS OF RHR, then go to Step 21.***

1 Check If RHR Pumps Should Be Stopped

Action/Expected Response	Response Not Obtained
a. RHR pumps - ANY RUNNING	a. Go to Step 2.
b. RCS LEVEL - ADEQUATE FOR PLANT CONDITIONS • Drain Down Level 1) LI-3-6421 - GREATER THAN 23% 2) LI-3-6423 - GREATER THAN 23% **OR** • Pressurizer Level, LI-3-462 - GREATER THAN 10%	b. Perform the following: 1) Stop both RHR pumps **AND** place them in standby. 2) Go to Step 2.
c. Check RCS Level - STABLE **OR** INCREASING	c. Perform the following: 1) Maintain RCS inventory using the following methods while continuing with this procedure: a) Charging flow (Step 13). b) RWST Gravity Feed (Step 14). c) VCT Overpressure Feed (Step 15).
d. RHR flow - LESS THAN 3000 GPM	d. Reduce RHR flow to 3000 gpm.
e. RHR pumps - CAVITATING	e. Perform the following: 1) **IF** level stable or increasing, **THEN** go to appropriate plant procedure as determined by the Shift Manager. 2) **IF** level decreasing, **THEN** go to Step 1f.
f. Stop both RHR pumps and place them in standby.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	**Isolate Letdown And Known Drain Paths**	
	a. Excess letdown isolation valves – CLOSED	
	1) HCV-3-137, Excess Letdown Flow Controller	1) Close HCV-3-137.
	2) CV-3-387, Excess Letdown Isolation Valve From Cold Leg To Excess Letdown Heat Exchanger	2) **WHEN** HCV-3-137 has been closed for 12 hours, **THEN** close CV-3-387.
	b. Normal Letdown isolation valves - CLOSED • CV-3-200A, LTDN Orifice Stop Valve - 45 gpm • CV-3-200B, LTDN Orifice Stop Valve - 60 gpm • CV-3-200C, LTDN Orifice Stop Valve - 60 gpm • LCV-3-460, High Pressure Letdown Isolation From Loop B Cold Leg	b. Manually close valves.
	c. RHR letdown Isolation Valves - CLOSED • HCV-3-142, RHR LTDN to CVCS	c. Manually close valve.
3	**Check RCS Operating In A Drain Down OR Reduced Inventory Configuration Prior To The Event**	Go to Step 5.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***Personnel working in containment should be warned prior to refilling the RCS to avoid inadvertent contamination of personnel working near RCS openings.***
- ***The Shift Manager SHALL evaluate the necessity of evacuating non-essential personnel inside containment.***
- ***Only borated water should be added to the RCS to maintain adequate shutdown margin.***

10 **Check RCS Level**

a. RCS level

* Drain Down Level

* LI-3-6421 - LESS THAN **OR** EQUAL TO 23%

OR

* LI-3-6423 - LESS THAN **OR** EQUAL TO 23%

OR

* Pressurizer Level, LI-3-462- LESS THAN **OR** EQUAL TO 50%

a. Perform the following:

1) Maintain RCS inventory using the following method(s) while continuing with this procedure:

a) Charging flow (Step 13).

b) RWST Gravity Feed (Step 14).

c) VCT Overpressure Feed (Step 15).

2) Go to Step 16.

b. RCS level

* Drain Down Level

* LI-3-6421 LESS THAN **OR** EQUAL TO 14%

OR

* LI-3-6423 LESS THAN **OR** EQUAL TO 14%

OR

* Pressurizer Level, LI-3-462- LESS THAN **OR** EQUAL TO 10%

b. Go to Step 13

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

11 **Refill The RCS As Follows**

a. Align at least one High-head SI pump for hot leg injection as follows

1) Align at least one train of safety injection as follows

* Verify the following A train SI equipment aligned for injection

a) SI To Hot Leg, MOV-3-869, - OPEN

b) Loop A Hot Leg Safety Injection, MOV-3-866A - OPEN

* Verify the following B train SI equipment aligned for injection

a) SI To Hot Leg, MOV-3-869, - OPEN

b) Loop B Hot Leg Safety Injection, MOV-3-866B - OPEN

b. Start at least one HI-Head SI pump

c. Refill RCS with HHSI until either of the following conditions satisfied

* RHR cooling - RESTORED

<u>OR</u>

* LI-3-462, PZR Level Cold Cal - GREATER THAN 50%

12 **Go To Step 16**

QUESTION 8

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027	AK2.03
	Importance Rating	2.6	

Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners

Proposed Question: RO Question # 8

Given the following conditions:

- Unit 3 Reactor power is 100%.
- Pressurizer Pressure Control is in automatic.

If PC-3-444J, Pressurizer Pressure Controller, is inadvertently SET to maintain Pressurizer Pressure at 2475 psig, which ONE of the following describes the IMMEDIATE response of the Pressurizer Pressure Control System?

A. Pressurizer PORV, PCV-3-456, opens.

B. Both Pressurizer Spray Valves open to 100%.

C. Backup Group A and B Pressurizer Heaters energize.

D. Control Group Pressurizer Heaters reduce to minimum current.

Proposed Answer: C

Explanation (Optional):

A. Incorrect because PCV-456 will only open with high pressure sensed on PT-3-445. Plausible because PCV-455C <u>will</u> only open when the PZR Pressure Controller rises above setpoint.

B. Incorrect because a demand will exist to raise pressure, not lower it via spray valve operation. Plausible because the spray valves <u>will</u> open if actual PZR pressure rises

to 2475 psig.

C. CORRECT.

D. Incorrect because a demand will exist to raise pressure, not lower it via turning off any energized heaters. Plausible because ALL heaters are controlled via PC-3-444J, Pressurizer Pressure Controller and it is a common misconception that the PZR PCS works to raise pressure as demand increases.

Technical Reference(s): SD-009, Pressurizer & Relief System (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902109A, Obj. 7 & 8 (As available)

Question Source: Bank # WTSI 65121

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2008

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 7

55.43

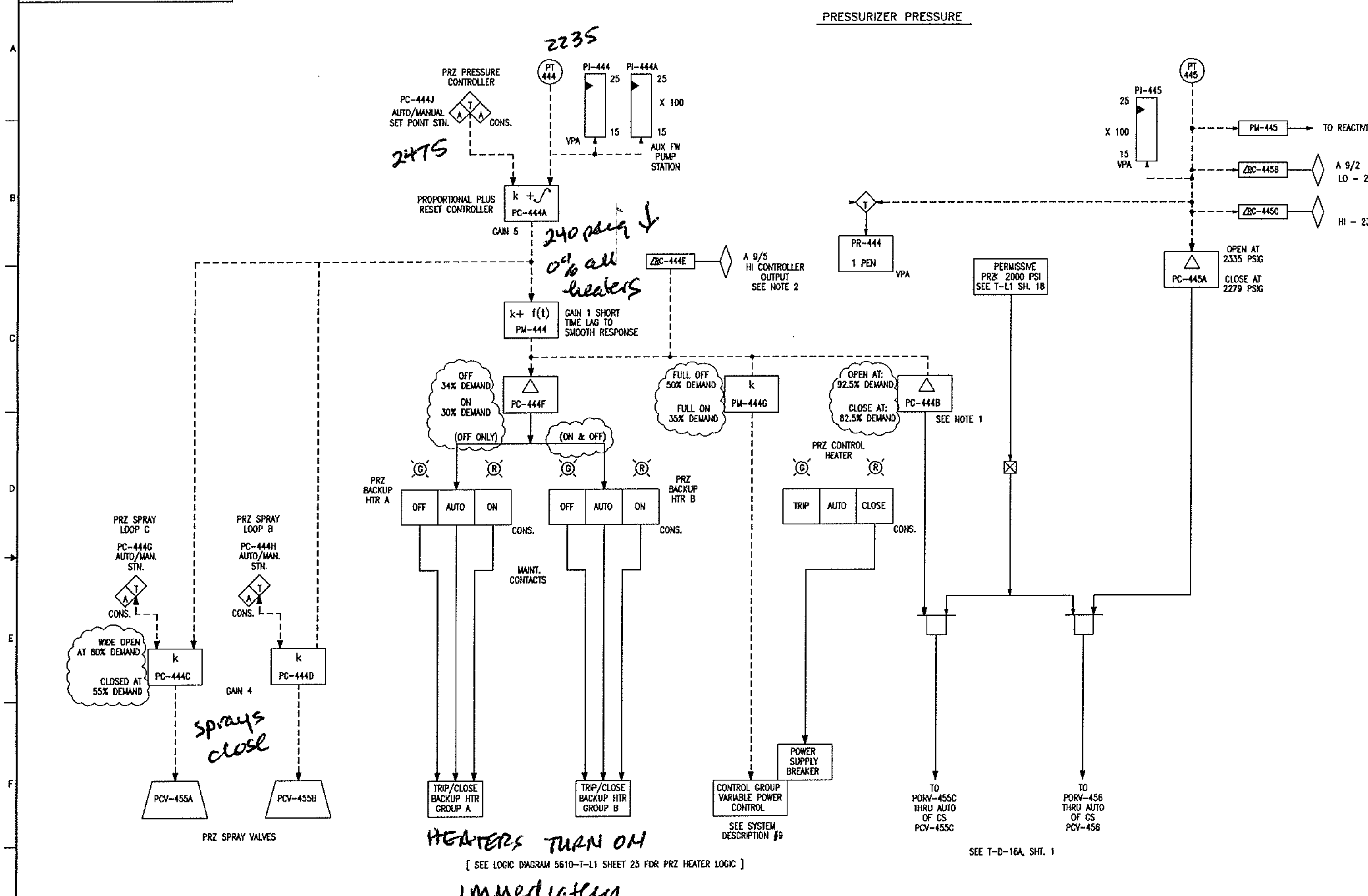
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

K/A Match Justification:

Question matches the K/A in that it tests the response of the PZR PCS components to a malfunction within the Master Pressurizer Pressure Controller.

PRESSURIZER PRESSURE



SET POINTS PRZ PRESSURE - PSIG

if 2475

2485	SAFETY VALVES OPEN
2385	REACTOR TRIP
2335	PORV'S OPEN
2310	SPRAYS WIDE OPEN
	HIGH PRESSURE ALARM
2260	SPRAYS START TO OPEN
2250	CONTROL HEATERS FULL OFF
2235	NORMAL OPERATION
2220	CONTROL GP FULL ON
>2210	BACKUP HEATERS OFF (REF. ONLY)
2210	BACKUP HEATERS ON
2185	LO PRESSURE ALARM
2000	PERMISSIVE TO OPEN PORV'S
1835	REACTOR TRIP
1730	SAFETY INJECTION

- sprays close
- HTRS OFF
- AT 2335 PCV-456 opens

NOTES:

1. PORV 456 WILL OPEN AT 2335 PSIG. PORV 455C WILL OPEN AT 92.5% DEMAND OUTPUT OF PC-444A.
2. ALARM SOUNDS AT 80% DEMAND OUTPUT OF PC-444A OR ABOUT 2310 PSIG. HOWEVER PC-444A IS A PROPORTIONAL PLUS RESET CONTROLLER AND ALARM MAY SOUND PRIOR TO 2310 PSIG.
3. "% DEMAND" REFERS TO CONTROLLER PC-444A OUTPUT AS DISPLAYED ON THE AUTO/MANUAL SETPOINT STATION.

REV	DATE	REVISION	BY	CH	APP	APP
12	12-8-06	ISSUED AS-BUILT PER CRN-I-4500 (PC/M 05-073).	RV	BB	-	TS
11	8-5-97	ISSUED AS-BUILT PER CRN-I-3578(PC/M 97-019)	RV	RH	BP	TS
10	5-18-95	INCORPORATED CRN-I-3176 FOR PC/M 95-042.	RV	RH	JJN	JCW
9	3-8-94	INCORPORATED CRN-I-1142 FOR PC/M 93-221.	RV	RH	RAS	JCW
8	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	MSP	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

CONTROL SYSTEM DIAGRAM

PRESSURIZER PRESSURE CONTROL

POD

DRAWING NUMBER 5610-T-D-16B

SYS

SHEET 1

REV 12

CAD FILE: 5610-T-D-16B_SH 1.DWG

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-041.5

<u>Title</u>:

Pressurizer Pressure Control Malfunction

(Continuous Use)

<u>Safety Related Procedure</u>	
Responsible Department:	Operations
Revision Approval Date:	12/17/07

RTSs *92-1254P, 92-1142P, 92-2143P, 93-1420P, 96-0981P, 96-1011P, 98-0918P, 01-0622P, 07-1080P*
PC/Ms *87-011, 90-528, 92-031*
***OTSC**: 0408-97*

W97:TNM/dt/bc/nw

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Foldout page is required to be monitored throughout this procedure.

CAUTION

The Master Controller should be operated carefully (Normal controller output for 2235 psig is 42.5 percent demand; 92 percent demand will open PCV-3-455C). If the following conditions are met, an excessive increase in controller output could cause Power Operated Relief Valve PCV-3-455C to open:

1. ***PCV-3-455C hand switch in AUTO.***
2. ***Pressurizer pressure is greater than or equal to 2000 psig, or OMS switch in LO Press Ops.***

1 Check PZR Pressure Control Instrument Loop Not Failed

a. Check PT-3-444 - NOT FAILED by comparison with adjacent pressure channels and known plant parameters

RESPONSE NOT OBTAINED:

a. Perform the following:

1) Verify PCV-3-455C **OR** MOV-3-536 CLOSED.

2) Take manual control of PC-3-444J, PZR PRESS CONTROL.

3) **IF** manual control of PC-3-444J is **NOT** effective, **THEN** perform the following:

* Take manual control of PZR spray valves.
* Take manual control of PZR heaters.

b. Check PT-3-445 - NOT FAILED by comparison with adjacent pressure channels and known plant parameters

RESPONSE NOT OBTAINED:

b. Perform the following:

1) Verify PCV-3-456 **OR** MOV-3-535 CLOSED.

QUESTION 9

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029	EK2.06
	Importance Rating	2.9	

Knowledge of the interrelations between the following and ATWS: Breakers, relays, and disconnects

Proposed Question: RO Question # 9

Given the following conditions:

- Unit 3 is at 90% power.
- RPS Testing is in progress.
- Reactor Trip Breaker A (RTA) is CLOSED.
- Reactor Trip Breaker B (RTB) is OPEN.
- Reactor Trip Bypass Breaker B (BYB) is Racked In and CLOSED.

During the testing the following occurs:

- The "A" RCP shaft seizes.
- A Reactor Trip signal is NOT generated by Protection Train B.
- Protection Train A generates a Reactor Trip signal as designed.
- The Reactor does NOT automatically trip.
- Manual Reactor Trip was successful.

Which ONE of the following identifies ALL the Reactor Trip Breaker Trip Coils and Reactor Trip Bypass Breaker Trip Coils that have changed state to trip the Reactor?

A. The RTA Undervoltage Trip Coil and the BYB Shunt Trip Coil.

B. The RTA Shunt Trip Coil and the BYB Shunt Trip Coil.

C. The RTA Undervoltage Trip Coil, the RTA Shunt Trip Coil, and the BYB Undervoltage Trip Coil.

D. The RTA Undervoltage Trip Coil, the RTA Shunt Trip Coil, the BYB Undervoltage Trip Coil, and the BYB Shunt Trip Coil.

Proposed Answer: C

Explanation (Optional):

A. Incorrect because the RTA UV coil and BYB shunt trip are not the only RTB breaker coils to actuate. Plausible because the Reactor *will* trip if the RTA UV coil were the only UV coil to deenergize or the BYB shunt trip coil were to energize if in service.

B. Incorrect because the RTA Shunt trip and the RTB Shunt Trip Coil are not the only RTB breaker coils to actuate. Plausible because the Reactor *will* trip if the RTA shunt trip coil were the only coil to energize or if the RTB Shunt Trip Coil were to energize.

C. CORRECT. According to SD-063, Page 35, Protection Train A actuates the RTA UV and Shunt Trip Coils and the BYB UV coil ONLY.

D. Incorrect because the BYB Shunt Trip Coil does not actuate on a Reactor Trip signal. According to Page 35 of SD-063, the BYB Shunt Trip Coil is used only when testing. Plausible because the first two coils are correct, because the complicated control scheme of the RTB coils is easily misunderstood, and because the bypass breaker shunt trip coils <u>*are*</u> used to trip the breakers when in the TEST position.

Technical Reference(s): SD-063, Reactor Protection and Safeguards Actuation System (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902163, Obj. 5 (As available)

Question Source: Bank # WTSI 59478

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2008 McGuire

Question Cognitive Level: Memory or Fundamental Knowledge (1I)

Comprehension or Analysis

Question Difficulty: Moderate (B)

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

QUESTION 10

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	038	EK1.02
Importance Rating	3.2	

Knowledge of the operational implications of the following concepts as they apply to the SGTR: Leak rate vs. pressure drop

Proposed Question: RO Question # 10

Given the following:

- A Steam Generator Tube Rupture has occurred on 3C S/G.
- RCS cooldown and depressurization are complete.
- Preparations are made to transition to 3-EOP-ES-3.1, Post SGTR Cooldown using Backfill.
- Pressurizer level is 38%.
- RCS subcooling is 40°F.
- 3C S/G narrow range level is 73% and slowly rising.

3C S/G level is slowly rising, in accordance with 3-EOP-E-3, Steam Generator Tube Rupture, which ONE of the following describes the required action and the reason?

A. Turn on the Pressurizer Heaters to raise RCS subcooling

B. Turn on the Pressurizer Heaters to minimize RCS leakage

C. Open a Pressurizer Spray Valve to minimize RCS leakage

D. Open a Pressurizer Spray Valve to lower RCS subcooling

Proposed Answer: C

Explanation (Optional):

A. Incorrect because turning on the Pressurizer Heaters does raise RCS Subcooling, but will increase primary to secondary leakage.

B. Incorrect because turning on the Pressurizer Heaters increases pressure in the RCS. Plausible since this action would correct if S/G level was decreasing.

C. Correct

D. Incorrect because opening a Pressurizer Spray Valve does lower RCS Subcooling, but it is not correct reason.

Technical Reference(s): LP 6902919, TAA SGTR (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902119, Obj. 2 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Farley

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3PEO)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

RCS and ruptured S/G(s) pressures must be maintained less than the ruptured S/G(s) steam dump to atmosphere setpoint.

NOTES

- *ATTACHMENT 2 may be removed from the procedure and used by the Unit Reactor Operator to control RCS-to-secondary leakage during this procedure.*
- *When RCS depressurization is required, normal spray should be used whenever possible. If normal spray is NOT available and letdown is in service, auxiliary spray should be used. If normal spray and auxiliary spray are NOT available, one PRZ PORV should be used.*

34 **Control RCS Pressure AND Charging Flow To Minimize RCS-To-Secondary Leakage**

a. Perform appropriate action(s) from table (ATTACHMENT 2)

PRZ LEVEL	RUPTURED S/G(s) LEVEL		
	INCREASING	DECREASING	OFFSCALE HIGH
LESS THAN 29%[50%]	• Increase RCS Charging Flow • Depressurize RCS. Refer to note prior to this step	Increase RCS Charging Flow	• Increase RCS Charging Flow • Maintain RCS And Ruptured S/G(s) Pressures Equal
BETWEEN 29%[50%] and 50%	Depressurize RCS. Refer to note prior to this step.	Turn On PRZ Heaters	Maintain RCS And Ruptured S/G(s) Pressures Equal
BETWEEN 50% and 71%[50%]	• Depressurize RCS. Refer to note prior to this step. • Decrease RCS Charging Flow	Turn On PRZ Heaters	Maintain RCS And Ruptured S/G(s) Pressures Equal
GREATER THAN 71%[50%]	Decrease RCS Charging Flow	Turn On PRZ Heaters	Maintain RCS And Ruptured S/G(s) Pressures Equal

Question #10

Facility: Farley

Vendor: WEC

Exam Date: 2007

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	038	EK1.02
	Importance Rating	3.2	3.5

Knowledge of the operational implications of the following concepts as they apply to the SGTR: Leak rate vs. pressure drop

Proposed Question:

Given the following:

- A Steam Generator Tube Rupture has occurred on Unit 1.
- RCS cooldown and depressurization are complete.
- The crew is maintaining the plant stable while preparing to transition to ESP-3.1, Post SGTR Cooldown using Backfill.
- The ruptured SG narrow range level is 73% and slowly decreasing.
- PRZR level is approximately 38% and slowly increasing.
- The OATC turns on PRZR heaters IAW the guidance in EEP-3, Steam Generator Tube Rupture.

Which ONE of the following describes the reason the OATC turned on the PRZR heaters?
To maintain pressurizer saturation temperature

A: corresponding to ruptured SG pressure to minimize SG leakage into the RCS

B: above the intact SG pressure to maintain adequate secondary heat sink with intact SGs

C: above the corresponding ruptured SG pressure to ensure RCS Subcooling is maintained

D: corresponding to intact SG pressure to ensure RCS Subcooling is maintained

Proposed Answer: A

Explanation (Optional):

correct. Attempting to maintain an inventory balance between RCS and ruptured SG prior to ruptured SG cooldown

Background document for EEP-3 page 83 of 119

A:

Purpose: To control RCS pressure and charging flow to maintain an indicated pressurizer level while
minimizing primary-to-secondary leakage.
Basis: In order to explain the basis for the guidance provided in this step, consider again equilibrium
conditions between leakage through the failed SG tube and charging flow, as shown in Figure
30. For primary system pressures greater than the ruptured steam generator pressure (PSG),
primary-to-secondary leakage will occur so that excess charging flow, i.e., greater than letdown and coolant shrinkage, is necessary to maintain pressurizer inventory. Conversely,
for letdown flows greater than charging flow, the equilibrium RCS pressure is less than the
ruptured steam generator pressure and secondary-to-primary leakage will occur. The ideal
conditions, shown by Point B, occur when charging flow exactly compensates for letdown
and coolant shrinkage so that RCS pressure and the ruptured steam generator pressure equalize. For these conditions both the pressurizer and ruptured steam generator inventories
will remain constant. Obviously fluctuations about these ideal conditions will occur due to
variations in ruptured steam generator pressure, cooldown rates, and letdown flows. Consequently, the operator must continuously adjust RCS pressure and charging flow to control pressurizer and ruptured steam generator inventories. This step provides guidance for
performing these actions in the form of a table. Figure 30 can be divided into four different
regions which are characterized by pressurizer and ruptured steam generator level behavior.
For primary pressures greater than the ruptured steam generator pressure, leakage into the
steam generator will increase steam generator water level (LSG). Alternatively, water level
will decrease for RCS pressures less than the ruptured steam generator pressure. Similarly,
pressurizer level (LPRZR) will increase for RCS pressures less than equilibrium. This

leads
to the four regions illustrated in Figure 30. The steps one performs to stabilize the plant at
the ideal, equilibrium conditions depend on the pressurizer inventory and ruptured steam generator water level behavior. For example, if pressurizer level is low, region II or region III
must be entered to increase pressurizer level. This requires one to increase charging flow or
decrease RCS pressure, as shown in Figure 30. The further into these regions, the more rapidly pressurizer level will increase. Of course, if pressurizer level is high, the opposite response would be necessary. However, the ruptured steam generator water level must also
be considered. STEP DESCRIPTION TABLE FOR E-3Step29 If the steam generator water
level is increasing, RCS pressure must be reduced to stop primary_to_secondary leakage. If
the steam generator water level is decreasing, primary pressure should be increased by energizing pressurizer heaters to minimize leakage into the RCS. Note that in some cases,
actions which address pressurizer level conflict with those which address steam generator
level. For example, if steam generator level is increasing one must decrease RCS pressure.
Since this will also increase pressurizer level, the pressurizer could fill with water if level is
initially high. However, by reducing charging flow, pressurizer level will decrease. Since this
will also decrease RCS pressure if heaters are not energized, steam generator level will also
stabilize. Hence, for this situation the preferred action is to reduce charging flow

B: incorrect because if intact SG pressure was higher than RCS pressure, EEP-3.0 would not be the governing procedure, ECP-3.1 would

C: incorrect. Cooldown is to ensure subcooling. RCS and ruptured SG will act like 2 pressurizers. Subcooling is not the issue

D: incorrect. Cooldown is to ensure subcooling. RCS and ruptured SG will act like 2 pressurizers. Subcooling is not the issue

Technical Reference(s): EEP-3 and background documents page 83 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

this meets the KA in that the question tests the concept of why PRZR pressure and temp affect the leak rate into or out of the Przr

high miss during validation. Most not aware of the reason in bases for the reason for energizing the heaters for this condition

QUESTION 11

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E12	2.1.30
	Importance Rating	3.4	

Conduct of Operations: Ability to locate and operate components, including local controls.

Proposed Question: RO Question # 11

Given the following conditions:

- A steamline break has occurred on Unit 3 Turbine Deck.
- The Unit 3 crew has entered 3-EOP-ECA-2.1, Uncontrolled Depressurization of All Steam Generators.
- The crew notes that the Main Steamline Isolation Valves did NOT automatically close.
- Attempts to manually close the valves from their control switches/ pushbuttons were NOT successful.

In accordance with 3-EOP-ECA-2.1, which ONE of the following describes the NEXT directed action to take and where it is accomplished?

A. Dispatch an operator to pull fuses for ONE train of solenoids for each MSIV behind the Console

B. Dispatch an operator to simultaneously pull fuses for BOTH trains of solenoids for all MSIVs behind the Console

C. Dispatch an operator to pull fuses for ONE train of solenoids for each MSIV at the Alternate Shutdown Panel

D. Dispatch an operator to simultaneously pull fuses for BOTH trains of solenoids for all MSIVs at the Alternate Shutdown Panel

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Per 3-EOP-ECA-2.1, when the MSIVs fail to automatically close, RNO Step 1 directs the crew to first try to manually close the valves (from the control switches on the console). If that is unsuccessful, the Step then directs the crew to pull fuses behind the console.

B. Incorrect since fuses for only one train of solenoids need to be pulled. Plausible because the 2nd part is correct. Also plausible because there are two sets of solenoids that operate to close the valves and two trains of MSIS will actuate the solenoids. (See LP 6902117, Pages 30 and 36-37 for a description of the MSIV control switches and solenoids).

C. Incorrect since the fuses are located behind the console, not at the ASP. Plausible because the 1st part is correct. Also plausible because there are control switches for the MSIVs on the ASP (See LP 6902117, Pages 30 and 36-37 for a description of the MSIV control switches and solenoids).

D. Incorrect since fuses for only one train of solenoids need to be pulled. Also incorrect since the fuses are located behind the console, not at the ASP. Plausible because there are two sets of solenoids that operate to close the valves and two trains of MSIS will actuate the solenoids. Also plausible because there are control switches for the MSIVs on the ASP. (See LP 6902117, Pages 30 and 36-37 for a description of the MSIV control switches and solenoids).

Technical Reference(s): 3-EOP-ECA-2.1, Uncontrolled Depressurization of All Steam Generators

LP 6902117, Main and Extraction Steam System

(Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: LP 6902335, Obj. 3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1B)

Comprehension or Analysis

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests location of control power fuses to the MSIV solenoid valves, during an uncontrolled depressurization of all SGs.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-EOP-ECA-2.1

Title:

Uncontrolled Depressurization of All Steam Generators

(Continuous Use)

Safety Related Procedure

Responsible Department:	Operations
Revision Number	4
Issue Date	8/10/11
Revision Approval Date:	8/8/11

***ARs** 564941, 1617263, 1672328*
***PCRs** 08-4390, 08-4390, 10-1667*
***PC/Ms** 87-025, 87-264, 90-440, 90-524, 95-028, 96-022, 08-006, 07-008*

W2002/cls/cls/mr

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Foldout page is required to be monitored throughout this procedure.

1	**Check Secondary Pressure Boundary** a. Main steamline isolation and bypass valves - CLOSED	Manually close valves. **IF** valves can **NOT** be closed, **THEN** dispatch operator to pull fuses for one train of solenoids for each MSIV (behind console) **OR** locally close ~~valves or block valves one loop at a time.~~
	b. Feedwater control and bypass valves - CLOSED	
	c. Feedwater isolation valves - CLOSED • MOV-3-1407 • MOV-3-1408 • MOV-3-1409	Correct Answer (A)
	d. S/G steam dump to atmosphere valves - CLOSED	
	e. S/G blowdown isolation valves - CLOSED	
	f. S/G sample valves - CLOSED	

INSTRUCTOR ACTIVITY

OBJECTIVES 4	NSO/LPRO/LPSO

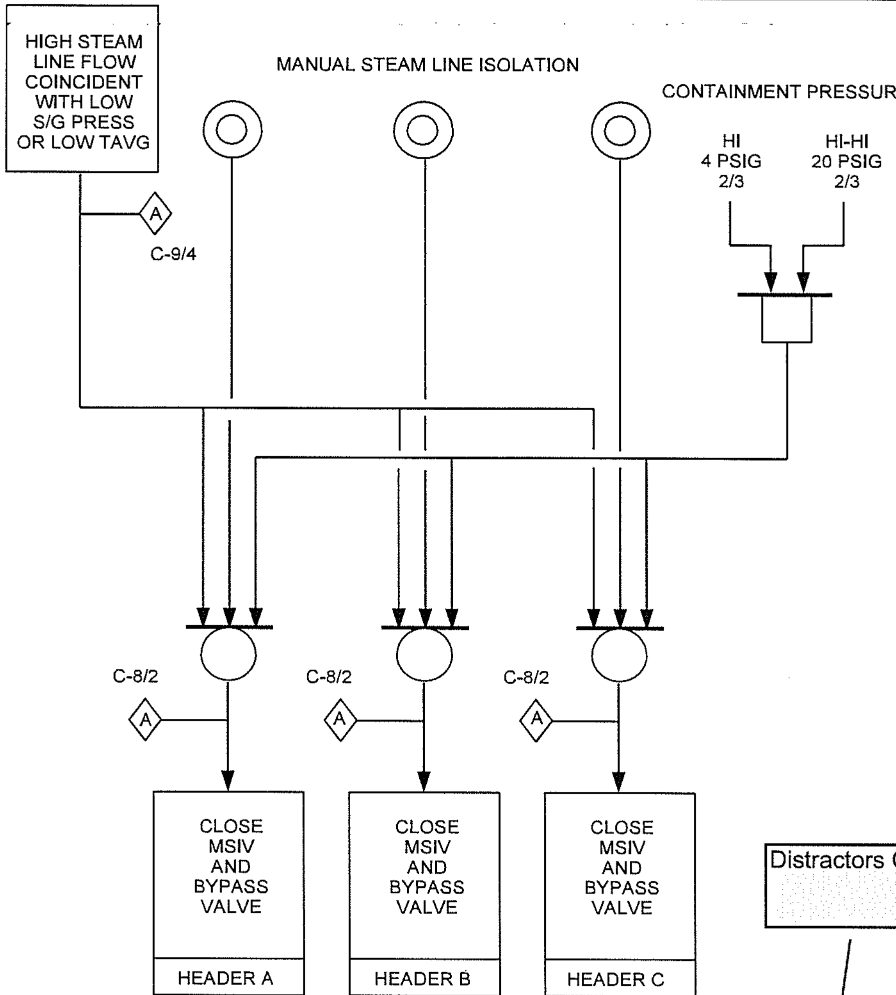
General Description	Slide(s) 39 - 42

2.4 Main Steam Isolation Valve

1. Air operated stop check valve in each steam line.
2. Isolates steam flow from S/G
3. Operation
   a. Held open by air pressure against spring pressure
   b. Air from instrument air system
   c. Four solenoid valve arrangement for controlling opening and closing
      1) Two valves in series for opening and in parallel for closing operation
      2) Insure reliability
   d. Valve disc opens into the steam flow
   e. Two additional solenoids for venting while valve is open
4. Valve control
   a. Main steam stop control switch
      1) Located on console
         a) OPEN / AUTO / CLOSE
         b) Controls four solenoid valves
   b. ASP
      1) OPEN / CLOSE
      2) Transfer SW
   c. Automatic closing
      1) High (≥ 4 PSIG) and high-high (≥ 20 PSIG) containment pressure, **OR**
      2) High steam flow (40% rated steam flow at 0 to 20% load linear to 114% rated steam flow at 100% load) coincident with either:
         a) Low $T_{AVG} \leq 543°F$, **OR**
         b) Low steam line pressure ≤ 614 PSIG

Distractors B & D

Distractors C & D



Distractors C & D

A Manual Steam Line Isolation may be initiated for each individual steam generator from the three Steam Line Isolation Pushbuttons on VPB.

The high steam flow setpoint is variable with turbine load and is basically a mismatch signal between the required steam flow for the turbine load and the actual steam flow. Refer to LP-6900163/(Sys. 049, 063) for more details.

The MSIVs can also be closed manually from the Alternate Shutdown Panel (ASP). Each MSIV has a two-position switch OPEN/CLOSE at the ASP. This switch is normally in the CLOSE position. When the Alternate Shutdown Panel is placed into service, the LOCAL/REMOTE switches are placed in the LOCAL position to close the MSIVs. Operation from the ASP is accomplished after transfer to the ASP has been completed.

Initiation of a steam line isolation signal will also cause the STEAM LINE ISOLATION alarm to annunciate on annunciator C, window 8/2. If the steamline isolation is due to the high steam line flow circuit there will also be a HI STM FLOW W/LOW TAVE/LOW STM PRESS SI alarm on annunciator C, window 9/4.

In the event of a steam line rupture, the isolation valves and non- return check valves are required to quickly interrupt steam flow in either direction, thus preventing an uncontrolled blowdown from more than one steam generator. The isolation valves are located outside the containment as close to the building as possible and downstream of the relief and safety valves. The isolation valves provide protection for a steam break outside containment downstream of the isolation valve and for a steam generator tube rupture. The most limiting accident is the break inside containment. For this accident the MSIVs and non-return checks close within five seconds to ensure against the uncontrolled blowdown of the two unaffected generators and limit the containment pressure rise to an acceptable value below design pressure of the containment building. On a break outside containment upstream or downstream of the isolation valve, containment pressurization is not a concern; however, uncontrolled blowdown of more than one steam generator must still be prevented.

A fast acting valve is not required for the steam generator tube rupture. The isolation valve in this case serves to limit primary leakage after shutdown when primary pressure has been reduced below shell side design pressure of the steam generator.

2.5 MSIV Air Operator Air Accumulator Backup System (Unit 4 Only)

The Air Operator Air Accumulator Backup System was installed due to a design deficiency that could result in the original MSIV closure time being greater than the five seconds required by technical specifications. Additionally, the MSIVs were designed for high flow, steam break conditions. However under certain low steam flow conditions, without instrument air, the accumulator air volume was not sufficient to close the MSIVs. This postulated steam flow could exceed the capacity of the Auxiliary Feedwater System resulting in all three Steam Generators boiling dry and the eventual loss of Main Steam to drive the auxiliary feed pump turbines and thus a loss of heat sink prior to achieving RHR conditions.

The Air Operator Air Accumulator Backup System is a dedicated, safety related system with independent pneumatic circuits, redundant electric solenoid valves, and independent high pressure gas reserves. This system ensures MSIV closure in $\leq$ 5 seconds and ensures MSIV closure can be maintained for a minimum of 1 hour without any operator action.

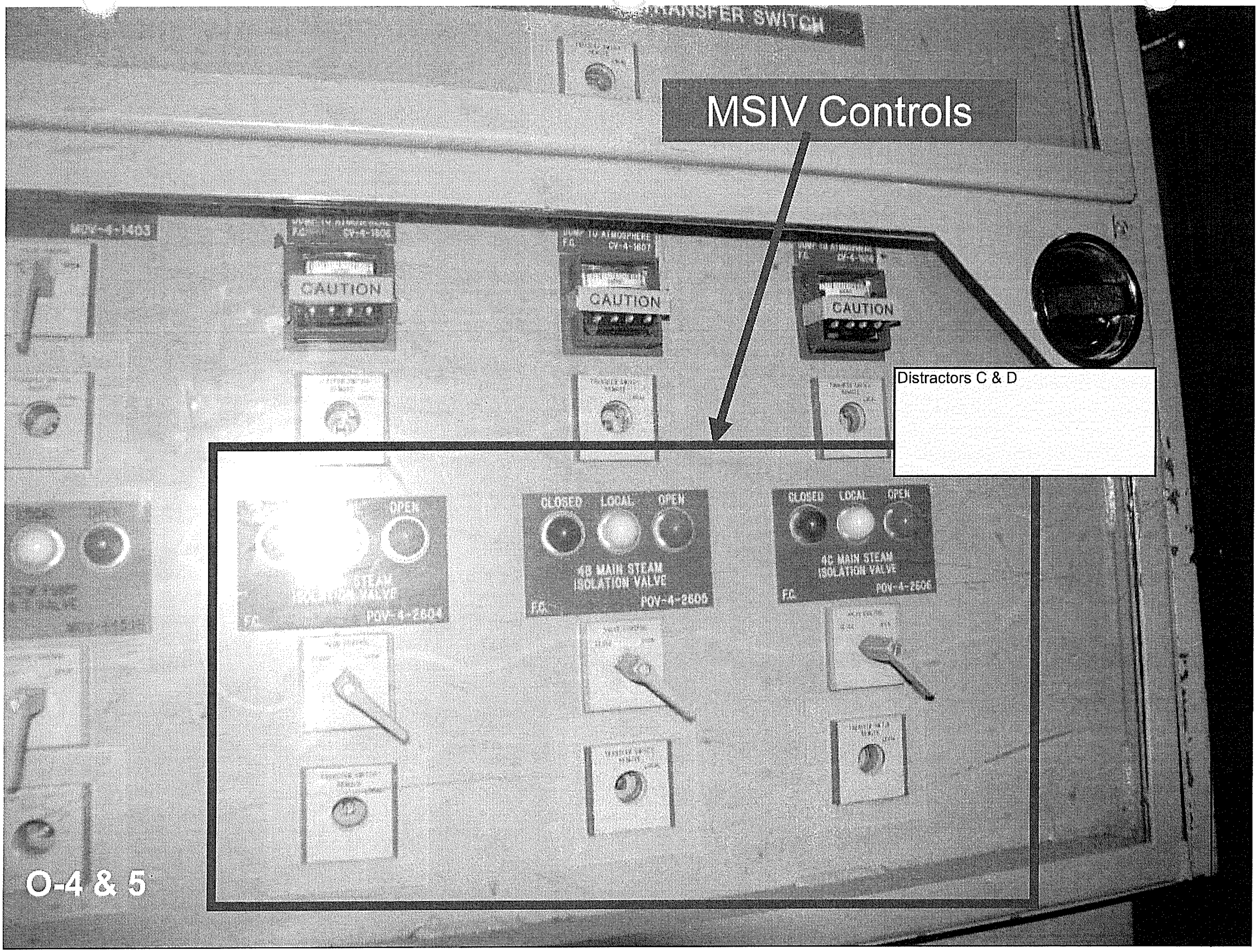
A large accumulator, consisting of two large interconnected tanks, is installed in the instrument air closing piping for each MSIV. Additionally, the manual valve in the instrument air opening piping for each MSIV is throttled and locked in position.

INSTRUCTOR ACTIVITY

OBJECTIVES 4	NSO/LPRO/LPSO

General Description	Slide(s) 50 - 53

3. Instrument Air
   a. Normal MSIV operating gas
   b. Supplied from normal plant Inst Air system
   c. Large Accumulator Air Reserve Tank (per MSIV)
   d. Air used to open and close MSIVs
   e. Open and close supply to MSIV operating piston is via 2 solenoid valves (Two Trains)
   f. Two solenoid valves on closure supply continuously vent top of MSIV operating piston when MSIV is open
   g. 3/32" orifice in vent line maintains constant backpressure on operating piston
   h. Inherent leakby on operating piston ensures some gas pressure on piston
   i. Vent solenoids close on MSIV closure signal
4. MSIV closure
   a. Control switch (CR or ASP)
   b. Main steam isolation
   c. Opening solenoids de-energize and position to vent
   d. Both closing solenoids energize to supply air to piston top
   e. Venting solenoids energize to close
5. Miscellaneous Items
   a. Globe valve in opening supply is throttled to limit dynamic response while venting for MSIV closure
   b. Check valve around train 'A' opening solenoid is provided to allow vent path via 'B' if 'A' fails to position to vent
   c. Check valve also blocks air supply to MSIV if Train 'B' solenoid fails open during MSIV closure
6. Vital 125 Volt DC
   a. Supplies operating solenoids
   b. Train 'A' solenoids from 3D01 breaker 21 and 4D23 breaker 11
   c. Train 'B' solenoids from 3D23 breaker 14 and 4D01 breaker 21



MSIV Controls
Distractors C & D
CAUTION
CAUTION
CAUTION
CLOSED LOCAL OPEN
4B MAIN STEAM ISOLATION VALVE
POV-4-2605
CLOSED LOCAL OPEN
4C MAIN STEAM ISOLATION VALVE
POV-4-2606
OPEN
POV-4-2604
O-4 & 5

QUESTION 12

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	054	AK3.02
Importance Rating	3.4	

Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Matching of feedwater and steam flows

Proposed Question: RO Question # 12

Given the following:

- Unit 4 is at 70% power.
- SGFPs A and B are in service.
- Alarm SGFP A/B MOTOR OVERLOAD TRIP (D 6/1) is received.
- Turbine Load remains stable.

Which ONE of the following describes the action to be taken by the operating crew and the reason for this action?

A. Take manual control of the Main Feedwater Regulating Valves and raise feedwater flow to match steam flow to stop the decrease in S/G levels.

B. Take manual control of the Main Feedwater Regulating Valves and raise feedwater flow to match steam flow to prevent the automatic controls from "overshooting" the program level for the S/Gs.

C. Manually reduce turbine load until first stage pressure is at 45 percent load to prevent the automatic controls from "overshooting" the program level for the S/Gs.

D. Manually reduce turbine load until first stage pressure is at 45 percent load to match steam flow to feedwater flow and stop the decrease in S/G levels.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Based on the equipment lineup given, available feedwater flow is not sufficient for 70% power; therefore, SG levels will not stabilize. Plausible because this would be the correct actions, per 4-ARP-097-CR.C, (1/1, 1/2, 1/3) for power below 55%.

B. Incorrect. Based on the equipment lineup given, available feedwater flow is not sufficient for 70% power; therefore, SG levels will not rise and 'overshoot.' Plausible because overshoot is a common response of control circuits as they take time for stabilization. The applicant assumes taking manual control will alleviate the low feedwater flow.

C. Incorrect. Turbine load reduction occurs until first stage pressure is at 45 percent load to match steam flow to feedwater flow and stop the decrease in S/G levels, not to prevent an overshoot. Plausible because overshoot is a common response of control circuits as they take time for stabilization.

D. CORRECT. With the amount of feed flow limited to one pump, the success path is to limit turbine load to 45% first stage pressure.

Technical Reference(s): 4-GOP-301, Hot Standby to Power Operation (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902248, Obj. 4 (As available)

Question Source: Bank # WTSI 66606

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2008 South Texas

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 5

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

C1

SG A
NARROW RANGE
LO/LO-LO
LEVEL

CAUSES:
1. Steam Generator Level Control Malfunction
2. Instrument Failure

DEVICE:	SETPOINT:		LOCATION:
• LC-474A	LO 35%	LO-LO 10%	N/A
• LC-475A	LO 35%	LO-LO 10%	
• LC-476A	LO 35%	LO-LO 10%	
• LC-474B	LO-LO 10%		
• LC-475B	LO-LO 10%		

PROMPT ACTIONS

1. IF malfunctioning SG level controls, THEN:
   - **TAKE** manual control of level.
   - **RETURN** SG levels to normal.

Distractor A

ALARM CONFIRMATION

1. **CHECK** LI-4-474, 475, 476, A STM GEN LEVEL less than 35% for LO alarm on VPA.
2. **CHECK** LI-4-474, 475, 476 less than 10% for LO-LO alarm on VPA.
3. IF LI-4-474 less than 10%, THEN **CHECK** LC474A1, SG A LO-LO LEVEL and LC474B1, SG A LO LEVEL bistable status light LIT on VPB.
4. IF LI-4-475 less than 10%, THEN **CHECK** LC475A1, SG A LO-LO LEVEL and LC475B1, SG A LO LEVEL bistable status light LIT on VPB.
5. IF LI-4-476 less than 10%, THEN **CHECK** LC476A1, SG A LO-LO LEVEL bistable status light LIT on VPB.
6. **CHECK** recorder FR-4-478, A STEAM GENERATOR on console.

OPERATOR ACTIONS

IF alarm is due to instrument failure, THEN **REFER TO** 4-ONOP-049.1, Deviation Or Failure Of Safety Related Or Reactor Protection Channels.

REFERENCES:
1. FPL Logic Diagram 5610-T-L1, Sheet 19
2. FPL Control System Diagram 5610-T-D-17
3. Tech Specs Sections 3/4.3.1, 3/4.3.3

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 4

4-ONOP-089

CAUTION

Performance of this procedure may affect core reactivity.

Title:

Turbine Runback

(Continuous Use)

Non-Safety Related Procedure	
Responsible Department:	Operations
Revision Approval Date:	11/14/07C

RTSs *86-1181P, 87-1308, 88-1233, 88-1650, 88-2233, 91-2289, 92-1268P, 93-0415P, 95-1093, 96-0492P, 00-0195P, 00-0669P, 07-0947P*
OTSC *10421*
PC/Ms *84-211, 92-031, 92-181, 95-100, 04-069*

W97:*/sw/ln/ln

FPL	TURKEY POINT UNIT 4 **ANNUNCIATOR RESPONSE PROCEDURE** QUALITY RELATED **CONTINUOUS USE**	Procedure No. **4-ARP-097.CR.C** Revision No. **0** Effective Date **04/15/10**

Title:

CONTROL ROOM RESPONSE - PANEL C

Responsible Department: **OPERATIONS**

Special Considerations:

This is a New Procedure. Initial use should include increased awareness because of the potential technical and/or sequential changes to the procedure. After initial use of this procedure provide comments back to the Procedure Upgrade Project.

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED__________ INITIAL__________

Revision	Approved By	Approval Date
0	Michael Murphy	03/23/10

UNIT #	UNIT 4
DATE	
DOCT	PROCEDURE
DOCN	4-ARP-097.CR.C
SYS	OPS
STATUS	COMPLETED
REV	0
# OF PGS	

4.0 **IMMEDIATE OPERATOR ACTION**

4.1 Verify the automatic actions listed in 3.0 are functioning to stabilize and maintain plant conditions, or assume manual control.

CORRECT ANSWER

5.0 **SUBSEQUENT OPERATOR ACTIONS**

5.1 Determine the cause of the runback initiation **AND** refer to the appropriate ONOP for specific recovery instructions.

5.2 Verify the following conditions:

5.2.1 Steam generator levels and pressures stabilized.

5.2.2 Steam dumps closed.

5.2.3 Tavg matches Tref.

5.2.4 Pressurizer levels and pressures stabilized.

5.3 Notify the Load Dispatcher **AND** the Plant General Manager in accordance with 0-ADM-115, Notification of Plant Events.

5.4 **IF** possible, **THEN** mark Control Room charts with the date, time, and cause of the incident.

5.5 Complete operator logs.

5.6 Notify the Shift Manager to review the requirements of PI-AA-100-1002, Failure Investigation Process (FIP), to determine if a FIP Team should be activated.

5.7 **IF** reactor power has changed by greater than or equal to 15 percent, **THEN** notify the Chemistry Department that RCS Sampling is required by Technical Specifications Table 4.4-4, Item 6.b.

FPL	TURKEY POINT UNIT 4 ANNUNCIATOR RESPONSE PROCEDURE QUALITY RELATED **CONTINUOUS USE**	Procedure No. **4-ARP-097.CR.D** Revision No. **2** Effective Date **05/03/11**

Title:

CONTROL ROOM RESPONSE - PANEL D

Responsible Department: **OPERATIONS**

Special Considerations:

> This is a New Procedure. Initial use should include increased awareness because of the potential technical and/or sequential changes to the procedure. After initial use of this procedure provide comments back to the Procedure Upgrade Project.

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED_________ INITIAL__________

Revision	Approved By	Approval Date
0	Robert Hess	06/20/10
2	B Stamp	03/08/11

UNIT #	UNIT 4
DATE	
DOCT	PROCEDURE
DOCN	4-ARP-097.CR.D
SYS	OPS
STATUS	COMPLETED
REV	2
# OF PGS	

REVISION NO.: 2	PROCEDURE TITLE: CONTROL ROOM RESPONSE - PANEL D	PAGE: 34
PROCEDURE NO.: 4-ARP-097.CR.D	TURKEY POINT UNIT 4	WINDOW: 6/1 (Page 1 of 1)

CAUSES: 1. Motor malfunction
2. Low bus voltage

D6

SGFP A/B MOTOR OVERLOAD TRIP

DEVICE:	**SETPOINT:**	**LOCATION:**
Relay 174/TDDO	2160 amperes	Bkrs 4AA03 & 4AC14

ALARM CONFIRMATION

CHECK feed pump indications on console.

OPERATOR ACTIONS

1. **ENSURE** the following automatic actions have occurred:
   - Turbine runback if power was greater than or equal to 45 percent
   - Automatic start of idle SGFP if applicable
   - Possible automatic start of AFW
2. IF turbine runback applicable, THEN **GO TO** 4-ONOP-089, Turbine Runback.
3. **MONITOR** S/G levels.
4. **DISPATCH** operator to check Feed Pump Breakers 4AA03 or 4AC14 for any targets.
5. **DISPATCH** operator to check feed pump locally for any abnormal indications.

CORRECT ANSWER

REFERENCES: 1. FPL DWG 5610-T-L1 Sh 25A
2. FPL DWG 5610-E-26 Sh 1C

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 4

This procedure may be affected by a T.C. (Temporary Change) Verify information prior to use.
Date verified________ Initials________

4-GOP-301

CAUTION
Performance of this procedure may affect core reactivity.

Title:

Hot Standby to Power Operation

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Number:	4
Issue Date:	4/20/11
Revision Approval Date:	4/19/11

ARs *574483, 579118, 1600751, 581376, 1614039, 1639210*
PCRs *08-3777, 08-4135, 08-3669, 08-3794, 08-4231, 09-0459, 08-4094, 09-0548, 09-0646, 09-1351, 09-1289, 09-1175, 09-2945, 09-1650, 09-3803, 10-0094, 09-3879*
PC/Ms *83-200, 86-007, 86-201, 87-257, 87-263, 87-264, 87-266, 88-178, 88-487, 89-169, 90-441, 92-018, 92-073, 92-178C, 92-181, 93-053, 95-028, 96-013, 96-022, 96-086, 97-012, 97-014, 98-050, 99-016, 99-045, 01-063, 02-085, 04-026, 04-113, 04-163, 05-084, 08-004, 08-011, 09-051*

INIT

NOTE

If the second feedwater pump is not going to be placed in service, then reactor power can be raised to 55 percent with a single feedwater pump in service. Power must be reduced to 45 to 50 percent when it is desired to place the second feedwater pump in service.

Distractors B + C

CAUTION

Due to recent design changes to the FRVs and the on-going fine tuning of the control circuits, the FRVs may not respond as expected after the second feed pump is started. Manual feed control may be necessary if the control system does not respond as expected after starting the second feedwater pump.

____ 5.79 **IF** desired and steam generator level is on program, **THEN** place the second S/G Feedwater Pump in service using 4-NOP-074, Steam Generator Feedwater System.

____ 5.79.1 Record MWE SGFP placed in service: ____________________

____ 5.79.2 Verify both running pumps have approximately equal running amps.

____ 5.79.3 Place the S/G FD Pump Turbine Runback switch to the NORMAL position.

5.80 **WHEN** Reactor Power is greater than 45 percent, **THEN** perform the following:

____ 5.80.1 Verify that the POWER BELOW P-8 status light on VPA is OFF.

____ 5.80.2 Verify Turbine Overspeed Protection amber 20 percent Load light is out.

NOTE

Reactor Protection System design precludes testing of the RCP Breaker Loss of Flow (1/3) logic function in accordance with 4-OSP-049.1 (STP 2387 and 2388) prior to entering the applicable mode and plant condition (Mode 1, above P-8).

____ 5.80.3 **IF** the RCP Breaker Loss of Flow (1/3) logic function surveillance (STP 2387 and/or 2388) is expired, **THEN** log the surveillance as a missed surveillance **AND** comply with Technical Specification. 4.0.3. (Reference CR 2008-20641)

QUESTION 13

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055	EA2.03
	Importance Rating	3.9	

Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power

Proposed Question: RO Question # 13

Given the following:

- A station blackout occurred on Unit 3.
- The operating crew is performing 3-EOP-ECA-0.0, Loss of All AC Power.
- NEITHER Unit 3 EDG can be started.
- BOTH Unit 4 EDGs are operating and supplying their respective 4 KV Busses.
- Off-Site power availability is NOT expected within the next 2 hours.

Which ONE of the following describes the actions that are necessary to restore power?

Align 4KV Bus…

A. 3A OR 3B ONLY using the Station Blackout Tie Line in accordance with 3-ONOP-004.1, System Restoration Following Loss of Offsite Power.

B. 3A OR 3B ONLY using the Station Blackout Tie Line in accordance with 3-ONOP-004.2 (4.3), Loss of 3A (3B) 4KV Bus.

C. 3A AND 3B using the Station Blackout Tie Line in accordance with 3-ONOP-004.1, System Restoration Following Loss of Offsite Power.

D. 3A AND 3B using the Station Blackout Tie Line in accordance with 3-ONOP-004.2 (4.3), Loss of 3A (3B) 4KV Bus.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because ONLY 1 bus will be energized through the Station Blackout tie line. Also plausible because the ONOP-004.1 does give guidance for restoration and energizing busses assuming one source of power to Unit 3 is available.

B. CORRECT. Per Steps 8-14 of ONOP-004.2 (4.3).

C. Incorrect because both busses cannot be energized from the station blackout tie line without violating procedures. Also plausible because the ONOP-004.1 does give guidance for restoration and energizing busses assuming one source of power to Unit 3 is available.

D. Incorrect because both busses cannot be energized from the station blackout tie line without violating procedures. Plausible because correct procedure is referenced

Technical Reference(s): 3-EOP-ECA-0.1, Step10
3-ONOP-004.2, Steps 8-14
3-ONOP-004.1
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902349, Obj. 3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

Question matches the K/A in that it tests actions required for restoring power to a 4KV Bus during a blackout.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	**Check If AC Power Has Been Restored**	
	a. Check the 3A and 3B 4KV buses - AT LEAST ONE ENERGIZED	a. Perform the following: 1) Restore AC power using the following procedures: • 3-ONOP-004.2, LOSS OF 3A 4KV BUS • 3-ONOP-004.3, LOSS OF 3B 4KV BUS 2) **WHEN** power is restored to the 3A or 3B 4KV bus, **THEN** observe the CAUTIONS prior to Step 32 and go to Step 32 to perform recovery actions. 3) Observe CAUTION prior to Step 11 **AND** continue with Step 11.
	b. Verify required safeguards equipment – OPERATING	b. Manually start equipment as required.
	c. Check if 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES being monitored FOR INFORMATION ONLY prior to entering 3-EOP-ECA-0.0, LOSS OF ALL AC POWER	c. Implement FRPs as required, unless this procedure was directly entered from outside the EOP network.
	d. Return to procedure **AND** step in effect	

CAUTION

When power is restored to 3A or 3B 4KV bus, recovery actions should continue by observing CAUTIONS prior to Step 32 and then performing Step 32.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	**Place Non-Running Equipment Switches In PULL-TO-LOCK Or STOP As Follows** • Unit 3 high-head SI pumps – PTL • Containment spray pumps – PTL • Emergency containment coolers – STOP • Emergency containment filter fans – STOP **AND** OPEN Breaker 30806, Emergency Containment Filter Fan 3B, on MCC 3D • RHR pumps – PTL • CCW pumps – PTL	

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-004.1

Title:

System Restoration Following Loss of Offsite Power

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Approval Date:	9/4/07

RTSs *91-1561P, 91-2743P, 93-0173P, 93-1679P, 95-0234P, 95-0498P, 95-0918P, 97-1375P, 01-0529P, 01-0695P, 03-0204P, 04-0967P, 07-0478P*
PC/M *87-258, 87-263, 87-264, 87-265, 94-059, 01-009*

W97:DH/ev/mrg/ln

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Check If Offsite Power Should Be Restored To 3A 4KV Bus At This Time**	
	a. Consult with the Shift Manager to determine desired order of offsite power restoration * 3A 4KV bus followed by 3B 4KV bus **OR** * 3B 4KV bus followed by 3A 4KV bus	Distractor A
	b. Check desired order of offsite power restoration - 3A 4KV BUS FOLLOWED BY 3B 4KV BUS	b. Go to Step 35.
6	**Check 3A 4KV Bus - ENERGIZED**	Go to Step 28.
7	**Check 3A 4KV Bus - ENERGIZED BY 3A EMERGENCY DIESEL GENERATOR**	Go to Step 11.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-004.2

Title:

Loss of 3A 4KV Bus

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Number:	1
Issue Date:	4/11/11
Revision Approval Date:	4/8/11

ARs *1635876*
PCRs *09-1723, 09-2751*
PC/Ms *87-258, 87-263, 87-264, 87-265, 94-059, 01-009*

W97:BC/ln/nw/cls

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

The Station Blackout Tie Line may be used only when both the 3A and 3B 4KV buses are deenergized.

NOTE

If the 3A and 3B 4KV buses are both deenergized because offsite power and Unit 3 Emergency Diesel Generators are NOT available, power needs to be restored to at least one of these 4KV buses within 10 minutes to satisfy station blackout requirements.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	**Determine If Station Blackout Tie Line May Be Used** • Check 3B 4KV bus – DEENERGIZED • Check 4A and 4B 4KV buses – AT LEAST ONE ENERGIZED	Perform the following: a. Determine if the Shift Manager wants to energize 3A 4KV bus from 3C 4KV bus using ATTACHMENT 2, while continuing with this procedure. b. Continue efforts to reenergize 3A 4KV bus from the following: * 3A Emergency Diesel using Steps 4 and 5. **OR** * Unit 3 Startup Transformer using Step 6. **OR** * Unit 4 Startup Transformer using Step 7. c. **WHEN** 3A 4KV bus is energized, **THEN** go to Step 16.
9	**Check 3D 4KV Bus Lockout Relay - RESET**	Perform the following: a. Reset 3D 4KV bus lockout relay. b. **IF** 3D 4KV bus lockout relay can **NOT** be reset, **THEN** go to Step 15.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	**Check 3D 4KV Bus - ALIGNED TO 3A 4KV BUS** • Supply From 4KV Bus 3A, 3AD01 – CLOSED • Feeder To 4KV Bus 3D, 3AA17 – CLOSED	Perform the following: a. Open Feeder To 4KV Bus 3D, 3AB19 b. Open Supply From 4KV Bus 3B, 3AD06 c. Close Supply From 4KV Bus 3A, 3AD01 d. Close Feeder To 4KV Bus 3D, 3AA17 e. **IF** 3D 4KV bus can **NOT** be aligned to 3A 4KV bus, **THEN** go to Step 15.
11	**Verify Station Blackout Permissive Blue Light For Station Blackout Breaker, 3AD07 – ON**	Perform the following: a. Open the following breakers: • 3AA02, Auxiliary Transformer 3A 4KV Bus Supply • 3AA05, Startup Transformer 3A 4KV Bus Supply • 3AA20, 3A Emergency Diesel To 3A 4KV Bus • 3AA22, 3A 4KV Bus Emergency Tie To Unit 4 Startup Transformer • All load breakers on 3A and 3D 4KV buses b. **IF** station blackout permissive can **NOT** be satisfied, **THEN** go to Step 15.
12	**Check 4D 4KV Bus – ENERGIZED**	Perform the following: a. Request Unit 4 RO to reenergize 4D 4KV bus using 4-ONOP-004.5, LOSS OF 4D 4KV BUS. b. **IF** 4D 4KV bus can **NOT** be energized, **THEN** go to Step 15.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***When a station blackout condition exists, loading on each Unit 4 Emergency Diesel Generator shall be limited to 3095 KW.***
- ***If the Unit 4 4KV bus supplying power to the 4D 4KV bus is energized by an EDG AND Station Blackout Breaker 4AD07 is closed, non-running safeguards equipment on the bus supplying power should be placed in PULL-TO-LOCK or STOP to prevent autostart and possible overload of the EDG.***

13 **Check 4KV Bus Supplying Power To 4D 4KV Bus - ENERGIZED BY OFFSITE POWER**

RESPONSE NOT OBTAINED:

Perform the following:

a. IF only one Unit 4 4KV bus is energized AND from an EDG, THEN perform one of the following:

1) Check that the Unit 4 RO has completed Step 2 of Attachment 2 of 4-EOP-ES-0.1, AND go to Step 14.

OR

2) Check that Unit 4 RO has completed Step 3 of Attachment 2 of 4-ONOP-004, AND go to Step 14.

b. Have the Unit 4 RO place non-running safeguards equipment in PULL-TO-LOCK or STOP on the Unit 4 4KV bus supplying the 4D 4KV Bus.

c. IF loads can NOT be reduced, THEN go to Step 15.

CAUTION

If offsite power to the Unit 4 4KV bus supplying power to the 4D 4KV Bus is lost after Station Blackout Breaker 4AD07 is closed, the associated EDG output breaker will NOT close until 4AD07 has been opened.

14 **Try To Re-energize 3A 4KV Bus From Station Blackout Tie Line**

a. Close Station Blackout Breaker 3AD07 using keylock switch (Key Number 82)

RESPONSE NOT OBTAINED: a. Go to Step 15.

b. Direct Unit 4 RO to close Station Blackout Breaker 4AD07 using keylock switch (Key Number 82)

QUESTION 14

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	056	AA2.54
Importance Rating	2.9	

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Breaker position (remote and local)

Proposed Question: RO Question # 14

Given the following:

- The station experiences a Loss of Offsite Power.
- One undervoltage relay for 4B 4KV Bus fails to actuate.
- All other Bus undervoltage relays operate properly.
- All Load Center undervoltage and degraded voltage relays operate properly.

For Unit 4, which ONE of the following describes (1) the bus-stripping response and (2) the response of the EDG(s) Output Breakers?

A. (1) Bus stripping will occur ONLY on Bus 4A;
(2) ONLY the 'A' EDG output breaker will close.

B. (1) Bus stripping will occur on Bus 4A AND on Bus 4B;
(2) ONLY the 'A' EDG output breaker will close.

C. (1) Bus stripping will occur ONLY on Bus 4A;
(2) BOTH the 'A' EDG AND the 'B' EDG output breakers will close.

D. (1) Bus stripping will occur on Bus 4A AND on Bus 4B;
(2) BOTH the 'A' EDG AND the 'B' EDG output breakers will close.

Proposed Answer: D

Explanation (Optional):

A. Incorrect since both EDGs will start and load and both Bus 4A and 4B will load shed. Plausible because this would be true if the 2/2 UV relays on the 4KV Busses were the only relays which initiate load shedding and start of the respective EDG.

B. Incorrect since both EDGs will start. Plausible because the 1st part is correct. Also plausible because the 2nd part would be true if the 2/2 UV relays on the 4KV Busses were the only relays which start the respective EDG.

C. Incorrect since both Bus 4A and 4B will load shed. Plausible because the 2nd part is correct. With a start of the EDG and given no other failures, the EDG output breakers should close as designed. Also plausible because this would be true if the 2/2 UV relays on the 4KV Busses were the only relays which initiate load shedding.

D. CORRECT. When offsite power is lost, the 4160 V Vital busses 4A and 4B will sense the UV and actuate the UV relays. 2/2 UV relays must actuate on the bus to start the respective EDG. See SD-140, Main Power Distribution, Pages 58 states: "4.16KV bus A and B each have voltage sensing relays that will actuate bus associated Emergency Load Sequencer to strip the bus loads. Additionally, the vital 480 volt load centers associated with each bus also have voltage sensing relays that on degraded voltage will input the Emergency Load Sequencer to cause the same bus stripping actions. Detailed information may be found in System 024, Emergency Load Sequencer Lesson and Logic Sheets 12, 12A, 12B, 13 and 13A." Additionally, SD-140, Page 59 states: "The four emergency diesel generators (EDGs), located in two emergency diesel buildings, automatically provide power to 4.16kV safety-related (ESF) buses A and B within 15 seconds following a loss of power to those buses. Should either A or B bus experience a complete loss of voltage or a sustained undervoltage (degraded voltage) condition, as sensed on A, B, C, or D load center, the associated bus Emergency Load Sequencer will strip the buses, send a start signal to its EDG and a close signal to the EDG output breaker, which will close when the bus is cleared. LP 6902136, Page 197 states: "A and B 4kV buses each have two loss of voltage sensing relays, both of which must see approximately 30 to 50% of nominal bus voltage (2975 volts) for 1 second or more to pick up their associated X2 (for "A" bus) and/or Z2 (for "B" bus) BUS STRIPPING RELAYS." With a start of the EDG and given no other failures, the EDG output breakers should close as designed.

Technical Reference(s): SD-140, Main Power Distribution
LP 6902136, Emergency Diesel Generator & Auxiliaries (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902136, Obj. 10 (As available)

Question Source: Bank # WTSI 56489

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2005 Davis Besse

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

3.6 AUTOMATIC START SIGNALS

OBJECTIVE # 7 AND 14

Correct Answer (D)

Auto Start Signals

Loss of Voltage (LV) ~~on A or B 4kV buses.~~

LV on "A" 4kV bus starts the "A" EDG

LV on "B" 4kV bus starts the "B" EDG

A and B 4kV buses each have two loss of voltage sensing relays, both of which must see approximately 30 to 50% of nominal bus voltage (2975 volts) for 1 second or more to pick up their associated X2 (for "A" bus) and/or Z2 (for "B" bus) BUS STRIPPING RELAYS.

The X2 relay will start the "A" EDG. (See logic sheet 13, 13A and 9A for additional details.) Loss of voltage on any single safety-related 4kV bus will start its associated EDG.

Any Safety Injection ~~signal starts all four EDGs.~~ (ANY UNIT/ANY TRAIN)

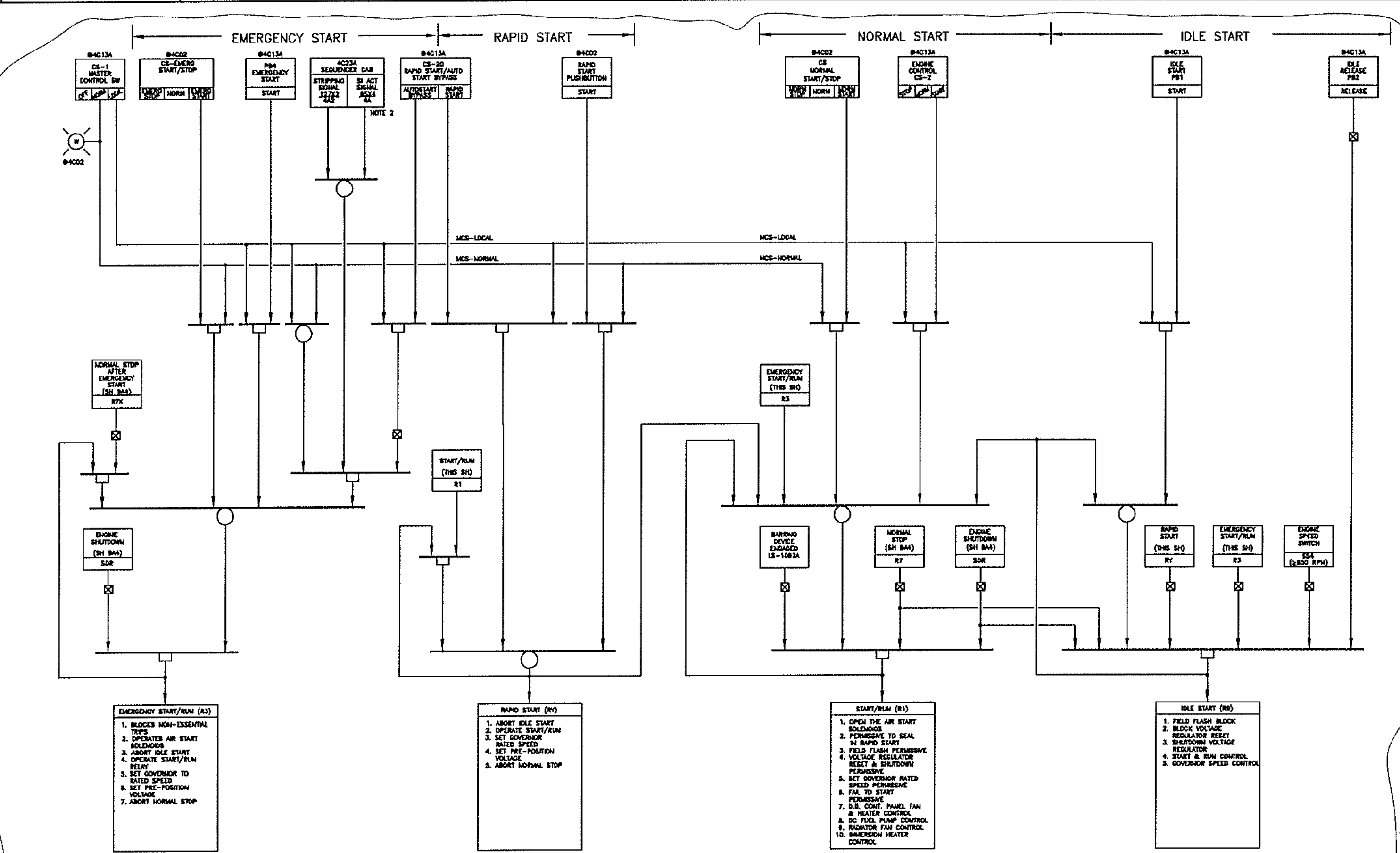
Degraded voltage condition on the associated Load Centers A or C for "A" Bus EDG also on B or D for "B" Bus EDG.

> ***NOTE:*** *Undervoltage sensed on A, B, C or D load centers, by 2/2 UV relays on each load center, will also cause bus stripping and start the appropriate EDG. The setpoints and time delays vary in each case (See Logic Sheets for additional details.)*

Auto Start Prerequisites

1. Control Power, at engine panel, must be "ON"

2. Even with control power available, the generator panel-mounted LOCAL-NORMAL-OFF switch must be in "NORMAL" or "LOCAL".



EMERGENCY START/RUN	RAPID START	NORMAL START	IDLE START
THE EDG STARTS IN RESPONSE TO EMERGENCY START MANUAL OR AUTOMATIC SIGNALS AND ACCELERATES TO RATED SPEED (900 RPM) AND VOLTAGE (4160 V) WITHIN 15 SECONDS. IN THIS MODE OF OPERATION THE NON-ESSENTIAL ENGINE AND GENERATOR PROTECTIVE FEATURES ARE BYPASSED. ONLY THE GENERATOR DIFFERENTIAL PROTECTION AND ENGINE OVERSPEED ARE CAPABLE OF AUTOMATICALLY TRIPPING THE EDG. MANUAL OR AUTOMATIC EMERGENCY START SIGNALS WILL TERMINATE IDLE RUN AND PREPOSITION THE GOVERNOR AND VOLTAGE REGULATOR.	THE EDG RAPID START IS PRIMARILY PROVIDED TO VERIFY THE EDG ABILITY TO START AND ACCELERATE TO RATED SPEED (900 RPM) AND VOLTAGE (4160 V) WITHIN 15 SECONDS, WHILE MAINTAINING ALL THE GENERATOR AND ENGINE PROTECTIVE FEATURES IN PLACE. A MANUAL RAPID START SIGNAL WILL TERMINATE IDLE RUN.	THE EDG NORMAL STARTS IN RESPONSE TO MANUAL SIGNALS AND ACCELERATE TO IDLE SPEED (450 RPM) FOR A 60 SECOND PERIOD. AT THE END OF THIS PERIOD THE EDG WILL ACCELERATE TO RATED SPEED (900 RPM) AND VOLTAGE (4160 V). THE 60 SECOND WARM-UP PERIOD WILL BE TERMINATED IF A MANUAL RAPID START SIGNAL OR A MANUAL OR AUTOMATIC EMERGENCY START/RUN SIGNAL ARE RECEIVED, AND THE EDG WILL ACCELERATE TO RATED SPEED AND VOLTAGE.	EDG IDLE START, STARTS AND ACCELERATE THE EDG TO IDLE SPEED (450 RPM) FOR AN INDEFINITE PERIOD. DURING IDLE RUN THE GENERATOR WILL NOT BE FLASHED, THEREFORE NO EDG VOLTAGE WILL BE PRESENT. THE IDLE RUN WILL BE TERMINATED IF A MANUAL RAPID START, MANUAL IDLE RELEASE OR A MANUAL OR AUTOMATIC EMERGENCY START ARE RECEIVED, AND THE EDG WILL ACCELERATE TO RATED SPEED AND VOLTAGE.

NOTES:

1. LOGIC SHOWN FOR DG4A. LOGIC FOR DG4B IS IDENTICAL WITH APPROPRIATE A TO B CHANGES TO COMPONENT/PANEL NUMBERS REFER TO SCHEMATIC DIAGRAMS FOR CHANGES.
2. A SI SIGNAL ON EITHER UNIT, EITHER TRAIN WILL GIVE START SIGNAL TO UNIT 3 & 4 EDG'S.

NOTE: THIS DWG. WAS MADE FROM 5614-T-L1 SH.9A1/87-284 REV 4

NUCLEAR SAFETY RELATED

REV	DATE	REVISION	BY	CH	APP	APP

REV	DATE	REVISION	BY	CH	COR	APP
1	9/29/91	RE-ISSUED AS-BUILT FOR PC/M 87-284	RH	LH	-	FCJ

FPL

TURKEY POINT NUCLEAR UNIT 4

LOGIC DIAGRAM
EDG START

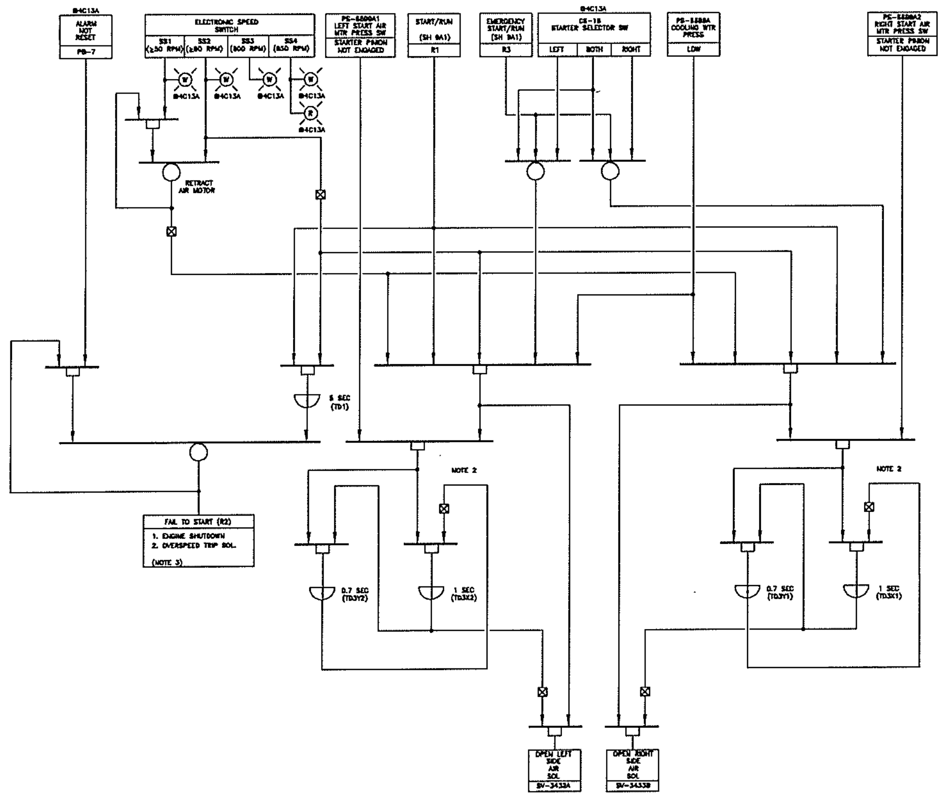
POD

DRAWING NUMBER 5614-T-L1

SHEET 9A1

SYS -

REV 1



NOTES:

1. LOGIC SHOWN FOR DG 4A. LOGIC FOR DG 4B IS IDENTICAL WITH APPROPRIATE A TO B CHANGES TO COMPONENT/PANEL NUMBERS. REFER TO SCHEMATIC DIAGRAMS FOR CHANGES.
2. THERE ARE THREE 1 SECOND START/PINION ENGAGEMENT ATTEMPTS WITHIN THE 5 SECOND START FAIL TIME, WITH A .7 SECOND RESET TIME BETWEEN EACH PINION ENGAGEMENT ATTEMPT
3. FAIL TO START (R2) OCCURS WHEN ENGINE SPEED DOES NOT EXCEED 90 RPM WITHIN 5 SECONDS OF A START SIGNAL. MUST BE LOCALLY RESET.

ELECTRONIC SPEED SWITCH FUNCTIONS	
SS1 50 RPM	1. PERMISSIVE TO TRIP LOCKOUT RELAY 2. PERMISSIVE FOR NORMAL STOP COAST-DOWN TIMER 3. SHUTDOWN PERMISSIVE 4. GOVERNOR AND VOLTAGE REG. PREPOSITION PERMISSIVE 5. AIR START SOLENOIDS PERMISSIVE
SS2 90 RPM	1. AIR START SOLENOIDS PERMISSIVE 2. SHUTDOWN PERMISSIVE
SS3 600 RPM	1. READY TO START INDICATION 2. GENERATOR FIELD FLASH SIGNAL 3. VOLTAGE REGULATOR RESET PERMISSIVE 4. RADIATOR FAN CONTROL 5. CIRC. AND TURBO OIL PUMP CONTROL
SS4 850 RPM	1. IDLE START INTERLOCK OR PERMISSIVE 2. GENERATOR FIELD FLASH PERMISSIVE 3. VOLTAGE REGULATOR RESET PERMISSIVE 4. PERMISSIVE FOR LOW COOLING WATER PRESSURE EMERG SHUTDOWN

EDG MANUAL CONTROL MODES

MODE	OPERATOR	LOCATION	LOCAL MASTER CONTROL SWITCH POSITION	FUNCTIONAL DESCRIPTION
EMERGENCY START	SELECTOR SWITCH	CONTROL ROOM CONSOLE 4C02	N	STARTS EDG AND ACCELERATES TO FULL SPEED, BYPASSING IDLE PERIOD. PROTECTIVE INTERLOCKS ARE BYPASSED.
	PUSHBUTTON	ENGINE CONTROL PANEL	L	
RAPID START	PUSHBUTTON	CONTROL ROOM VERTICAL PANEL 4C02	N	STARTS EDG AND ACCELERATES TO FULL SPEED, BYPASSING IDLE PERIOD. PROTECTIVE INTERLOCKS ARE NOT BYPASSED.
	SELECTOR SWITCH	EDG CONTROL PANEL	L	
NORMAL START	SELECTOR SWITCH	CONTROL ROOM CONSOLE 4C02	N	STARTS EDG AND ACCELERATES TO IDLE SPEED (450 RPM) FOR 80 SECONDS BEFORE RELEASING TO ACCELERATE TO FULL SPEED (900 RPM).
	SELECTOR SWITCH	EDG CONTROL PANEL	L	
IDLE START	SELECTOR SWITCH	EDG CONTROL PANEL	L	STARTS EDG AND ACCELERATES TO IDLE SPEED (450 RPM) FOR INDEFINITE TIME PERIOD OR UNTIL IDLE RELEASE IS ENGAGED, RAPID START IS ACTUATED, OR THERE IS A MANUAL/AUTOMATIC EMERGENCY START SIGNAL.
EMERGENCY STOP	SELECTOR SWITCH	CONTROL ROOM CONSOLE 4C02	N	DECELERATES EDG TO STOP BYPASSING BOTH IDLE INTERVAL AND COAST DOWN TIMER TO IMMEDIATELY STOP THE EDG.
	PUSHBUTTON	EDG CONTROL PANEL	B	
	PUSHBUTTON	EDG SKID	B	
NORMAL STOP	SELECTOR SWITCH	CONTROL ROOM CONSOLE 4C02	N	DECELERATES THE EDG IN THE NON-EMERGENCY MODE FROM FULL SPEED TO IDLE SPEED FOR TWENTY MINUTES, THEN COASTS DOWN TO STOP.
	SELECTOR SWITCH	EDG CONTROL PANEL	L	
IDLE RELEASE	SELECTOR SWITCH	EDG CONTROL PANEL	L	RELEASES THE EDG FROM INDEFINITE IDLE (AS ACTUATED BY ENGAGING IDLE START) AND ALLOWS ACCELERATION TO FULL SPEED. (NOTE: EDG MUST HAVE IDLED FOR A MINIMUM 80 SECONDS).
AUTO START BYPASS	SELECTOR SWITCH	EDG GENERATOR CONTROL PANEL	L	INHIBITS ALL EDG AUTO STARTS (UNDERVOLTAGE, LOOP, OR SI).

LOCAL MASTER CONTROL SWITCH (CS-1) POSITION AS FOLLOWS:

N: SWITCH OPERABLE WHEN MASTER CONTROL SWITCH IS IN "NORMAL" ONLY (REMOTE/CONTROL ROOM CONTROL CAPABILITY)
L: SWITCH OPERABLE WHEN MASTER CONTROL SWITCH IS IN "LOCAL" ONLY
B: SWITCH OPERABLE WHEN MASTER CONTROL SWITCH IS IN EITHER "NORMAL" OR "LOCAL" BUT NOT IN "OFF."

NOTE: THIS DWG. WAS MADE FROM 5614-T-L1 SH.9A2/87-264 REV 4

NUCLEAR SAFETY RELATED

REV	DATE	REVISION	BY	CH	APP	APP
2	8/9/00	ISSUED AS-BUILT PER CRN-M-10066(PC/M 99-051)	RV	RH	BP	BAP
1	9/29/91	RE-ISSUED AS-BUILT FOR PC/M 87-264 & INCORP. CRN-I-0892	RH	LH	-	FCJ

FPL

TURKEY POINT NUCLEAR UNIT 4

LOGIC DIAGRAM
EDG ENGINE START

POD

DRAWING NUMBER 5614-T-L1

SHEET 9A2

SYS -

REV 2

FILE: UTE01757.DWG

Figure 1A – Electrical Distribution: Switchyard and Unit 3

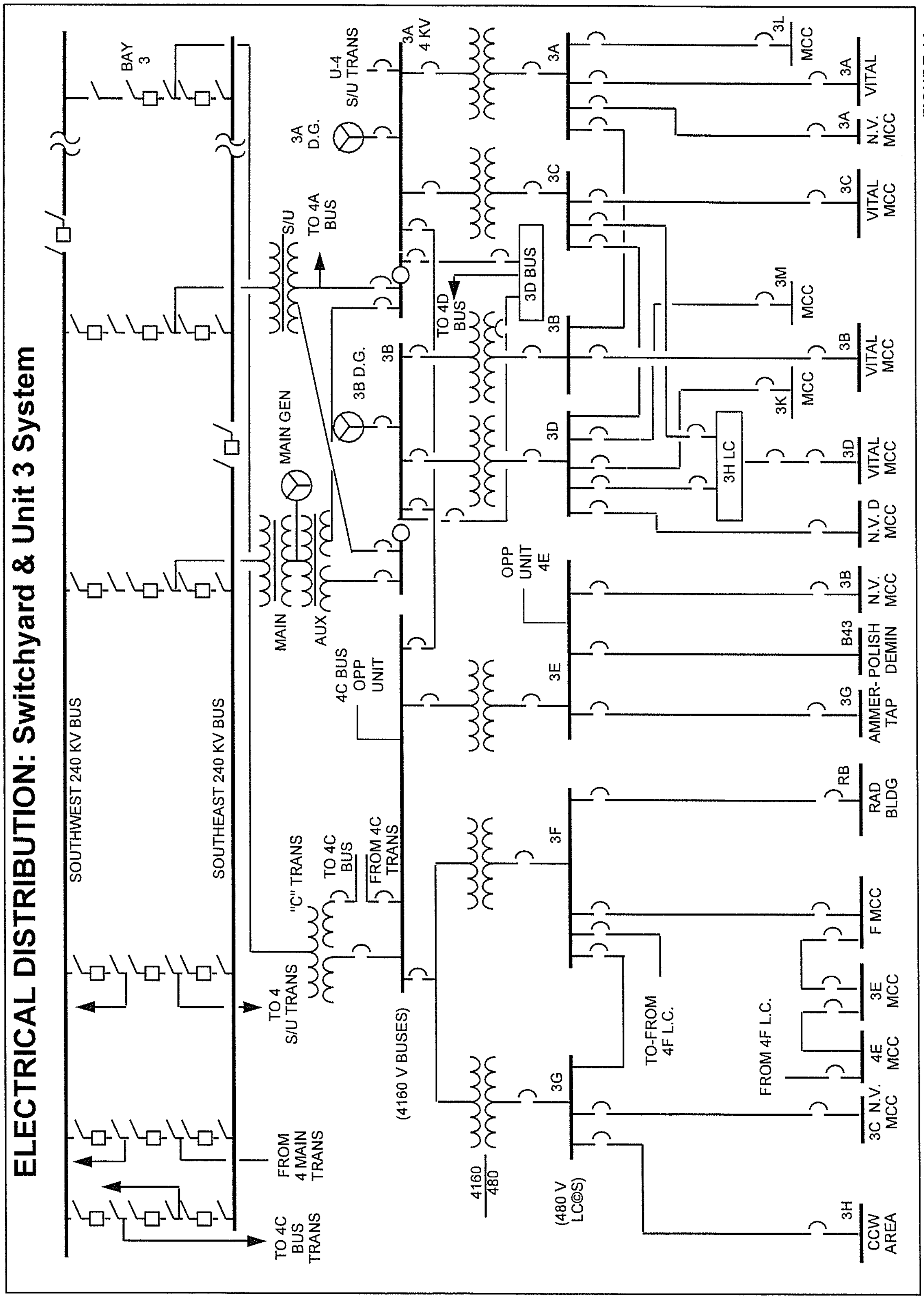


FIGURE 1A
Rev.9:7/18/07

SD 140

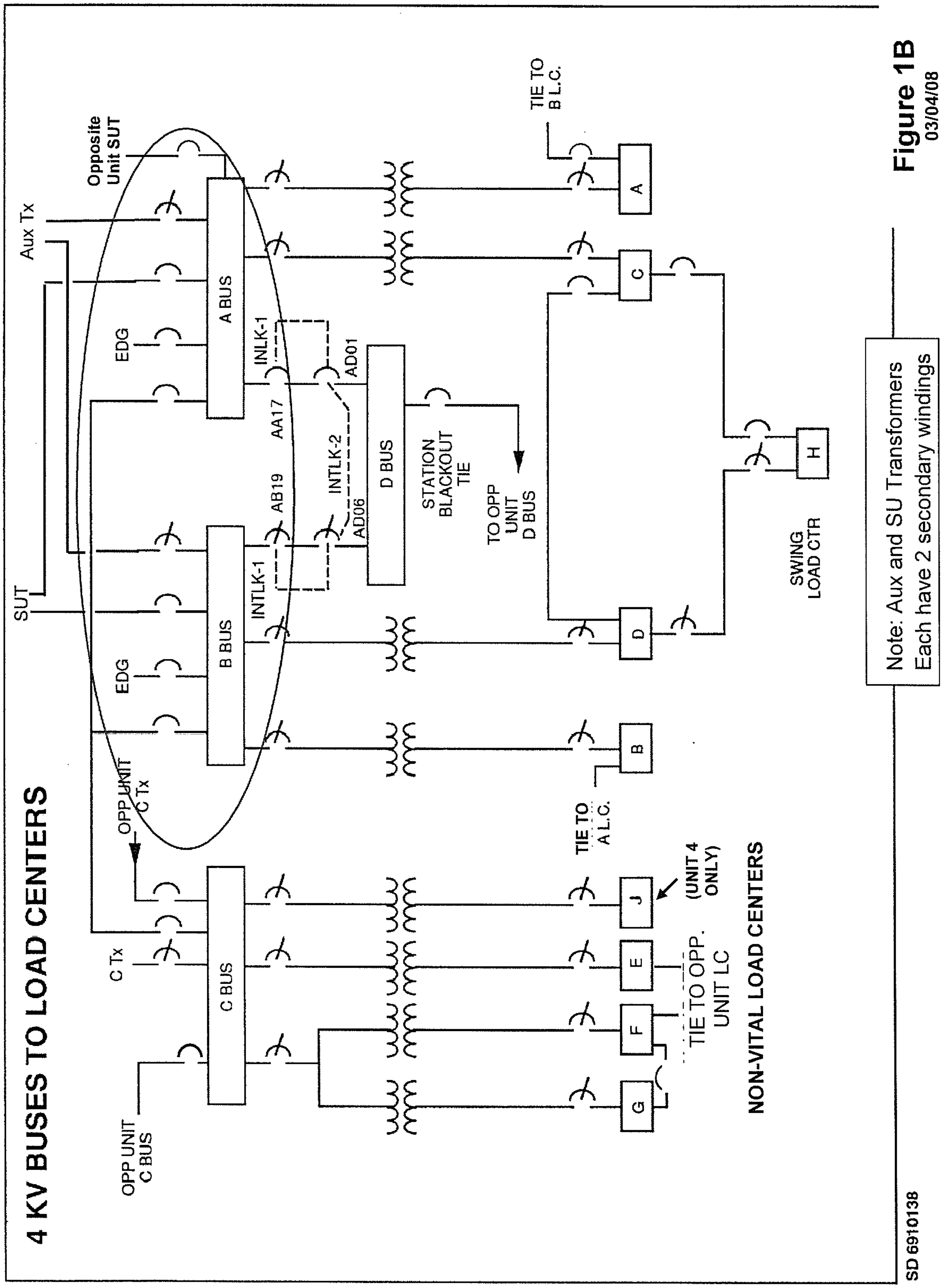
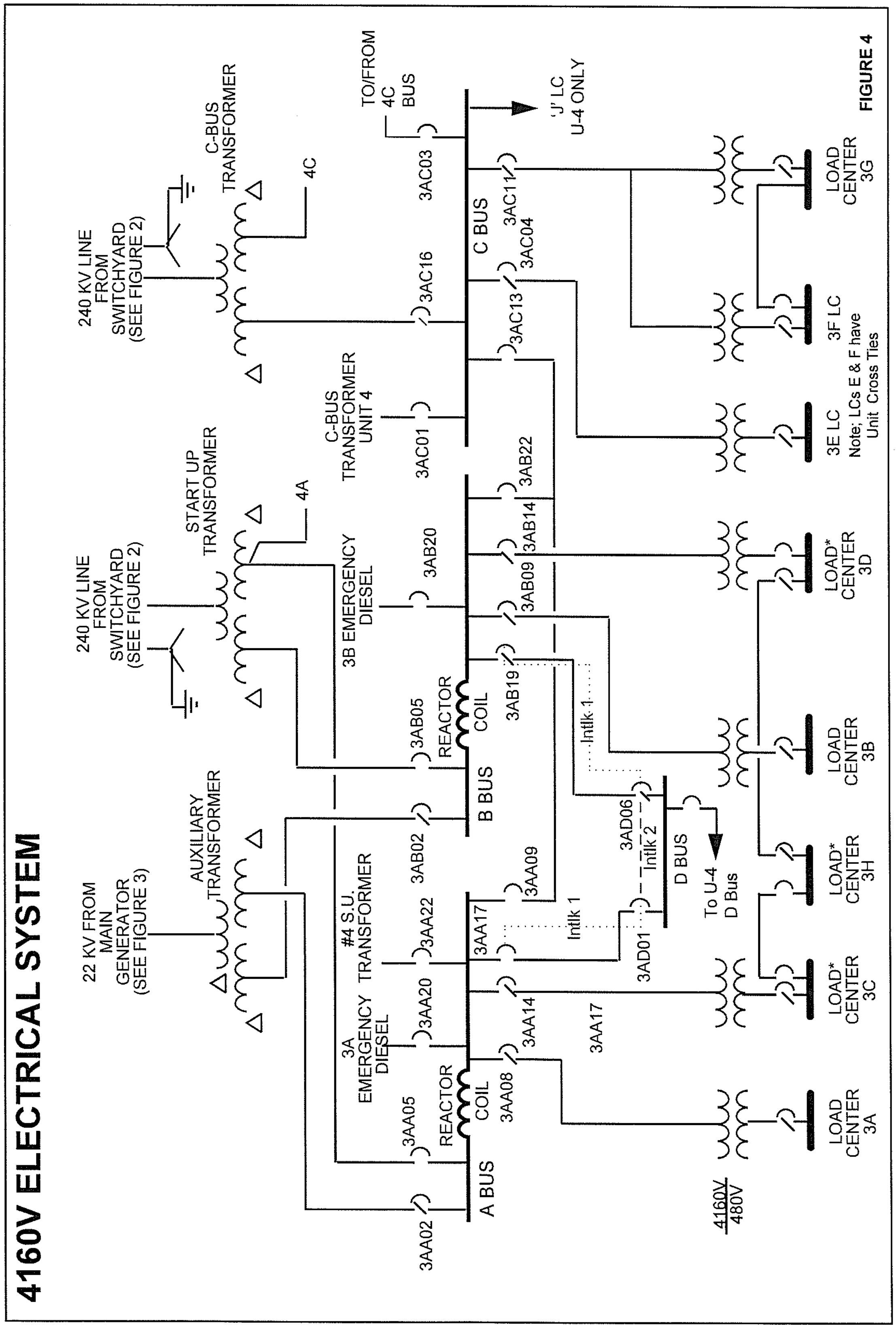


Figure 1B
03/04/08

This simplified drawing is the drawing that Operations personnel are expected to be able to draw.

Figure 4 - 4160V Electrical System



CORRECT ANSWER

A.2.11.5 4.16kV Undervoltage and Underfrequency Trips

4.16KV bus 'A' and 'B' each have voltage sensing relays that will actuate bus associated Emergency Load Sequencer to strip the bus loads. Additionally, the vital 480 volt load centers associated with each bus also have voltage sensing relays that on degraded voltage will input the Emergency Load Sequencer to cause the same bus stripping actions. *Detailed information may be found in System 024, Emergency Load Sequencer Lesson and Logic Sheets 12, 12A, 12B, 13 and 13A.* The undervoltage reactor trip discussion that follows should not be confused with the loss of voltage/undervoltage conditions that initiate bus stripping.

The undervoltage reactor trip and the underfrequency reactor coolant pump trip provide reactor core protection against DNB as a result of a loss of voltage or underfrequency. A low flow trip (<90% flow) is sensed at the elbow of each reactor coolant loop between the steam generators and the reactor coolant pumps. The reactor coolant pump underfrequency trip strips the reactor coolant pump(s) from the affected 4.16kV bus to prevent negating the flywheel action (flow coastdown). Above 45% load (P-8) loss of one RCP trips the reactor. Loss of two pumps between 45% (P-8) and 10% (P-7) load also results in reactor trip.

A.2.11.6 Undervoltage Reactor Trip

The undervoltage condition is sensed on each bus by two undervoltage relays. Bus 'A' undervoltage is sensed by undervoltage relays 127-3A3 & 127-3A4 [127-4A3 & 127-4A4]. Bus 'B' undervoltage is sensed by relays 127-3B3 & 127-3B4 [127-4B3 & 127-4B4].

If an undervoltage condition (71.5% of normal voltage or 2975 volts) is sensed by at least 1 out of 2 relays on both buses 'A' & 'B', the UV relays de-energize. *See 5610-T-L1 sheet 20.* This condition energizes a 1.0 second time delay relay. If the undervoltage condition exists for greater than 1.0 seconds, a reactor trip is generated if power is greater than 10% (P-7).

A.2.11.7 Underfrequency Reactor Coolant Pump Trip

The underfrequency condition is sensed on each bus by two underfrequency relays. Bus 'A' underfrequency is sensed by relays 181-3A1 and 181-3A2 [181-4A1 and 181-4A2]. Bus 'B' underfrequency is sensed by relays 181-3B1 and 181-3B2 [181-4B1 and 181-4B2].

If underfrequency (56.1 Hz) is sensed a trip signal is sent to each reactor coolant pump breaker. *Logic Diagram 5610-T-L1 sheet 20 illustrates the trip scheme.* The loss of reactor coolant pumps results in a reactor trip as detailed in paragraphs below.

A.2.11.8 Reactor Coolant Pump Breaker Position Trip

In addition to the underfrequency and undervoltage trips on the RCPs, a reactor trip generated by RCP breaker position is provided. This is to supply anticipatory reactor core protection against DNB resulting from the opening of one RCP breaker with power above 45% (P-8); or the opening of two or more breakers with power between 10% (P-7) and 45% (P-8). *Reference: 5610-T-L1 sheet 20.*

The initiating logic for a trip signal above 45% (P-8) reactor power is a 1-out-of-3 (1/3) "Breaker Open" logic for a reactor trip. If reactor power is between 10% (P-7) and 45% (P-8) the initiating logic for a reactor trip is 2-out-of-3 (2/3). Below 10% (P-7) reactor power, the RCP breaker position trip is not in effect.

CORRECT ANSWER

A.2.12 Emergency Diesel Generators

The four emergency diesel generators (EDGs), located in two emergency diesel buildings, automatically provide power to 4.16kV safety-related (ESF) buses 'A' and 'B' within 15 seconds following a loss of power to those buses. Should either 'A' or 'B' bus experience a complete loss of voltage or a sustained undervoltage (degraded voltage) condition, as sensed on A, B, C, or D load center, the associated bus Emergency Load Sequencer will strip the buses, send a start signal to its EDG and a close signal to the EDG output breaker, which will close when the bus is cleared.

The EDGs are rated at 4.16kV, 3 phase, 60 Hz, 900 rpm with a continuous load rating of 2500 KW for 3A & 3B EDG and 2874 KW for 4A & 4B EDG.

EDG 3A and 4A supply buses 3A and 4A through breakers 3AA20 and 4AA20, respectively. EDG 3B and 4B supply buses 3B and 4B through breakers 3AB20 and 4AB21, respectively. These breakers are 1200 amp air circuit breakers and are located in their respective switchgear. The breakers are operated either from the console in the control room or locally at the Diesel Generator Panel. A Local-Remote Switch on the 'A' Bus panels will remove breaker control from the control room when placed in the LOCAL position. A Normal/Isolate switch on the 'B' and 'D' bus panels removes breaker control from the control room when placed in the Isolated position.

If the EDG is tied to the bus while the main generator is operating and a main generator lockout occurs, the EDG breakers will trip to prevent overloading the EDGs, but the engine continues to run. However, to allow automatic breaker closing by the Emergency Load Sequencer, this trip signal is removed when the bus stripping relays energize.

In addition to the loss of voltage signals previously mentioned a safety injection signal will directly input EDGs starting circuitry causing all EDGs to start.

Detailed information may be found in System 024, Emergency Load Sequencer Lesson and Logic Sheets 12, 12A, 12B, 13 and 13A.For more information on the diesel generators and auxiliaries see SD-137and logic diagrams, 5610-T-L1, sheets 9A1 - 9A7.

QUESTION 15

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	057	2.4.50
	Importance Rating	4.2	

Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: RO Question # 15

Unit 3 is operating at 100% power when VITAL AC BUS INVERTER TROUBLE (F 1/2) alarms.

Subsequently, the following annunciators are received (NOT all inclusive):

- POWER RANGE LOSS OF DETECTOR VOLTAGE (B 6/5)
- INTERM RANGE N-35 LOSS OF COMP VOLTAGE (B 5/3)
- SEQUENCER 3B TROUBLE (X1/4)

Using the subsequent alarms, which vital panel lost power and what is the INITIAL operator response when SG A NARROW RANGE HI LEVEL (C 2/1) alarms?

A. 3P08 lost power; control 3A S/G level manually

B. 3P08 lost power; ensure an automatic Turbine trip has occurred

C. 3P06 lost power; control 3A S/G level manually.

D. 3P06 lost power; ensure an automatic Turbine trip has occurred

Proposed Answer: C

Explanation (Optional):

A. Incorrect since a failure of panel 3P08 has not occurred. Plausible because the 2nd part is

correct. Also plausible because the VITAL AC BUS INVERTER TROUBLE (F 1/2) and POWER RANGE LOSS OF DETECTOR VOLTAGE (B 6/5) annunciators are common to a failure of both panels.

B. Incorrect since according to 3-ONOP-003.6, Step 5, the implication is that manual control of 3A S/G level is possible and an immediate plant trip is NOT required. Plausible because, if Panel 3P08 had been lost, reducing Feedwater flow on 3C Steam Generator will lead to a Reactor Trip caused by Hi-Hi level on 3A Steam Generator. Also plausible because the VITAL AC BUS INVERTER TROUBLE (F 1/2) and POWER RANGE LOSS OF DETECTOR VOLTAGE (B 6/5) annunciators are common to a failure of both panels.

C. CORRECT. The INTERM RANGE N-35 LOSS OF COMP VOLTAGE (B 5/3) will alarm. S/G level control is affected and if manual actions are not performed a SG A NARROW RANGE HI LEVEL (C 2/1) can occur. This is indicative of a loss of 3P06 vs. 3P08. According to 3-ARP-097.CR.C, 2/1, the first Prompt Action is: "TAKE manual control of level." This is also required by 3-ONOP-003.6, Loss of 120V Vital Instrument Panel 3P06, Step 5.

D. Incorrect since, according to 3-ONOP-003.6, Step 5, the implication is that manual control of 3A S/G level is possible and a plant shutdown is NOT required. Plausible because the 1st part is correct. Plausible because the action for HI-HI SG LVL TURB TRIP / FEEDWATER ISOLATION (E2/6) is to ensure an automatic Turbine trip has occurred.

Technical Reference(s):

3-ONOP-003.8, Loss of 120V Vital Instrument Panel 3P08

3-ONOP-003.6, Loss of 120V Vital Instrument Panel 3P06,

3-ARP-097.CR.C, 2/1, SG A NARROW RANGE HI LEVEL
3-ARP-097.CR.E, 2/6, HI-HI SG LVL TURB TRIP/FEEDWATER ISOLATION

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900260, Obj. 4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-003.6

<u>Title</u>:

Loss of 120V Vital Instrument Panel 3P06

(Continuous Use)

<u>Safety Related Procedure</u>	
Responsible Department:	Operations
Revision Number:	0A
Issue Date:	8/17/10
Revision Approval Date:	8/9/10

ARs *570674*
PCRs *08-1707, 09-3293, 10-1367*
PC/Ms *93-005, 94-034, 95-102, 97-036, 98-025*

W97:/nw/mr/cls

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***Reducing feed flow to less than steam flow by 665,000 lbs/hr will result in a reactor trip due to low level trip logic on Channel 1 of each steam generator.***
- ***Steam Generator 3A level controls are in MANUAL and 3A FW Bypass Valve fails closed.***
- ***3A Steam Generator Level Recorder is DE-ENERGIZED.***
- ***Steam Generator 3C level controls are in AUTO LOCKUP.***
- ***Main Generator load should be maintained as stable as possible until all FW Control Valves are restored to Automatic control.***

NOTE

3B Steam Generator Level Controller should remain in AUTOMATIC.

5 **Control Steam Generator Water Levels As Follows:**

- 3A Steam Generator by manual control of Feedwater flow
- 3C Steam Generator by adjusting the following parameters:
  - Blowdown flow
  - Feed flow
  - Turbine load
  - Steam Flow

Correct Answer (C)

6 **Maintain The Following Plant Parameters - STABLE:**

- Tavg
- Reactor power
- Pressurizer Pressure
- Pressurizer Water level
- Steam Generator Water level

RESPONSE NOT OBTAINED: **IF** any reactor trip setpoint is approached or exceeded, **THEN** manually trip the reactor **AND** perform 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.

ENCLOSURE 1

(Page 1 of 4)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON LOSS OF 3P06

FUNCTIONS, Operating

Lock up of Pressurizer Pressure Controllers causing spray valves to stay as is
Lose Auto and Manual Control of C Feedwater Control Valve, FCV-3-498
Lose Auto Control of A Feedwater Control Valve, FCV-3-478 ← Loss of AUTO ONLY
Lose Auto ~~and Manual 3A Charging Pump Control~~ causing Auto Lock-up
Lose Auto Speed Control of 3B and 3C Charging Pumps
Lose the Auto Makeup Control to the Volume Control Tank
Lose power to Control Relay from MOV-3-115C which opens LCV-3-115B
Letdown Isolation
Pressurizer heaters de-energize
Lose Auto and Manual control of PCV-3-145, Letdown Pressure Controller
Loss of 3B Diesel Load Sequencer, 3C23B-1 deenergized
Lose AMSAC A Processor
Lose the Ability to Block the Source Range Trip
Lose Feedwater Isolation signal (Reactor Trip with Tavg ≤554°F)
Loss of power to hand/auto station for CV-3-1607 which fails closed

NOTES

- *The following conditions exist which affect Pressurizer Pressure control:*
  - *Pressurizer Pressure Controller PC-444J - AUTO LOCKUP*
  - *PZR Spray Valve Controllers - AUTO LOCKUP*
  - *PZR heaters deenergized*
  - *Letdown isolation*
  - *3A charging pump - AUTO LOCKUP*
  - *3B AND 3C Charging pump loss of auto speed control*

ENCLOSURE 1
(Page 2 of 4)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON LOSS OF 3P06

NOTES

- *With vital panel 3P06 deenergized, 3B bus sequencer is out of service resulting in the following Tech Spec implications:*

  1. *AFW actuation from bus stripping on 3B 4KV bus will NOT be generated, placing the unit in a shutdown action statement (Tech Spec 3.3.2, Table 3.3-2, functional unit 6.d action 23 invokes Tech Spec 3.0.3.)*

  2. *Loss of Power signals are lost via the 3B bus sequencer, placing the unit in a shutdown action statement (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 7a, b and c)*

  3. *Bus stripping will NOT automatically occur, 3B EDG will NOT automatically close in on the bus and is out service; actions of Tech Spec 3.8.1.1 apply.*

INDICATORS

TI-3-401	RX Vessel Leak of Temp
TI-3-133	Seal Water Return Temp
TI-3-139	Excess LTDN HX Temp
PI-3-121	Charging Pumps Disch Press
TI-3-123	Regen Hx Outlet Temp
TI-3-141	LTDN Relief To PRT Temp
TI-3-143	Non-Regen HX LTDN Temp
FI-3-150	Low Pressure Letdown Flow Indication
FR-3-154B	#1 Seal Leakoff Recorder Low Range (Fails As Is)
FR-3-154A	#1 Seal Leakoff Recorder High Range (Fails As Is)
PI-3-154	C RCP Seal ΔP
PI-3-128A	B RCP Thermal Barrier ΔP
PI-3-402	RCS Press NR
PI-3-403	RCS Press WR
TI-3-465	Pzr Safety Valve Temp
TI-3-467	Pzr Safety Valve Temp
TI-3-469	Pzr Safety Valve Temp
TI-3-463	PZR Relief Temp
TI-3-452	PZR Spray Loop B Temp
TI-3-451	PZR Spray Loop C Temp
TI-3-412B	A Loop Ovpwr ΔT
TI-3-412A	A Loop ΔT
TI-3-412C	A Loop Ovtemp ΔT

ENCLOSURE 1
(Page 3 of 4)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON LOSS OF 3P06

INDICATORS

TI-3-412D	A Loop Temp Avg
PI-3-455	PZR Press Ch I
LI-3-459A	PZR Level Prot/Cont.
FI-3-414	RCS Flow Loop A
FI-3-424	RCS Flow Loop B
FI-3-434	RCS Flow Loop C
TR-3-412	Delta-T Recorder
NR-3-46	NIS Overpower Recorder
LI-3-474	A Stm Gen Level
LI-3-484	B Stm Gen Level
LI-3-494	C Stm Gen Level
LR-3-477	Stm Gen Wide Range Level (Fails As Is)
FR-3-478	3A Steam Generator Recorder
LI-3-470	Pzr Relief Tk Level
TI-3-471	Pzr Relief Tk Temp
PI-3-472	PZR Relief Tank Pressure
PC-3-444H	Auto Manual Station for Pzr Spray Valve PCV-3-455B
PC-3-444G	Auto Manual Station for Pzr Spray Valve PCV-3-455A
PC-3-444J	Auto Manual Station for Pressurizer Pressure Controller
SC-3-151A	Auto Manual Station Charging Pump A Control
PC-3-145B	Auto Manual Station Low Pressure Letdown Pressure
FC-3-113A	Auto Manual Station Boric Acid to Blend System
NI-3-6649B1/B2	Gammametrics Backup NIS
N-3-31	Source Range Counts and Source Range Startup Rate
N-3-35	Inter Range Current Current and Startup Rate
N-3-41	Power Range and Axial Flux Difference
LC-3-478A	Stm Gen A Control Valve Controller, Lose Inst, MAN Control Only
FCV-3-479	3A FW Bypass Valve
U-3	Pressurizer Safety Valve Acoustic Monitoring System
NIS Rack Ch 1	(N-31, N-35, N-41)
RI-3-6311B	Containment High Radiation
TI-3-610B	CCW Pump Inlet Temp
TI-3-607B	B CCW HX Outlet Temp
FI-3-613B	B CCW HX HDR Flow
RI-3-6311B	Containment High Radiation (VPC)

ALARMS

A 1/1, RCP THERMAL BARR COOLING WATER HI FLOW
A 1/5, RCP SEAL LEAKOFF HI FLOW (C RCP only)
A 6/4, RCP SEAL WATER LO DP (C RCP)
C 6/1, SG A LEVEL DEVIATION
G 9/2 SI PUMP 3B LO SUCTION PRESSURE
H 1/2, SFP HI TEMP
H 7/3, RHR PP A COOLING WATER LO FLOW
H 7/5, CSP A/B COOLING WATER LO FLOW
X 3/6, SI PP COOLING WATER LO FLOW
UNIT 4 G 9/2 SI PUMP 3B LO SUCTION PRESSURE

1.0 PURPOSE

This procedure provides instructions to be followed upon receipt of Loss of 120V Vital Instrument Panel 3P06.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 Indications

2.1.1 Power Range N-41 Failure (NIS Racks Channel I Lights Out)

2.1.2 Loss of Channel I Vital Instrumentation/Indications

2.1.3 Transfer of Feedwater Control from Automatic to Manual for Steam Generator A

2.1.4 Loss of Power to Pressurizer pressure control Auto/Manual Station (auto lockup)

2.1.5 Loss of Power to the Pressurizer Spray Valve Auto/Manual Station (auto lockup)

2.1.6 Loss of Pressurizer Heaters (Control and Backup)

2.1.7 Isolation of CVCS Letdown Flow

2.1.8 Loss of Power to Pressurizer Level Auto/Manual Station (auto lockup)

2.1.9 Loss of Power to 3A Charging Pump Auto/Manual Station (auto lockup)

2.1.10 PCV-3-456 Auto Open Disabled (if in OMS LOW PRESSURE OPS)

2.1.11 Loss of Power to Steam Generator C Auto/Manual Station (auto lockup)

2.2 Alarms

2.2.1 F 1/2, VITAL AC BUS INVERTER TROUBLE

2.2.2 B 6/5, POWER RANGE LOSS OF DETECTOR VOLTAGE

2.2.3 B 7/1, NIS/RPI ROD DROP ROD STOP

2.2.4 C 6/1, SG A LEVEL DEVIATION

2.2.5 A 1/5 RCP SEAL LEAKOFF HI FLOW

2.2.6 A 6/4, RCP SEAL WATER LO DP

2.2.7 A 7/6, RCP C SEAL WATER BYPASS LO FLOW (if CV-3-307 Open)

Correct Answer (C)

Annunciators unique to a loss of Panel 3P06

2.2.8 H 7/5, CSP A/B COOLING WATER LO FLOW

2.2.9 A 1/1, RCP THERMAL BARR COOLING WATER HI FLOW

2.2.10 H 6/2, RHR HX HI/LO FLOW

2.2.11 X 4/1, ARMS HI RADIATION

Correct Answer (C)

Annunciator unique to a loss of Panel 3P06

2.2.12 H 1/2, SFP HI TEMP

2.2.13 G 9/2 SI PUMP 3B LO SUCTION PRESSURE

2.2.14 UNIT 4: G 9/2 SI PUMP 3B LO SUCTION PRESSURE

2.3 General

2.3.1 Loss of the 120V Vital Instrument Panel 3P06 results in a loss of automatic feedwater control, and a loss of power to all channel I instrumentation. ENCLOSURE 1 of this procedure contains a list of instrumentation lost in the Control Room due to the loss of Vital Instrument Panel 3P06.

2.3.2 As with any loss of a vital AC panel, early diagnosis and recovery is of greatest assistance toward unit restoration.

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 Technical Specifications

1. Section 3.3.2, ESFAS Instrumentation
2. Section 3.4.3, Pressurizer
3. Section 3.8.1.1, Diesel Generators
4. Section 3.8.3.1, Onsite Power Distribution
5. Section 3.4.9.3, Overpressure Mitigating Systems

3.1.2 FSAR

1. Section 8.2-7, Electrical

3.1.3 Plant Drawings

1. 5610-E-855, Breaker List

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-003.8

<u>Title</u>:

Loss of 120V Vital Instrument Panel 3P08

(Continuous Use)

<u>Safety Related Procedure</u>	
Responsible Department:	Operations
Revision Number:	0A
Issue Date:	8/17/10
Revision Approval Date:	8/10/10

ARs *570679*
PCRs *08-1709, 09-3296, 09-2756*
PC/M *93-005, 99-048, 02-085*

1.0 **PURPOSE**

1.1 This procedure provides instructions to be followed upon receipt of Loss of 120V Vital Instrument Panel 3P08.

2.0 **SYMPTOMS OR ENTRY CONDITIONS**

2.1 Indications

2.1.1 Power Range N-43 Failure (NIS Racks Channel III Lights Out)

2.1.2 Loss of Channel III Vital Instrumentation/Indications

2.1.3 Transfer of Feedwater Control from Automatic to Manual for Steam Generator C

2.1.4 Loss of Power to 3C Charging Pump Auto/Manual Station (auto lockup)

2.1.5 Possible loss of Pressurizer Heaters (control and backup)

2.1.6 Possible Isolation of CVCS Letdown Flow

2.1.7 PCV-3-455C Auto Open Disabled (if OMS in LOW PRESSURE OPS)

2.1.8 PCV-3-456 Auto Opens (if OMS in LOW PRESSURE OPS)

2.1.9 Loss of Power to Steam Generator A Auto/Manual Station (auto lockup)

2.2 Alarms

2.2.1 F 1/2, VITAL AC BUS INVERTER TROUBLE

2.2.2 B 6/5, POWER RANGE LOSS OF DETECTOR VOLTAGE

2.2.3 B 7/1, NIS/RPI ROD DROP ROD STOP

2.3 General

2.3.1 Loss of the 120V Vital Instrument Panel 3P08 results in a loss of automatic feedwater control, and a loss of power to all channel III instrumentation. Enclosure 1 of this procedure contains a list of instrumentation lost in the Control Room due to the loss of Vital Instrument Panel 3P08.

2.3.2 As with any loss of a vital AC panel, early diagnosis and recovery is of greatest assistance toward unit restoration.

Correct Answer (C)

Indications of a loss of Panel 3P08 are limited to these 3 annunciators.

ENCLOSURE 1
(Page 1 of 3)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON FAILURE OF VITAL INSTRUMENT PANEL 3P08

FUNCTIONS, OPERATING

Loss of Auto and Manual control of a Feedwater Control Valve, FCV-3-478
~~Loss of Auto control of C Feedwater Control Valve, FCV-3-498~~
PRMS MONITORS due to loss of power
- Lose S/G Blowdown, causes Steam Generator level to increase
- Loss of power to R-11 and R-12, initiates Control Room **AND** Containment Ventilation Isolation
- Lose Liquid/Gas Release

Steam dump to condenser valves receive trip open signal but no Arming Signal
Lose Auxiliary Feedwater Train 2 Controllers (3)
3C Charging Pump controller locks up as is
Lose automatic operation of PORV PCV-3-456 (if OMS in normal OPS)
Disarms AMSAC due to loss of PT-446 (after six minute time delay)
Loss of 3B QSPDS (If 3B Inverter and CVT lost)
Possible loss of power to hand/auto station for CV-3-1608 if aligned to 3P08 and would fail closed

Loss of AUTO & MAN

INDICATORS

FI-3-110	Emerg Borate Flow
TI-3-116	VCT Temperature
PI-3-117	VCT Pressure
PI-3-156	A RCP Seal ΔP
TI-3-432B	C Loop OVPWR ΔT
TI-3-432A	C Loop ΔT
TI-3-432C	C Loop OVTEMP ΔT
TI-3-432D	C Loop Temp Avg
PI-3-457	Pzr Pressure
LI-3-461	Pzr Level/Cont Ch III
FI-3-416	A Loop RCS Flow
FI-3-426	B Loop RCS Flow
FI-3-436	C Loop RCS Flow
PI-3-445	Pzr Pressure

ENCLOSURE 1
(Page 2 of 3)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON FAILURE OF VITAL INSTRUMENT PANEL 3P08

INDICATORS (Cont):

PI-3-475	A Stm Gen Pressure CH II
PI-3-485	B Stm Gen Pressure CH III
PI-3-495	C Stm Gen Pressure CH III
LI-3-476	A Stm Gen Level CH III
LI-3-486	B Stm Gen Level CH III
LI-3-496	C Stm Gen Level CH III
FI-3-474	A Stm Gen Stm Flow CH III
FI-3-484	B Stm Gen Stm Flow CH III
FI-3-494	C Stm Gen Stm Flow CH III
FI-3-477	A Stm Gen Feedflow
FCV-3-498	S/G C Feedwater Flow Control Valve Goes To MANUAL
FI-3-487	B Stm Gen Feedflow
FI-3-497	C Stm Gen Feedflow CH III
PI-3-466	STM HDR Press
PI-3-446	First Stage Pressure
SC-3-153	3C Charging Pump Controller
NR-3-45A	Nuclear Instrumentation (NIS) Recorder
HCV-3-137	Excess LTDN Controller
N-3-43	Power Range
N-3-43	Axial Flux Difference
FR-3-498	SG C Feedflow Recorder
FCV-3-499	C S/G Bypass Valve Fails Closed
LI-3-6384B	CST Level
NIS Rack	Channel III N-43
LR-3-6308B	Containment Pressure
LI-3-6583B	RWST Level
HIC-3-758	RHR HX Outlet Flow
PI-3-405	RCS Pressure
PI-3-6425B	Containment Pressure
PI-3-6306B	Cont Pressure
3C05	Status Lights for Channel III Go Out
LI-3-6309B	Cont Sump Level
LI-3-6308B	Containment Level
3C05	Status Lights for Channel IV Go Out

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***Reducing Feedwater flow on 3C Steam Generator will lead to a Reactor Trip caused by Hi-Hi level on 3A Steam Generator.***
- ***Steam Generator 3A level controls are in AUTO LOCKUP.***
- ***3C Steam Generator Level Recorder is de-energized.***
- ***Main Generator load should be maintained as stable as possible until all FW Control Valves are restored to automatic control.***

NOTES

- *3A Steam Generator level will slowly increase due to a loss of PRMS R-19 and Steam Generator blowdown. If S/G Level can NOT be controlled a reactor trip may occur in approximately 10 minutes due to HI S/G Level.*
- *3C Steam Generator level controller is in MANUAL.*
- *3B Steam Generator level controller should remain in AUTOMATIC.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Control 3A S/G Levels As follows:**	
	a. Check power level less than 90%	a. <u>**IF**</u> power is greater than 90% <u>**AND**</u> power will <u>**NOT**</u> be restored to 3PO8 within 10 minutes, <u>**THEN**</u> make preparations to trip the plant.
	b. Control 3A Steam Generator Water Level by Adjusting the Following Parameters: • Feed Flow using FCV-3-479 • Turbine load • Steam Flow	Distractors B & D
	c. Control 3C S/G level using MANUAL control	

1.0 **PURPOSE**

1.1 This procedure provides instructions to be followed upon receipt of Loss of 120V Vital Instrument Panel 3P08.

2.0 **SYMPTOMS OR ENTRY CONDITIONS**

2.1 Indications

2.1.1 Power Range N-43 Failure (NIS Racks Channel III Lights Out)

2.1.2 Loss of Channel III Vital Instrumentation/Indications

2.1.3 Transfer of Feedwater Control from Automatic to Manual for Steam Generator C

2.1.4 Loss of Power to 3C Charging Pump Auto/Manual Station (auto lockup)

2.1.5 Possible loss of Pressurizer Heaters (control and backup)

2.1.6 Possible Isolation of CVCS Letdown Flow

2.1.7 PCV-3-455C Auto Open Disabled (if OMS in LOW PRESSURE OPS)

2.1.8 PCV-3-456 Auto Opens (if OMS in LOW PRESSURE OPS)

2.1.9 Loss of Power to Steam Generator A Auto/Manual Station (auto lockup)

2.2 Alarms

2.2.1 F 1/2, VITAL AC BUS INVERTER TROUBLE

2.2.2 B 6/5, POWER RANGE LOSS OF DETECTOR VOLTAGE

2.2.3 B 7/1, NIS/RPI ROD DROP ROD STOP

2.3 General

2.3.1 Loss of the 120V Vital Instrument Panel 3P08 results in a loss of automatic feedwater control, and a loss of power to all channel III instrumentation. Enclosure 1 of this procedure contains a list of instrumentation lost in the Control Room due to the loss of Vital Instrument Panel 3P08.

2.3.2 As with any loss of a vital AC panel, early diagnosis and recovery is of greatest assistance toward unit restoration.

Correct Answer (C)

Indications of a loss of Panel 3P08 are limited to these 3 annunciators.

QUESTION 16

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	062	AA1.03
Importance Rating	3.6	

Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water: SWS as a backup to the CCWS

Proposed Question: RO Question # 16

Given the following:

- At Unit 3, a loss of Component Cooling Water (CCW) occurs and the operating crew implements 3-ONOP-030, Component Cooling Water Malfunction.
- Emergency cooling water is being aligned to the operating 3A Charging Pump's Oil Cooler per Attachment 1, Control of Emergency Cooling Water to Charging Pumps, of 3-ONOP-030.
- Subsequently, a Loss of Offsite Power occurs and the Diesel Driven Service Water pump cannot be started.

Which ONE of the choices below completes the following statement regarding 3A Charging Pump operation in response to the Loss of Service Water?

In accordance with 3-ONOP-030, the 3A Charging Pump is to be operated at ___(1)___ speed until Attachment 1 is complete. When hydraulic coupling oil temperature (indicated temperature at the oil cooler outlet) reaches 195°F, the required action is to ___(2)___ .

A. (1) MINIMUM
(2) stop 3A Charging Pump

B. (1) MINIMUM
(2) reduce 3A Charging Pump speed if running above MINIMUM speed

C. (1) MAXIMUM
(2) stop 3A Charging Pump

D. (1) MAXIMUM
(2) reduce 3A Charging Pump speed if running above MINIMUM speed

Proposed Answer: C

Explanation (Optional):

A. Incorrect since the charging pump should be operated at maximum speed, not minimum. Plausible because the 2nd part is correct. Also plausible because a novice applicant may believe that it is better to operate at minimum speed because it would generate less heat at minimum speed.

B. Incorrect since the charging pump should be operated at maximum speed, not minimum. Also incorrect since the first temperature requiring a trip is 195°F, not 220°F. Plausible because the 2nd part is the correct value for actual temperature in the hydraulic coupling (see excerpt below). Also plausible because a novice applicant may believe that it is better to operate at minimum speed because it would generate less heat at minimum speed.

<u>Reference</u>
3-ONOP-030, Attachment 1

NOTE
Maximum charging pump oil temperature is 220°F to prevent oil break down.

C. CORRECT.

<u>Reference</u>
3-ONOP-030, Attachment 1

D. Incorrect since the first temperature requiring a trip is 195°F, not 220°F. Plausible because the 2nd part is the correct value for actual temperature in the hydraulic coupling (see excerpt below). Also plausible because the 1st part is correct.

<u>Reference</u>
3-ONOP-030, Attachment 1

NOTE
Maximum charging pump oil temperature is 220°F to prevent oil break down.

Technical Reference(s): 3-ONOP-030, Component Cooling Water Malfunction
LP 6902229, Component Cooling Water Malfunction
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	LP 6902229, Obj. 4, 8	(As available)
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge (1P)	
	Comprehension or Analysis	
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41 7	
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-030

<u>Title</u>:

Component Cooling Water Malfunction

(Continuous Use)

<u>Safety Related Procedure</u>	
Responsible Department:	Operations
Revision Number:	2
Issue Date:	6/15/11
Revision Approval Date:	3/25/11

ARs *580245*
PCRs *08-3724, 08-2161, 10-0619*
OTSCs *10906, 0543-00, 0301-02, 0195-03*
PC/Ms *91-064, 92-031, 92-108, 93-034, 94-096, 96-092, 96-093, 00-016, 04-112, 06-018*

ATTACHMENT 1
(Page 1 of 5)

CONTROL OF EMERGENCY COOLING WATER TO CHARGING PUMPS

NOTES

- *Emergency cooling water SUPPLY hose has a quick disconnect fitting on one end and a cam lock fitting on the other end.*
- *Loss of off-site power in coincidence with a loss of CCW will require the diesel driven service water pump to be in service in order to provide emergency cooling water to the charging pumps.*

1. Connect cam lock fitting end of emergency cooling water supply hose to Service Water Connection Inside Unit 3 Charging Pump Room, 3-70-179A.
2. Consult with Unit 3 Reactor Operator to determine desired charging pump.
3. Verify desired charging pump is stopped **OR** running at maximum speed.

Correct Answer

4. Connect quick disconnect fitting end of emergency cooling water supply hose to emergency hose connection on desired charging pump.
   a. Emergency Hose Connection to Charging Pump A Oil Cooler, 3-10-291

   OR

   b. Emergency Hose Connection to Charging Pump B Oil Cooler, 3-10-289

   OR

   c. Emergency Hose Connection to Charging Pump C Oil Cooler, 3-10-299

NOTE

Emergency cooling water OUTLET hose has a quick disconnect fitting on one end and no fitting on the other end.

5. Connect quick disconnect fitting end of emergency cooling water outlet hose to emergency hose connection on desired charging pump.
   a. Emergency Hose Connection to Charging Pump A Oil Cooler, 3-10-290

   OR

   b. Emergency Hose Connection to Charging Pump B Oil Cooler, 3-10-288

   OR

   c. Emergency Hose Connection to Charging Pump C Oil Cooler, 3-10-298

ATTACHMENT 1
(Page 2 of 5)

CONTROL OF EMERGENCY COOLING WATER TO CHARGING PUMPS

6. Remove cover from floor drain to be used in Charging Pump Room.

7. Route open end of emergency cooling water outlet hose to floor drain being used in Charging Pump Room.

8. Isolate CCW to hydraulic oil cooler on desired charging pump:

a. Close CCW to A Charging Pump Oil Cooler Inlet, 3-825A

OR

b. Close CCW to B Charging Pump Oil Cooler Inlet, 3-825C

OR

c. Close CCW to C Charging Pump Oil Cooler Inlet, 3-825E

9. Isolate CCW from hydraulic oil cooler on desired charging pump:

a. Close CCW from A Charging Pump Oil Cooler Inlet, 3-825B

OR

b. Close CCW from B Charging Pump Oil Cooler Inlet, 3-825D

OR

c. Close CCW from C Charging Pump Oil Cooler Inlet, 3-825F

10. Open Service Water Connection Inside Unit 3 Charging Pump Room Root Valve, 3-70-179.

11. Open Service Water Connection Inside Unit 3 Charging Pump Room, 3-70-179A.

12. Establish service water to desired Charging Pump:

a. Open Emergency Hose Connection to Charging Pump A Oil Cooler, 3-10-291

OR

b. Open Emergency Hose Connection to Charging Pump B Oil Cooler, 3-10-289

OR

c. Open Emergency Hose Connection to Charging Pump C Oil Cooler, 3-10-299

ATTACHMENT 1

CONTROL OF EMERGENCY COOLING WATER TO CHARGING PUMPS

13. Adjust service water flow from desired charging pump to provide maximum flow.

a. Open Emergency Hose Connection to Charging Pump A Oil Cooler, 3-10-290

OR

b. Open Emergency Hose Connection to Charging Pump B Oil Cooler, 3-10-288

OR

c. Open Emergency Hose Connection to Charging Pump C Oil Cooler, 3-10-298

14. **IF** service water flow is not obtained, **THEN** have the Service Water System placed in service using 0-NOP-012, SERVICE WATER SYSTEM, using any available pump including the diesel driven SWP D.

15. Notify Unit 3 Reactor Operator that emergency cooling water has been established to desired charging pump.

NOTE

Maximum charging pump oil temperature is 220°F to prevent oil break down. The installed temperature indicators only indicate up to 200°F. Some indicators are located on the cooler inlet and others on the cooler outlet. Maximum expected ΔT across the cooler is 20°F. At 195°F on the cooler outlet (oil to the hydraulic coupling), this would equate to 215°F on the cooler inlet (oil from the hydraulic coupling).

16. Monitor oil temperatures on running charging pump.

17. **IF** hydraulic coupling oil temperature on running charging pump exceeds 195°F, **THEN** perform the following:

a. Notify Unit 3 Reactor Operator that operating charging pump should be stopped.

b. Consult with Unit 3 Reactor Operator to determine if emergency cooling water should be realigned to a different charging pump.

c. **IF** Unit 3 Reactor Operator determines that emergency cooling water must be realigned to a different charging pump, **THEN** go to Step 20 of this attachment.

18. **IF** Unit 3 Reactor Operator determines that emergency cooling water to charging pumps is no longer required, **THEN** go to Step 20 of this attachment.

19. Return to Step 16 of this attachment.

20. Verify charging pump being supplied with emergency cooling water is stopped.

QUESTION 17

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	065	2.4.21
	Importance Rating	4.0	

Emergency Procedures / Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: RO Question # 17

Given the following:

- Unit 3 is in MODE 4, cooling down to MODE 5.
- 3B RCP is running.
- ONE train of Unit 3 RHR is operating in the cooldown mode.
- Unit 4 is operating at 100% power.

Subsequently:

- A dual unit loss of instrument air occurs and the operating crews implement 0-ONOP-013, Loss of Instrument Air.
- Attempts to start Service Air compressors and the Temporary Diesel Instrument Air Compressor have been unsuccessful.
- Instrument Air pressure is slowly lowering.
- Unit 3 RCS temperature is 280°F and lowering.
- Unit 4 Main Feed Flow to S/Gs is lowering.

In accordance with 0-ONOP-013, which ONE of the following identifies the action(s) to take for controlling the Unit 3 cooldown and for operating Unit 4?

	UNIT 3	**UNIT 4**
A.	Start / Stop RHR pumps	Trip the Reactor
B.	Start / Stop RHR pumps	Take MANUAL control of Main Feed Reg. Valves and restore S/G levels to program
C.	Close MOV-3-749A/B, RHR Hx 3A/B CCW Outlet	Trip the Reactor

D.	Close MOV-3-749A/B, RHR Hx 3A/B CCW Outlet	Take MANUAL control of Main Feed Reg. Valves and restore S/G levels to program

Proposed Answer: A

Explanation (Optional):

A. CORRECT.

0-ONOP-013, Foldout Page:

A dual unit loss of instrument air (less than 60 psig) results in the loss of the additional functions:

- 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.

Maintain Instrument Air Available To The Auxiliary Building Greater Than 65 PSIG Perform the following on both Unit 3 and Unit 4.

IF operating, THEN trip Unit 3 and Unit 4 reactors AND perform the appropriate EOP-E-0, Reactor Trip or Safety Injection while continuing with this procedure.

B. Incorrect since Unit 4 must be tripped, not maintained at program. Plausible because the 1st part is correct. Also plausible because the 2nd part is similar to one of the actions (for AFW FCVs) in the Foldout Page for 0-ONOP-013 if IA pressure had just gone below 90 psig. The main FRVs are also discussed in the Foldout Page; therefore, it is plausible that a novice applicant could combine the two (AFW FCVs in MAN & discussion of Main FRVs) into one action.

0-ONOP-013, Foldout Page:

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):

- Feedwater Reg Valves
- Feedwater Bypass Valves

C. Incorrect since it is not proceduralize to close MOV-3-749A/B, RHR Hx 3A CCW Outlet, on a loss of Instrument Air to RHR. Plausible because the 2nd part is correct. Also plausible because closing MOV-3-749A/B, RHR Hx 3A/B CCW Outlet would eventually stop the cooldown.

D. Incorrect since it is not in procedure to close MOV-3-749A/B, RHR Hx 3A CCW Outlet, on a loss of Instrument Air to RHR. Also incorrect since Unit 4 must be tripped, not maintained at program. Plausible because the 2nd part is similar to one of the actions (for AFW FCVs) in the Foldout Page for 0-ONOP-013 if IA pressure had just gone below 90 psig. The main FRVs are also discussed in the Foldout Page; therefore, it is

plausible that a novice applicant could combine the two (AFW FCVs in MAN & discussion of Main FRVs) into one action. Also plausible because closing MOV-3-749A/B, RHR Hx 3A/B CCW Outlet would eventually stop the cooldown.

Technical Reference(s):	0-ONOP-013, Loss of Instrument Air LP 6902286, Loss of Instrument Air	(Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective:	LP 6902286, Obj. 7	(As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	(3SPK)

Question Difficulty Level: B

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that, during a Loss of IA, it tests knowledge of parameters (IA pressure and RCS temperature) related to safety systems (RHR) during implementation of abnormal procedures (3-ONOP-013, Loss IA).

Florida Power & Light Company

Turkey Point Nuclear Plant

0-ONOP-013

Title:

Loss of Instrument Air

(Continuous Use)

Non-Safety Related Procedure	
Responsible Department:	Operations
Revision Number:	1
Issue Date:	6/22/11
Revision Approval Date:	6/21/11

ARs *564767*
PCRs *09-0159, 10-2475*
PC/Ms *85-135, 88-245, 89-119, 85-135, 92-031, 93-109, 95-133, 07-072, 08-004*

W97:JWG/ln/emc/cls

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13	**Maintain Instrument Air Available To The Auxiliary Building Greater Than 65 PSIG**	Perform the following on both Unit 3 and Unit 4. 1. **IF** operating, **THEN** trip Unit 3 and Unit 4 reactors **AND** perform the appropriate EOP-E-0, Reactor Trip or Safety Injection while continuing with this procedure. 2. Locally align the Charging Pump Suction from the RWST by performing the following: a. Locally open the Charging Pump Suction from RWST, 3-358 **AND** 4-358. b. Hold close control switch for LCV-3-115C **AND** LCV-4-115C until its associated breaker is opened. c. Locally open breakers 30669 **AND** 40669. d. **IF** Letdown orifice(s) is (are) still aligned for service, **THEN** isolate the in service orifice(s). e. **IF** RCS cooldown in progress, **THEN** monitor RCS cooldown **AND** start or stop the RHR Pump(s) as necessary to maintain the cooldown/heatup within limits f. Verify Reactor Coolant Pump Thermal Barrier Isolation Valves, MOV-3-626 **AND** MOV-4-626 are OPEN to maintain RCP seal cooling. g. Start and stop Charging Pump(s) as necessary to maintain Pressurizer level between 22 and 50 percent.

FOLDOUT PAGE

UNIT TRIP CRITERIA

IF a loss of Instrument Air results in any of the following, **THEN** manually trip the reactor on the affected unit(s) **AND** enter the appropriate EOP network while continuing to restore Instrument Air to the affected unit(s).

- Instrument Air System cannot be maintained greater than 65 psig. CORRECT ANSWER
- Required isolation of any of the following air headers:
  - Auxiliary Building
  - Containment Building
  - Turbine Building
- Inability to maintain S/G levels **AND** Instrument Air Compressors are unable to restore pressure.

MAJOR COMPONENT IMPACTS

1. Instrument Air pressure 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
   - Train 2 of AFW (same unit), operate FCV's in MANUAL to conserve N_2. Distractors B+D
   - Train 1 of AFW (opposite unit), operate FCV's in MANUAL to conserve N_2.
   - For dual unit loss of Instrument Air, operate all AFW FCVs (both units) in MANUAL to conserve N_2.
   - Unit 3 EDG Fuel Oil Transfer capability (Unit 3 Only)
2. Instrument Air pressure less than 65 psig results in the loss or partial loss of function depending on the spring bench setting of the following valve(s), in addition to those functions already lost in Step 1 above:
   - Letdown Isolation Valves
   - Feedwater Reg Valves
   - Feedwater Bypass Valves
   - Unit 3A and B Emergency Diesel Generator Day Tank Level Control Valve
3. Instrument Air pressure less than 60 psig results in the loss of the following functions, in addition to those functions already lost in Steps 1 and 2 above:
   - 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 18

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E05	EK2.1
	Importance Rating	3.7	

Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO Question # 18

Unit 3 has experienced a unit trip and safety injection with a failure of Auxiliary Feedwater to supply feed flow to S/Gs. The crew has entered 3-EOP-FR-H.1, Response to Loss of Secondary Heat Sink.

Other plant conditions are as follows:

<u>Containment Conditions</u>

- Atmospheric Air Temperature: 155°F
- Pressure: 0.5 psig
- Radiation Levels: Have increased slightly post trip

<u>S/G Wide Range (WR) Levels</u>

- 3A S/G: 24%
- 3B S/G: 34%
- 3C S/G: 25%

In accordance with 3-EOP-FR-H.1, which ONE of the following describes (1) the MINIMUM required actions to initiate Main Feedwater flow after resetting the Safety Injection Signal to the S/Gs and (2) if any flow restrictions apply?

A. (1) Start a Main Feedwater Pump and open Feedwater Bypass Valves 5-10%
(2) Main Feedwater flow may be set to desired flow rate

B. (1) Push Feedwater Bypass Isolation Reset, open Feedwater Bypass Valves 5-10% and start a Main Feedwater Pump
(2) Main Feedwater flow may be set to desired flow rate

C. (1) Start a Main Feedwater Pump and open Feedwater Bypass Valves 5-10%
(2) Main Feedwater flow is limited to 25 gpm

D. (1) Push Feedwater Bypass Isolation Reset, open Feedwater Bypass Valves 5-10% and start a Main Feedwater Pump
(2) Main Feedwater flow is limited to 25 gpm

Proposed Answer: B

Explanation (Optional):

A. Incorrect because the sequence is wrong for starting the MFP, then opening the Main Feedwater Isolation Bypass Valves, Also, the resetting of the Feedwater Bypass Isolation Reset is omitted. The applicant believes this signal is reset when Safety Injection resets. Plausible because this is the logic for the Feedwater Isolation Signal. Also, 2nd part is correct.

B. CORRECT. 3-EOP-FR-H.1 performs the following with NO flow restrictions: Reset SI, Push Feedwater Bypass Isolation Reset, open Feedwater Bypass Valves 5-10% and start a Main Feedwater Pump

C. Incorrect because the sequence is wrong for starting the MFP, then opening the Main Feedwater Isolation Bypass Valves, Also, the resetting of the Feedwater Bypass Isolation Reset is omitted. The applicant believes this signal is reset when Safety Injection resets. Plausible because this is the logic for the Feedwater Isolation Signal. Also, 2nd part is incorrect, but would be the action taken in 3-EOP-ECA-2.1.

D. Incorrect. Plausibility - 1^{st} part is correct. 2nd part is incorrect, but would be the action taken in 3-EOP-ECA-2.1.

Technical Reference(s): 3-EOP-FR-H.1, Response to Loss of Secondary Heat Sink. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902337, Obj. 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests how feedwater is manually controlled during a Loss of Secondary Heat Sink.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-EOP-FR-H.1

Title:

Response to Loss of Secondary Heat Sink

(Continuous Use)

Safety Related Procedure

Responsible Department:	Operations
Revision Number	4
Issue Date	8/10/11
Revision Approval Date:	8/8/11

ARs *574252, 1617263, 1672876*
PCRs *08-4206, 08-5812, 10-2031*
PC/Ms *87-025, 87-194, 87-213, 87-264, 88-345, 90-440, 90-524, 94-059 95-028, 96-022, 04-145, 07-008*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- *If a loss of offsite power occurs or the opposite unit experiences a SI signal after SI has been reset, manual actions may be required to restore safeguards equipment to the required configuration.*
- *Low range flow indication is NOT available when using main feedwater instrumentation and an alternate source of feedwater. Changes in RCS temperature and S/G level may be used to control feedwater flow.*
- *Feed flow is required to be initiated at a rate NOT to exceed 100 gpm to avoid excessive RCS cooldown and to limit thermal stress in the S/G(s) with wide range level less than 10%[33%]. This 100 gpm feed flow limit applies to each individual S/G and only while that S/G wide range level is less than 10% [33%].*

4 **Try To Establish Main Feedwater Flow To At Least One S/G**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Verify SI - RESET	a. Reset SI.
CORRECT ANSWER b. Reset feedwater bypass isolation using pushbuttons on VPB	Distractor A
c. Manually **OR** locally open feedwater bypass valve(s) between 5% and 10%	c. **IF** no feedwater bypass valve can be opened, **THEN** go to Step 10.
d. Check condensate system – IN SERVICE	d. Try to place condensate system in service. **IF** condensate system can **NOT** be placed in service, **THEN** go to Step 6.
e. Establish main feedwater flow 1) Start S/G feed pumps 2) Adjust feedwater bypass valves to desired flow	e. **IF** main feedwater flow can **NOT** be established, **THEN** go to Step 6.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-EOP-ECA-2.1

Title:

Uncontrolled Depressurization of All Steam Generators

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Number	4
Issue Date	8/10/11
Revision Approval Date:	8/8/11

ARs *564941, 1617263, 1672328*
PCRs *08-4390, 08-4390, 10-1667*
PC/Ms *87-025, 87-264, 90-440, 90-524, 95-028, 96-022, 08-006, 07-008*

W2002/cls/cls/mr

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- *A minimum feed flow of 25 gpm must be maintained to each S/G with a narrow range level less than 6%[32%].*
- *Low range flow indication is NOT available when using main feedwater instrumentation and an alternate source of feedwater. Changes in RCS temperature and S/G level can be used to control feedwater flow.*
- *Feed flow is required to be initiated slowly to avoid excessive RCS cooldown and to limit thermal stress in S/Gs.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Control Feed Flow To Minimize RCS Cooldown**	
	a. Check cooldown rate in RCS cold legs - LESS THAN 100°F/HR	a. Decrease feed flow to 25 gpm to each S/G. Go to Step 3c.
	b. Check narrow range level in all S/Gs - LESS THAN 50%	b. Control feed flow to maintain narrow range level less than 50% in all S/Gs.
	c. Check RCS hot leg temperatures - STABLE OR DECREASING	c. Control feed flow or dump steam to stabilize RCS hot leg temperatures. <u>**IF**</u> adequate feed flow to stabilize hot leg temperatures or 345 gpm is <u>**NOT**</u> available, <u>**THEN**</u> go to 3-EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, Step 1.
4	**Monitor If RCPs Should Be Stopped**	
	a. RCPs - ANY RUNNING	a. Go to Step 5.
	b. High-head SI pumps - AT LEAST ONE RUNNING	b. Go to Step 5.
	c. RCS subcooling - LESS THAN 25°F[65°F]	c. Go to Step 5.
	d. Stop all RCPs	
5	**Check CST Level - GREATER THAN 10%**	Add makeup to CST using 3-NOP-018.01, CONDENSATE STORAGE TANK (CST).

Distractor C + D

QUESTION 19

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	003	AK1.02
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: Effects of turbine-reactor power mismatch on rod control

Proposed Question: RO Question # 19

Given the following:

- Unit 4 is operating at 40% power.
- Tavg is 558°F and stable.
- Tref is 559°F and stable.
- All control systems are aligned in automatic.
- Control Bank D Control Rods are at 150 steps.

Which ONE of the following describes the plant response after a dropped rod and the required operator action in accordance with 4-ONOP-028.3, Dropped RCC?

A. The difference between Tavg and Tref is smaller.
Rods are placed in Manual.

B. The difference between Tavg and Tref is smaller.
Rods are left in Auto.

C. The difference between Tavg and Tref is larger.
Rods are placed in Manual.

D. The difference between Tavg and Tref is larger.
Rods are left in Auto.

Proposed Answer: C

Explanation (Optional):

A. Incorrect Plausibility – 1st part is incorrect. However, the applicant understands when Tavg lowers, then Pstm going to the Turbine lowers (Tref). The effect of the lowering Tavg will be greater than Tref. Also plausible because the immediate action for a Dropped RCC is to place the rods in MAN.

B. Incorrect Plausibility – 1st part is incorrect. However, the applicant understands when Tavg lowers, then Pstm going to the Turbine lowers (Tref). The effect of the lowering Tavg will be greater than Tref. Also plausible because the rod control demand is to have rods step out, but due to this control being disabled the rods do not move. Without rod movement, the applicant leaves rods in Auto which is incorrect.

C. CORRECT. On a dropped rod, Tavg will lower. Rods are placed in manual to stabilize the plant and ensure SDM is maintained

D. Incorrect Plausibility – 1st part is correct. Also, plausible because the rod control demand is to have rods step out, but due to this control being disabled the rods do not move. Without rod movement, the applicant leaves rods in Auto which is incorrect.

Technical Reference(s): LP 6900105, Full Length Rod Control (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900105, Obj. 10 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3PEO)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests the impact that a dropped rod has on the power mismatch circuit of the Full Length Rod Control System.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-028.3

C A U T I O N
Performance of this procedure may affect core reactivity.

Title:

Dropped RCC

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Number:	3
Issue Date:	2/1/11
Revision Approval Date:	1/27/11

ARs *590548, 1611268*
PCRs *08-5732, 09-0989, 10-0931*
PC/Ms *93-005, 09-006*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If this procedure was entered as a result of performing either 3-PTP-028.2, ROD POSITION INDICATION SYSTEM REPLACEMENT TESTING, PHASE 2, MODE 3 TESTS, or 3-SMI-028.03, RPI HOT CALIBRATION, CRDM STEPPING TEST AND ROD DROP TEST, AND the RCS boron concentration is greater than or equal to test requirements, this procedure shall NOT be performed.

NOTES

- *Foldout page is required to be monitored throughout this procedure.*
- *Misaligned rods are addressed by 3-ONOP-028.1, RCC Misalignment.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check Number Of RCCs DROPPED - More Than One**	
	a. Check the following * More than one rod bottom light - ON **OR** * More than one rod position indicator – AT ZERO	a. Go to Step 2.
	b. Manually trip the Reactor	
	c. Go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION	
2	**Check Reactor In Mode 1**	**IF** in Mode 2 or Mode 3, **THEN** perform the following: a. Stop all rod motion and dilution. b. Ensure reactor power is stable or decreasing. 1) **IF** startup rate is positive and/or reactor power is increasing, **THEN** manually trip the Reactor and go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION. c. Commence boration to the ARO Hot Standby RCS boron concentration specified in the Plant Curve Book, Section 3, Figure 7.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Place Rod Motion Control Selector Switch To MANUAL**	CORRECT ANSWER

CAUTIONS

- ***Do NOT dilute the RCS while performing this procedure until the SHUTDOWN MARGIN calculation has been performed using 0-OP-028.2, SHUTDOWN MARGIN CALCULATION.***
- ***Do NOT increase reactor power while performing 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
4	**Verify Automatic Controls Are Functioning To Stabilize The Unit <u>AND</u> No Transient Is 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.
5	**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 as defined in the Plant Curve Book, Section 5, Figure 1	Within 30 minutes, reduce reactor power to less than 50% using 3-ONOP-100, FAST LOAD REDUCTION, while continuing with this procedure.
6	**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%**	
7	**Declare The Dropped RCC Inoperable**	

QUESTION 20

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	005	AA2.03
	Importance Rating	3.5	

Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable

Proposed Question: RO Question # 20

Given the following:

- Unit 4 is at 88% power.
- During a load reduction, it was determined that 2 rods are mechanically bound.
- One Control Rod in Bank D Group 1 is stuck at 196 steps.
- One Control Rod in Bank D Group 2 is stuck at 196 steps.
- All other Control Bank D Rods are at 192 steps.

In accordance with Technical Specification 3.1.3.1, Movable Control Assemblies - Group Height, which ONE of the following describes the action required within ONE hour for the stuck rods?

A. Determine TS 3.1.1.1, SHUTDOWN MARGIN – Tavg GREATER THAN 200°F, is satisfied.

B. Align the remainder of rods in the affected bank with the stuck rods.

C. Determine that QPTR requirements are satisfied or enter the applicable action statement.

D. Verify that the Axial Flux Difference is within the limits specified in the Core Operating Limits Report.

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Technical Specifications 3.1.3.1, *Movable Control Assemblies - Group Height*, Action a, requires SDM verification within 1 hour.

B. Incorrect since a load reduction is in progress, rods are moving in. Would not withdraw remainder of rods to align with rods that are stuck out

C. Incorrect since a QPTR verification is not required for an inoperable rod. Plausible since rod misalignment can cause power peaks within the core and severe misalignment can cause QPTR to exceed the limits of TS 3.2.4.

D. Incorrect since an AFD verification is not required for an inoperable rod. Plausible since, per 0-ADM-536, *Technical Specification Bases Control Program*, Page 37, rod alignment is related to AFD. Also plausible since some AFD T.S. Actions are less than 1 hour.

Technical Reference(s): Tech Spec 3.1.3.1, *Movable Control Assemblies - Group Height* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902521, Obj. 3 (As available)

Question Source: Bank #

Modified Bank # 67498 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 5

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

2008 DC Cook significantly modified

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:

CORRECT ANSWER

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.

5.0 **SUBSEQUENT ACTIONS**

5.1 Immovable RCC

5.1.1 **DO NOT** increase reactor power without permission from the Reactor Engineering Supervisor and the Shift Manager.

5.1.2 Maintain steady state condition as follows:

1. Maintain Tavg equal to Tref.

    a. Borate/dilute as necessary

    OR

    b. Change turbine load as necessary.

2. **IF** possible, **THEN** avoid insertion of the control rods.

5.1.3 Notify the following:

1. Reactor Engineering Supervisor or designee.
2. I&C Supervisor to verify RPI indication and to investigate CRDM System for possible failure.

5.1.4 **IF** one or more RCC is inoperable due to being immovable because of excessive friction or mechanical interference **OR** known to be untrippable, **THEN** proceed as follows:

1. Determine that the shutdown requirement of Technical Specification 3.1.1.1 is satisfied within 1 hour **AND**;
2. Be in Hot Standby within 6 hours, in accordance with 3-GOP-103, Power Operation to Hot Standby.

5.1.5 **WHEN** more than one full length rod is inoperable **OR** misaligned from the group step counter demand position by more than plus or minus 12 steps, **AND** RTP is greater than 90 percent, **THEN** within 1 hour, perform the following:

1. Restore all indicated rod positions to within the Allowed Rod Misalignment (plus or minus 12 steps),

    OR

2. Reduce RTP to less than 90 percent **AND** confirm that all indicated rod positions are within the Allowed Rod Misalignment (plus or minus 18 steps),

    OR

3. Be in Hot Standby within 6 hours.

5.1.6 **WHEN** more than one full length rod is inoperable **OR** misaligned from the group step counter demand position by more than plus or minus 18 steps **AND** RTP is equal to or less than 90 percent, **THEN** within 1 hour perform the following:

1. Restore all indicated rod positions to within the Allowed Rod Misalignment (plus or minus 18 steps),

OR

2. Be in Hot Standby within 6 hours.

5.1.7 **WHEN** one RCC is trippable but inoperable due to causes other than addressed in Step 5.1.4, **THEN** power operation may continue provided that within 1 hour:

1. The RCC is restored to an operable status

OR

2. The RCC is declared inoperable and the remainder of the RCCs in the bank with the inoperable RCC are aligned to within 12 steps with power greater than 90 percent RTP, **OR** within 18 steps when power is less than or equal to 90 percent RTP of the inoperable rod while not exceeding the rod sequence **AND** insertion limits using the Plant Curve Book, Section VII Figure 3 **AND**,

   a. The thermal power level shall be restricted in accordance with Technical Specification 3.1.3.6 during subsequent operation,

OR

3. The RCC is declared inoperable and the shutdown margin requirement of Technical 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 percent within 1 hour, and within the next 4 hours the power range high neutron flux trip setpoint is reduced to less than or equal to 85 percent of rated thermal power, **AND**;

   b. The shutdown margin requirement of Technical Specification 3.1.1.1 is determined at least once per 12 hours, **AND**;

   c. A power distribution map is obtained from the incore movable detectors and $F_Q(Z)$ and $F_{\Delta H}N$ are verified to be within the limits within 72 hours, **AND**;

   d. A re-evaluation of each accident analysis listed in Enclosure 1 is performed within 5 days to confirm the previously analyzed results of these accidents remain valid under these conditions.

Florida Power & Light Company

Turkey Point Nuclear Plant

This procedure may be affected by a T.C. (Temporary Change) Verify information prior to use.
Date verified_________ Initials________

0-ADM-536

Title:

Technical Specification Bases Control Program

(Information Use)

Responsible Department:	Licensing
Revision Number:	4
Issue Date:	7/11/11
Revision Approval Date:	6/14/11

ARs *578453, 585189, 572662, 1639312, 564275*
PCRs *08-3002, 08-5461, 09-0368, 09-2817, 09-3278, 09-3877, 10-0503, 09-3103, 10-1047, 10-2294*
PC/Ms *06-049, 04-123, 07-010, 08-030, 09-006*

W2003:DPS/mr/cls/mr

ATTACHMENT 1
(Page 26 of 114)

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.

QUESTION #20

Facility: DC Cook

Vendor: WEC

Exam Date: 2008

Exam Type: R

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	1
Group #	2	2
K/A #	005	AA2.03
Importance Rating	3.5	4.4

Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable

Proposed Question:

Given the following plant conditions on Unit 2:

- The unit is at 100% power
- One Control Rod in Bank D Group 1 was found stuck at 190 steps.
- While aligning the remainder of the rods in Bank D to 190 steps an additional Control Rod in Bank D Group 2 was found stuck at 210 steps.
- It has been determined that both rods are mechanically bound.

In accordance with Technical Specifications, which ONE of the following describes the action required within one hour?

A: Determine that Shutdown Margin requirements are satisfied.

B: Determine that QPTR requirements are satisfied or enter the applicable action statement.

C: Verify all peaking factors are within acceptable limits.

D: Align the remainder of rods in the affected banks within 12 steps of the stuck rods.

Proposed Answer: A

Explanation (Optional):

A: Technical Specifications 3.1.4, Rod Group Alignment Limits, Condition A, requires SDM verification within 1 hour.

B: Plausible since rod misalignment can cause skewed QPTR in the core.

C: Plausible since rod misalignment can cause power peaks within the core.

D: Plausible since this is TS relates to rod alignment limits.

Technical Reference(s): Tech Spec 3.1.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: RO-C-AOP-D8/#RO-C-AOP0240412-T1 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Requires the ability to determine the actions that are required per Tech Specs within one hour when TWO Control Rods become inoperable.

QUESTION 21

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	024	2.2.42
Importance Rating	3.9	

Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question: RO Question # 21

Given the following:

- Unit 3 is at 90% power.
- Boron Concentration is 450 ppm

When B 8/2, ROD BANK A/B/C/D EXTRA LO LIMIT, is lit, the Shift Technical Advisor (STA) calculates Shutdown Margin (SDM) to be 1.25% $\Delta k/k$ due to CB D insertion.

Which ONE of the following describes the EARLIEST actions required, if any, in accordance with Technical Specification 3.1.1.1, Shutdown Margin - Tavg Greater than 200°F to restore SDM?

REFERENCE PROVIDED

A. No action is required ~~because determined SDM meets minimum SDM~~.

B. Immediately initiate and continue boration of ≥ 16 gpm until the SDM is restored.

C. Immediately initiate emergency boration of ≥ 45 gpm in accordance with 3-ONOP-046.1, Emergency Boration, until the SDM is restored.

D. Be in HOT STANDBY in 1 hour due to loss of Shutdown Margin.

Proposed Answer: B

Explanation (Optional):

A. Incorrect because boration is required since the SDM is not in the "Acceptable Operation" region of T.S. Figure 3.3-1. Plausible because the given conditions place the SDM near the "Acceptable Region" and the graph could be misread.

B. CORRECT. For the given conditions, the SDM is slightly below the "Acceptable Operation" region. The Action for TS 3.1.1.1 states: "With the SHUTDOWN MARGIN less than the applicable value shown in Figure 3.1-1, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored."

C. Incorrect since the emergency boration must begin after 1 hour if SDM is not restored. Plausible because the boration rate is high enough.

D. Incorrect because boration is required since the SDM is not in the "Acceptable Operation" region of T.S. Figure 3.3-1. Plausible because the applicant understands SDM will improve with a plant trip. However, to be in Hot Standby in 1 hour is not the required action.

Technical Reference(s):

Technical Specification 3.1.1.1, Shutdown Margin - Tavg Greater Than 200°F

TS 3.1.1.2, Shutdown Margin - Tavg Less Than Or Equal To 200°F

3-ONOP-046.1, Emergency Boration

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: T.S. Figure 3.3-1

Learning Objective: LP 6902521, Obj. 3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

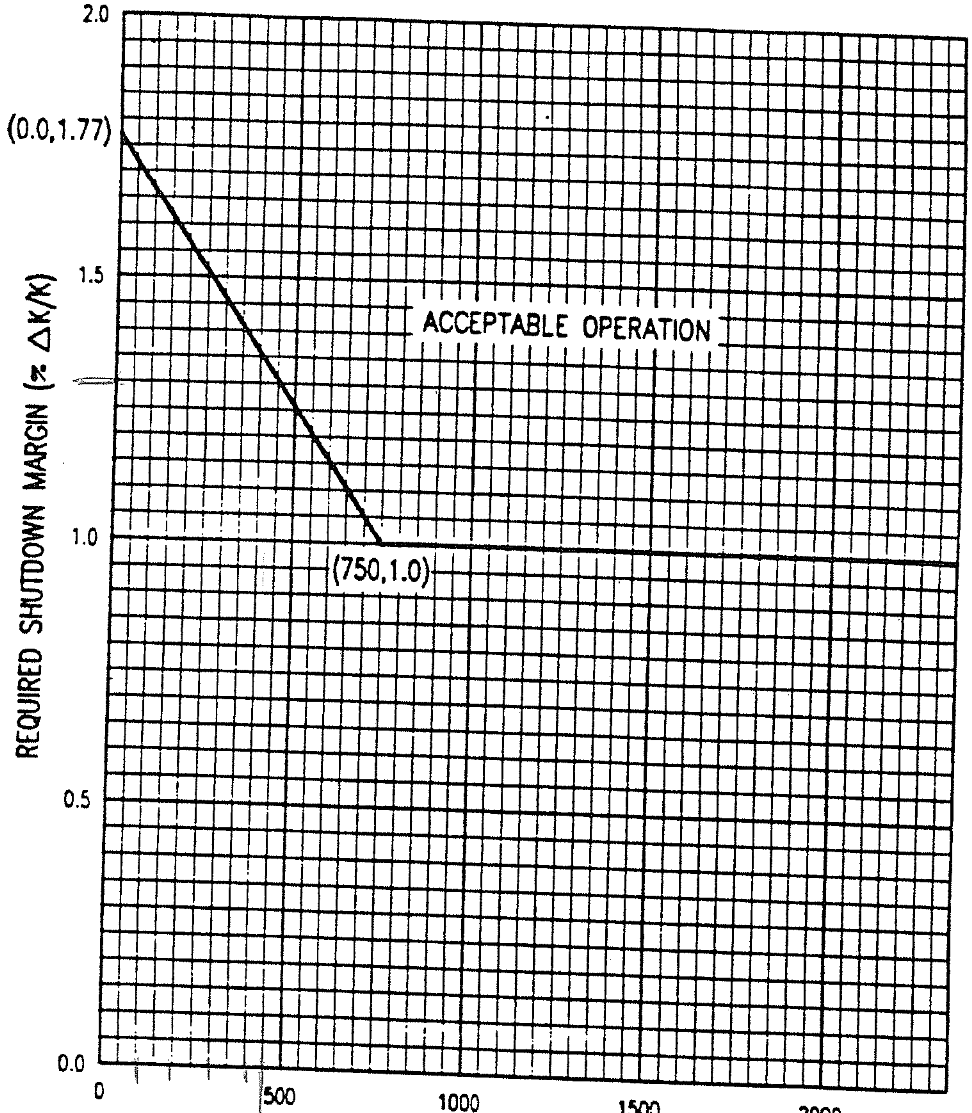
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	(2DR)
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	5
	55.43	

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests a T.S. Action requiring immediate boration. While not an exact match, this question meets the intent of the K/A because "immediate" boration is, in a sense, "emergency" boration. At PTN, TS do not require "emergency" boration. 3-ONOP-046.1, *Emergency Boration*, contains only 1 Symptom/Entry Condition that relates to TS (2.4). In part, Section 2.4 states: "Two or more control or two or more shutdown rods not fully inserted after a reactor shutdown or trip, and shutdown margin is not confirmed to be in compliance with Figure 3.1-1 of Technical Specifications." So, the only relationship between TS and 3-ONOP-046.1 is through SDM, which requires "immediate" boration, not necessarily "emergency" boration.



RCS BORON CONCENTRATION (PPM)

Figure 3.1-1
Required Shutdown Margin vs Reactor Coolant
Boron Concentration

CAUSES: Control bank A, B, C or D inserted to or below its low low limit

B17

ROD BANK A/B/C/D EXTRA LO LIMIT

DEVICE:
Temperature Comparators:
- TC-409D
- TC-409E
- TC-409F
- TC-409L

SETPOINT:
Banks A and B - Fixed at 203 steps

Banks C and D - Variable, dependent on RCS loop ΔT.

LOCATION:
Control Racks 22 and 28

NOTE

Expected alarm during reactor startup or shutdown when rods are below the low low insertion limit.

ALARM CONFIRMATION

1. **CHECK** Control Rod Position - Insertion Limit recorders on VPA.
2. **CHECK** RPI and step counters on console.

OPERATOR ACTIONS

1. **RESTORE** the control rods back above the low limit AND **RESTORE** shutdown margin by:
   A. **STOP** driving control rods in.
   B. **PERFORM** immediate boration equal to or greater than 16 gpm.
2. **CHECK** for load increase with **NO** rod movement.
3. **CHECK** for inadvertent dilution due to valve misalignment in CVCS System.
4. IF a control rod malfunction, THEN **REFER TO** the following as appropriate:
   - 3-ONOP-028, Reactor Control System Malfunction
   - 3-ONOP-028.1, RCC Misalignment
   - 3-ONOP-028.2, RCC Position Indication Malfunction
   - 3-ONOP-028.3, Dropped RCC
5. IF control rods are **NOT** above the rod insertion limit within one hour, THEN **PERFORM** emergency boration using 3-ONOP-046.1, Emergency Boration.

REFERENCES:
1. FPL Drawing 5610-T-D-12B Sheet 1
2. Tech Spec Section 3.1.1.1, 3.1.3.6

1.0 PURPOSE

1.1 This procedure provides instructions for unplanned and uncontrolled increases in reactivity requiring immediate injection of concentrated boric acid solution when the unit is shutdown (EOPs not in use), in refueling, or when directed by the EOPs.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 Excessive control rod insertion for greater than 1 hour for a critical reactor

2.1.1 B 8/2, ROD BANK A/B/C/D EXTRA LO LIMIT

2.1.2 Control rod position indicators are below the bank insertion limit (Plant Curve Book, Sect 7, Fig 3)

2.2 An uncontrolled decrease in any RCS Cold Leg temperature below 525°F for any reason and the RCS not borated to cold shutdown concentration as required by 3-GOP-305, Hot Standby to Cold Shutdown.

2.2.1 Decreasing pressurizer level

2.2.2 Decreasing pressurizer pressure

2.3 Uncontrolled reactivity increase

2.3.1 Uncontrolled increase in reactor power and RCS temperature with no control rod movement.

2.3.2 Uncontrolled increase in source range count rate when subcritical.

2.3.3 Chemistry sample indicates less than required RCS boron concentration for a shutdown reactor.

2.4 Two or more control or two or more shutdown rods not fully inserted after a reactor shutdown or trip, and shutdown margin is not confirmed to be in compliance with Figure 3.1-1 of Technical Specifications.

2.4.1 Rod position indicators indicate rods are not fully inserted and rod bottom lights are out.

2.4.2 Rod position indicators and bottom lights are out-of-service.

2.5 EOP network requires Emergency Boration.

K/A Match Justification

Correct Answer (B)

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to the applicable value shown in Figure 3.1-1.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than the applicable value shown in Figure 3.1-1, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the applicable value shown in Figure 3.1-1:

Distractor C

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

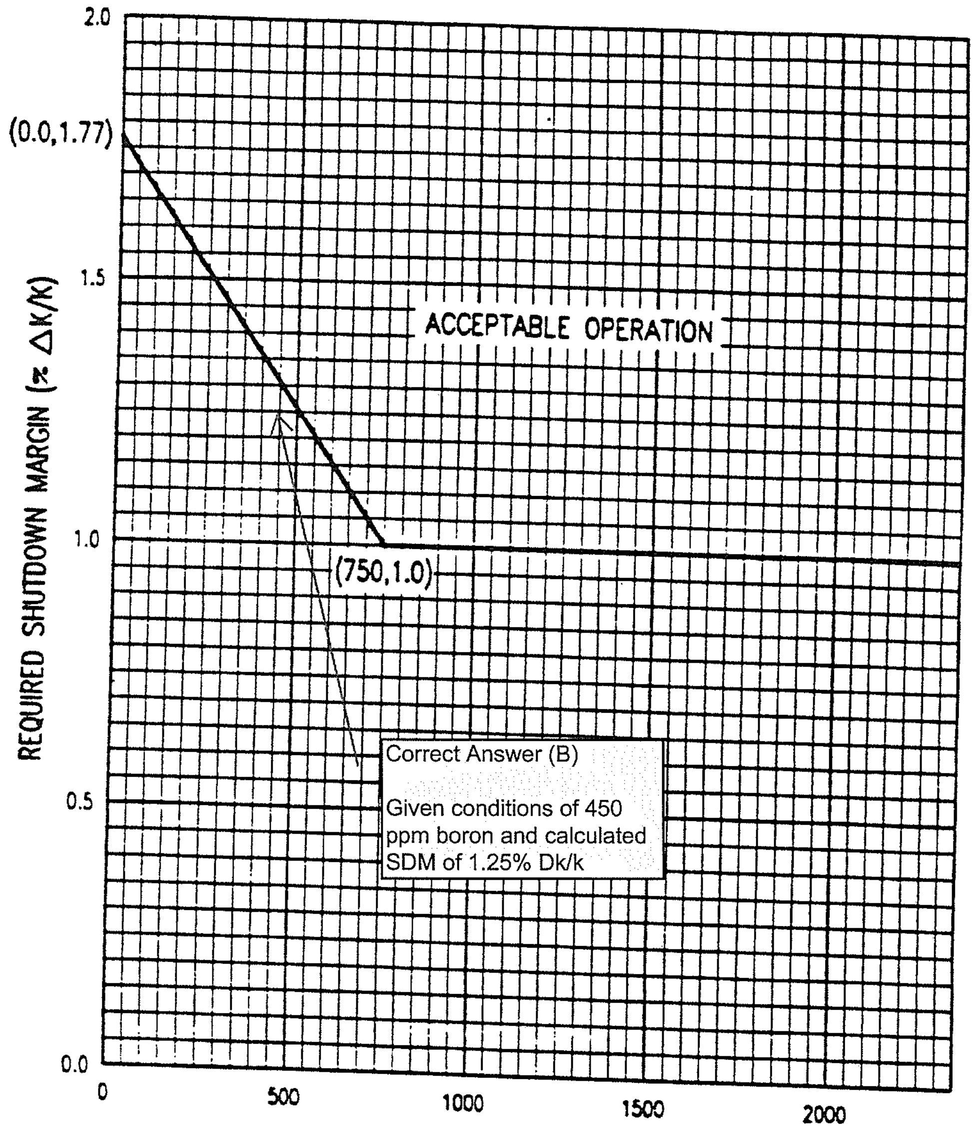
3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 When in Mode 1 or 2, the overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.



RCS BORON CONCENTRATION (PPM)

Figure 3.1-1
Required Shutdown Margin vs Reactor Coolant
Boron Concentration

QUESTION 22

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	032	AK3.01
Importance Rating	3.2	

Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Startup termination on source-range loss

Proposed Question: RO Question # 22

Given the following:

- A Reactor Startup is in progress on Unit 4.
- Unit 4 has entered MODE 2 operation.
- Control Bank "D" is at 50 steps.
- Reactor Power is in the Source Range only.
- Source Range Channel N-31 fails LOW.

Which ONE of the following describes the required action(s) in accordance with Technical Specifications and the reason?

A. CONTINUE Startup;
Use N-32 Source Range Channel and Gammametrics for the Reactor Startup

B. PLACE Source Range Channel N-31 in Level Trip Bypass;
CONTINUE the Reactor Startup with N-32 Source Range Channel available

C. CONTINUE Startup;
ONLY Intermediate Range Flux and the Power Range Flux-Low Trips are required

D. SUSPEND the Startup;
TWO Source Range Channels are required for a Reactor Startup

Proposed Answer: D

Explanation (Optional):

A. Incorrect for raising power below P-6. For a startup, 2 are required to continue.
Plausible – Gammametrics is acceptable to monitor Reactor Power in lower modes of operation.

B. Incorrect for raising power below P-6. For a startup, 2 are required to continue. Plausible – Placing Source Range Channel N-31 in Level Trip Bypass does bypass the protection signals from the failed instrument.

C. Incorrect for raising power below P-6. For a startup, 2 are required to continue. Plausible - Both Intermediate Range Flux (non-credited) and the Power Range Flux-Low (credited) Trips provide protection which is a true statement, however more protection is required through the SR Channels.

D. Correct

Technical Reference(s): TS 3.3.1 and TS Table 3.3-1 (Attach if not previously provided)

LP 6900104 section 2.9

Proposed References to be provided to applicants during examination: None

Learning Objective: 6900104, Obj. 14a, 15 (As available)

Question Source: Bank # WTSI 66211

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Ginna

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 7 & 10

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Ginna 2007 NRC Exam

Matches KA because it tests the reason for termination of the startup, while also testing whether the RO knows that startup must be terminated under conditions where P-6 is not satisfied IAW TS 3.3.1

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)

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.

QUESTION 23

Facility: Turkey Point
Vendor: WEC
Exam Date: 2011
Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	033	AA1.01
Importance Rating	2.9	

Ability to operate and / or monitor the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Power-available indicators in cabinets or equipment drawers

Proposed Question: RO Question # 23

Given the following:

- Unit 3 is at 30% power and raising power.
- Annunciator INTERM RANGE N-35 LOSS OF COMP VOLTAGE (B 5/3) alarms.
- The operating crew determines B 5/3 is a valid alarm.

<u>Intermediate Range N-35 Drawer Indications</u>
- N-35 Drawer – LOSS OF DETECTOR VOLT light is ON.
- N-35 Drawer – LOSS OF COMP. VOLT light is ON.

For the given indications, which ONE of the following describes the indications on the INTERMEDIATE RANGE N-35 Drawer and status of the Reactor Trip Breakers?

	Intermediate Range N-35 Drawer Indications	Reactor Trip Breakers
A.	CONTROL POWER ON status light is OFF	are tripped
B.	CONTROL POWER ON status light is OFF	NOT tripped
C.	INSTRUMENT POWER ON status light is OFF	are tripped
D.	INSTRUMENT POWER ON status light is OFF	NOT tripped

Proposed Answer: D

Explanation (Optional):

A. Incorrect because the conditions presented would not result in a reactor trip. Additionally, control power is still available as indicated by drawer lights being illuminated. Plausible because there are 2 power supplies to the instrument.

B. Incorrect because drawer indication would be out if control power was lost. Plausible because the condition presented would not result in a reactor trip.

C. Incorrect because loss of compensating voltage would not result in a reactor trip at 30% power. Plausible because if the unit was at lower power, this could result in a reactor trip.

D. Correct. Indications are consistent with a loss of instrument power. Because reactor power is at 30%, Intermediate Range Neutron Flux Trip will remain blocked, so a reactor trip will not occur.

Technical Reference(s): SD-004, Excore Nuclear Instrumentation

LP 6900104, Excore Nuclear Instrumentation

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900104, Obj. 7 (As available)

Question Source: Bank # WTSI 68971

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 TMI

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

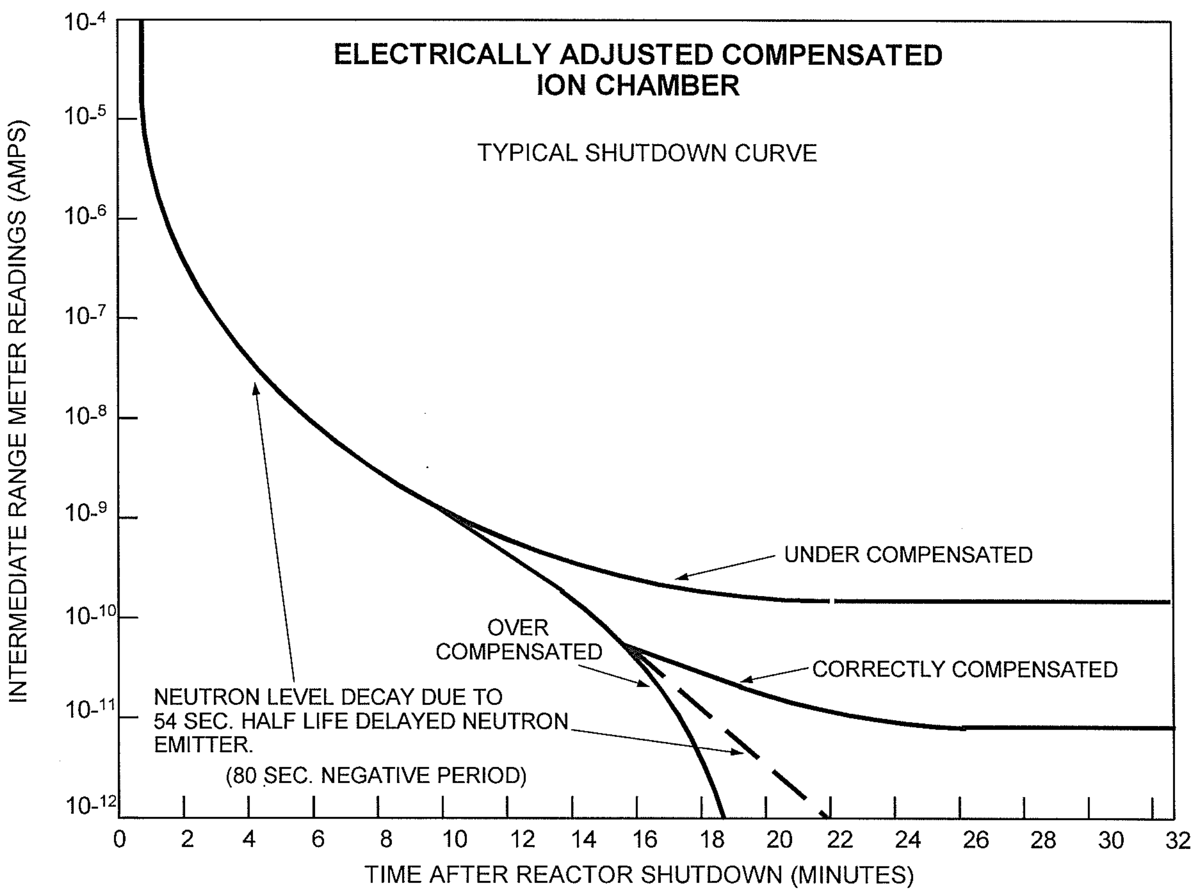
Design, components, and function of control and safety systems, including instrumentation,

signals, interlocks, failure modes, and automatic and manual features.

Comments:

EXCORE NUCLEAR INSTRUMENTATION

CIC GAMMA COMPENSATION



NR-45 TYPICAL RECORDING CHART TRACE (TRIP FROM FULL POWER)

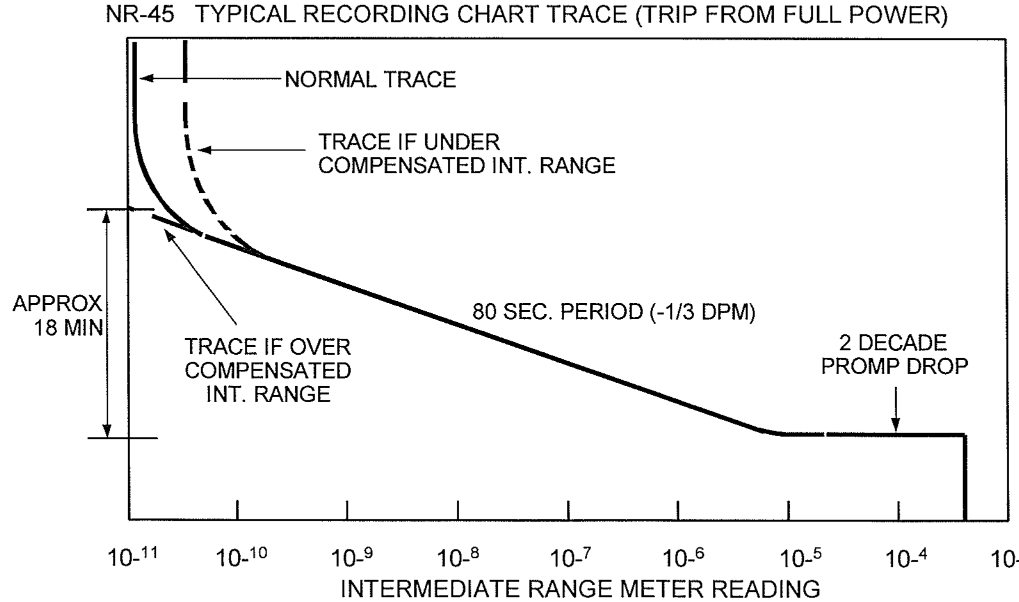


FIGURE 23
Rev 3:10/14/97

EXCORE NUCLEAR INSTRUMENTATION

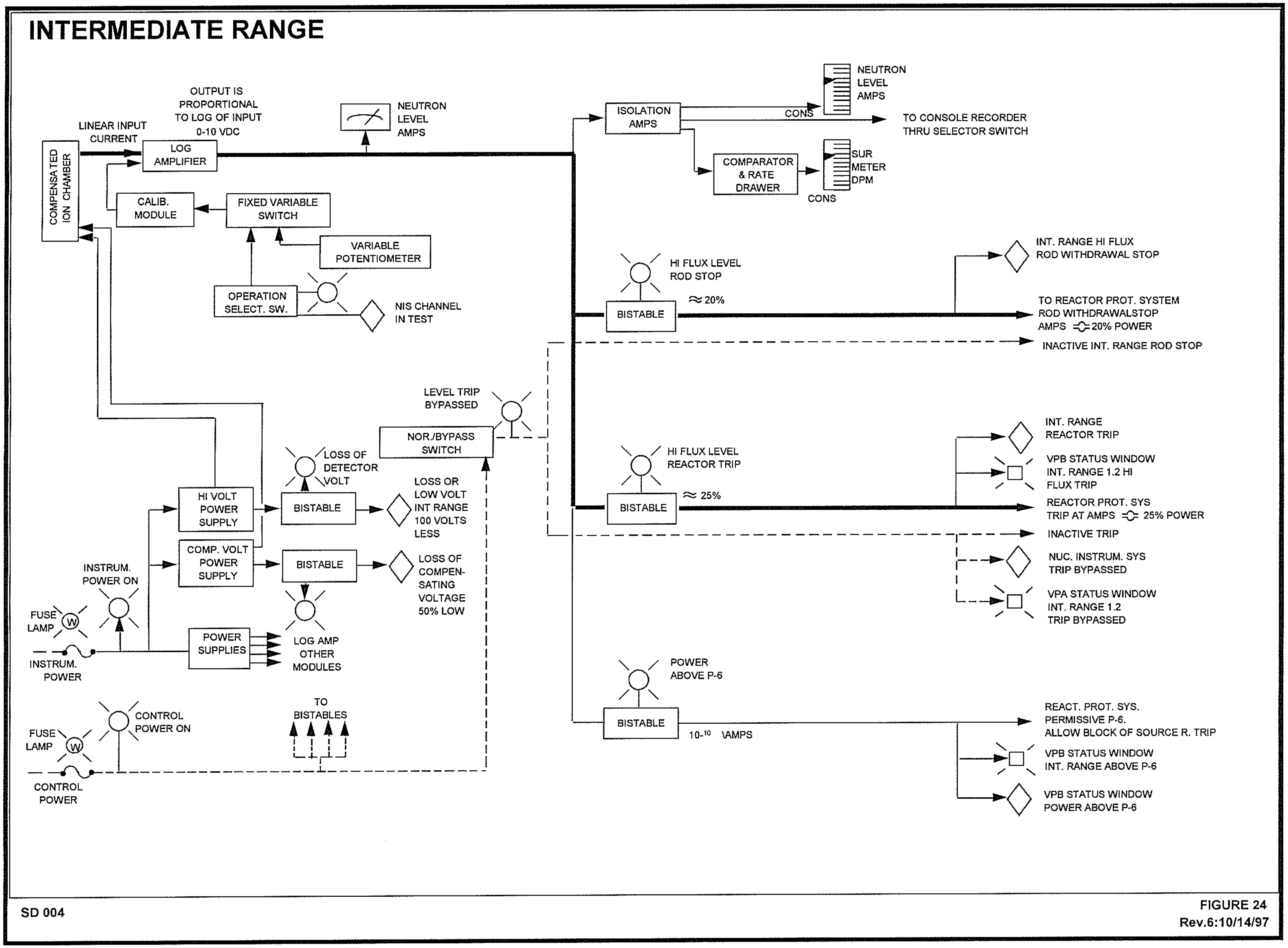


FIGURE 24
Rev.6:10/14/97

EXCORE NUCLEAR INSTRUMENTATION

INTERMEDIATE RANGE DRAWER

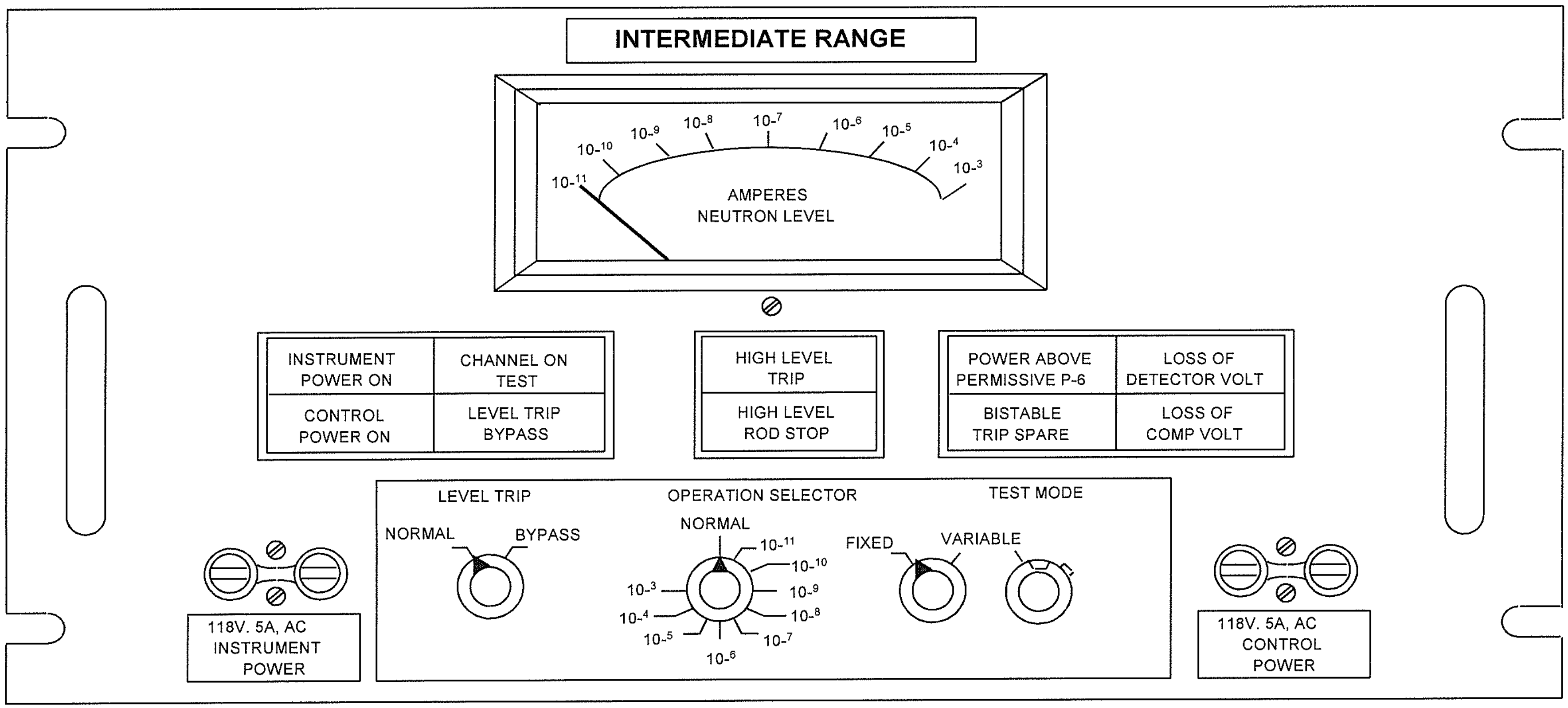


FIGURE 27
Rev.4:7/25/94

EXCORE NUCLEAR INSTRUMENTATION

In the outer boron-lined chamber, the neutrons react with the boron causing ionization as in the case of the Source Range.

$$\frac{1}{0}n + \frac{10}{5}B \rightarrow \left(\frac{11}{5}B\right)* \rightarrow \frac{7}{3}Li + \frac{4}{2}He + energy$$

The ions produced are collected on the detector poles having an opposite charge. This produces a signal which is representative of the neutron flux. Electrons are also collected on the outer can wall from the gamma radiation which interacts with the outer gas volume. This additional signal is representative of the gamma flux and is additive to the neutron flux signal. The outer chamber operates in the ionization region thus all the charged particles produced are collected on the electrodes.

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

Gamma Compensation

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

In order to obtain this true neutron only signal, the two opposing gamma signals must be canceled exactly. Since it is physically impossible for both the inner and outer volumes manufactured to be identically sensitive to the gamma flux present under all operating conditions, the problem of how to

EXCORE NUCLEAR INSTRUMENTATION

ensure exact compensation arises. It is impractical due to the physical inaccessibility of the inner volume to physically adjust the inner volume. However, by grooving the inner electrode and applying a variable negative voltage, the size of the inner volume can be adjusted electrically. The inner volume of the CIC operates in the recombination region of the detector characteristic curve and by adjusting the compensating voltage, only a small fraction of the total ionization is collected.

The intermediate range drawer monitors reactor power over a range of eight decades between 10^{-11} and 10^{-3} ion chamber amperes. Indications of level and startup rate (SUR) are provided at the NIS cabinets and on the console.

Because neutron events are occurring rapidly enough, no signal conditioning is necessary prior to the log level amplifiers as was required in the Source Range.

Log Current Amplifier

This assembly receives current from the detector in the range between 10^{-11} and 10^{-3} amperes. Refer to Figure 24. The assembly provides a logarithmic voltage output, 0 to +10 volts DC, proportional to a linear input current. With the use of the log amplifier, a wide-range input current is compressed logarithmically to a usable voltage suitable for metering and the generation of trip signals. The output from the log amplifier is simultaneously coupled to an isolation amplifier and three bistable relay driver assemblies. The output voltage is also displayed on the neutron level meter calibrated in amperes between 10^{-11} and 10^{-3}.

Internal switches and potentiometers are provided for setting and adjusting the log current amplifiers. Both fixed and variable signals can be injected into the log amplifiers for test and calibration purposes. This is accomplished by the use of switches located on the front panel of the drawer and a calibrate module located inside of the Intermediate Range drawer assembly.

An "idling current" of 10^{-11} amps is maintained through the log current amplifier. This idling current ensures the proper current flow through the amplifier is maintained to help prevent damage to the log current amplifier. The idling current also helps stabilize the output of the log current amplifier at very low detector current levels where they may fluctuate.

EXCORE NUCLEAR INSTRUMENTATION

Because of the idling current, there is an apparent zero offset in the module when it is correctly calibrated. This means that when the detector is supplying 10^{-11} amps to the intermediate range, the drawer should indicate 2 x 10^{-11} amps (to account for idling current). In the first decade, there is a maximum error of up to 100 percent due to idling current alone. This error diminishes by a factor of 10 in each successive decade (e.g., 10^{-11} amps is only 0.1 percent of 10^{-8} amps) until at full scale the idling error is only 10^{-6} percent. This is one of the reasons that 10^{-8} amps is picked for taking "critical data".

Isolation Amplifier

The 0 to +10 volt DC log-level output from the log current amplifier is applied to an isolation amplifier for the generation of DC voltages for use by the reactor plant computers, indicators and recorders. It is identical to the amplifier used in the Source Range channels and isolates the Intermediate Range protective signals from faults which may occur in the equipment after the isolation amplifier.

Bistable Relay Drivers

The 0 to +10 volt log-level DC voltage from the log amplifier is applied simultaneously to three bistable relay driver assemblies. These assemblies produce an indication when the input voltage exceeds a preset value. When tripped, the bistable removes an AC control signal from relays located in the reactor protection racks and the miscellaneous relay racks.

The relay modules are:

1. High Level Rod Stop (20% power)
2. P-6 Signal (>10^{-10} amperes)
3. High Level Trip signal (25% power)

Additionally there are two other bistable relay drivers (not controlled from the level amplifier).

1. Receives a signal from the compensating voltage power supply (20 VDC).

EXCORE NUCLEAR INSTRUMENTATION

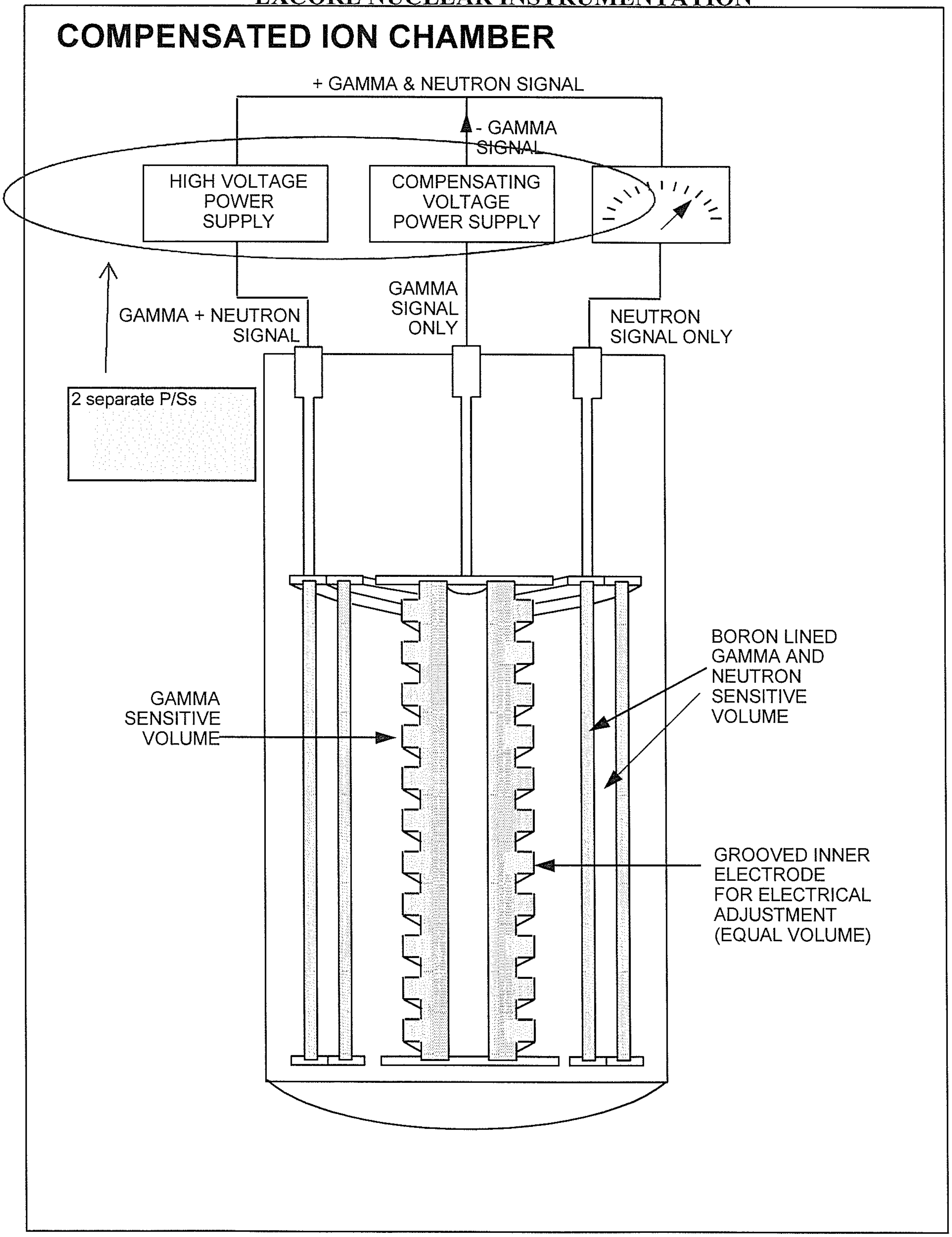


FIGURE 22
Rev 1: 7/20/93

QUESTION 24

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	061	AK2.01
	Importance Rating	2.5	

Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following: Detectors at each ARM system location

Proposed Question: RO Question # 24

EACH of the following ~~Area Radiation~~ Monitors provide input to the Control Room annunciator ARMS HI RADIATION on Panel X, with the **<u>EXCEPTION</u>** of:

A. U-4 New Fuel Storage Area.

B. Spent Fuel Pit Exhaust Duct.

C. U-3 Containment High Range Radiation Monitors.

D. U-3 Sample Room.

Proposed Answer: C

Explanation (Optional):

A. Incorrect because this location inputs to the common ARMS HI RADIATION (4/1) alarm on Annunciator Panel X in the Control Room (Per SD-068, Pages 17 & 18). Plausible because there are other panels that receive high radiation alarms, such as Annunciator Panels X and WB.

B. Incorrect because this location inputs to the common ARMS HI RADIATION (4/1) alarm on Annunciator Panel X in the Control Room (Per SD-068, Pages 17 & 18). Plausible because there are other panels that receive high radiation alarms, such as Annunciator Panels X and WB.

C. CORRECT. Per SD-028, Pages 16 & 17, the high alarm actuates annunciator window H-1/5 (CHRMS HI RADIATION) on the control board.

D. Incorrect because this location inputs to the common ARMS HI RADIATION (4/1) alarm on Annunciator Panel X in the Control Room (Per SD-068, Pages 17 & 18). Plausible because there are other panels that receive high radiation alarms, such a Annunciator Panels X and WB.

Technical Reference(s): 3-ARP-097.CR.H and .X (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902168, Obj. 4 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Turkey Point 2005 – Not last 2

6.5 Enclosures

6.5.1 Enclosure 1, Page number:

1. LS-3568 HI SUMP LEVEL
2. SPARE
3. SPARE
4. WASTE HOLDUP TANK 02 HI-HI LEVEL LS-1078A
5. WASTE HOLDUP TANK 02 HI LEVEL LS-1078A
6. WASTE HOLDUP TANK 02 LO LEVEL LS-1078B

Distractors A, B, & D

7. RADIATION MONITOR CHANNEL FAILURE
8. RADIATION MONITOR LEVEL ALERT
9. RADIATION MONITOR AREA ALARM
10. WASTE EVAP #2 DISTILLATE ACTIVITY HI LEVEL RE021A
11. WASTE EVAP #3 DISTILLATE ACTIVITY HI LEVEL RE022A
12. HIGH LEVEL RADIATION MONITOR ACTIVITY RE-023
13. WASTE MONITOR TANK A HI LEVEL LS-1082A
14. WASTE MONITOR TANK B HI LEVEL LS-1088A
15. WASTE MONITOR TANK C HI LEVEL LS-1089A
16. WASTE MONITOR TANK A LO LEVEL LS-1082A
17. WASTE MONITOR TANK B LO LEVEL LS-1088A
18. WASTE MONITOR TANK C LO LEVEL LS-1089A
19. HEAT TRACING FAILURE
20. DC POWER FAILURE DIVERT VALVE
21. FLASHER

END OF TEXT

REVISION NO.: 0	PROCEDURE TITLE: WASTE/BORON SOUTH PANEL ANNUNCIATOR RESPONSE	PAGE: 3
PROCEDURE NO.: 0-ARP-097.WB.B	TURKEY POINT PLANT	

ANNUNCIATOR PANEL SOUTH WASTE/BORON

	1	2	3	4	5	6	7	8	9
1	1/1 HOLDUP TANK A HI-LO LEVEL PAGE 4	2/1 HOLDUP TANK B HI-LO LEVEL PAGE 7	3/1 HOLDUP TANK C HI-LO LEVEL PAGE 11	4/1 HOLDUP TANKS HI PRESSURE PAGE 14	5/1 MONITOR TANK A HI-LO LEVEL PAGE 18	6/1 MONITOR TANK B HI-LO LEVEL PAGE 22	7/1 SPARE PAGE 25	8/1 SPARE PAGE 28	9/1 VENT HEADER HI PRESSURE PAGE 31
2	1/2 MOISTURE SEPARATOR NO. 1 LO LEVEL PAGE 5	2/2 MOISTURE SEPARATOR NO. 1 HI LEVEL PAGE 8	3/2 MOISTURE SEPARATOR NO. 2 LO LEVEL PAGE 12	4/2 MOISTURE SEPARATOR NO. 2 HI LEVEL PAGE 15	5/2 PLANT STACK HI RADIATION PAGE 20	6/2 REACTOR COOLANT DRAIN TANK UNIT 3 HI LEVEL PAGE 23	7/2 REACTOR COOLANT DRAIN TANK UNIT 3 LO LEVEL PAGE 26	8/2 REACTOR COOLANT DRAIN TANK UNIT 4 HI LEVEL PAGE 29	9/2 REACTOR COOLANT DRAIN TANK UNIT 4 LO LEVEL PAGE 32
3	1/3 REACTOR COOLANT DRAIN TANK UNIT 3 HI PRESSURE PAGE 6	2/3 REACTOR COOLANT DRAIN TANK UNIT 4 HI PRESSURE PAGE 10	3/3 REACTOR COOLANT DRAIN TANK UNIT 3 HI TEMPERATURE PAGE 13	4/3 REACTOR COOLANT DRAIN TANK UNIT 4 HI TEMPERATURE PAGE 17	5/3 WASTE LIQUID HI RADIATION PAGE 21	6/3 SPARE PAGE 24	7/3 SPARE PAGE 27	8/3 SPARE PAGE 30	9/3 SPARE PAGE 33

Distractors A, B, & D

ANNUNCIATOR PANEL H

	1	2	3	4	5	6	7	8	9
1	H1/1 SFP LO LEVEL PAGE 4	H2/1 ACCUM A HI/LO PRESS PAGE 10	H3/1 SI PP 3A TRIP PAGE 16	H4/1 SI PP 3A MOTOR OVERLOAD PAGE 22	H5/1 CNTMT HI-HI/HI PRESS PAGE 28	H6/1 CSP A/B MOTOR OVERLOAD PAGE 34	H7/1 RHR PP A HI PRESS PAGE 40	H8/1 CCW PP A/B/C TRIP PAGE 46	H9/1 RCP A MOTOR BEARING HI TEMP PAGE 52
2	H1/2 SFP HI TEMP PAGE 5	H2/2 ACCUM A HI/LO LEVEL PAGE 11	H3/2 SI PP 3B TRIP PAGE 17	H4/2 SI PP 3B MOTOR OVERLOAD PAGE 23	H5/2 CNTMT ISOLATION ACTIVATED PAGE 29	H6/2 RHR HX HI/LO FLOW PAGE 35	H7/2 RHR PP B HI PRESS PAGE 41	H8/2 CCW PP A/B/C MOTOR OVERLOAD PAGE 47	H9/2 RCP B MOTOR BEARING HI TEMP PAGE 53
3	H1/3 SFP HI LEVEL PAGE 6	H2/3 ACCUM B HI/LO PRESS PAGE 12	H3/3 SI PP 4A TRIP PAGE 18	H4/3 SI PP 4A MOTOR OVERLOAD PAGE 24	H5/3 AFW/CNTMT ISOLATION CABINET FUSE FAIL PAGE 30	H6/3 RHR PP A/B MOTOR OVERLOAD PAGE 36	H7/3 RHR PP A COOLING WATER LO FLOW PAGE 42	H8/3 CCW PP HEADER LO PRESS PAGE 48	H9/3 RCP C MOTOR BEARING HI TEMP PAGE 54
4	H1/4 PRMS HI RADIATION PAGE 7	H2/4 ACCUM B HI/LO LEVEL PAGE 13	H3/4 SI PP 4B TRIP PAGE 19	H4/4 SI PP 4B MOTOR OVERLOAD PAGE 25	H5/4 CNTMT ISOLATION IN TEST PAGE 31	H6/4 RHR PP A/B TRIP PAGE 37	H7/4 RHR PP B COOLING WATER LO FLOW PAGE 43	H8/4 CCW PP SUCTION HI TEMP PAGE 49	H9/4 RCP MOTOR BRG COOLING WATER HI TEMP PAGE 55
5	H1/5 CHRMS HI RADIATION PAGE 8	H2/5 ACCUM C HI/LO PRESS PAGE 14	H3/5 RHR PUMP/HX DISCHARGE HI/LO TEMP PAGE 20	H4/5 SPARE PAGE 26	H5/5 SIS LOGIC IN TEST PAGE 32	H6/5 RWST LO LEVEL PAGE 38	H7/5 CSP A/B COOLING WATER LO FLOW PAGE 44	H8/5 CCW HX OUTLET HI TEMP PAGE 50	H9/5 RCP MOTOR BRG COOLING WATER LO FLOW PAGE 56
6	H1/6 PRMS CHANNEL FAILURE PAGE 9	H2/6 ACCUM C HI/LO LEVEL PAGE 15	H3/6 PRMS R11/R12 BYPASSED/ WARNING ACTUATED PAGE 21	H4/6 SPARE PAGE 27	H5/6 SIS POWER FAILURE PAGE 33	H6/6 RWST LO-LO LEVEL PAGE 39	H7/6 SURGE TANK LO LEVEL PAGE 45	H8/6 CCW HEAD TANK HI/LO LEVEL PAGE 51	H9/6 RCP A/B/C PUMP/MOTOR HI TEMP PAGE 57

Correct Answer (C)

REVISION NO.:	PROCEDURE TITLE:	PAGE:
1	CONTROL ROOM RESPONSE - PANEL H	8
PROCEDURE NO.:		WINDOW:
3-ARP-097.CR.H	TURKEY POINT UNIT 3	1/5 (Page 1 of 1)

CAUSES:
1. High radiation in containment
2. Loss of power
3. Control module removed from racks
4. Loss of signal from detector

H1/5

CHRMS HI RADIATION

DEVICE:
- RAD-3-6311A
- RAD-3-6311B

SETPOINT: 1.3×10^4 R/HR

LOCATION: N/A

Correct Answer (C)

ALARM CONFIRMATION

1. **CHECK** the following:
   - Indication on Panel 3C10
   - Control module in Rack QR-81/82 for level indication and for power to each module

OPERATOR ACTIONS

1. **CHECK** ARMS channels inside containment for increased levels.
2. **PERFORM** channel check test using 3-OSP-201.1, RO Daily Logs.
3. **NOTIFY** Shift Manager.
4. **REFER TO** TS 3.3.3 for additional actions.
5. **REFER TO** 0-EPIP-20101, Duties of Emergency Coordinator.

REFERENCES: Tech Spec Section 3.3.3

REVISION NO.: 0	PROCEDURE TITLE: CONTROL ROOM RESPONSE - PANEL X	PAGE: 22
PROCEDURE NO.: 3-ARP-097.CR.X	TURKEY POINT UNIT 3	WINDOW: 4/1 (Page 1 of 1)

CAUSES:
1. Local area high radiation
2. Failed ARMS detector

X4/1
ARMS HI RADIATION

DEVICE: Area Rad Monitors

SETPOINT: N/A

Correct Answer (C)

LOCATION: Channels on ARMS rack

ALARM CONFIRMATION

IDENTIFY alarming channel(s) by noting individual channel(s) HIGH alarm light(s) illuminated on Area Radiation Monitoring System Control Panel R-30.

OPERATOR ACTIONS

1. **CHECK** valid alarm by performing 0-ONOP-066, High Area Radiation Monitoring System Alarm.
2. IF any ARM channels 1 through 6 is determined to have failed while in Mode 5 or 6, THEN **NOTIFY** Radiation Protection to install a portable area radiation monitor, with alarm.
3. IF any ARM channels 21 through 24 is determined to have failed, THEN **NOTIFY** Radiation Protection to install a portable area radiation monitor capable of meeting requirement of 10 CFR 50.68(b).

REFERENCES:
1. FPL Dwg 5610-J-822
2. Victoreen Tech Manual V000452

ANNUNCIATOR PANEL X

	1	2	3	4	5	6	7	8	9
1	X1/1 DC LC 3A TROUBLE PAGE 4	X2/1 4KV BUS 3A LO VOLTAGE PAGE 10	X3/1 SU XFMR 3 DIFFERENTIAL CURRENT TRIP PAGE 16	X4/1 ARMS HI RADIATION PAGE 22	X5/1 VITAL DC BATTERY CHARGER FAILURE PAGE 28	X6/1 CONT BLDG EMERG VENT LO FLOW PAGE 34	X7/1 SU XFMR 4 DIFFERENTIAL CURRENT TRIP PAGE 40	X8/1 4KV BUS 4A LO VOLTAGE PAGE 46	X9/1 DC LC 4B TROUBLE PAGE 52
2	X1/2 DC LC 3A GROUND PAGE 5	X2/2 4KV BUS 3A SUPPLY BKR FAN FAIL PAGE 11	X3/2 SU XFMR 3 FAULT PRESS PAGE 17	X4/2 AUX/RADWASTE BUILDING SUPPLY FAN TRIPPED PAGE 23	X5/2 CWP LUBE WATER STORAGE TANK LOW LEVEL PAGE 29	X6/2 RADWASTE BLDG PANEL C46 TROUBLE PAGE 35	X7/2 SU XFMR 4 FAULT PRESS PAGE 41	X8/2 4KV BUS 4A SUPPLY BKR FAN FAIL PAGE 47	X9/2 DC LC 4B GROUND PAGE 53
3	X1/3 SEQUENCER 3A TROUBLE PAGE 6	X2/3 4KV BUS 3B LO VOLTAGE PAGE 12	X3/3 SU XFMR 3 GROUND FAULT TRIP PAGE 18	X4/3 AUX/RADWASTE BUILDING EXHAUST FAN TRIPPED PAGE 24	X5/3 SERVICE WTR/ FIRE PP/ RWT/ WTP TROUBLE PAGE 30	X6/3 WASTE BORON PANEL TROUBLE PAGE 36	X7/3 SU XFMR 4 GROUND FAULT TRIP PAGE 42	X8/3 4KV BUS 4B LO VOLTAGE PAGE 48	X9/3 SEQUENCER 4A TROUBLE PAGE 54
4	X1/4 SEQUENCER 3B TROUBLE PAGE 7	X2/4 4KV BUS 3B SUPPLY BKR FAN FAIL PAGE 13	X3/4 SU XFMR 3 PANEL TROUBLE PAGE 19	X4/4 SPARE PAGE 25	X5/4 LUBE WATER PUMP AUTO XFER/ TRIP PAGE 31	X6/4 WASTE HOLDUP TANK ROOM SUMP HI LEVEL PAGE 37	X7/4 SU XFMR 4 PANEL TROUBLE PAGE 43	X8/4 4KV BUS 4B SUPPLY BKR FAN FAIL PAGE 49	X9/4 SEQUENCER 4B TROUBLE PAGE 55
5	X1/5 DC LC 3B TROUBLE PAGE 8	X2/5 4KV BUS 3A/3B GROUND PAGE 14	X3/5 SU XFMR 3 BREAKER OVERCURRENT TRIP PAGE 20	X4/5 4KV BUS 3A/3B LOSS OF VOLT RELAY FUSE FAIL PAGE 26	X5/5 SPARE PAGE 32	X6/5 4KV BUS 4A/4B LOSS OF VOLT RELAY FUSE FAIL PAGE 38	X7/5 SU XFMR 4 BREAKER OVERCURRENT TRIP PAGE 44	X8/5 4KV BUS 4A/4B GROUND PAGE 50	X9/5 DC LC 4A TROUBLE PAGE 56
6	X1/6 DC LC 3B GROUND PAGE 9	X2/6 PAGE/ SITE EVAC LOSS OF AC VOLTAGE PAGE 15	X3/6 SI PP COOLING WATER LO FLOW PAGE 21	X4/6 3 GEN / 3 SU XFMR / BUS 3A/3B/3D LOR FUSE FAIL PAGE 27	X5/6 BAST TECH SPEC LO LEVEL PAGE 33	X6/6 4 GEN / 4 SU XFMR / BUS 4A/4B/4D LOR FUSE FAIL PAGE 39	X7/6 BORIC ACID AMBIENT TEMP MONITORING TROUBLE PAGE 45	X8/6 WASTE HOLDUP TANK HI LEVEL PAGE 51	X9/6 DC LC 4A GROUND PAGE 57

CONTAINMENT POST ACCIDENT EVALUATION SYSTEM

DETAILED DESCRIPTION

Component/Function Description

Radiation monitors installed inside the containment building provide redundant indication in the Control Room of radiation levels in the containment building. Refer to Figures 6, 7, & 1. These monitors are area monitors (Gamma Ionization Chambers) which indicate in R/Hr. The purpose of the monitors is to provide the Control Room operator and/or supervisory personnel with radiation levels inside containment under all conditions.

The CHRRMS channel is comprised of components manufactured by the General Atomics Corporation. For each unit there are two separate and redundant channels which are designated RI-6311 A & B. Channel RI-6311A is located at elevation 25' 10" within 1 1/2' of the containment outside wall close to the personnel hatch. Channel RI-6311B has its detector located on the "A" steam generator shield wall at elevation 64' near the pressurizer along with Unit R-2 ARMS for Unit 3 and R-5 ARMS for Unit 4.

Protective Action Recommendations (PARS) made by the Emergency Coordinator in accordance with 0-EPIP-20101, "Duties of Emergency Coordinator", utilize the CHRRMS indications.

An accident resulting in a CHRRMS reading of 4.1×10^4 R/Hr will result in a site boundary dose of 50 mr/Hr. If the CHRRMS reading increased by a factor of 10 to 4.1×10^5 R/Hr the site boundary dose would also increase by a factor of 10 to 500 mr/Hr, assuming poor dispersion conditions and a design leak rate from the containment building of 0.25%/Day.

Instrumentation and Control

Indication from the Containment High Range Radiation Monitors is provided in the Control Room on Vertical Panel "C" and on racks QR 81/82. See Figures 6, 7 & 1. Historical records are also provided by strip chart recorders located on racks QR 81/82.

Two alarms are associated with the CHRRMS. The low range alarm (high alarm) is received as an indicated radiation level of 1.3×10^4 R/Hr and the high range alarm (hi-hi alarm) is received

CONTAINMENT POST ACCIDENT EVALUATION SYSTEM

when an indicated radiation level of 1.3×10^5 R/Hr is achieved. The high alarm actuates annunciator window H-1/5 (CHRMS HI RADIATION) on the control board and lights the yellow light on the module. The HI-HI alarm only lights the red light on the module. If power is lost to the monitors this annunciator window will also be actuated.

The high radiation alarm corresponds to a Site Area Emergency and the Hi-Hi alarm corresponds to a General Emergency in accordance with 0-EPIP-20101, "Duties of Emergency Coordinator."

RADIATION MONITORING AND PROTECTION

Installed batteries, which are charged when the timer is in operation, will maintain the program to within 1 percent for a period of 8 hours provided the batteries have been allowed to charge at least 8 hours.

The detector response to leaks is proportionate to the RCS activity level. Considering background radiation levels in the vicinity of the detector, reactor vessel head leaks as low as 0.01 GPM could be detected.

Loss of this detection system is not considered to have any adverse impact on the plant operation, and is not relied upon for any accident mitigation for safe shutdown.

AREA RADIATION MONITORING SYSTEM

Area Radiation Monitoring Cabinets, Channels 1-24 only (PC/M 88-462)

Channels 1 through 24 are located in a cabinet (3QR30) in the control room. Channels 26 through 30 are located in a cabinet in the radwaste building control room. The control room cabinet receives power from AC distribution panel 3P06 switch 16. Power for the radwaste building cabinet (QR-31) is from AC distribution panel DP-66 switches 2 and 3. The cooling fan for the ARMS cabinet is powered from lighting panel LP33 breaker 15.

Control Room Cabinet

All of the remote indicators are centralized in one cabinet which is located in the control room. The cabinet houses 24 Universal Digital Rate meters and a 30 point recorder. The cabinet also has 12 blank panels for future expansion of the system. Each Digital Rate meter (remote indicator) provides all power to its respective channel including the local indicator, preamplifier, detector, alarm light, and the horn. The recorder sequentially records the outputs from the remote indicators at a chart speed of two inches per hour and a print rate of 10 seconds per point.

Universal Digital Ratemeter (UDR)

RADIATION MONITORING AND PROTECTION

This is a cabinet mounted module which accepts the signal from the preamplifier via a digital circuit. The signal is processed to provide two visual displays, one current output and an alarm relay output. Other outputs are available for future modifications. Visual displays are a five digit display of the radiation value and a multi-color bargraph indicator which covers the range of 10^{-1} to 10^{7} mR/hr. The bargraph has three LED segments per decade. The bargraph will change color in the event of an alarm condition. Front panel alarm indicators and rear panel output relays for alarm annunciation are also included. Front panel control pushbuttons are provided to turn power on/off, to acknowledge alarms, and to activate an electronic check source function. Momentary pushbuttons are also available on the front panel for HIGH alarm setpoint and WARN alarm setpoint. Analog outputs of 4-20 mA is connected for recorder and computer monitoring. Analog output of 0 to 10 VDC is available but not connected. A communication loop transfers data between the remote indicator and the preamplifier.

Four LEDs are used to provide visual indication of alarm status on the front face of the remote indicator. They are as follows:

HIGH Alarm	Red LED
WARN Alarm	Amber LED (Not Used)
Fail Alarm	Red LED
Range Alarm	Red LED

Only the high alarm is connected to annunciator window X-4/1 on the common panel. The high alarm will flash until acknowledged. The warn setpoint is not used. The fail and range alarms are not connected to annunciation.

The check source LED is lit when the check source pushbutton is pressed to activate an electronic check source.

The alarm acknowledge pushbutton is depressed to acknowledge alarms and will cause the alarm LEDs to transition to a steady state indication.

The Detector

RADIATION MONITORING AND PROTECTION

INDICATIONS, CONTROLS, AND ALARMS

Area Radiation Monitoring System (See 0-OSP-066, Attachment 1 For Latest Setpoints.)

CHANNEL	INDICATION (LOCATION)	INSTRUMENT NUMBER	RANGE mRem/hr.	ALARM SETPOINT mRem/hr.
R-1	U3 Containment Mezzanine (near personnel air lock – 30'6" elev.)	RI-3-1401B	10^{-1} to 10^{7}	100 (5)[†]
R-2	U-3 Containment Operating Floor (58' elev.)	RI-3-1402B	10^{-1} to 10^{7}	150 (10)[†]
R-3	U-3 Containment In-Core Drive (34'6"elev.)	RI-3-1403B	10^{-1} to 10^{7}	150 (20)[†]
R-4	U4 Containment Mezzanine (near personnel air lock - 30'6" elev.)	RI-4-1404B	10^{-1} to 10^{7}	100 (5)[†]
R-5	U-4 Containment Operating Floor (58' elev.)	RI-4-1405B	10^{-1} to 10^{7}	150 (10)[†]
R-6	U-4 Containment In-Core Drive (34'6" elev.)	RI-4-1406B	10^{-1} to 10^{7}	100 (20)[†]
R-7	U-3 SFP Canal Area (58' elev.)	RI-3-1407B	10^{-1} to 10^{7}	40
R-8	U-4 SFP Canal Area (58' elev.)	RI-4-1408B	10^{-1} to 10^{7}	50
R-9	Aux. Bldg. Tank & Pump ~~(4' elev.)~~	RI-1409B	10^{-1} to 10^{7}	10
R-10	Aux. Bldg. Chem Storage Room (18' elev.)	RI-1410B	10^{-1} to 10^{7}	40
R-11	~~U-4 Cask Wash Area (18' elev.)~~	RI-4-1411B	10^{-1} to 10^{7}	10
R-12	U-3 Cask Wash Area (18' elev.)	RI-3-1412B	10^{-1} to 10^{7}	10
R-13	U-3 Sample Room (18' elev.)	RI-3-1413B	10^{-1} to 10^{7}	10
R-14	U-4 Sample Room (18' elev.)	RI-4-1414B	10^{-1} to 10^{7}	10
R-15	Aux. Bldg. North N/S Corridor (18' elev.)	RI-3-1415B	10^{-1} to 10^{7}	5
R-16	Aux. Bldg. South N/S Corridor (18' elev.)	RI-4-1416B	10^{-1} to 10^{7}	5
R-17	Aux. Bldg. (18' elev.) East E/W Corridor	RI-1417B	10^{-1} to 10^{7}	5

Distractor D

Area Radiation (Continued)

[†] Alarm setpoints when unit is in extended shutdown.

Distractor B

RADIATION MONITORING AND PROTECTION

CHANNEL	INDICATION	CHANNEL	RANGE mRem/hr.	ALARM SET-POINT mRem/hr.
R-18	Aux. Bldg. West E/W Corredor (18' elev.)	RI-1418B	10^{-1} to 10^{7}	5
R-19	SFP Exhaust Duct	RI-3-1419B	10^{-1} to 10^{7}	15
R-20	Unit 3 & 4 Control Room	RI-1420B	10^{-1} to 10^{7}	2
R-21	U-3 SFP North Wall	RI-3-1421B	10^{-1} to 10^{7}	20‡
R-22	U-4 SFP South Wall	RI-4-1422B	10^{-1} to 10^{7}	20‡
R-23	U-3 New Fuel Storage Area	RI-3-1423B	10^{-1} to 10^{7}	20‡
R-24	U-4 New Fuel Storage Area	RI-4-1424B	10^{-1} to 10^{7}	10‡
N/A	Area Dose Rate CHRRMS	RAD-*-6311A	10^{0} to 10^{8}*	1.3×10^{4}* E-Plan Site Area Emergency
N/A	Area Dose Rate CHRRMS	RAD-*-6311B	10^{0} to 10^{8}*	1.3×10^{4}* E-Plan Site Area Emergency

ALARM	SENSING DEVICE	ANNUNCIATOR
ARMS HI RADIATION	ARMS channels 1 through 24	X-4/1
RADWASTE BUILDING ARMS HI RADIATION	ARMS channel 26, 27, 28, 29 & 30 (normally locked in)	X-6/2
CHRMS HI-HI/HI RADIATION	RAD-6311A and RAD-6311B (Setpoint 1.3x 10^4/1.3x10^5R/hr)	H-1/5

Distractor A

Correct Answer (C)

Process Radiation

* Rem/hr (See SD-028/(SYS. 094) for description.)

‡ ARMS channels 21, 22, 23 and 24 were installed to satisfy 10 CFR 70.24, Criticality Accident Requirements. The alarm setpoints for these channels are required to be not less than 5 millirems per hour (in order to avoid false alarms) not more than 20 millirems per hour per 10 CFR 70.24.

QUESTION 25

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	067	2.1.30
	Importance Rating	4.4	

Conduct of Operations: Ability to locate and operate components, including local controls.

Proposed Question: RO Question # 25

Given the following:

- Annunciator X6/6, XFMR/HYDROGEN SEAL OIL DELUGE OPERATING is LIT in the Control Room.
- The Fire Brigade Team Leader reports the Unit 3 Main Transformer are on fire and the Main Transformer Cooling Fans and Oil Pumps are tripped.
- The ANPO reports the Electric Fire Pump seized on start and no Fire Pumps are running.

The Control Room directs the ANPO to perform a Emergency/Manual Start of the Diesel Driven Fire Pump by ....

A. throttling closed the Gauge Test Line Drain, 10-1054.

B. throttling open the DDFP Mercoid Sensing Line Isolation Valve 10-769.

C. ensuring Battery 1 & 2 Switches are ON, placing the Control Switch to MANUAL, and pushing the CRANK 1 pushbutton.

D. ensuring Battery 1 & 2 Switches are ON and pushing the CRANK 1 & CRANK 2 pushbuttons simultaneously.

Proposed Answer: C

Explanation (Optional):

A. Incorrect since these actions are for surveillance operation, not an emergency start.

Plausible because this valve is operated during surveillance testing to locally start this pump.

B. Incorrect since these actions are for surveillance operation, not an emergency start. Plausible because this valve is operated during surveillance testing to locally start this pump.

C. CORRECT. Per Section 7.8 of 0-OP-016.1, Fire Protection Water

D. Incorrect since these actions are not complete for an emergency start. Also, the start does not require both pushbuttons. Plausible because these components are operated during the emergency start process to locally start this pump.

Technical Reference(s): LP 6902143, *Fire Protection* (Attach if not previously provided)

3-OP-016.1

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902143, Obj. 11 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests local controls (Panel C286) and operation of equipment (Main/Reserve switch) during a fire in the Cable Spreading Room.

Florida Power & Light Company

Turkey Point Nuclear Plant

This procedure may be affected by a T.C. (Temporary Change) Verify information prior to use. Date verified________ Initials________

0-OP-016.1

Title:

Fire Protection Water System

(Continuous Use)

Quality Related Procedure

Responsible Department:	Operations
Revision Number	2
Issue Date	10/21/10
Revision Approval Date:	10/12/10

ARs *571738, 571323*
TCs *09-011*
PC/Ms *86-079, 87-053, 87-212, 87-259, 87-260, 87-261, 87-264, 87-333, 87-380, 87-503, 87-601, 88-364, 88-503, 89-164, 90-076, 90-430, 92-004, 92-107, 93-099, 96-078, 97-031, 97-058, 00-008, 02-052, 02-099, 03-002, 04-124, 05-012, 05-041, 08-022, 08-042, 07-069, 08-072, 08-049, 10-023*

INITIALS
CK'D VERIF

NOTE

When engine is starting and oil pressure is not yet up to full pressure, the Low Oil Pressure Light will light at the Diesel Fire Pump Controller Panel, but the alarm bell will not ring. When oil pressure builds up and switch opens, the Low Oil Pressure Light will go out.

7.8 Emergency/Manual Start of the Diesel Driven Fire Pump

7.8.1 Verify both Battery 1 and Battery 2 switches are ON.

7.8.2 Place control switch to MANUAL.

7.8.3 Push the CRANK 1 pushbutton.

7.8.4 **IF** engine does NOT start, **THEN** push the CRANK 2 pushbutton.

QUESTION 26

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	E13	EA1.2
Importance Rating	3.0	

Ability to operate and / or monitor the following as they apply to the (Steam Generator Overpressure) Operating behavior characteristics of the facility.

Proposed Question: RO Question # 26

Unit 3 experienced a Loss of Offsite Power (LOOP) with the following:

- All MSIVs and MSIV Bypass Valves were verified closed.
- S/G Pressures: S/G A - 1010 psig; S/G B - 1070 psig; S/G C - 1170 psig
- S/G NR Levels: S/G A - 45%; S/G B - 55%; S/G C - 65%
- Total Auxiliary Feedwater Flow is 100 gpm to each S/G.

In accordance with 3-EOP-FR-H.2, Response to Steam Generator Overpressure, which ONE of the choices below both identify actions to limit the overpressure condition?

A. Open 3B S/G Steam Dump to Atmosphere Valve OR isolate AFW to the 3B S/G

B. Open 3C S/G Steam Dump to Atmosphere Valve OR isolate AFW to the 3C S/G

C. Open 3B S/G Steam Dump to Atmosphere Valve OR prepare for 3B S/G Blowdown for operation

D. Open 3C S/G Steam Dump to Atmosphere Valve OR prepare for 3C S/G Blowdown for operation

Proposed Answer: B

Explanation (Optional):

A. Incorrect because 3B S/G only meets entry for 3-EOP-FR-H.4. Plausible because these are the correct actions to take for entry into 3-EOP-FR-H.2, and S/G pressure is beyond normal band for pressure.

B. CORRECT. 3-EOP-FR-H.2 allows for different methods of limiting pressure in the affected S/G. Two of these methods are to open a S/G Steam Dump to Atmosphere

Valve OR to isolate AFW to the affected S/G.

C. Incorrect since using S/G Blowdown is a strategy in 3-EOP-FR-H.3. Plausible because the first part is correct for entry into 3-EOP-FR- H.4. Also, the second method would remove mass from the S/G, but this is provided in 3-EOP-FR-H.3 when narrow range level is greater than 80%.

D. Incorrect. Plausible because the first part is Correct. Plausible because the second method would remove mass from the S/G, but this is provided in 3-EOP-FR-H.3 when narrow range level is greater than 80%.

Technical Reference(s): 3-EOP-FR-H.2, Response to Steam Generator Overpressure (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902337, Obj. 4 (As available)

Question Source: Bank # 70872

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 VC Summer

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

This was modified from VCS 2009 exam but left as bank item because the bank contains several similar items

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-EOP-FR-H.2

Title:

RESPONSE TO STEAM GENERATOR OVERPRESSURE

Safety Related Procedure

Responsible Department:	Operations
Revision Approval Date:	04/15/99
Periodic Review Due:	04/14/04

RTS 94-1286P, 97-1327P, 98-1130P

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

*Throughout this procedure **affected** refers to any S/G in which pressure is greater than 1130 psig.*

1 Identify Affected S/G(s)

a. Any S/G pressure - GREATER THAN 1130 PSIG

a. Return to procedure **AND** step in effect.

2 Verify Feedwater Isolation To Affected S/G(s)

a. Feedwater control valve(s) - CLOSED

a. Manually close valve(s).

b. Feedwater bypass valve(s) - CLOSED

b. Manually close valve(s).

c. Close feedwater isolation valve(s)

* MOV-3-1407 for S/G A
* MOV-3-1408 for S/G B
* MOV-3-1409 for S/G C

c. Locally close valve(s).

3 Check Affected S/G(s) Narrow Range Level - LESS THAN 90%[72%]

Go to 3-EOP-FR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL, Step 1.

CAUTION

If affected S/G narrow range level increases to greater than 90%[72%], steam should NOT be released from the affected S/G(s).

4 Try To Dump Steam From The Affected S/G(s)

* S/G steam dump to atmosphere valve

OR

* MSIV Bypass valves

OR

* Steam supply valves to AFW pump

Observe CAUTION prior to Step 6 **AND** go to Step 6.

CORRECT ANSWER

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

This procedure may be affected by an O.T.S.C. (On The Spot Change) verify information prior to use
Date verified________ Initials________

3-EOP-FR-H.3

Title:

Response to Steam Generator High Level

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Approval Date:	4/2/09

PCRs *08-5384*
RTSs *94-1287P, 96-0079P, 97-1328P, 98-1130P*
PC/M *95-168, 05-026*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9 Prepare For Blowdown

Go to Step 11.

DISTRACTORS C+D

a. Place blowdown keylock switch for affected S/G(s) in DRAIN/FILL position

* HIS-3-1427X for S/G A
* HIS-3-1426X for S/G B
* HIS-3-1425X for S/G C

b. Verify S/G sample valve on affected S/G(s) - OPEN

* MOV-3-1427 for S/G A
* MOV-3-1426 for S/G B
* MOV-3-1425 for S/G C

c. Verify blowdown flow control valves - CLOSED

- FCV-3-6278A
- FCV-3-6278B
- FCV-3-6278C

d. Locally close S/G Blowdown Manual Containment Isolation on affected S/G(s)

* SGB-3-007 for S/G A
* SGB-3-008 for S/G B
* SGB-3-009 for S/G C

e. Place Blowdown Isolation Valve Control Switch on affected S/G(s) in OPEN

* CV-3-6275A for S/G A
* CV-3-6275B for S/G B
* CV-3-6275C for S/G C

f. **WHEN** Blowdown Isolation Valve on affected S/G(s) is open, **THEN** locally open S/G Blowdown Manual Containment Isolation

* SGB-3-007 for S/G A
* SGB-3-008 for S/G B
* SGB-3-009 for S/G C

QUESTION 27

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	E16	EK3.2
	Importance Rating	2.9	

Knowledge of the reasons for the following responses as they apply to the (High Containment Radiation) Normal, abnormal and emergency operating procedures associated with (High Containment Radiation).

Proposed Question: RO Question # 27

- Containment High Range Area Radiation Monitors (CHRRMS) are 6E5 R/HR.

In accordance with Response to High Containment Radiation Level, 3-EOP-FR-Z.3, which ONE of the following describes (1) the bases for starting Emergency Containment Filter Fans and (2) the reason for re-energizing the Containment Purge Isolation Valves?

A. (1) To reduce the iodine concentration in the Containment atmosphere
(2) To ensure Containment Ventilation Isolation

B. (1) To reduce the radioactive noble gas concentration in the Containment atmosphere
(2) To lower Containment pressure when normal methods are unavailable

C. (1) To reduce the iodine concentration in the Containment atmosphere
(2) To lower Containment pressure when normal methods are unavailable

D. (1) To reduce the radioactive noble gas concentration in the Containment atmosphere
(2) To ensure Containment Ventilation Isolation

Proposed Answer: A

Explanation (Optional):

A. CORRECT. The SI signal starts the emergency containment filtration (ECF) system filter fans. During the accident, iodine removal is by the ECF system filters. Also, the steps of 3-EOP-FR-Z.3 re-energize the Containment Purge Isolation Valves to verify Containment Ventilation Isolation to limit any release.

B. Incorrect because the Emergency Filters are started specifically to remove iodine. Plausible because it could be easily assumed that the Containment Spray System operation is sufficient to remove iodine and that the Emergency Filters were required to remove other radioactive products. Also, the applicant believes steps of 3-EOP-FR-Z.3 re-energize the Containment Purge Isolation Valves to lower Containment pressure when normal methods are unavailable which is not correct per this guidance.

C. Incorrect. Plausible – The first part is correct. Also, the applicant believes steps of 3-EOP-FR-Z.3 re-energize the Containment Purge Isolation Valves to open the valves for purging Containment to lower Containment pressure when normal methods are unavailable which is not correct per this guidance.

D. Incorrect because the Emergency Filters are started specifically to remove iodine. Plausible because it could be easily assumed that the Containment Spray System operation is sufficient to remove iodine and that the Emergency Filters were required to remove other radioactive products. Also, the second part is correct.

Technical Reference(s): 3-EOP-FR-Z.3, Response to High Containment Radiation Level (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902129, Obj. 2
LP 6902338, Obj. 20 (As available)

Question Source: Bank # 70885

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 VC Summer

Question Cognitive Level: Memory or Fundamental Knowledge (1B)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Verify Containment And Control Room Ventilation Isolation**	
	a. Unit 3 containment purge exhaust and supply fans - OFF	a. Manually stop fans.
	b. Containment Purge Supply and Exhaust Isolation valves - CLOSED • POV-3-2600 • POV-3-2601 • POV-3-2602 • POV-3-2603	b. Manually close valve(s). **IF** any valve can **NOT** be closed, **THEN** behind VPB, pull fuse for any open valve(s). * XEP for POV-3-2600 * XLAG for POV-3-2601 * XEQ for POV-3-2602 * XLAH for POV-3-2603
	c. Containment Instrument Air Bleed Isolation valves - CLOSED • CV-3-2819 • CV-3-2826	c. Manually or locally close valves.
	d. Verify Control Room ventilation status panel - PROPER EMERGENCY RECIRCULATION ALIGNMENT	d. Manually align equipment for Control Room emergency recirculation.
2	**Verify Emergency Containment Filter Fans - AT LEAST TWO RUNNING**	Manually start filter fans to ESTABLISH AT LEAST TWO RUNNING FANS.
3	**Notify Plant Personnel To Monitor Radiation Levels While Performing Local Actions**	
4	**Notify TSC Staff Of Containment Radiation Levels To Obtain Recommended Action**	
5	**Return To Procedure AND Step In Effect**	

END OF TEXT

FINAL PAGE

BASIS DOCUMENT

WOG Procedure Step 2 **PTN Procedure Step** 2

Verify Emergency Containment Filter Fans - AT LEAST TWO RUNNING

BASIS:

This step instructs the operator to place the containment atmosphere filtration system in service if possible. This system, which is part of the Turkey Point design, will reduce by filtration the radioactivity of the containment atmosphere.

STEP DEVIATIONS FROM WOG GUIDELINES:

TYPE DESCRIPTION

2 At Turkey Point the emergency filtration system consisting of running at least two emergency containment filter fans which are automatically started on a SI signal. This step was rewritten to again verify this automatic action. This was required to address this difference from the reference plant and satisfy the intent of the WOG step.

Correct Answer (A)

PLANT SPECIFIC SETPOINTS:

N/A

Abnormal Operations

Any of the following conditions actuate a trip of Containment Purge System supply fans, exhaust fans, associated dampers and containment isolation valves: (containment ventilation isolation signal)

a. High containment activity on Containment Radiation Monitor, R-11

b. High containment gases activity on Containment Radiation Monitor, R-12

c. Safety Injection Signal (automatic or manual).

d. Phase A isolation or Phase B isolation

Precautions/Limitations

Refer to current revision of 0-OP-053 for the detailed list of Precautions and Limitations.

4.0 EMERGENCY CONTAINMENT COOLING AND FILTERING SYSTEMS (055, 056)

4.1 PURPOSE/FUNCTIONS

The functions of the emergency containment cooling and filtering system are as follows:

1. Provides cooling of the Containment atmosphere following a DBA, thus reducing the Containment pressure.
2. Filters iodine from the Containment atmosphere following a DBA.

Correct Answer (A)

INSTRUCTOR ACTIVITY

PPT Slides 48 - 52

4.2 DESIGN BASES

A. Helps to prevent exceeding containment design pressure and temperature following a DBA

B. Reduction of iodine concentration following a DBA, ensuring off-site radiation doses at the site boundary do not exceed guidelines of 10 CFR 100.

Correct Answer (A)

C. One train of containment spray or two of the three ECCs can provide the heat removal capability (120 Million BTU/hr) to maintain the post accident containment temperature and pressure below the design values, however, the design and licensing basis LOCA analysis assumes the use of both redundant systems. This design basis was used for equipment qualification.

D. One or two of the three ECCs work in conjunction with one train of containment spray to maintain the containment temperature and pressure within the design basis equipment qualification envelopes.

Background Information

The design basis for containment heat removal, and the basis for containment pressure transient calculations in the FSAR chapter 14 safety analyses, assumes that one of the three ECCs and one train of containment spray actuate for post-LOCA heat removal and that a second ECC will be running within the first 24 hours following the accident. The temperature and pressure profiles generated by the chapter 14 analysis were used as the basis for equipment qualification.

Long term equipment qualifications are ensured by making sure the second ECC is running within 24 hours of the accident.

The design basis containment pressure is 55 PSIG. This was conservatively established for the design basis LOCA analysis, which bounds the MSLB analysis.

During the licensing process for Turkey Point, some hypothetical, beyond-the-licensing basis scenarios were developed for the design basis LOCA event. These resulted in a peak calculated containment pressure of 58.5 PSIG. To accommodate these hypothetical, beyond-the-licensing basis scenarios, the containment structure was designed to withstand a pressure of 59 PSIG.

To sum it all up, the containment is physically designed to withstand 59 PSIG, the containment design basis pressure is 55 PSIG, and the containment pressure calculated for a design basis LOCA is 49.9 PSIG. Containment design maximum temperature is 283°F while equipment qualification limits temperature to 276°F.

4.2 DESIGN BASES

The emergency containment cooling and filtering system works in conjunction with the containment spray system under accident conditions to prevent exceeding containment design pressure and temperature as a result of a DBA. The cooling system itself is designed principally to remove sufficient heat from the containment following a DBA to keep containment pressure from exceeding design pressure. The filtering system is designed to reduce the iodine concentration in the containment atmosphere following a DBA to levels ensuring that the off-site radiation dose will not exceed the guidelines of 10 CFR 100 at the site boundary. The design capacity of the filtering system was based on the following conditions:

1. Postulated iodine release to the Containment was calculated with the ORIGEN2 code using TID 1484 release fractions at a power level of 2346 mw(t) based on a 24,000 mwd/mtu 2 region equilibrium cycle equilibrium fission product inventory.
2. Twenty-five percent of the total core iodine inventory is available for leakage from the containment. This assumes 50% of the total core iodine is released to containment and 50% of this activity immediately plates out on the containment walls.
3. 0.25% per day Containment leak rate for the first 24 hours and 0.125% per day thereafter.
4. 4% of total iodine in Containment atmosphere is methyl iodine, 91% is elemental iodine and 5% is particulate iodine.
5. Two of the three filtering units in operation for 2 hours.
6. 90% filter efficiency for elemental iodine, 95% filter efficiency for particulate iodine, and 30% filter efficiency for methyl iodine.

The design basis for containment heat removal, and the basis for containment pressure transient calculations in the FSAR chapter 14 safety analyses, assumes that one of the three ECCs and one train of containment spray actuate for post-LOCA heat removal and that a second ECC will be running within the first 24 hours following the accident. The temperature and pressure profiles generated by the chapter 14 analysis were used as the basis for equipment qualification.

The emergency containment cooling system is used during post accident conditions in conjunction with the containment spray system to remove heat and iodine from the containment atmosphere which could be a result of the LOCA or steam break accident. The system consists of three ECCs for heat removal and three ECFs for iodine removal. The ECCs and ECFs start on a safety injection signal.

Correct Answer (A)

QUESTION 28

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003	A3.01
	Importance Rating	3.3	

Ability to monitor automatic operation of the RCPS, including: Seal injection flow

Proposed Question: RO Question # 28

Given the following:

- Unit 3 is operating at 100% power.
- Charging Pump 3A is running in Automatic.
- HCV-3-121, Charging Flow to Regen Hx, is at 50% demand.
- HCV-3-121, Charging Flow to Regen Hx, fails due to an air leak.

Which of the following correctly describes (1) the failure position of HCV-3-121, Charging Flow to Regen Hx, and (2) the effect on Seal Injection Flowrate?

(Assume NO operator action with controls in automatic.)

A. (1) opens
(2) Seal Injection flow rises

B. (1) opens
(2) Seal Injection flow lowers

C. (1) closes
(2) Seal Injection flow rises

D. (1) closes
(2) Seal Injection flow lowers

Proposed Answer: B

Explanation (Optional):

A. Incorrect since Seal Injection flow rises. Plausible – The applicant assumes HCV-3-121, Charging Flow to Regen Hx, fails closed which would protect the RCP Seals by sending more flow to them.

B. CORRECT. HCV-3-121, Charging Flow to Regen Hx, fails open which allows less restriction to flow to the RCS and more Charging flow.

C. Incorrect since HCV-3-121 fails open. Plausible – The applicant assumes HCV-3-121, Charging Flow to Regen Hx, fails closed which would force more flow to the RCP Seals.

D. Incorrect since HCV-3-121 fails open and Seal Injection flow rises. Plausible – The applicant assumes HCV-3-121, Charging Flow to Regen Hx, fails closed which causes total charging flow to lower.

Technical Reference(s): LP 6902108, RCPs (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902108, Obj 4.a (As available)

Question Source: Bank # 47503

Modified Bank # (Note changes or attach parent)

New

Question Difficulty Level: B

Question History: Last NRC Exam: 2003 Point Beach

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

CAUSES:
1. Improper adjustment of HCV-3-121
2. Loss of seal injection flow
3. Number 1 seal failure

A42

RCP LABYRINTH SEAL LO ΔP

DEVICE:
- PC-3-131 for A RCP
- PC-3-128 for B RCP
- PC-3-125 for C RCP

SETPOINT:
10" H_2O

LOCATION:
N/A

ALARM CONFIRMATION

1. **CHECK** the following less than 10" H_2O on VPA:
   - PI-3-131A, A RCP THERMAL BARRIER D/P
   - PI-3-128A, B RCP THERMAL BARRIER D/P
   - PI-3-125A, C RCP THERMAL BARRIER D/P

OPERATOR ACTIONS

1. **ENSURE** charging flow adequate for plant conditions.

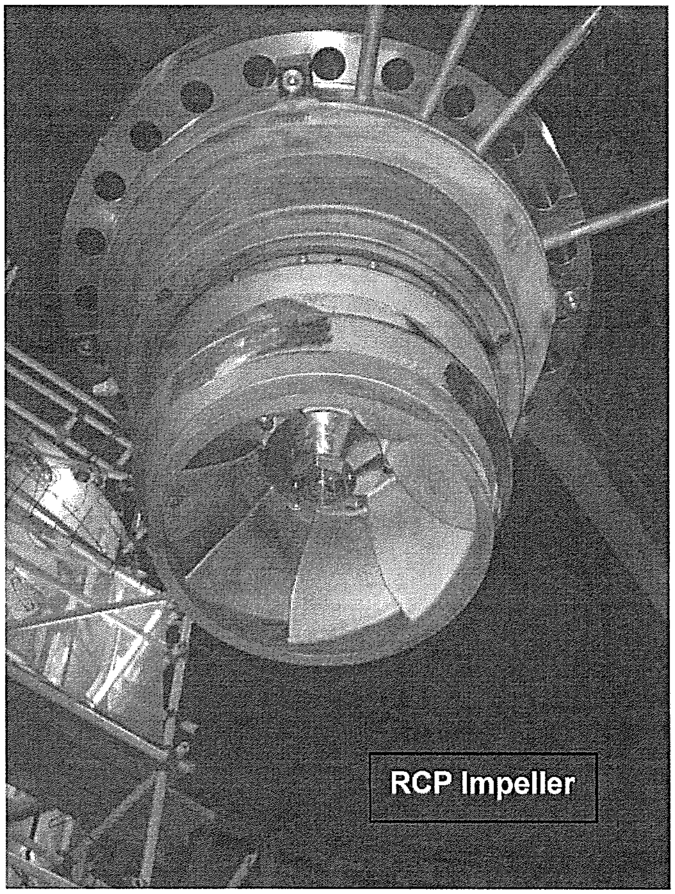
> **CAUTION**
>
> Care must be exercised when throttling HCV-3-121 in the closed direction. Throttling this valve completely closed can cause the Charging Pump discharge relief valve to lift resulting in a possible loss of charging if the relief valve fails to reseat.

2. **ADJUST** HCV-3-121, CHARGING FLOW TO REGEN HX to increase seal injection flow.
3. IF seal injection flow to all 3 RCPs are **NOT** balanced, THEN **ADJUST** the following valves locally to obtain a flow rate between 8 to 20 gpm for each pump:
   - 3-297A for A RCP.
   - 3-297B for B RCP.
   - 3-297C for C RCP.
4. IF charging flow is lost, THEN **ENSURE** MOV-3-626, RCP THERMAL BARRIER CCW OUTLET is OPEN to supply CCW to the RCP Thermal Barriers.
5. IF #1 seal is damaged, THEN:
   A. **CHECK** for high seal leak off flow, FR-3-154A, RCP SEAL LEAK OFF - WIDE RANGE on VPA.
   B. **REFER TO** 3-ONOP-041.1, Reactor Coolant Pump Off Normal.
6. **REFER TO** Tech Spec 3.4.1

REFERENCES:
1. FPL Drawing 5613-M-3047, Sheet 3, CVCS – Seal Water Injection to RCP
2. Tech Spec 3.4.1.1, 3.4.1.2, 3.4.1.3, and 3.4.1.4

INSTRUCTOR ACTIVITY

RCPs take suction on the 31" I.D. intermediate leg and discharge to the 27.5" I.D. cold leg. From a hydraulic design standpoint it is ideal to have approximately three to five pipe diameters of straight pipe before a pump suction. This is not possible because of the limitation to minimize coolant piping volume to limit the release of radioactivity in the event of a pipe rupture and also to limit the size of the required containment structure. Instead, a larger pipe diameter was selected to reduce pressure drop and turbulence at the pump suction.



The major components of the RCP pump consist of the pump casing, impeller, diffuser, thermal barrier, lower radial bearing, pump shaft, and pump flange.

The rotating impeller draws coolant up axially through the bottom of the casing. The impeller imparts velocity to the fluid and directs it radially into the diffuser. The diffuser consists of a series of stationary vanes with expanding cross-sectional area. The fluid gains pressure as it expands through the vanes with a corresponding loss of velocity. After passing through the diffuser the coolant enters the main discharge bowl of the pump casing and exits through the discharge nozzle.

The pump impeller is shrunk and keyed to the bottom of the shaft. Above the impeller is the thermal barrier containing a heat exchanger. This limits heat transfer between the reactor coolant (547°F) and the seal injection water (130°F). The seal 'O' rings are not designed to withstand normal RCS operating temperatures. Therefore, some deterioration may occur above 350°F. The cooling medium is component cooling water.

High pressure seal injection water from the discharge of the charging pumps is admitted just above the thermal barrier. At this point the flow splits with 5 GPM flowing down past the thermal barrier and into the RCS. The remaining 3 GPM of the total 8 GPM per pump flows upward along the pump shaft to cool and lubricate the lower radial bearing. From the radial bearing, injection flow continues upward to the mechanical seals. This seal injection flow split is important to radial bearing and seal operation.

OBJECTIVE 4.a	NSO
OBJECTIVE 4.b	LPRO/LPSO

General Description	Slide(s) 34

General Description

- High pressure seal injection just above the thermal barrier at 8 GPM.
  - 5 GPM down past the thermal barrier into the RCS
  - 3 GPM up along pump shaft to cool the shaft and lubricate the lower radial bearing. – ***this flow is critically important!***
- A function of the thermal barrier is to help achieve this flow distribution.
- Thermal barrier heat exchanger also provided to cool the RCS flowing up through the shaft in the event of a loss of seal injection flow.
- The thermal barrier also acts as a low pressure seal during seal maintenance.
- The seal system has:
  - #1 seal – a film riding seal with controlled leakoff – 1–3 GPM leakoff
  - #2 seal – face rubbing seal that has its leakage diverted to the RCP Standpipe – standpipe provides backpressure on the seal. – 2–3 GPH
  - #3 seal – also a face rubbing seal that "leaks" to the containment sump – 3–100 cc/hr.
- RCP motors are air cooled with the upper and lower bearing oil coolers cooled by CCW - Class B thermoplastic epoxy insulated, squirrel cage induction.
  - Upper bearing – Kingsbury type thrust and radial guide bearing
  - Lower – pivoted-pad radial bearing
  - Top of the motor is a flywheel and anti-reverse rotation device.

Q: What is the difference between film riding and face rubbing seal?

A: Film riding - seals are not in contact, separated by a thin film of water.
Face rubbing - contact between seal ring and runner.

INSTRUCTOR ACTIVITY

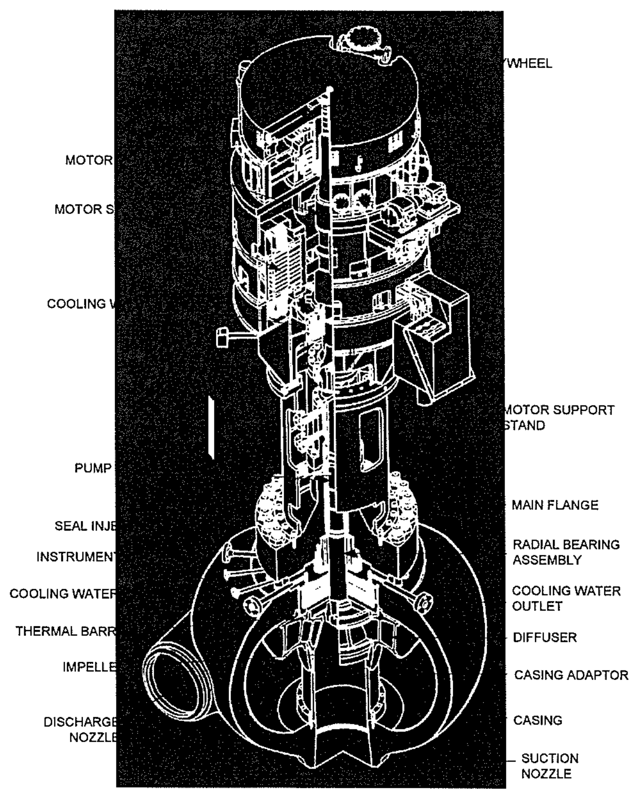
This function of the thermal barrier is to help achieve the correct flow split by providing flow resistance to the seal water entering the RCS. To accomplish this function a labyrinth seal is machined into the surface of the thermal barrier housing adjacent to the pump shaft.

A related function of the thermal barrier is to cool the upward flow of reactor coolant to the radial bearing and seals in the event of a loss of seal injection flow. The labyrinth seals will help limit the RCS flow up to the shaft seals.

The lower inside diameter of the flanged casing is beveled and overlaid with a stellite surface. When the motor is uncoupled, the shaft and impeller drop slightly to form a seal with a similar stellited surface on a shoulder of the pump shaft.

The seal system controls the upward flow of high pressure injection water. The system consists of three mechanical seals. #1 seal is a film-riding seal with controlled leakoff of approximately 3 GPM going to the volume control tank in the CVCS system. Numbers 2 & 3 seals are face rubbing seals. #2 seal leak-off, approximately 3 GPH, diverts to the reactor coolant drain tank via a standpipe. The standpipe provides back-pressure for #2 seal. #3 seal leakoff, approximately 100cc/hr, goes directly to the containment sump.

REACTOR COOLANT PUMP



SD 008 FIGURE 1 Rev.1:3/12/91

The RCP motors are air-cooled, Class B Thermoplastic Epoxy-insulated, squirrel cage induction motors. The air cooled rotor and stator are of standard construction. Six resistance temperature detectors (RTD) are located throughout the stator windings. A double acting Kingsbury thrust bearing and radial guide bearing are located above the stator. These two bearings are oil cooled and lubricated. The oil is circulated through an external oil cooler cooled by component cooling water. A separately lubricated and cooled radial bearing is located below the stator. The heat exchanger for this bearing also uses component cooling water and is located in the bearing oil pot.

INSTRUCTOR ACTIVITY

OBJECTIVE 3	**NSOI. LPRO, LPSO**
OBJECTIVE 9	**NSOI, LPRO, LPSO**

General Description:

Review Figure 1

- Low pressure letdown valve PCV-3(4)-145 (O.C.).
- High temperature demineralizer diverting valve TCV-3(4)-143 (O.C.).
- One of the two paths (O.C.).
  a. Demineralizers (Purification Section) and then to the Reactor Coolant Filters and then VCT or
  b. Directly to Reactor Coolant Filters (3) then VCT
- Volume Control Tank (VCT) Level Control Valve LCV-3(4)-115A (3- way valve).
  a. Normal path-VCT
  b. Alternate (high level in VCT) -CVCS holdup tanks.

1.3.2 Charging (feed) - Two Main Suction Paths:

- **Path one:** VCT Outlet Isol Vlv LCV-3(4)-115C
- **Path two:** Refueling Water Storage Tank (RWST) to Chg Suction Vlv (LCV-3(4)-115B) or manual valve 3(4)-358

Charging Pumps (3) to either of two discharge paths

- **Path one:** Back pressure vlv HCV-3(4)-121 to the Regenerative Heat Exchanger (tube side)
- **Path two:** Seal water inj filters to the RCP Seal injection lines via 297A, B, or C

OBJECTIVE 3.a, 9.a

After leaving the non-regenerative heat exchanger letdown flow undergoes a second pressure reduction. This pressure reduction is accomplished by modulation of the letdown pressure control valve, PCV-3(4)-145.

After leaving the pressure control valve, letdown flow passes through the purification portion of CVCS. This subsystem consists of five mixed bed demineralizers, and three particulate filters. The demineralizers remove impurities from the water by ion exchange and mechanical filtration. Along with impurities certain fission products will undergo ion exchange.

If the non-regenerative heat exchanger outlet temperature reaches 135°F as sensed by TT-3(4)-143, TCV-3(4)-143 will automatically divert flow around the demineralizers to the volume control tank via the Reactor Coolant System (RCS) filters.

The letdown flow leaving the mixed bed demineralizer normally passes directly through two of the three, ≤25 micron, reactor coolant filters. The filters are used to collect resin fines and other particulate matter.

Regardless of the purification flow path selected, the filtered letdown flow next passes through a diversion valve LCV-3(4)-115A. It then enters the volume control tank (VCT) through a spray nozzle in the gas space of the tank.

The diversion valve LCV-3(4)-115A, positions as a function of VCT level to either pass flow to the VCT or divert it to the CVCS holdup tanks. This is the normal method for decreasing RCS water inventory should it become excessive.

1.3.2 Charging:

The purified, hydrogenated coolant flows from the bottom of the VCT through a motor operated isolation valve, LCV-3(4)-115C to the suction of the charging pumps.

Charging pump suction may also be aligned from one of two paths from the RWST via LCV-3(4)-115B or manual valve 3-(4)-358.

There are three positive displacement charging pumps rated at 77GPM each. Normally only one pump is operated to makeup for letdown flow and supply seal injection water to the RCPs.

A manual flow control valve, HCV-3(4)-121 located in the charging line is positioned to provide sufficient backpressure to cause proper RCP seal injection flow.

Prior to branching off to the pumps, the water passes through one of the two full capacity seal water injection filters. Then through the throttled seal injection valves 3(4)-297A, B, & C.

Downstream of HCV-3(4)-121 charging water penetrates containment and passes through the tube side of the regenerative heat exchanger. It is preheated from ≈130°F to ≈493°F. The charging line then branches into three parallel flow paths.

QUESTION 29

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004	A1.07
	Importance Rating	2.7	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: Maximum specified letdown flow

Proposed Question: RO Question # 29

Given the following:

- Unit 4 is at 100% power.
- Pressurizer level is 55% and stable following a charging pump IST.
- CV-4-200C is in service.
- Excess Letdown is in service.
- VCT level is 30% and stable.

The crew places another letdown orifice in service, CV-4-200A to lower Pressurizer level. No further operator action is taken.

Which ONE of the following describes (1) if a letdown design operating limit is exceeded when CV-4-200A is placed in service and (2) the consequence of the Low Pressure Letdown Valve, PCV-4-145, failing OPEN?

A. (1) Maximum letdown flow HAS been exceeded.
(2) Flow initially increases along with pressure lowering.

B. (1) Maximum letdown flow has NOT been exceeded.
(2) Flow initially increases along with pressure lowering.

C. (1) Maximum letdown flow HAS been exceeded.
(2) Flow initially decreases along with pressure rising.

D. (1) Maximum letdown flow has NOT been exceeded.
(2) Flow initially decreases along with pressure rising.

Proposed Answer: B

Explanation (Optional):

A. Incorrect since the maximum letdown flow (per 4-OP-047, Precaution 4.2) of 120 gpm has not been exceeded. When CV-4-200A was placed in service, this raised letdown flowrate from a nominal 60 gpm to a nominal 105 gpm. Plausible because the 2nd part is correct. Also plausible because placing CV-4-200B in service, vs CV-4-200A, would place nominal letdown flow *at* the maximum limit. Placing all three letdown orifice isolation valves in service *would* exceed the limit.

B. CORRECT. The maximum letdown flow (per 4-OP-047, Precaution 4.2) of 120 gpm has not been exceeded. When CV-4-200A was placed in service, this raised letdown flowrate from a nominal 60 gpm to a nominal 105 gpm. If the LP Letdown valve fails open, flow rises in the letdown line, and because this valve controls backpressure, the pressure in the line will lower

C. Incorrect since the maximum letdown flow (per 4-OP-047, Precaution 4.2) of 120 gpm has not been exceeded. Also incorrect because pressure actually lowers, does not rise. Plausible if the applicant does not realize that this is a reverse acting backpressure control valve. This is the effect if it failed closed

D. Incorrect pressure actually lowers, does not rise. Plausible if the applicant does not realize that this is a reverse acting backpressure control valve. This is the effect if it failed closed. Also plausible because first part is correct

Technical Reference(s): 4-OP-047, *CVCS - Charging and Letdown*

LP 6902113, *CVCS*

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902113, Obj. 9 (As available)

Question Source: Bank # 67185

Modified Bank # (Note changes or attach parent)

New

Question History:	Last NRC Exam:	2009	Millstone 3

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	(2DR)
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	5
	55.43	

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

4.0 PRECAUTIONS/LIMITATIONS

4.1 Before changing system status, Technical Specifications should be consulted for system requirements for that plant mode.

4.2 Design restrictions on demineralizer operation require the letdown flow rate to be maintained below 120 gpm, and the temperature of the water entering the inlet header to be less than 140°F.

4.3 Explosive mixtures of hydrogen and oxygen concentration shall be avoided at all times. The oxygen concentration in the VCT shall be maintained less than or equal to 2 percent by volume when hydrogen is greater than 4 percent.

4.4 The CVCS demineralizers are required to be bypassed prior to adding Hydrazine to the CVCS **EXCEPT** a demineralizer with PRC-01.

4.5 All work performed in the Radiation Controlled Area shall be performed in accordance with the requirements of the Radiation Work Permit and ALARA program.

4.6 When aligning remotely operated valves (i.e., chain operated, reach rods, etc.), the position shall be verified by local valve position. This requirement may be waived by the Shift Manager in cases of significant radiation exposure, which are those areas designated as high radiation areas, or areas deemed inaccessible by the Shift Manager.

4.7 Letdown flow should be maintained through the CVCS demineralizers to maintain system cleanliness. Securing letdown during plant cooldown may result in high dose rates in the RHR System. The RP Supervisor and the Radiochemist shall be notified if letdown is to be secured.

4.7.1 Letdown orifices should not be changed during delithiation operations. **IF** letdown flow has to be changed, **THEN** notify Chemistry so that the delithiation bed run time can be recalculated.

4.8 If a charging pump exhibits primary packing leakage symptoms as described below, then issue a PWO to Mechanical Maintenance Department to repack the pump.

4.8.1 Primary packing leakage of greater than 0.05 gpm: place on Plant Status Sheet and repack within 4 weeks.

4.8.2 Primary packing leakage of greater than 0.08 gpm: place on Plant Status Sheet and repack within 2 weeks.

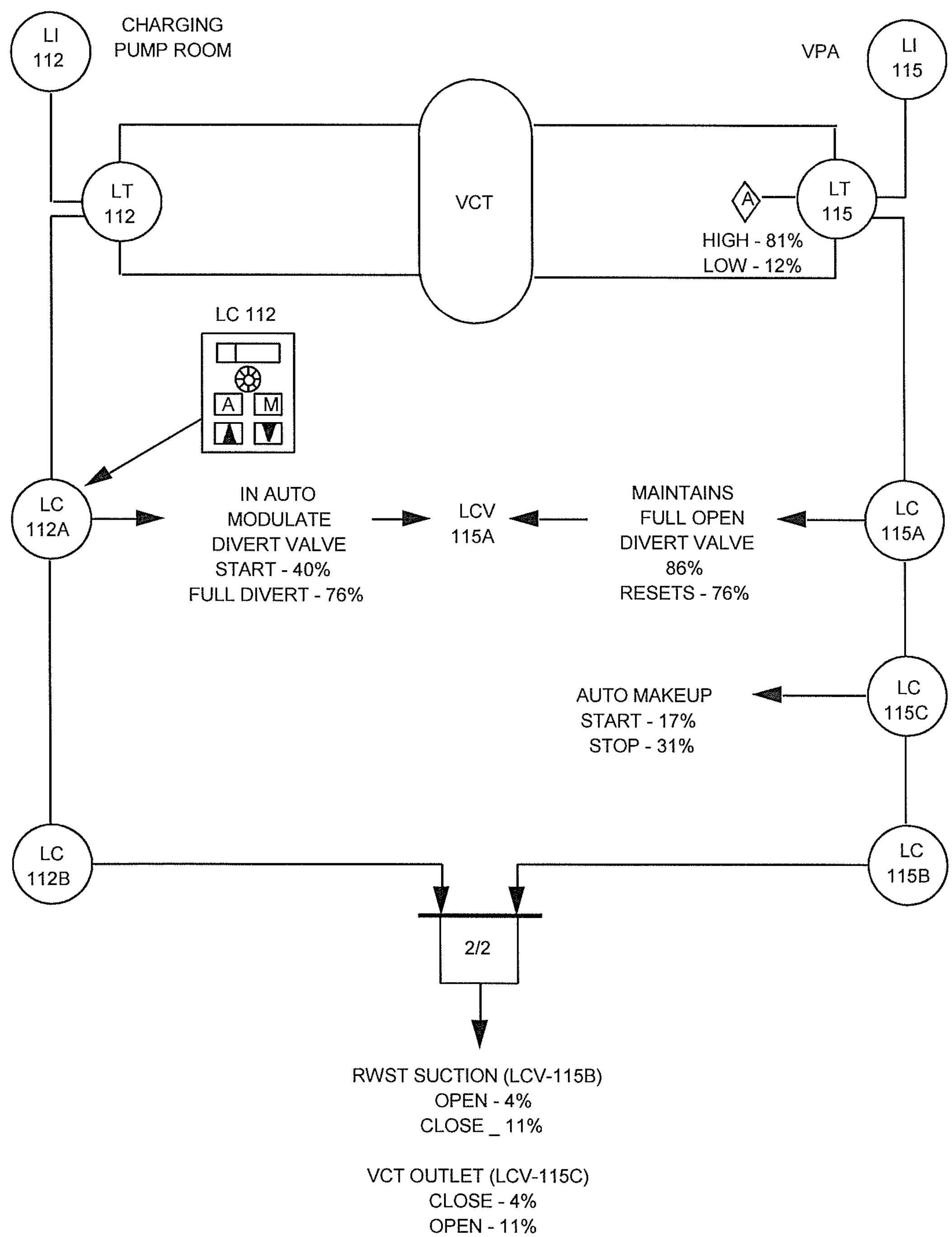
4.8.3 Abnormally high airborne gas concentration in the charging pump room.

4.9 If a charging pump exhibits secondary packing leakage symptoms as described below, then issue a PWO to Mechanical Maintenance Department to repack the pump.

4.9.1 Decreasing seal pot level that requires shiftly seal pot fills.

4.9.2 A steady stream of water leaking out any one of the plungers in the charging pump plunger well.

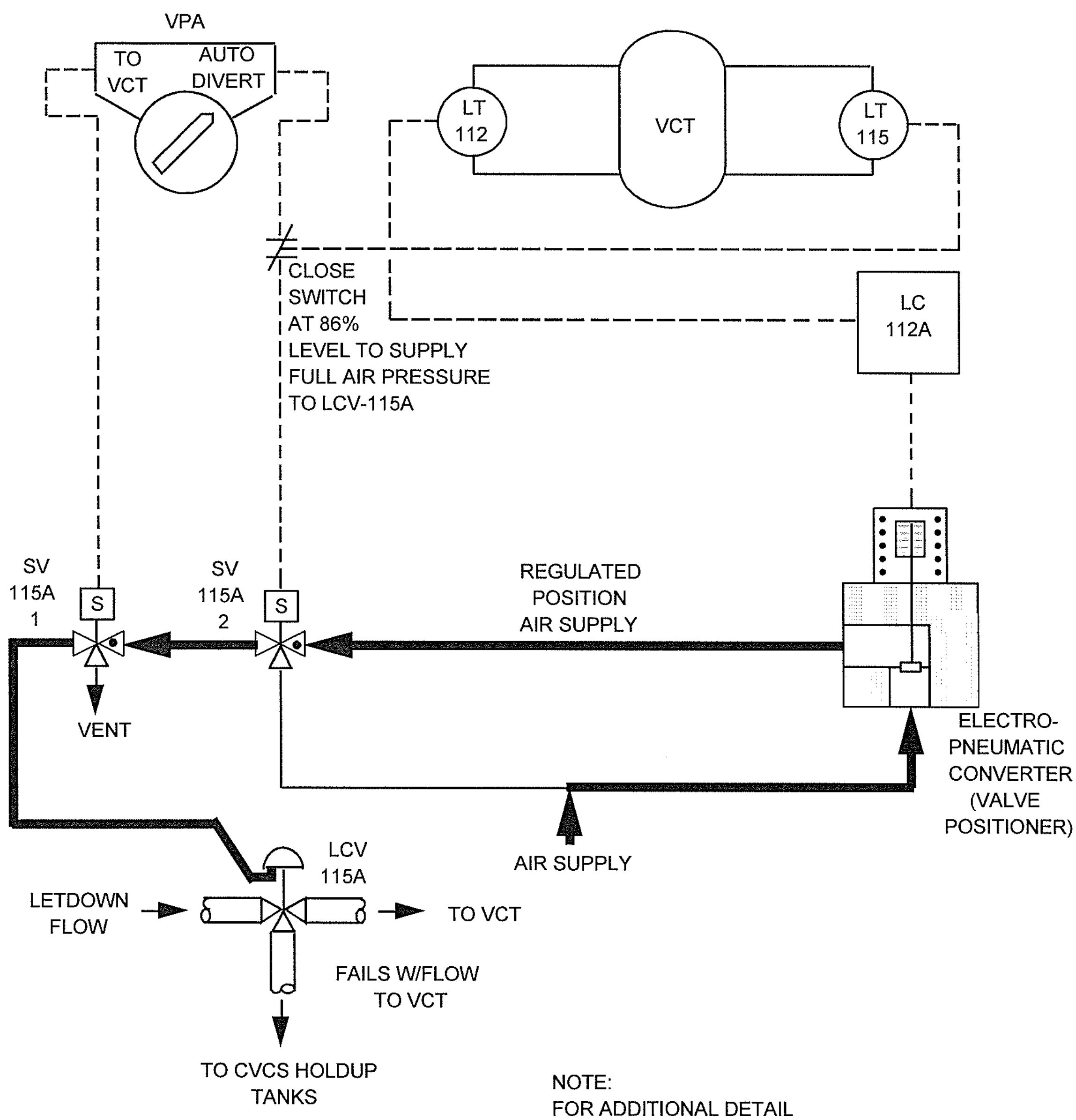
VCT LEVEL CONTROL



OBJECTIVE 3b

FIGURE 19
Rev.6:9/19/96

VCT/HOLDUP TANK DIVERTING VALVE (LCV-115A)

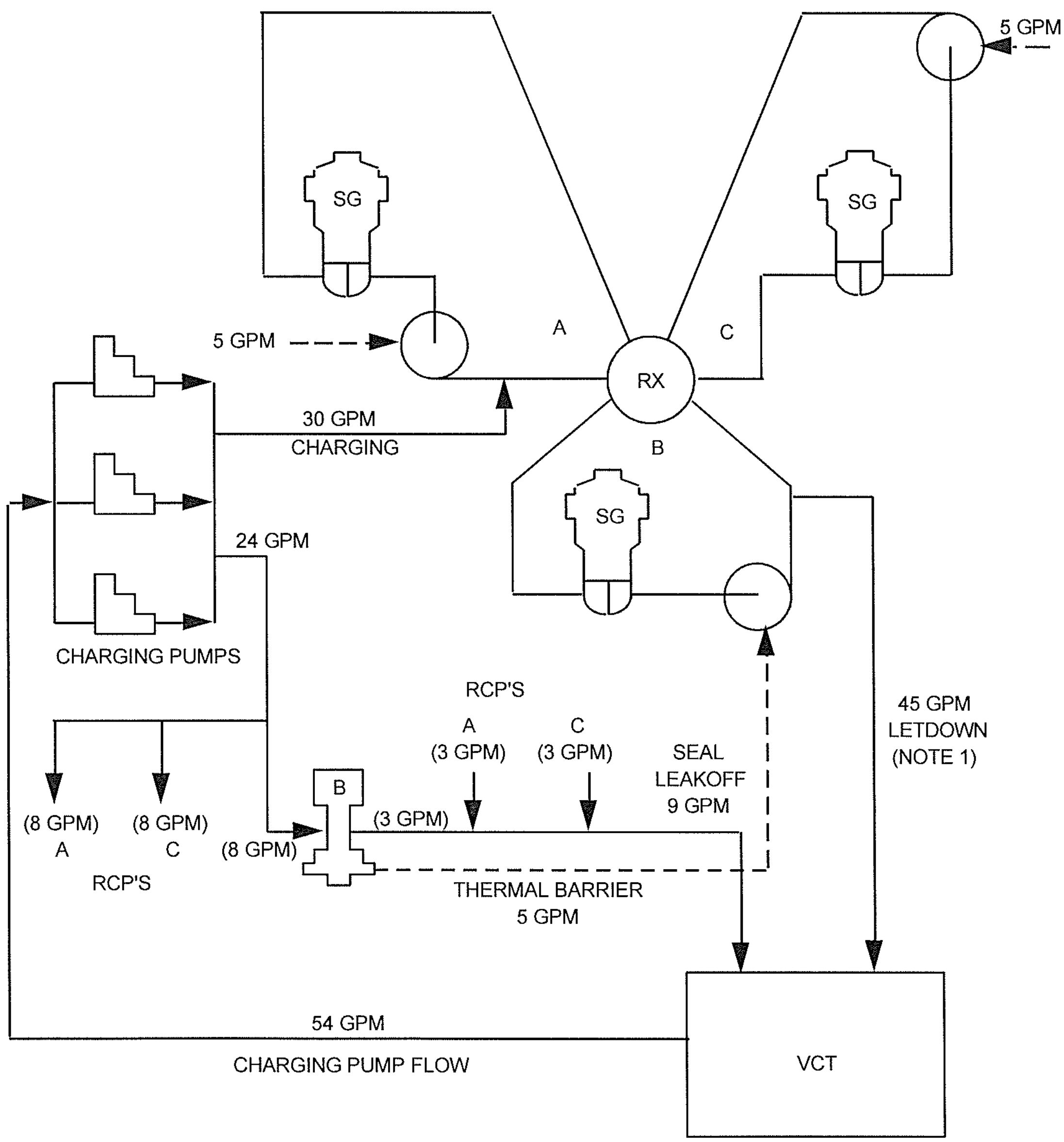


NOTE:
FOR ADDITIONAL DETAIL
SEE 5610-T-D-19

SD 013

FIGURE 20
Rev.6:3/27/96

BASIC CVCS FLOW BALANCE



NOTE 1: 45 GPM ORIFICE IN SERVICE TO MINIMIZE WEAR ON CHARGING PUMPS

SD 013

FIGURE 21
Rev.3:12/7/90

OBJECTIVE 5.a

The letdown flow stream passes through the shell side and the charging flow returns through the tube side. This arrangement places a lower design pressure limit on the shell as is typical of shell and tube heat exchangers. The unit is of stainless steel all welded construction. The heads and tube sheets are welded to the shells and the tubes are welded to the tube sheets. It is designed for 200 cycles of shell side fluid temperature changes from 560°F to 140°F over a 20-hour period at a maximum rate of 100°F/hr. This is based on 5 cold shutdowns per year for the 40-year plant life. The regenerative heat exchanger is located behind the Bio-wall inside containment.

The Regenerative Heat Exchanger tube side is protected from thermally induced overpressure transients by a ½" bypass line around CV-3(4)-310A. This protection is required in the abnormal event of charging line isolation CV-3(4)-310A, 310B, and 311 closed while letdown flow still exists. The bypass line contains a restricting orifice, RO-3(4)-311, which limits bypass flow when CV-3(4)-310A is closed to approximately 6.5 gpm. The bypass line also contains a normally open manual isolation valve, 3(4)-385, to allow for isolation for maintenance purposes.

2.3 Letdown Orifices and Orifice Isolation Valves

OBJECTIVE 5.a, 8.b

The three letdown orifices and their downstream isolation valves are located inside containment. Two of the orifices are rated at 60 GPM at normal RCS pressure, and the third orifice is rated at 45 GPM (See Figure 4 for typical orifice design). Normally one 45 GPM orifice is in operation with the other two in standby. One or both of the standby orifices may be used in parallel with the normally operating orifice for either flow control when the RCS pressure is less than normal, or for greater letdown flow during maximum purification or heatup. Each orifice consists of an assembly which provides for permanent pressure loss without recovery. Maximum letdown flow for purification purposes is 120 GPM based on the design capacity of the demineralizers and filters.

Each orifice is placed in and removed from service by operation of its associated isolation valve. The isolation valves are 2", air to open, fail closed, solenoid actuated, globe valves. The valves are designated CV-3(4)-200A, B, and C. CV-3(4)-200A isolates flow through the 45 GPM orifice~~s~~. The valves are manually operated by individual "CLOSE-AUTO-OPEN" switches on VPA.

2.9 Low Pressure Letdown Control Valve

OBJECTIVE 5.b, 6, 7.a, 8.d

PT-3(4)-145 senses letdown pressure downstream of the non-regenerative heat exchanger. The pressure signal is used to automatically position PCV-3(4)-145 via an "AUTO-MANUAL" setpoint station for the second system pressure reduction. PCV-3(4)-145 is normally set to maintain a ≈275 PSIG backpressure on the letdown orifices. This setpoint is designed to prevent flashing downstream of the letdown orifices. Flashing would cause excessive erosion of the orifices.

PCV-3(4)-145 is an air to close, fail open, solenoid actuated, 2" globe valve located in the charging pump room. It is provided with manual inlet, outlet, and bypass valves. The control station is located on the console. The pressure signal from PT-3(4)-145 is also used for indication on VPA and to trigger the LOW PRESSURE LETDOWN LINE HIGH FLOW HIGH PRESSURE alarm on annunciator panel A, window 5/5 at 490 PSIG↑.

2.10 Letdown Flow Measurement

OBJECTIVE 6, 7.a

An orifice is installed in the line between the non-regenerative heat exchanger and PCV-3(4)-145 to generate a DP for FT-3(4)-150. The flow signal generated by FT-3(4)0-150 provides indication on VPA via FI-3(4)-150 and a high flow alarm on annunciator panel A, window 5/5, at 130 GPM increasing. (High pressure and flow utilize the same annunciator.)

2.11 Low Pressure Letdown Line

There are four penetrations to the low pressure letdown line between the non-regenerative Heat Exchanger and high temperature diverting valve, TCV-3(4)-143. Refer to operating diagram 5613(14)-M-3047, Sheet 1.

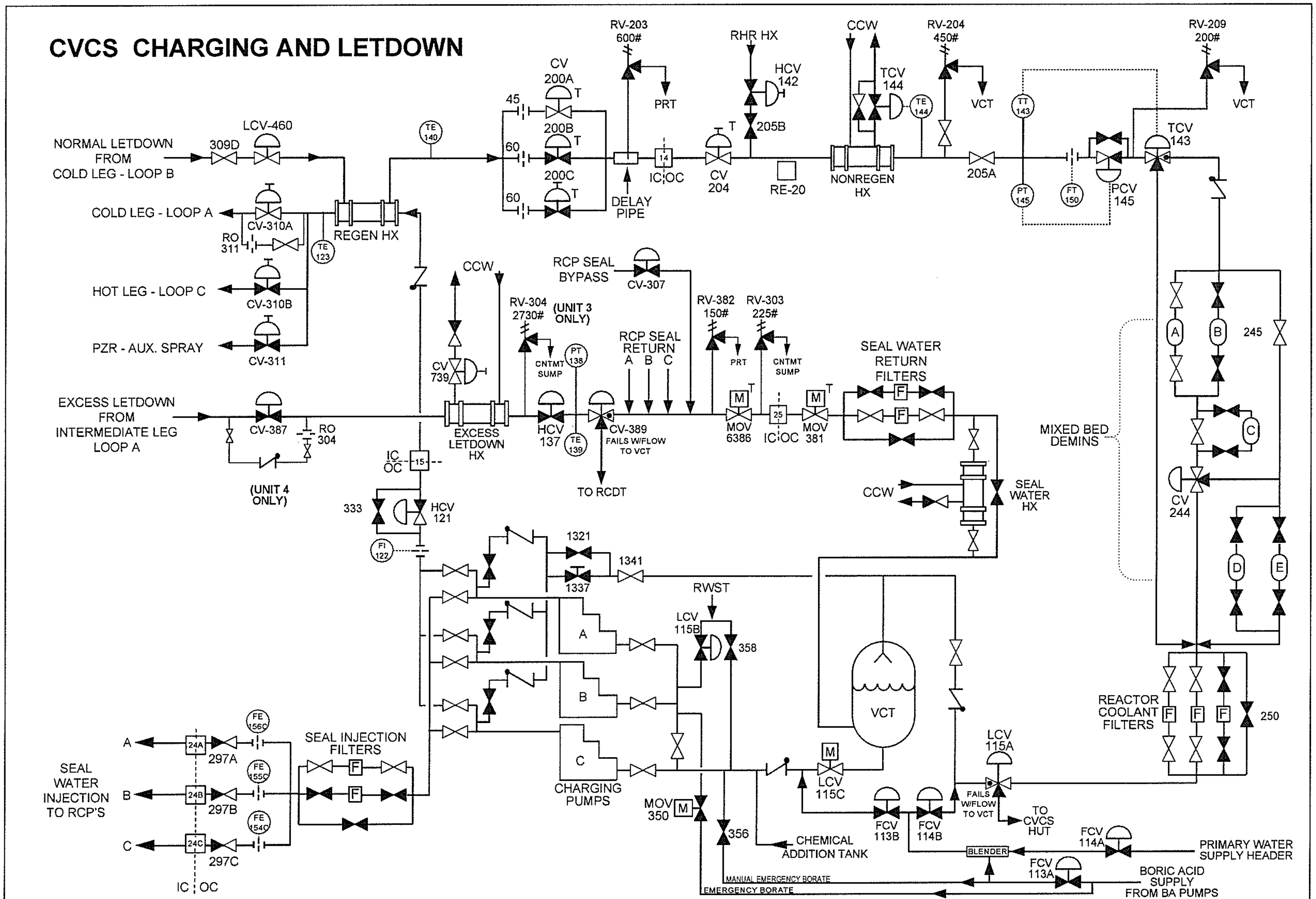
1. One of the penetrations is a tap to the sample system.

OBJECTIVE 5.b, 8.d

2. Another penetration is for a 2" line for the low pressure relief valve, RV-3(4)-209. This valve protects the low pressure piping, demineralizers, and reactor coolant filter from overpressure. The capacity of RV-3(4)-209 is equal to the maximum flow rate through all letdown orifices. It relieves to the VCT when the pressure exceeds 200 PSIG.

3. A 2" penetration is available for purifying the contents of the RWST using the mixed bed demineralizers and the refueling water purification pump. The water is then either returned to the RWST or aligned to the RCS hot leg high head safety injection line during refueling periods.

CVCS CHARGING AND LETDOWN



OBJECTIVE 3a

FIGURE 1

QUESTION 30

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005	K6.03
	Importance Rating	2.5	

Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger

Proposed Question: RO Question # 30

Given the following conditions:

- Unit 3 is in Mode 4 and shutting down for refueling.
- 3B RHR Cooling Train is in service.
- A tube leak occurs in the 3B RHR Exchanger.

Which ONE of the following describes (1) the correct symptom and (2) assuming no operator action(s), the initial effect on the RHR System temperature downstream the 3B Heat Exchanger?

A. (1) CCW Head Tank level lowers
(2) Temperature rises

B. (1) CCW Head Tank level lowers
(2) Temperature lowers

C. (1) RCS level lowers
(2) Temperature lowers

D. (1) RCS level lowers
(2) Temperature rises

Proposed Answer: D

Explanation (Optional):

A. Incorrect since CCW Head Tank level rises. Plausible because the applicant believes CCW discharge pressure is higher than RHR. Also plausible because 2nd part is

correct.

B. Incorrect since CCW Head Tank level rises. Plausible because the applicant believes CCW discharge pressure is higher than RHR. Also plausible because this effect is true if CCW were to leak into the RHR System.

C. Incorrect. Plausible because 1st part is correct. Also plausible because this effect is true if CCW were to leak into the RHR System.

D. Correct.

Technical Reference(s): 3-5613-M-3050 Sheet 1, Unit 3 RHR System (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900121A, Obj. 11 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

QUESTION 31

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	006	K3.02
Importance Rating	4.3	

Knowledge of the effect that a loss or malfunction of the ECCS will have on the following Fuel

Proposed Question: RO Question # 31

Given the following conditions:

- Unit 4 at 100 % power
- The isolation valve (4-867) upstream of MOV-4-843A and B was misaligned closed.

Which ONE of the following describes the impacts of 4-867 being closed when a 300 gpm Small Break Loss of Coolant Accident occurs on a RCS Cold Leg?

(Assume NO operator action.)

A. (1) HHSI flow to the Cold Legs is available.
(2) Core Exit Thermocouples will exceed 1200°F with expected fuel clad damage.

B. (1) HHSI flow to the Cold Legs is available.
(2) Core Exit Thermocouples will remain lower than 1200°F with NO expected fuel clad damage.

C. (1) HHSI flow to the Cold Legs is NOT available.
(2) Core Exit Thermocouples will exceed 1200°F with expected fuel clad damage.

D. (1) HHSI flow to the Cold Legs is NOT available.
(2) Core Exit Thermocouples will remain lower than 1200°F with NO expected fuel clad damage.

Proposed Answer: C

Explanation (Optional):

A. Incorrect since closure of 4-867 will isolate HHSI flow to RCS cold legs. Plausible because the applicant may not associate this valve closure with interrupting SI flow. Also, the applicant

understands SI Accumulators are not isolated. The effect with no initial HHSI flow will be Core Exit Thermocouples will increase greater than 1200°F with expected fuel clad damage.

B. Incorrect since closure of 4-867 will isolate HHSI flow to RCS cold legs. Plausible because the applicant may not associate this valve closure with interrupting SI flow. Also, the applicant understands SI Accumulators are not isolated. The effect with no initial SI flow will be Core Exit Thermocouples will remain lower than 1200°F due to Accumulator injection.

C. Correct. RCS pressure is too high to allow the Accumulators to inject during initial stages of a 300 GPM break. Plausible - Once the loop seal clears, the Accumulators inject. By the time of the Accumulator injection, CETs have climbed above 1200°F.

D. Incorrect since this will cause Core Exit Thermocouples will exceed1200°F with expected fuel clad damage. Plausible because the applicant may not associate this valve closure with loss of all HHSI flow at higher pressures with SI Accumulators available. Also, the applicant understands SI Accumulators will inject eventually and believes Core Exit Thermocouples will remain lower than 1200°F with NO expected fuel clad damage.

Technical Reference(s): LP 6902121, ECCS Safety Injection Accumulators; LP 6902918, Transient and Accident Analysis Loss of Coolant Accidents (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902121b, Obj. 6; LP 6902918, Obj. 5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

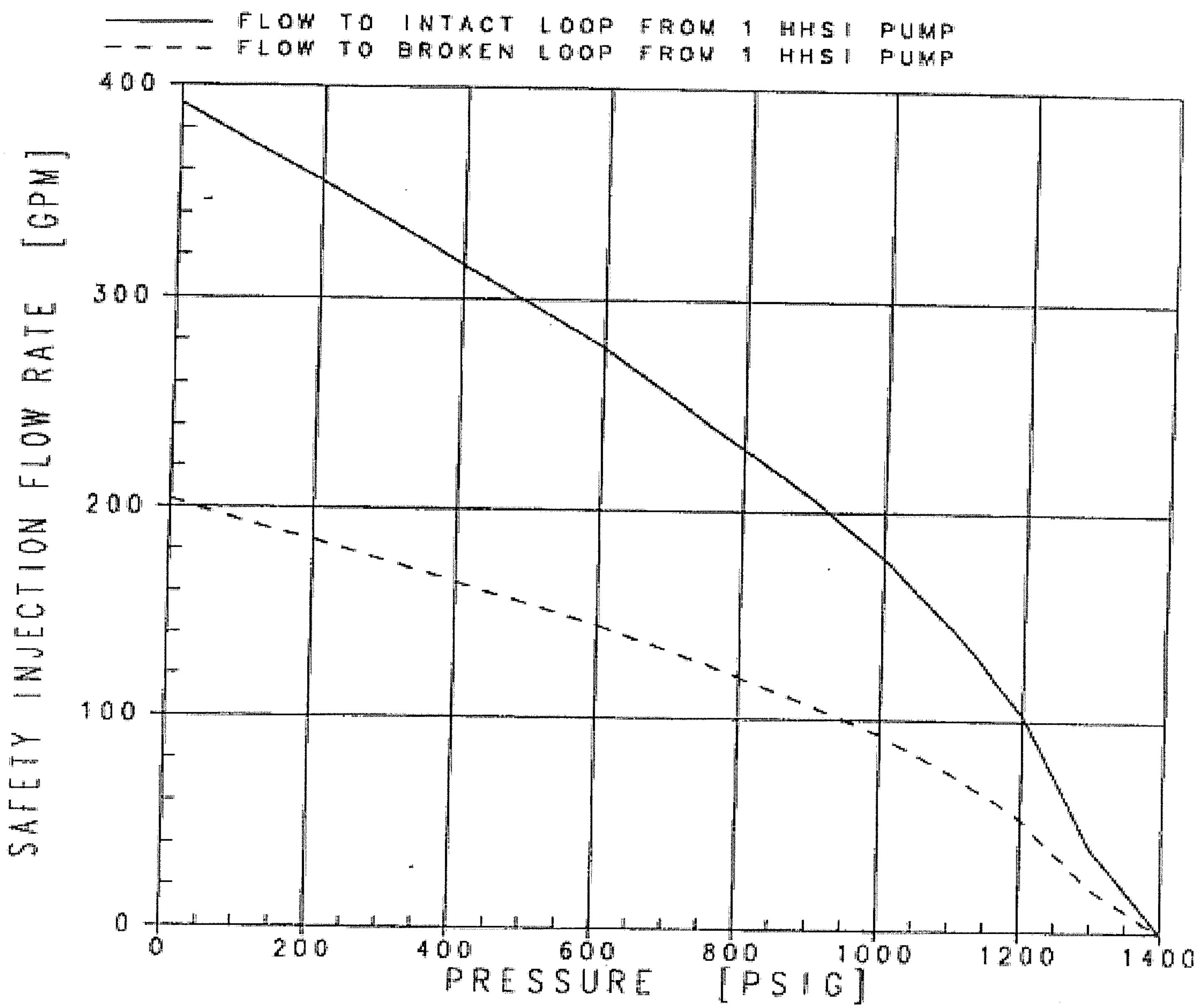
Question Difficulty Level: B

10 CFR Part 55 Content:	55.41	8
	55.43	

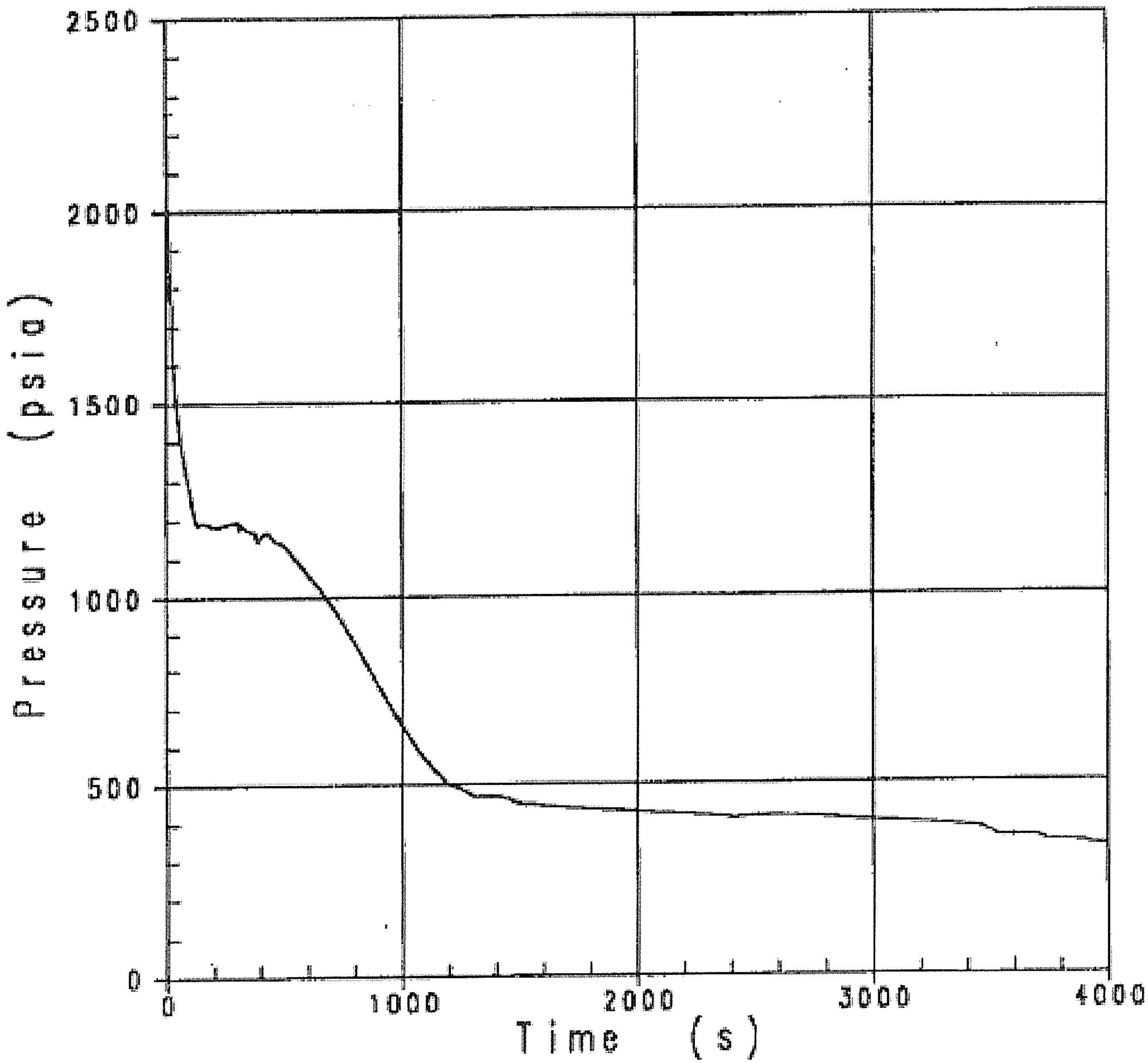
Components, capacity, and functions of emergency systems.

Comments:

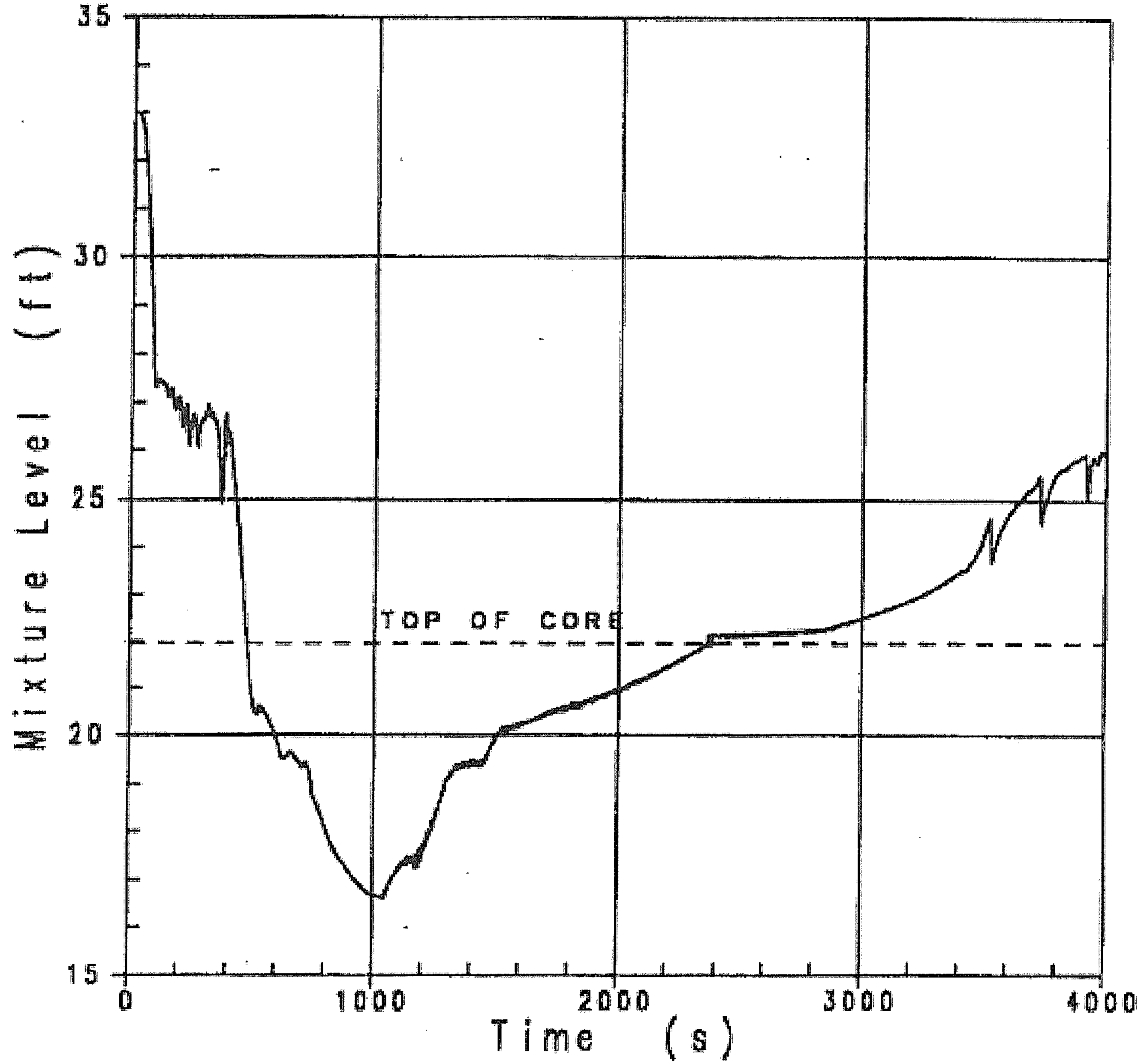
Small Break SI Flow Rate



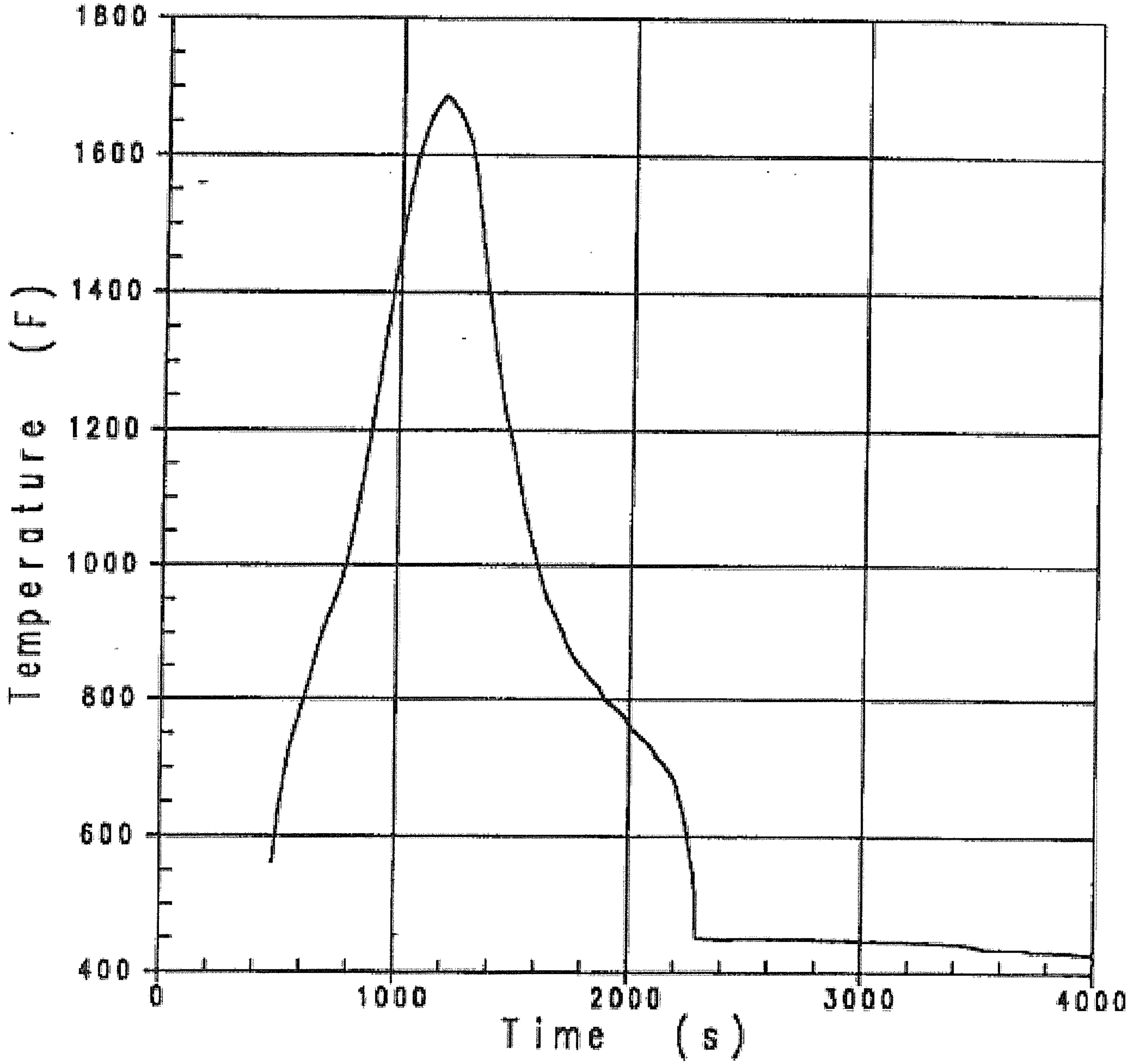
Small Break RCS Depressurization



Small Break Core Mixture Level



Small Break Peak Clad Temperature



Additional Break Cases

<u>Break Spectrum, (High T_{AVG})</u>

	BREAK SIZE		
	2-inch	3-inch	4-inch
Peak Clad Temperature (°F)	1656	1688[2]	1583
Peak Clad Temperature Location (ft)[1]	11.75	11.75	11.50
Peak Clad Temperature Time (sec)	2627	1188	668
Local Zr/H_2O Reaction, Max (%)	2.0188	1.5535	0.6679
Local Zr/H_2O Reaction Location (ft)[1]	11.75	11.50	11.25
Total Zr/H_2O Reaction (%)	< 1.0	< 1.0	< 1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)[1]	N/A	N/A	N/A

QUESTION 32

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	007	K3.01
Importance Rating	3.3	

Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment

Proposed Question: RO Question # 32

Given the following:

- Unit 3 was initially at 100% power.
- A Reactor Trip and Safety Injection (SI) occurred due to Pressurizer Safety Valve RV-3-551B failing open.
- PRT pressure was 85 psig and rising steadily.

Which ONE of the following predicts FUTURE Containment conditions within the next hour?

A. Containment Sump levels will remain constant
Containment Conditions will become ADVERSE

B. Containment Sump levels will rise
Containment Conditions will become ADVERSE

C. Containment Sump levels will remain constant
Containment Conditions will NOT become ADVERSE

D. Containment Sump levels will rise
Containment Conditions will NOT become ADVERSE

Proposed Answer: B

Explanation (Optional):

A. Incorrect. The PRT Rupture Discs will blow at 100 psig nominal pressure. On a vapor space event such as stuck open safety valve, it is predicted that the PRT will relieve to

Containment within 1 hour. The student may believe the vapor contents are directed to the Containment atmosphere in the form of steam with minimal condensation. Therefore, there is not a noticeable increase in the Containment Sump level. Along with this perception, the atmosphere is heated to adverse Containment conditions of 180°F.

B. Correct. The PRT Rupture Discs will blow at 100 psig nominal pressure. On a vapor space event such as stuck open safety valve, it is predicted that the PRT will relieve to containment within 1 hour. The wet vapor contents are directed to Containment atmosphere. Condensation of the wet vapor will collect in the Containment Sump. The atmosphere is heated to adverse Containment conditions of 180°F.

C. Incorrect: The PRT Rupture Discs will blow at 100 psig nominal pressure. On a vapor space event such as stuck open safety valve, the student may believe the vapor contents directed into the PRT are not sufficient to cause a PRT rupture after energy is removed from the quench volume in the tank. Therefore, there is no increase in Containment atmospheric conditions. Containment will not go ADVERSE.

D. Incorrect. The PRT Rupture Discs will blow at 100 psig nominal pressure. On a vapor space event such as stuck open safety valve, it is predicted that the PRT will relieve to containment within 1 hour. The student may believe the energy of wet vapor contents inside the PRT is not sufficient once the PRT is ruptured to cause the Containment conditions to go ADVERSE.

Technical Reference(s): LP 6902168, *Radiation Monitoring and Protection* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902109, Obj. 9, 11 (As available)

Question Source: Bank #

Modified Bank # 64069 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content:	55.41	8
	55.43	

Components, capacity, and functions of emergency systems.

Comments:

Modified from North Anna 2008

CAUSES:
1. RCS leakage
2. Instrument malfunction

G5/3

CNTMT LEVEL INCREASING > 1 GPM

DEVICE:
ERDADS Containment Sump Monitor (SUMPALM_A)

SETPOINT:
Sump level increasing at a rate > 1 gpm
Sump pump on for > 3 minutes
L1546 above alarm limit.

LOCATION:
N/A

ALARM CONFIRMATION

1. **CHECK** the following:
   - Cntmt Sump Recorders R-1418 on VPA, R-6308A and R-6308B behind the RCO desk
   - ERDADS point L1546_A or R.

OPERATOR ACTIONS

1. **MONITOR** RCS parameters for indications of a RCS leak.
2. **MONITOR** Component Cooling Water parameters for indication of a CCW System Leak.
3. **PERFORM** 3-OSP-041.1, Reactor Coolant System Leak Rate Calculation, to determine RCS leak rate.
4. **GO TO** 3-ONOP-041.3, Excessive Reactor Coolant System Leakage, and take actions as directed.
5. **REFER TO** Tech Spec 3.4.6.2.

REFERENCES: Tech Spec 3/4.4.6.2

PRESSURIZER AND RELIEF SYSTEM

Any steam or water discharged by the lifting of a PORV or code safety valve is passed to the pressuri; relief tank (PRT) through a common 12" discharge line that serves all five valves. Refer to Figures 1 a 16. The discharge line is large enough to prevent backpressure at the code safety valves from exceedi 20% of the setpoint pressure (2485 PSIG) at full flow. The function of the PRT is to condense and co (quench) any discharge from the PORV's or safety valves. Normally, the tank is partially filled with wa at or near containment ambient temperature and contains a predominantly nitrogen atmosphere maintain at a pressure of 6 to 8 PSIG by a nitrogen pressure regulator. Sparging nozzles located beneath the wa surface discharge steam into the water volume. The mixing that results condenses and cools t discharged steam. The PRT is equipped with a spray, supplied from the primary makeup water system drain to the suction of the reactor coolant drain tank (RCDT) pumps, a drain to the containment sump, a a connection to the vent header. See Drawing T-E-4501.

PRT Design

The PRT has an internal volume of 1300 cubic feet and is constructed of carbon steel with a corrosic resistant coating on the internal surface. The tank is sized to quench a discharge of about 500 cubic ft. pressurizer steam which is about the volume above the programmed pressurizer water level for full pov (53.3% level). The tank is not designed to accept a continuous discharge from the pressurizer. Desi temperature is 340°F; normal operating temperature is 120°F or less.

PRT Rupture Disks

The tank is protected from a discharge exceeding the design value by two rupture disks which relieve the containment atmosphere. The rupture disks have a combined capacity of 900,000 lbm/hr, saturat steam. This exceeds the combined capacity of the pressurizer safety valves. The tank design pressu 100 PSIG (and the rupture disk setting) is twice the calculated pressure resulting from the maximu safety valve discharge described above. This margin is to prevent deformation of the disks. The tank a rupture disk holder are also designed for full vacuum to prevent tank collapse if the tank contents co without nitrogen being added.

PRT Service Connections

The various PRT service connections are shown diagrammatically on T-E-4501.

upper section of VPB. If the valve was originally open at the time of a Phase A signal, and the Phase signal subsequently clears, the valve will not re-open unless OPEN is momentarily selected on the ha control switch.

PRT Indications and Alarms

Indication of PRT level, pressure, and temperature is provided on the console, section 1, vertical pan Additionally, a common PRESSURIZER RELIEF TANK HI TEMP/HI LEVEL HI PRESS/LOW LEVI alarm is provided on annunciator A, window 7-1. This common alarm is triggered by any one of t following conditions: PRT temperature 125°F ↑; PRT pressure 10 PSIG ↑; PRT level 83% ↑; PI level 68% ↓. PRT instrumentation, indication, and alarms are tabulated under the section "Indicatioı Controls and Alarms."

Sources of Discharge to PRT

[8]Aside from the safety valves and PORVs, the following potential sources of discharge are piped to t PRT:

1. Safety injection test line relief valve, RV-859
2. RCP seal water return line relief valve, RV-382
3. Letdown line relief valve, RV-203
4. RHRS return line (to RCS) relief valve, RV-706
5. Reactor vessel head vent system
6. [9]Valve packing leakoff collection for the following valves:
    1) 885, RHR return tie between A and B loops
    2) MOV-535, PORV block valve
    3) MOV-536, PORV block valve

INSTRUMENTATION AND CONTROL

[8] 5610-T-E-4501, Sheet 1; Reactor Coolant System
[9] PC/M 89-373, PC/M 89-375

Facility:	North Anna
Vendor:	WEC
Exam Date:	2008
Exam Type:	R

Original Item for Question 32

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	2
Group #	1	1
K/A #	007	K3.01
Importance Rating	3.3	3.6

Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment

Proposed Question:

Given the following:

- Unit 1 was initially at 100% power at the end of core life.
- Fuel failures have resulted in increased RCS activity (still within Tech Spec limits).
- A spurious Safety Injection (SI) occurred.
- One train of SI failed to reset from the control room.
- Multiple PRZR PORV lifts occurred during the recovery.
- PRT pressure is 115 psig.

Which ONE of the following identifies the current conditions in the Containment including the raditation monitor indications in the Main Control Room?

A: Containment Sump level constant;
Containment Particulate and Gaseous Radiation Monitors constant.

B: Containment Sump level increasing;
Containment Particulate and Gaseous Radiation Monitors increasing.

C: Containment Sump level increasing;
Containment High Range radiation Monitors constant.

D: Containment Sump level increasing;
Containment High Range Radiation Monitors increasing.

Proposed Answer: A

Explanation (Optional):

A: Correct. PRT rupture discs have not actuated at 100 psid as designed, thus energy and radioactive liquid released through the PORVs is still contained within the PRT. Therefore sump level and particulate and gaseous monitors will be constant for the given conditions. Further, the sample flow to the particulate and gaseous monitors is isolated automatically by Containment Phase A Isolation that occurs as a result of the SI.

B: Incorrect. First part is incorrect as noted above but plausible if applicant misunderstands that once ruptured, PRT pressure will drop to approximately Containment pressure based on the large size of the disc compared to the in-flow from periodic PORV cycling. Nominally the rupture disc should blow at 100 psi above typical Containment of 11 psia, but for the conditions given the rupture discs have not yet blown. Second part is incorrect but plausible since if the assumption is the rupture discs have actuated at the nominal pressure, airborne activity would increase in Containment. However the question clearly asks for control room indications, which for reasons noted above will not be increasing

C: Incorrect; First part is incorrect but plausible as discussed in Distractor B. Second part is incorrect but plausible since by virtue of the function (accident monitoring instrumentation) the candidate may assume that while sump level would be increasing, because of the scaling of the CHRRMs there would be no discernible change in high range R/M indication.

D: Incorrect. First part is incorrect but plausible as discussed in Distractor B. Second part is incorrect but plausible since the candidate may assume that sump level would be increasing, and that airborne radioactivity levels would also be increasing and would be detected on the high range R/Ms. The candidate may discount design limitations such as accuracy, sensitivity, statistical variance and the like, and conclude that since it is there, it must be capable of being "seen" on indications.

Technical Reference(s): RCS LP, AR, GFES (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41
	55.43

Comments:

QUESTION 33

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008	K4.02
	Importance Rating	2.9	

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Operation of the surge tank, including the associated valves and controls

Proposed Question: RO Question # 33

Which ONE of the following correctly completes the following statements?

With a small Component Cooling Water leak, ____(1)____ INITIALLY compensates for system leakage prior to operator action. During CCW operation, if R-3-17A or B, Component Cooling Water Monitors, reaches a high alarm setpoint, then RCV-3-609, CCW Head Tank Vent, is isolated ____(2)____.

A. (1) the makeup water valve
(2) manually

B. (1) the CCW Head Tank
(2) manually

C. (1) the CCW Head Tank
(2) automatically

D. (1) the makeup water valve
(2) automatically

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible since demineralized water is supplied through a manual valve, but only after the head tank is no longer maintaining CCW Surge Tank level. Vent valve automatically isolates on high radiation

B. Incorrect. First part is correct but Vent valve automatically isolates on high radiation

C. Correct.

D. Incorrect but plausible since demineralized water is supplied through a manual valve, but only after the head tank is no longer maintaining CC Surge Tank level. Second part is correct, CCW Surge Tank vent valve automatically closes on high radiation signal

3-ARP-097.CR.H

Technical Reference(s): LP 6902140, *Component Cooling Water System* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902140, Obj. 5 (As available)

Question Source: Bank #

Modified Bank # 70107 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Modified from VCS 2009

CAUSES:
1. High radiation in one of systems monitored by PRMS
2. PRMS system component failure

H1/4

PRMS HI RADIATION

DEVICE:	SETPOINT:	LOCATION:
• R-11 • R-12 • R-14 • R-15 • R-17A • R-17B • R-18 • R-19 • R-20	Variable with each PRMS channel	N/A

ALARM CONFIRMATION

1. **CHECK** the following:
   - Countrate meter on each PRMS drawer in Rack QR-66
   - Alarm indicators on each drawer in Rack QR-66

OPERATOR ACTIONS

1. IF 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 for expected automatic actions.
2. IF alarm is on R-15 or R-19, THEN **REFER TO** 3-ONOP-071.2, Steam Generator Tube Leakage for expected automatic actions.
3. IF alarm is on R-14, R-17A, R-17B, R-18, or R-19, THEN **CHECK** alarm valid as follows:
   A. **CHECK** FAIL/TEST light **NOT** LIT.
   B. **PUSH** FAIL/TEST light (meter reading of 288 or 289K)
   C. **PUSH** SOURCE CHECK light (should get meter increase).
   D. **PUSH** HIGH ALARM light to determine if meter level is above high alarm setpoint.
4. **ENSURE** required automatic actions.
5. IF 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.
6. IF alarm is on R-15 OR R-19, THEN **REFER TO** 3-ONOP-071.2, Steam Generator Tube Leakage.
7. **REFER TO** TS 3.3.3, 3.4.6, and 3.9.13 for additional required actions.

REFERENCES: Tech Spec Sections 3.3.3, 3.4.6, and 3.9.13
PC/M 07-055, R-15 Steam Jet Air Ejector Monitor Replacement

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

MOV-3-626 will close if return flow from RCP thermal barriers reaches 130 GPM due to RCS leak to CCW System at a thermal barrier.

29 **Check CCW System For High Activity**

a. Announce the high radiation alarm on page system and warn personnel to remain clear of all CCW piping

b. Verify RCV-3-609, CCW Head Tank vent Valve - CLOSED

c. Direct Chemistry Department to sample CCW System to determine its activity level

d. Route any known CCW system leakage to the WHUT floor drain.

CAUSES:
1. Leak out/in CCW System
2. Seal Water Heat Exchanger tube leak

H8/6

CCW HEAD TANK HI/LO LEVEL

DEVICE:	**SETPOINT:**	**LOCATION:**
LT-3-614	High - 85% Low - 10 %	CCW Head Tank on exterior containment wall

ALARM CONFIRMATION

CHECK CCW Head Tank Level Indication, LI-3-614A on VPB.

OPERATOR ACTIONS

1. IF any CCW pump is cavitating, THEN **REFER TO** the Foldout Page of 3-ONOP-030, Component Cooling Water Malfunction AND **STOP** the affected pump.
2. IF high level, THEN perform the following:
   A. **ENSURE** RCV-3-609, Head Tank Vent Valve, closes at 85% level.
   B. IF CCW Head Tank level greater than or equal to 100% on LI-3-614A, THEN **MAINTAIN** RCV-3-609, Head Tank Vent Valve, in CLOSED position until overflow piping has been drained.
   C. IF voiding is suspected, THEN **REFER TO** 3-NOP-030, Component Cooling Water, Section 5.22.
   D. IF there are **NO** abnormal radiation levels in CCW System, THEN **REFER TO** 3-NOP-030, Component Cooling Water.
3. IF low level, THEN **OPEN** MOV-3-832, Fill Valve AND **RESTORE** level to normal (10% to 85% in CCW Head Tank for normal CCW System requirements.)

NOTE

A Seal Water Heat Exchanger tube leak may be indicated by a low level concurrent with:
- Annunciator A-4/6 (VCT HI/LO LEVEL)
- Increased VCT level
- Unexplained slight power\Tave increase
- Increased sodium levels in the RCS

4. **CHECK** CCW System for possible leaks AND **ISOLATE** source of leakage.
5. **REFER TO** Tech Spec. 3.7.2 for any additional required actions.
6. IF CCW Head Tank level cannot be maintained, THEN **REFER TO** 3-ONOP-030, Component Cooling Water Malfunction.

REFERENCES:
1. FPL DWG 5613-M-3030
2. Tech. Spec. Section 3.7.2
3. PC/M 96-092, Addition of U-3 CCW Head Tank

CAUSES:
1. Leak out of CCW System
2. Seal Water Heat Exchanger tube leak

H7/6

SURGE TANK LO LEVEL

DEVICE:	SETPOINT:	LOCATION:
LT-3-613	Low - 45%	CCW Surge Tank in SFP Room

ALARM CONFIRMATION

CHECK CCW Surge Tank level indication, LI-3-613A on VPB.

OPERATOR ACTIONS

1. IF low level, THEN **OPEN** fill valve, MOV-3-832, and **RESTORE** level to normal (greater than 45% in CCW Surge Tank when CCW System is drained for maintenance).
2. **CHECK** CCW System for possible leaks and **ISOLATE** source of leakage.
3. IF CCW Surge Tank level (CCW System drained for maintenance) can **NOT** be maintained, THEN **REFER TO** 3-ONOP-030, Component Cooling Water

NOTE

A Seal Water Heat Exchanger tube leak may be indicated by a low level concurrent with:
- Annunciator A-4/6 (VCT HI/LO LEVEL)
- Increased VCT level
- Unexplained slight power\Tave increase
- Increased sodium levels in the RCS

4. **REFER TO** TS 3.7.2 for any additional required actions.

REFERENCES:
1. FPL DWG 5613-M-3030
2. PC/M 96-092, Addition of U-3 CCW Head Tank

OBJECTIVES 5f, 7g

The CCW head tank (see Figure 1A) is connected to the top of the CCW surge tank, which means that the surge tank is completely full of CCW and that the head tank is used for level indication and surge volume. A high level alarm is provided on the head tank at 85%. A low level alarm is also provided on the head tank at 10%. LT-614 was moved to the head tank and LT-613 was installed on the surge tank. New vacuum breakers were installed at the head tank location and the old vacuum breakers at the surge tank were removed. The Head Tank Vacuum Breakers open at 1.5 psig vacuum in the tank. Overflow from the head tank passes through RCV-609 to the waste holdup tank. RCV-609 auto closes on a high level of (85%) in the head tank as well as an R-17A & B alarm.

The head tank provides the following functions:

1. Provides the net positive suction head for the CCW pumps and an immediate source of makeup water to the system if out leakage develops.

2. A means for damping transient pressures developed in the system due to load changes and pump startup and shutdown.

3. A means of monitoring the volume of fluid in the system.

4. A means of providing for expansion and contraction of the fluid in the system due to temperature changes.

5. A source of pressure relief of the system through the relief valve to the waste holdup tank (WHT).

6. Expansion volume should reactor coolant in-leakage develop.

7. A method to prevent steam void formation in the emergency containment coolers following a LOCA with Loss Of Offsite Power.

OBJECTIVE 8d, 11f

The CCW head tank is equipped with a normally open, solenoid - operated, vent to the WHT, RCV-*-609. If Hi radiation is detected in the CCW return, the valve will automatically close. (Refer to 3/4-ONOP-067.)

QUESTION #33

Facility: Turkey Point

Vendor: WEC

Exam Date: 2009

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	008	K4.02
Importance Rating	2.9	

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following:
Operation of the surge tank, including the associated valves and controls

Proposed Question: RO Question # 33

Given the following plant conditions:
- 100% power
- The Demineralized Water System is isolated due to repeated filter clogging problems.
- B1 work week is in progress.
- The CCW Surge Tank level is normal.
- A leak has developed in the CCW Non-Essential Header.

Which ONE (1) of the choices below completes the following statement?
CCW Surge Tank:

A. 'A' side level will lower and the 'B' side will be stable

B. 'B' side level will lower and the 'A' side will be stable

C. level lowers on both sides and then only the 'A' side will lower

D. level lowers on both sides and then only the 'B' side will lower

Proposed Answer: C

Explanation (Optional):

A. Plausible if applicant does not understand CCW Surge Tank construction.

B. Plausible if applicant does not understand CCW Surge Tank construction and/or system

alignment.

C. CORRECT. Level lowers on bothe sides until the top of the divider plate, the only
lowers on the side with the leak (the loop supplying non-essentials).

D. Plausible if applicant believes the standby pump starts on the inactive loop.

Technical Reference(s): SOP-118, Page II.4 (Page 1), LP IB-2, Figure 2.1 (Page 79) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: (As available)

Question Source: Bank # WTSI 70107

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 VC Summer

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

QUESTION 34

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	010	K6.01
Importance Rating	2.7	

Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems

Proposed Question: RO Question # 34

Given the following:

- Unit 3 is operating at 50% power.
- ALL Pressurizer pressure controls are in AUTO.
- PT-3-444, Pressurizer Pressure Transmitter, fails LOW.

Which ONE of the choices below correctly completes the following sentence?

With no operator action over the next half hour, Unit 3 will ___(1)___ and RCS pressure will cycle around ___(2)___.

A. (1) trip;
(2) PORV PCV-3-456 Setpoint

B. (1) trip;
(2) PORV PCV-3-455C Setpoint

C. (1) remain at power;
(2) PORV PCV-3-456 Setpoint

D. (1) remain at power;
(2) PORV PCV-3-455C Setpoint

Proposed Answer: C

Explanation (Optional):

A. Incorrect since the plant will NOT trip and pressure will stabilize at 2335. Plausible because plant will trip when PT-444 fails HIGH. Also, the plant will cycle around the PORV PCV-3-456 Setpoint.

B. Incorrect since the plant will NOT trip and pressure will stabilize at 2335. PCV-3-455C signal is low due to PT-3-444, Pressurizer Pressure Transmitter, failing LOW. Plausible because plant will trip when PT-444 fails HIGH. Also, the plant will cycle around the PORV PCV-3-456 Setpoint, not PORV PCV-3-455C Setpoint.
.

C. CORRECT since pressure will stabilize at 2335 psig at PORV PCV-3-456 Setpoint.

D. Incorrect. PT-444 is failed LOW. This signal inputs as a low pressure to PCV-3-455C which will not open. Plausible since RCS pressure cycles at the PORV setpoint on PORV PCV-3-456.

Technical Reference(s): LP 6902109A, *Pressurizer Pressure Control* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902109A, Obj.8 (As available)

Question Source: Bank # 61060

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2006 Surrey

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3PEO)

Question Difficulty Level: B

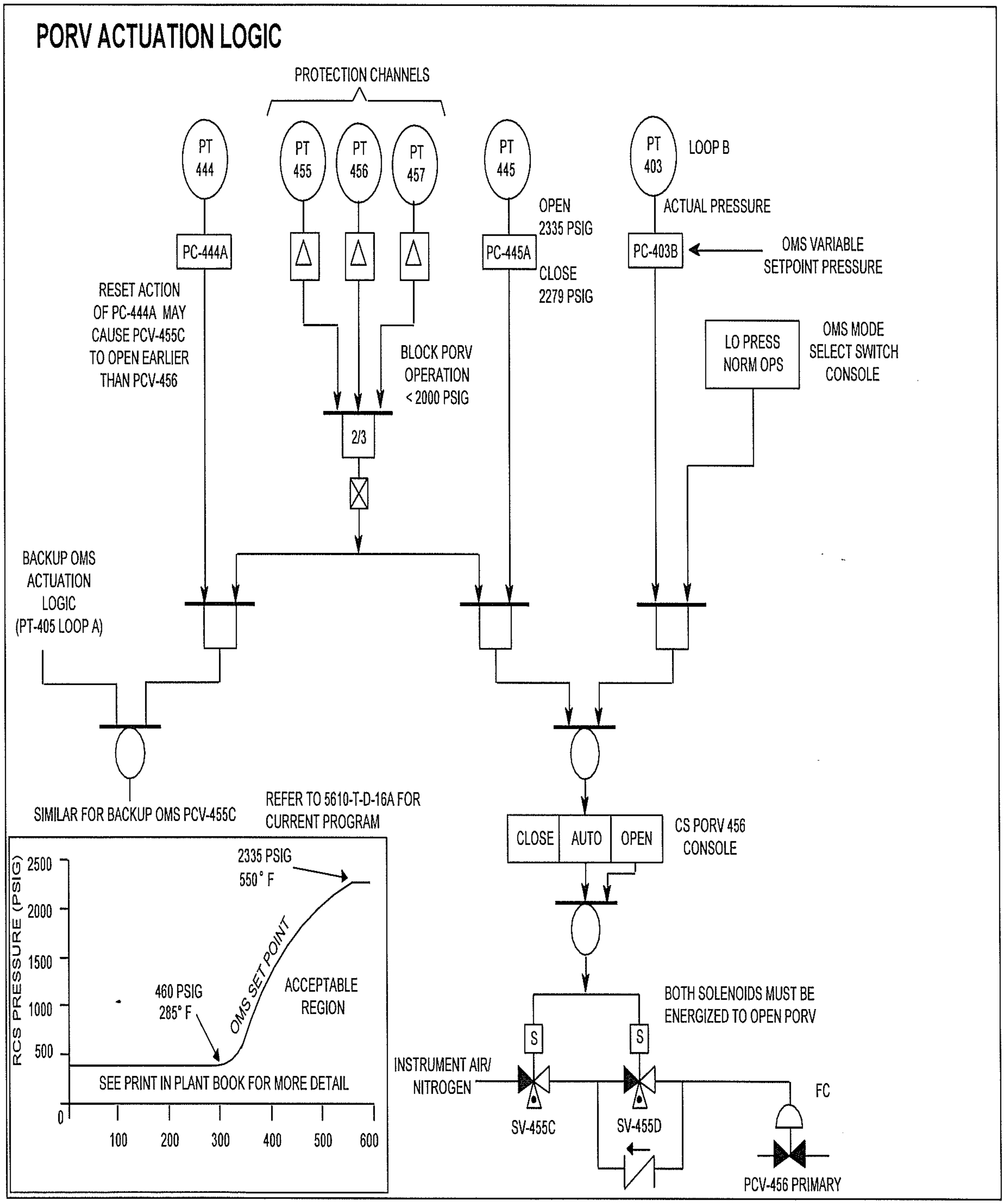
10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

INSTRUCTOR ACTIVITY



SD 009

FIGURE 21
01/29/02

PRESSURIZER PRESSURE CONTROL SETPOINTS

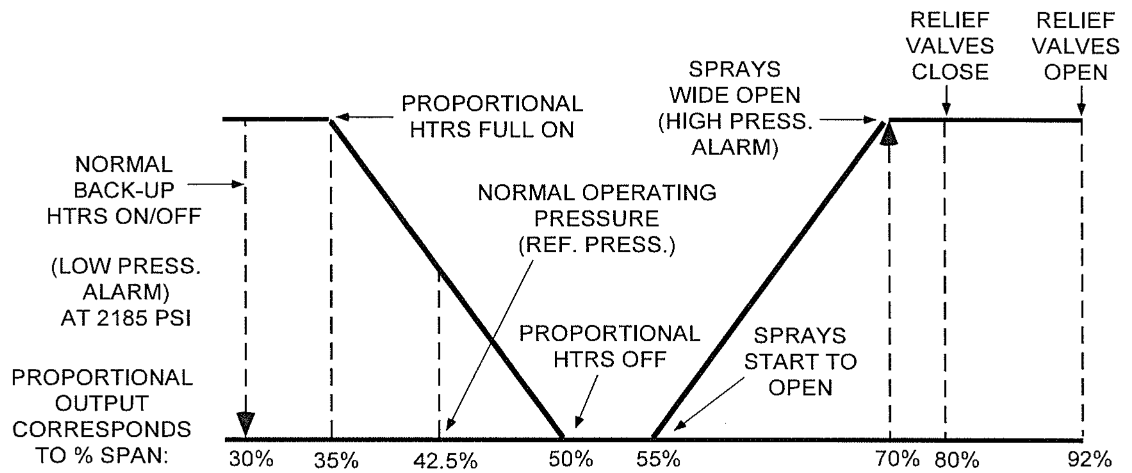
PRESSURE		ACTION
2485 psig		SAFETY VALVES OPEN
2385 psig		REACTOR TRIP
2335 psig	(92% OF SPAN)	PORV'S OPEN
2310 psig	(80% OF SPAN FOR PCV-455C)	SPRAY VALVES WIDE OPEN AND HIGH PRESS. ALARM
2279 psig	(70% OF SPAN FOR PCV-455C)	PORV'S CLOSED
2260 psig	(55% OF SPAN FOR PCV-455C)	SPRAY VALVES START OPEN
2250 psig	(50% OF SPAN)	CONTROL HEATERS FULL OFF
2235 psig	(42.5% OF SPAN)	NORMAL OP PRESSURE
2220 psig	(35% OF SPAN)	CONTROL HEATERS FULL ON
>2210 psig	(>30% OF SPAN)	BACKUP HEATERS OFF
≤2210 psig	(≤30% OF SPAN)	BACKUP HEATERS ON
2185 psig		LOW PRESSURE ALARM
2000 psig		PORV OPEN PERMISSIVE INTLK MANUAL SI BLOCK PERMISSIVE INTLK
1835 psig		REACTOR TRIP
1730 psig		SAFETY INJECTION

SD 009

FIGURE 22
10/31/05

INSTRUCTOR ACTIVITY

PRESSURIZER PRESSURE CONTROL SCHEME



THESE ARE THE SETPOINTS FOR NORMAL AT PRESSURE OPERATION (NOTE 2)

PRESSURIZER PRESSURE CONTROLLER PC-444A OPERATING DATA

PRESSURE ERROR	-85	-25	-15	0	+15	+25	+75	+100	+115
PERCENT CONTROLLER OUTPUT (Proportional Signal Only)	0	30	35	42.5	50	55	80	92	100

NOTE 1: THE CONTROLLER OUTPUT IS NORMALLY 42.5% WHEN PRESSURE ERROR SIGNAL (PRESSURIZER PRESSURE REF. PRESSURE) EQUALS ZERO.

NOTE 2: EXCEPT FOR PCV-456, CONTROL DEVICES OPERATE ON CONTROLLER OUTPUT. CONTROLLER OUTPUT IS A FUNCTION OF PRESSURE ERROR AND LENGTH OF TIME THE ERROR HAS EXISTED. THIS CAN CAUSE DEVICE TO FUNCTION AT A HIGHER OR LOWER PRESSURE THAN PROPORTIONAL ONLY.

444J TEN TURN LINEAR POTENTIONMETER CONTROLS PRESSURE IN THE RANGE OF 1500 PSI TO 2500 PSI AT A CHANGE OF 100 PSI PER TURN OF POTENTIONMETER. FOR EXAMPLE:

TO SET REFERENCE CONTROL POINT AT 2235 PSI:

$$\frac{(2235 \text{ PSI} - 1500 \text{ PSI})}{(100 \text{ PSI/TURN})} = 7.35 \text{ TURNS}$$

SD 009

FIGURE 26
10/31/05

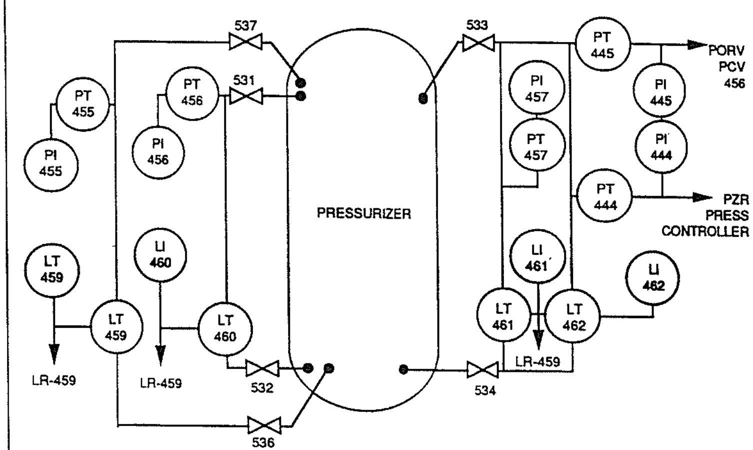
COMPONENT DESCRIPTION

OBJECTIVE 2

2.1 Pressurizer Pressure Transmitters

The (five) pressurizer pressure transmitters use the three steam side level taps to sense pressurizer pressure. See Figure 24.

PRESSURIZER LEVEL AND PRESSURE DETECTORS



SD 009

FIGURE 24
Rev.0:7/19/94

Three of the five transmitters supply pressure signals to the three protection channels. PT-455, PT-456, and PT-457 supply channels I, II, and III, respectively and are each on separate instrument columns. Signals generated by these three channels are used for reactor protection, safety injection, anticipatory alarms and indication on VPA. (See DWG 5610-T-D-16A sheet 1 for details.)

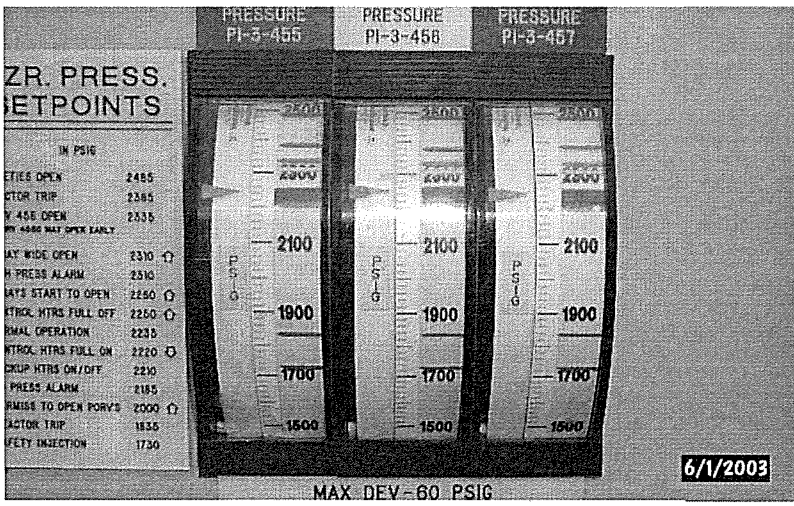
Two transmitters, PT-444 and PT-445, both on the same instrument column, are used for the pressurizer pressure control functions. PT-444 provides input to the heaters and sprays and PT-445 provides input to PORV-456. See DWG 5610-T-D-16B, sheet 1. They provide indication, PI-444 and PI-445, on VPA, and PT-444 also provides remote indication, PI-444A, at the auxiliary feedwater pump station. A selector switch on the console allows selection of either PT-445 or the output of pressurizer pressure controller PC-444A for recording on one-pen recorder, PR-444.

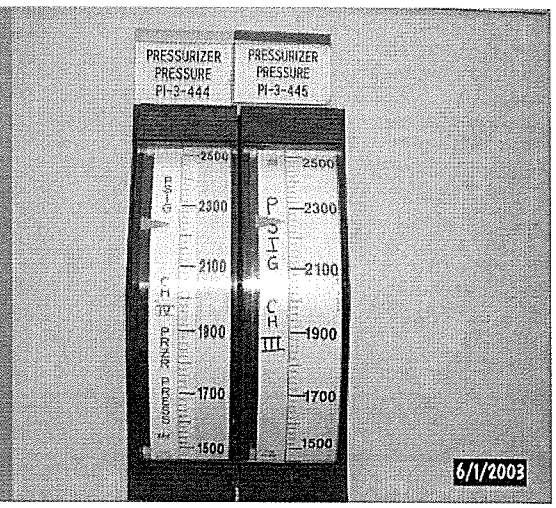
2.2 Pressurizer Pressure Instruments

OBJECTIVE 2, 7

Indication in the Control Room consists of five1500 – 2500 psig instruments, three for protection and two for control.

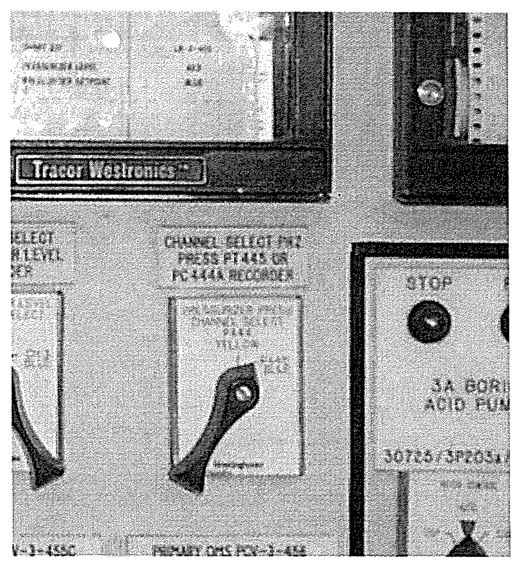
PROTECTION **CONTROL**





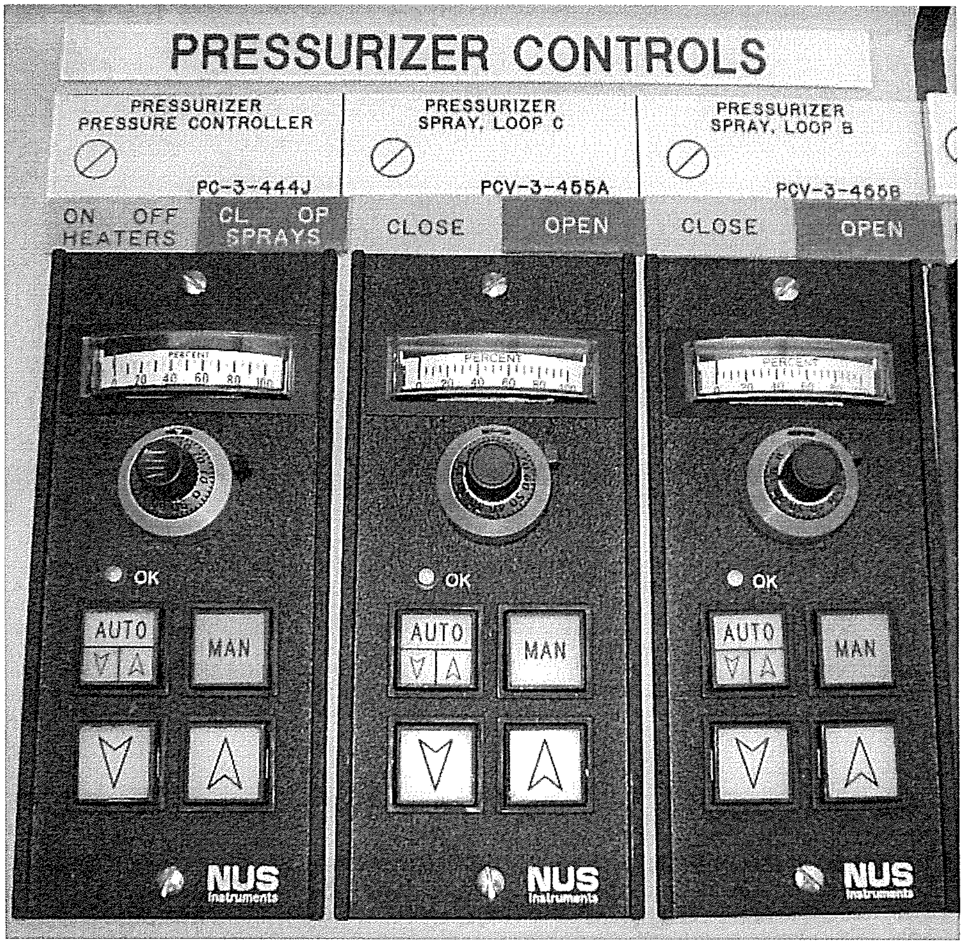
In addition to indication, the three protection channels provide for the following trips and interlocks: Pressurizer Pressure High Reactor Trip, Pressurizer Pressure Low Reactor Trip, Low Pressurizer Pressure Safety Injection, Safety Injection Block due to High Steam Line ΔP, and Block of Auto PORV operation.

The two control channels have separate functions. PT-444 provides for operation of the spray valves, heaters, and PORV -455C and also indication at the AFW station and VPA. PT-445 provides input for PORV-456 and indication on VPA.
A selector switch on the console allows selection of either PT-445 or the output-of pressurizer pressure controller PC-444A for recording on one-pen recorder, PR-444.



Channel Selector for Pressure Recorder

Pressurizer Master and Spray Controllers



OBJECTIVE 2,3b

PT-444 and its associated control circuitry contains the main pressurizer pressure controller PC-444A, which provides a compensated pressure error signal for the basic control scheme of pressurizer pressure. See Figure 26.

PT-445 is used to initiate a fixed high pressure alarm at 2310 PSIG ↑ and a fixed low pressure alarm at 2185 PSIG.

The main pressurizer pressure controller PC-444A, compares actual pressurizer pressure (P) to a setpoint pressure (P_{REF}). Setpoint pressure is determined by the setting of a ten-turn linear potentiometer PC-444J on the Console. The range of the potentiometer is 1500 PSIG to 2500 PSIG. Each turn equals 100 PSIG. During normal plant operations P_{REF} is set at 2235 PSIG (7.35 turns).

OBJECTIVE 8

The requirement for such an open permissive interlock protects the RCS from a severe depressurization incident should either (or both) pressurizer pressure control channels (PT-445 and/or PT-444) fail high. If such a failure occurs with the PORV in AUTO, the PORV will open and reduce RCS pressure unnecessarily. However, the open permissive interlock will shut the PORV as soon as pressure falls to 2000 PSIG. The ultimate purpose of interlocking the PORV's with the pressurizer pressure protection channels (PT-455, PT-456,and PT-457) is to keep the RCS in a subcooled condition and thereby preclude voiding in the core.

2.11 Over-Pressure Mitigation System

OBJECTIVE 2, 3c

The PORV's are also required to limit primary system pressure during heatup and cooldown cycles, when RCS component metal temperatures are low. The OMS system accomplishes this function by limiting RCS pressure excursions which keep the reactor vessel stresses below NDTT and DTT limits to prevent brittle fracture. The normal setpoint for the PORV's (2335 PSIG/ 92% of span for PCV-455C) is much too high to afford protection during periods of low RCS temperature. The OMS, when in service, opens the PORV's at much lower pressures, according to a variable setpoint which is continuously calculated (automatically) as a function of RCS loop temperature. The OMS compares actual pressure to this setpoint and operates the PORV's accordingly.

The OMS is designed to protect the RCS from over-pressurization due to the start of an idle reactor coolant pump with the steam generator water temperature ≥50°F above RCS cold leg temperature, or the start of a single high head safety injection pump and its injection into a water solid reactor coolant system.

Instrument Failures

The protection channel pressure transmitters, PT-455, PT-456, and PT-457 supply two of three coincidence circuits. A failure of one of the protective instruments would, in itself, not affect the pressure control system, but would affect the two out of three safeguards logic. As with the failure of any instrument supplying a reactor protection channel, the associated bistables would have to be manually tripped.

Malfunctions involving pressurizer pressure control are addressed in 3/4-ONOP-041.5, "Pressurizer Pressure Control Malfunction."

Failure high of PT-444 would cause all heaters to turn off and the spray valves to come full open. PORV PC-455C would open and pressure would decrease rapidly to 2000 PSIG. PC-455C would be shut at 2000 PSIG by the protective channels supplying the "block auto open" signal. Sprays would continue to decrease pressure until the reactor tripped at 1835 PSIG. Safety injection would be initiated at 1730 PSIG. To prevent the adverse effects of this instrument failure on the plant, the operator would place pressurizer pressure control in "MANUAL and control pressure by adjusting output.

Failure low of PT-444 would turn on all pressurizer heaters. Pressure would increase to the setpoint of PORV PCV-456 and would cycle around 2335 PSIG. The operator would select "MANUAL" pressure control and return system pressure to normal.

Failure of PT-445 high would cause PORV PCV-456 to open. Pressure would drop to 2000 PSIG and cycle around that setpoint. The only affect on pressure control of a failure low would be to cause PORV PC-456 to be inoperable in automatic.

QUESTION 35

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	012	K4.02
Importance Rating	3.9	

Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: Automatic reactor trip when RPS setpoints are exceeded for each RPS function; basis for each

Proposed Question: RO Question # 35

In accordance with 0-ADM-536, Technical Specification Bases Control Program, which ONE of the following Reactor Trip Setpoints provides reactor core protection against Departure from Nucleate Boiling (DNB)?

A. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

B. Reactor Coolant Pump Breaker Position Trip

C. Pressurizer Water Level

D. Power Range Neutron Flux – Low Range

Proposed Answer: B

Explanation (Optional):

A. Incorrect. This trip protects the RCS pressure boundary. Plausible because as the heat sink is lost, the plant heats up. The margin to DNB is reduced.

B. Correct. RCS flow is reduced as RCPs are tripped, and if there was no reactor trip on RCS low flow (or RCP breaker position) DNBR would be reduced

C. Incorrect. PZR water level is a trip on high level, and serves as a backup to PZR high pressure, which is protection of the RCS pressure boundary. Plausible because as the PZR water level is high due to a plant heatup and insurge, The margin to DNB is reduced.

D. Incorrect. This trip is designed to protect the RCS core, particularly from a rod withdrawal accident at low power. The reactor core safety limit is protected by this trip. Plausible because as neutron flux is higher, the margin to DNB is reduced.

Technical Reference(s): LP 6902163, *RPS and ESFAS* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902163, Obj. 7 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Components, capacity, and functions of emergency systems.

Comments:

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

2.2.1 (Cont'd)

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power, the Reactor Trip from the Turbine trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage pressure at approximately 10% of full power equivalent) and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System. Above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine first stage pressure at approximately 10% of full power equivalent) an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened. Above P-8 (a power level of approximately 45% of RATED THERMAL POWER) an automatic reactor trip will occur if one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7, an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is automatically blocked.

Underfrequency sensors are also installed on the 4.16 kV busses to detect underfrequency and initiate breaker trip on underfrequency. The underfrequency trip setpoints preserve the coast down energy of the reactor coolant pumps, in case of a grid frequency decrease so DNB does not occur.

QUESTION 36

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013	A3.01
	Importance Rating	3.7	

Ability to monitor automatic operation of the ESFAS including: Input channels and logic

Proposed Question: RO Question # 36

Given the following:

- Unit 3 is at 100% power.
- Unit 3 Pressurizer Pressure is at 2235 psig.
- Pressurizer Pressure Protection Channel PT-3-455 failed and was removed from service.
- All bistables were tripped in accordance with 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.

Which ONE of the following identifies (1) the status of the "Block Low Tave S.I." and the "Block Low Prz. Press. S.I." lights and (2) the requirement of additional Engineered Safety Features Actuation System (ESFAS) Channel(s) to initiate a Safety Injection on Pressurizer Low Pressure?

	Block Lights	Channels
A.	On	One
B.	On	Two
C.	Off	One
D.	Off	Two

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Block lights are off since defeating one channel wouldn't affect the block status (2/3 coincidence). Second half is correct, as placing one bistable in trip leaves only one more channel to trip for an actuation

B. Incorrect. Block lights are off since defeating one channel wouldn't affect the block status (2/3 coincidence). Second half incorrect but plausible because if applicant believes that channel is placed in bypass instead of trip, this would be correct.

C. Correct. Block lights are off because only 1 channel would be tripped, and block lights would only be on if 2 channels were below the permissive. Since one channel is placed in trip, only 1 more channel is required to cause the actuation.

D. Incorrect. Plausible because the first part is correct and also because second half would be correct if an applicant believed a channel was bypassed rather than tripped when it was taken out of service

Technical Reference(s): LP 6902163, RPS and ESFAS (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902163, Obj. 11 (As available)

Question Source: Bank # 66862

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010 Diablo Canyon

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Changed significantly but left as bank item

C. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM

The engineered safeguards system, in conjunction with the RPS, is provided to sense impending accident situations and avoid exceeding safety limits during anticipated transients by activating appropriate engineered safeguards features (ESF) as necessary and causing a reactor trip. The ESFs are designed to mitigate core damage in the event of a postulated accident. ESFs include the Emergency Core Cooling System, Containment Ventilation and Heat Removal Systems, and Component Cooling Water System.

The types of major postulated accidents for which the engineered safeguards system provides protection are:

1. Loss of primary coolant accident,
2. Loss of feedwater accident,
3. Main steamline break accident,
4. Other accidents resulting in the actuation of safeguards logic.

In order to provide protection against postulated accidents, several signals are generated by the engineered safeguards actuation system. These signals are:

1. Safety Injection
2. Phase A Containment Isolation
3. Containment High and High-High Pressure
4. Phase B Containment Isolation
5. Feedwater Isolation
6. Main Steam Line Isolation
7. Containment Ventilation Isolation

The required conditions for generation of the above engineered safeguards signals, along with the resulting actions, are given as follows:

Safety Injection

The safety injection signal results from any of the following conditions (see also Table 2 for setpoints):

1. Low pressurizer pressure SI at 1730 psig and lowering. Actuation of any two of the three pressurizer pressure channels (PC-455E, 456D, or 457D) protection circuits will result in a safeguards actuation. This actuation signal may be blocked below approximately 2000 PSIG. (PZR LO PRESS SI, panel C, window 3/6.)

SI actuation and Indication	95-96

8. **Engineered Safety Feature Actuation System**

   A. General

   1) Provided to sense impending accident situations and initiate necessary ESFs and trip the reactor.
   2) Designed to prevent or mitigate damage to reactor core.
   3) Engineered safety features:
      a. Emergency Core Cooling System
      b. Containment Spray System
      c. Emergency Containment Coolers
   4) Major postulated accidents ESF is designed for:
      a. Loss Of Coolant Accident (LOCA)
      b. Loss Of Feedwater Accident (LOFW)
      c. Main Steamline Break Accident (MSLB)
      d. Other miscellaneous accidents
   5) Engineered safeguards system signals
      a. Safety injection signal ("S" signal)
      b. Phase A containment isolation signal
      c. Phase B containment isolation signal
      d. Containment ventilation isolation signal
      e. Containment high-high pressure signal
      f. Feedwater isolation signal
      g. Main steam line isolation signal

   B. **Safety Injection Signal**

   1) Initiation signals Fig 15-18

Lo Press SI	97

      a. **Low Pressurizer Pressure**
         i. Coincidence - 2/3
         ii. Setpoint - 1730 PSIG ↓
         iii. Manual block below 2000 PSIG
         iv. Automatic reset above 2000 PSIG
         v. PZR LO PRESS SI alarm on C-3/6

5.11 **IF** a containment pressure channel has failed, **THEN** place the failed channel in the tripped condition by performing the following:

5.11.1 Remove fuses for failed channel using Attachment 7.

5.11.2 Verify channel is in tripped condition by observing corresponding status light (VPB) lit.

5.12 **IF** any other channel has failed, **THEN** perform the following to trip bistables for the failed channel.

5.12.1 **IF** plant conditions are such that all required bistables associated with the failed channel may be tripped without an undesired RPS or ESF actuation, **THEN** perform the following:

1. Place all bistable switches for the affected loop in test position using Attachment 4.
2. Verify bistables tripped by observing corresponding status light (VPB) lit.

5.12.2 **IF** plant conditions are such that all bistables associated with the failed channel may **NOT** be tripped due to an undesired RPS or ESF actuation, **THEN** perform the following:

1. Place only the bistables which will **NOT** cause an RPS or ESF actuation in the test/tripped position using Attachment 4.
2. Verify bistables tripped by observing corresponding status light (VPB) lit.
3. Follow action of Tech. Spec. 3/4.3 and/or 3.0.3 for those bistables which were **NOT** placed in the tripped condition.

5.13 **IF** any of the following channels are failed, **THEN** place the Bypass Switch(es) for the failed channel to Bypass position at the AMSAC panel using Attachment 5:

5.13.1 Any Steam Generator Level Channel I (LI-3-474, LI-3-484, or LI-3-494)

OR

5.13.2 Any Steam Generator Level Channel II (LI-3-475, LI-3-485, or LI-3-495)

OR

5.13.3 PT-3-446

OR

5.13.4 PT-3-447

QUESTION 37

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	022	K1.01
Importance Rating	3.5	

Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system

Proposed Question: RO Question # 37

Given the following:

- Unit 4 is at 100% power.
- Normal Containment Coolers are in the normal, at-power alignment.
- An automatic Safety Injection occurs.

Which ONE of the following describes (1) the status of the CCW flow to the NCCs and (2) the number of running Normal Containment Coolers after a Safety Injection?

A. (1) Unisolated
(2) All

B. (1) Isolated
(2) None

C. (1) Unisolated
(2) None

D. (1) Isolated
(2) All

Proposed Answer: B

Explanation (Optional):

A. Incorrect because the CCW valves will close on a Phase A. Plausible because the applicant may confuse the question with emergency containment coolers. Also plausible because the CCW inlet & outlet valves to the Emergency Containment Coolers are open after an SI.

B. CORRECT. Under normal operation, 3 of the 4 cooling units are operating with the fourth unit in standby. Each discharge damper opens/closes automatically with fan start/stop signals." The Phase „A" Containment Isolation (T) Signal automatically performs the following: a. Isolates the normal containment coolers and CRDM coolers by closing inlet (MOV-*-1417) and outlet (MOV-*-1418) valves."

C. I Incorrect since CCW to NCCs is isolated on Phase A, Plausible because all NCC Cooler fans are tripped on a SI.

D. Incorrect since NCCs are not running or started on SI, Emergency CCs are. Plausible since CCW to the NCCs is isolated after a Phase A.

Technical Reference(s): LP 6902129, Containment Ventilation and Heat Removal Systems; 4-NOP-057, Containment Normal Ventilation and Cooling System (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902129, Obj. 9 (As available)

Question Source: Bank # 68599

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2008 Harris

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

1.0 PURPOSE

This procedure provides instructions for operation of the following:

- Control Rod Drive Mechanism Coolers
- Normal Containment Coolers
- Penetration Cooling Fans.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

1. The Normal Containment Cooler Fans will trip and the component cooling water combined inlet and outlet to the coolers will close if Containment Isolation Phase A is actuated.

2. The Normal Containment Cooler Fans will trip off on loss of off-site power.

2.2 Limitations

1. On restoration of off-site power, the Normal Containment Cooler Fans will remain off until reset and can be manually restarted by placing the control switch to OFF/RESET and then to ON.

2. On restoration of off-site power, the Normal Containment Cooler Fans in standby mode will remain off until the control switch is taken to OFF/RESET and then to STBY. After reset, these fans become available as standby.

3. Containment average air temperature shall **NOT** exceed 120°F during normal plant operating conditions. If temperatures exceed this limit, then refer to Technical Specification 3.6.1.5 for required actions.

4. If the Normal Containment Cooler Fans trip as a result of Containment Isolation Phase A, then the NCC RESET pushbutton must be pressed after Phase A is reset to permit a start of the Normal Containment Coolers.

5. At least one Normal Containment Cooler Fan is required to be running for PRMS R-4-11 and R-4-12 to be OPERABLE.

QUESTION 38

E

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026	A1.02
	Importance Rating	3.6	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment temperature

Proposed Question: RO Question # 38

Given the following:

- Unit 4 is operating at 100% power.
- Unit 4 has experienced a Loss of Coolant Accident (LOCA) with a Loss of Offsite Power (LOOP).
- The crew has entered to 4-EOP-E-1, Loss of Reactor or Secondary Coolant.

In accordance with 4-EOP-E-1, Loss of Reactor or Secondary Coolant, which ONE of the following sets of conditions for Containment determines (1) when a Containment Spray Pump can be secured and (2) when MUST a 2nd Emergency Containment Cooler be started?

	(1)	(2)
A.	180°F	12 hrs
B.	122°F	24 hrs
C.	180°F	24 hrs
D.	122°F	12 hrs

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because as 180°F is the adverse Containment value for Containment temperature. Also, 12 hours is the number which is used for hot leg recirc. The student may assume this point is the same for starting a second Containment Cooler.

B. Correct.

C. Incorrect. Plausible because as 180°F is the adverse Containment value for Containment temperature. 24 hours is correct.

D. Incorrect. 122°F is correct. Plausible because 12 hours is the number which is used for hot leg recirc. The student may assume this point is the same for starting a second Containment Cooler.

Technical Reference(s): 3-EOP-E-1, steps 12 and 33 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902129, Obj. 9 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Components, capacity, and functions of emergency systems.

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12	**Check If Containment Spray Should Be Stopped**	
	a. Containment spray pumps – ANY RUNNING	a. Observe CAUTION prior to Step 13 **AND** go to Step 13.

	b. Check the following
	• Emergency Containment Filter Spray Valves - CLOSED
	  • 4A ECF Spray SV-4-2905, 2906
	  • 4B ECF Spray SV-4-2907, 2908
	  • 4C ECF Spray SV-4-2909, 2910
	• Containment temperature - LESS THAN 122°F
	• Containment pressure - LESS THAN 14 PSIG
		b. **WHEN** containment pressure less than 14 psig, **AND** containment temperature less than 122°F, **THEN** do Steps 12c through 12e. Observe CAUTION prior to Step 13 **AND** continue with Step 13.

	c. Reset containment spray signal	
	d. Stop both containment spray pumps **AND** place in standby	
	e. Close Containment Spray Isolation valves • MOV-4-880A • MOV-4-880B	

CORRECT ANSWER

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
30	**Verify Power To All SI System MOVs - AVAILABLE** • MOV-4-843A • MOV-4-843B • MOV-4-872 • MOV-4-869 • MOV-4-866A • MOV-4-866B • MOV-4-861A • MOV-4-861B • MOV-4-860A • MOV-4-860B • MOV-4-864A • MOV-4-864B • MOV-4-880A • MOV-4-880B • MOV-4-862A • MOV-4-862B • MOV-4-863A • MOV-4-863B • MOV-4-744A • MOV-4-744B • MOV-4-750 • MOV-4-751	Consult with TSC staff to determine how to provide power to any deenergized valve(s).
31	**Check SI System Valves – IN POSITION REQUIRED BY ATTACHMENT 2**	Consult with TSC staff to determine proper sequence for placing valves in desired position. **WHEN** proper sequence has been determined, **THEN** place valves in desired position.
32	**At 12 Hours After Event Initiation, Go To 4-EOP-ES-1.4, TRANSFER TO HOT LEG RECIRCULATION, Step 1**	Distractors A + D

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
33	**At 24 Hours After Event Initiation, Verify Two Emergency Containment Coolers In Operation**	CORRECT ANSWER
	a. Start ECCs as necessary to maintain two in operation	
34	**Check If Containment Spray Should Be Stopped**	
	a. Containment spray pumps – ANY RUNNING	a. Go to Step 35.

	b. Check the following
	• Emergency Containment Filter Spray valves - CLOSED
	  • 4A ECF Spray SV-4-2905, 2906
	  • 4B ECF Spray SV-4-2907, 2908
	  • 4C ECF Spray SV-4-2909, 2910
	• Containment pressure – LESS THAN 14 PSIG
	• Containment temperature - LESS THAN 122°F
		b. **WHEN** containment pressure less than 14 psig **AND** containment temperature less than 122°F, **THEN** do Steps 34c, 34d, and 34e. Continue with Step 35.

	c. Reset containment spray signal	
	d. Stop both containment spray pumps **AND** place in standby	
	e. Close containment spray isolation valves • MOV-4-880A • MOV-4-880B	
35	**Evaluate Long-Term Plant Status**	
	a. Consult TSC staff	

END OF TEXT

QUESTION 39

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039	K4.05
	Importance Rating	3.7	

Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Automatic isolation of steam line

Proposed Question: RO Question # 39

Which ONE of the following describes the Steam Line Isolation setpoint associated with High Main Steam Line Flow with Low Tavg?

A. Starts with 40% steam flow at 0% power, and then increases linearly to about 120% steam flow at 100% power.

B. Starts with 20% steam flow at 0% power, and then increases linearly to about 120% steam flow at 100% power.

C. Constant 20% steam flow until 20% power, and then increases linearly to about 120% steam flow at 100% power.

40% → 118%

D. Constant 40% steam flow until 20% power, and then increases linearly to about 120% steam flow at 100% power.

20% 100%

Proposed Answer: D

Tavg 543°F

Explanation (Optional):

A. Incorrect since linear increase does not start until 20%. Plausible because the numbers are the same as the values for the correct setpoint, but the constant value of 40% steam flow limit from 0% - 20% is left out of this choice.

B. Incorrect since linear increase does not start until 20%. Plausible because the numbers are similar to the correct setpoint, but the constant value of 40% steam flow limit from 0% - 20% is left out of this choice.

C. Incorrect since constant setpoint is 40% not 20%. Plausible because this choice correctly states that there is a constant value of Steam Flow setpoint up to a certain power. The setpoint at 100% power is correct, as is the fact that the setpoint increases nearly linearly.

D. CORRECT. 5610-J-844, Sheet 6A

Technical Reference(s): LP 6902163. *RPS and ESFAS* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902163, Obj. 8 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010 Farley

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

CAUSES:
1. Instrument failure
2. Steamline break

C7
SG A STEAMLINE HI FLOW

DEVICE:
- FC-474
- FC-475

SETPOINT:
1.28×10^6 lbs/hr at 0 20% Load.
Linear with 1st Stage Pressure to 3.84×10^6 lbs/hr from 20-100% Load.

LOCATION:
N/A

ALARM CONFIRMATION

1. **CHECK** FI-3-474, A STM GEN STM FLOW vs. program flow based on PI-3-446, FIRST STAGE PRESSURE on VPA.
2. **CHECK** FI-3-475, A STM GEN STM FLOW vs. program flow based on PI-3-447, FIRST STAGE PRESSURE on VPA.
3. **CHECK** bistables FC474, FC475, LOOP A HI STM FLOW status lights LIT on VPB.
4. **CHECK** recorder FR-3-478, A STEAM GENERATOR on console.

OPERATOR ACTIONS

1. IF alarm is due to instrument failure, THEN **REFER TO** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
2. IF alarm is **NOT** due to instrument failure, THEN **INVESTIGATE** for possible steam line break.

REFERENCES:
1. FPL Control System Diagram 5610-T-D-18B
2. Tech Spec Sections 3/4.3.1, 3/4.3.2

Hi Stm Flow with Lo Tavg/SG SI	100

d. **High Steam Flow coincident with either low steam line pressure or low T_{AVG}**

i. Coincidence - 1/2 high steam line flow signals in 2/3 lines in coincidence with either low steam generator pressure (2/3 S/Gs), or low T_{AVG} (2/3)

ii. Setpoint

a) High Steam Flow

- High steam flow setpoint is the ΔP that corresponds to 40% of rated steam flow from 0-20% turbine load, or
- Setpoint increases linearly to the ΔP that corresponds to 114% of rated steam flow at 100% turbine load.
- Turbine load is determined by first stage pressure.
  - Protection signal is not compensated as the VPA indication of steam flow is.

b) Low T_{AVG} - 543°F

c) Low S/G press. - 614 PSIG

- Discuss reasons for two different signals.
  - Break at HS/B vs. 100% power

iii. Manual block below 543°F

iv. HI STM FLO W/LOW TAVG/LOW STM PRESS SI alarm on C-9/4

Manual SI PB	101-102

e. **Manual Actuation**

i. One of two pushbuttons

a) Can't be blocked

ii. MANUAL SI alarm on C-2/4

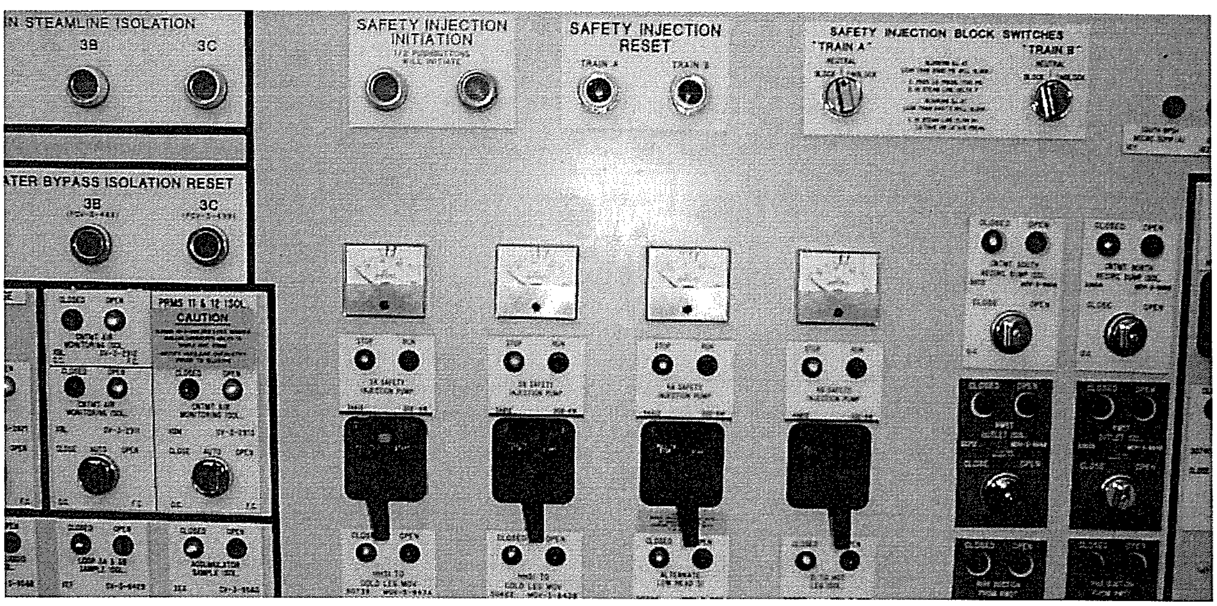
4. High steam flow coincident with either low steam generator pressure or low T_{AVG}. One out of two high steamline flow signals in two out of three lines combined with either low steam generator pressure in two out of three S/Gs, or low T_{AVG} in two out of three RCS loops will actuate this engineered safeguards function. (HI STM FLOW LOW T_{AVG} LO STM PRESS. SI, panel C, window 9/4.) This SI signal may be manually blocked when T_{AVG} is less than 543°F (2/3), and will be automatically unblocked when above this temperature. This SI provides protection against a steam break downstream of the MSIVs.

   The high steam flow setpoint is the ΔP which corresponds to 40% of Rated Steam Flow from 0-20% Turbine Load, then increases linearly to the ΔP which corresponds to 114% of rated steam flow at 100% turbine load. Turbine load is determined by First Stage Pressure (PT-446/447), (Refer to Figures 26 and 26A.)

5. Manual Actuation by pushing either Safety Injection pushbutton on VPB (1/2). (MANUAL SI, panel C, window 2/4.)

The logic for the safeguards actuation signals is given in Figures 15, 16, 17, and 18.

A complete list of actions which result from any of the above signals is given in **Table 5.**



MSLI	116-17

G. **Main Steam Line Isolation**

1) Initiation signals
   a. Manual
   b. High and Hi-Hi- containment pressure
   c. High steam line flow coincident with low steam generator pressure or low Tavg
      i. Cannot be blocked when below 543°F like safety injection can.
      ii. Steam flow signal to High Steam Flow comparator is not density compensated.
         a) As steam pressure drops during plant cooldown, the indicated steam flow at high steam flow actuation decreases.
         b) During cooldown following an accident, inadvertent main steam line isolation may occur, and increase the probability of release of radioactive substances and exposure of the general public.
2) Steamline Isolation alarm, C-8/2
3) Automatic closing of following valves
   a. Main steam stop valves (MSIV)
      i. S/G A - POV 2604, S/G B - POV 2605, S/G C - POV 2606
   b. Main steam stop bypass valve
      i. S/G A - MOV 1400, S/G B - MOV 1401, and S/G C - MOV 1402

Main Steam Line Isolation

Main steam line isolation results from any of the following signals:

1. Manual actuation from the control console or VPB. A control switch and pushbutton is provided for each stop valve.

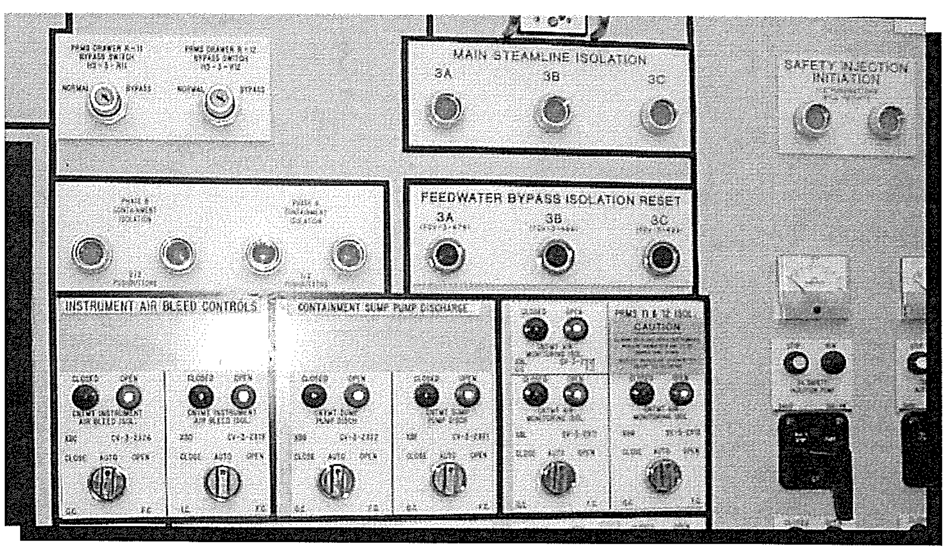
2. High and High-High containment pressure as previously described.

3. High steam line flow coincident with low steam generator pressure or low T_{AVG}. This signal is also used to actuate safety injection (previously discussed), but the main steam isolation signal cannot be manually blocked. The steam flow signal is not density compensated, and thus, the flow comparator sees higher than actual steam flow as T_{avg} and steam pressure decrease during plant cooldown. Since the control room steam flow is density compensated, main steam line isolation can result at indicated flow rates significantly less than the 0% power setpoint. (See Figures 27 and 27A)

The main steam line isolation circuitry is illustrated in Figure 15. The MAIN STEAMLINE ISOLATION alarm, C-8/2, is actuated by any of the signals listed above.

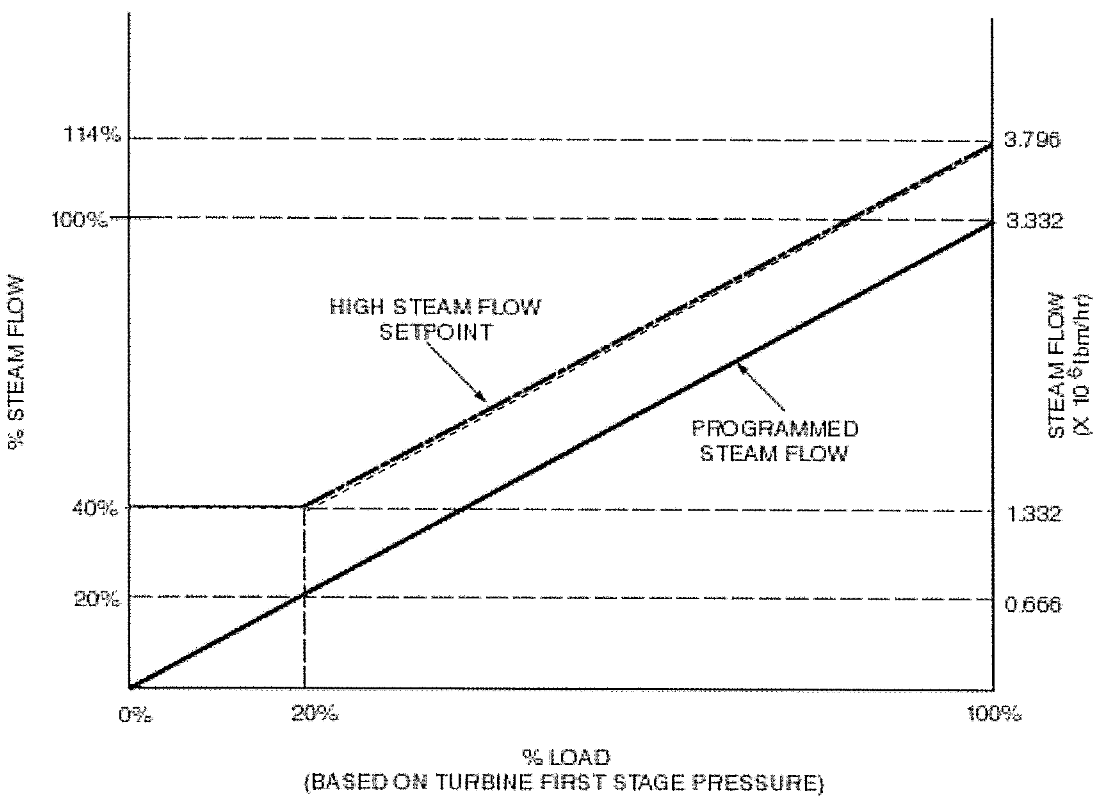
Valves which automatically close as a result of the main steam line isolation signals are listed below:

- POV-2604 main steam stop valve, S/G A
- POV-2605 main steam stop valve, S/G B
- POV-2606 main steam stop valve, S/G C
- MOV-1400 main steam stop bypass valve, S/G A
- MOV-1401 main steam stop bypass valve, S/G B
- MOV-1402 main steam stop bypass valve, S/G C



INSTRUCTOR ACTIVITIES

HIGH STEAM FLOW SETPOINT VS PROGRAMMED STEAM FLOW



SD-063

FIGURE 26
REV. 10/05/07

HIGH STEAM FLOW SETPOINT

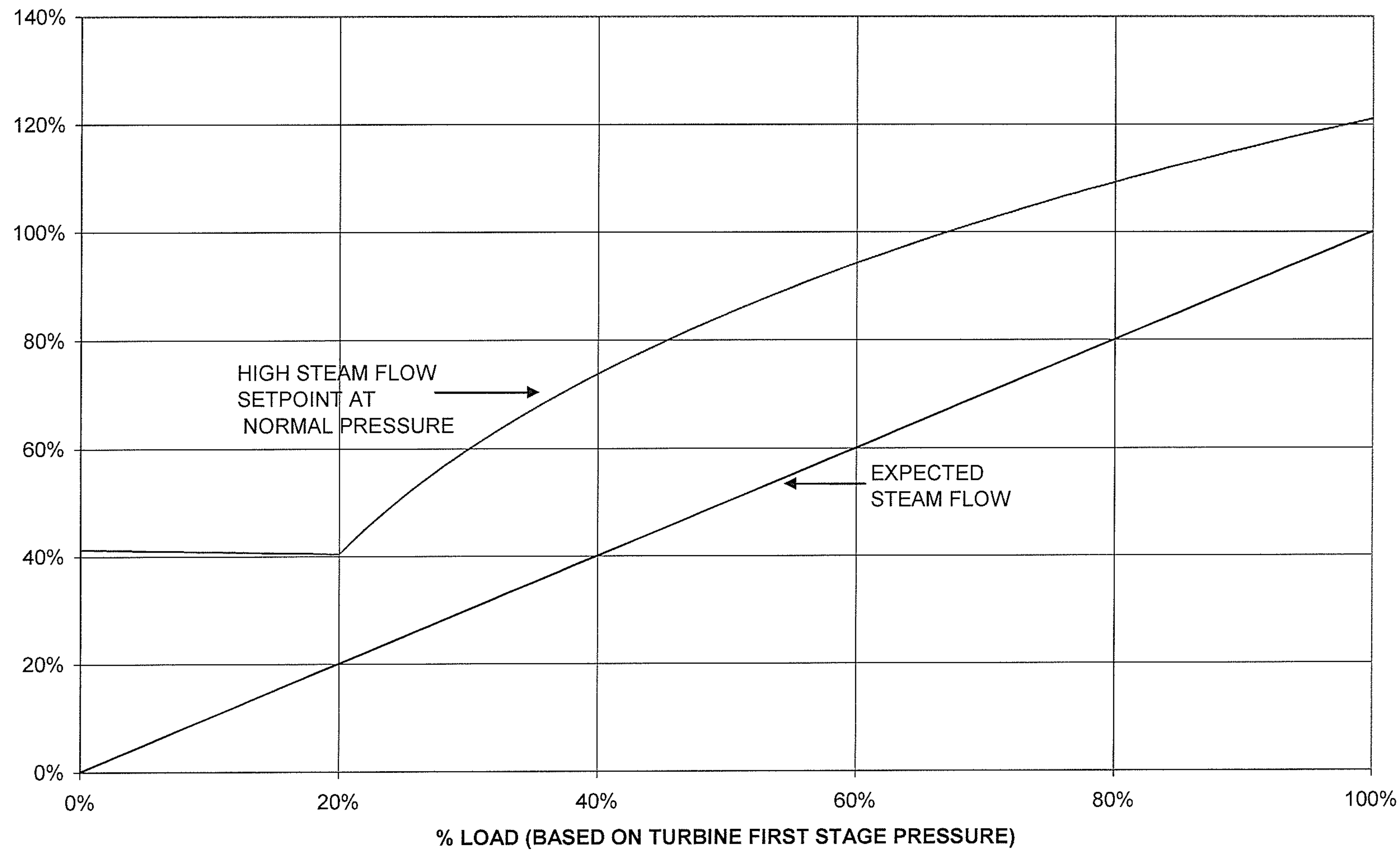


Figure 26A

QUESTION 40

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059	A4.11
	Importance Rating	3.1	

Ability to manually operate and monitor in the control room: Recovery from automatic feedwater isolation

Proposed Question: RO Question # 40

Given the following:

- Unit 4 experienced a Reactor Trip from 100% power due to a failed open Main Feedwater Regulating Valve.
- A Feedwater Isolation Signal was generated.

With repairs complete, Unit 4 is preparing for startup:

- S/G Narrow Range Levels are 45%, 55%, 68% and stable.
- Tave is 543°F and stable.
- Pressurizer Pressure is 2235 psig and stable.

Which ONE of the following describes the MINIMUM action to reset the Main Feedwater Regulating Valve's Slow Close Solenoid?

A. Restore S/G Narrow Range Levels to 50%

B. Close the Reactor Trip Breakers

C. Raise Tave to greater than 547°F

D. Push all Feedwater Bypass Isolation Reset pushbuttons

Proposed Answer: B

Explanation (Optional):

A. Incorrect since this will not reset the Main Feedwater Regulating Valve's Slow Close Solenoid. The applicant believes returning high S/G levels to program clears the initiating high S/G level signal. Plausible because this is part of the Fast Close circuitry associated with the Main Feedwater Regulating Valves.

B. CORRECT. Feedwater Isolation occurs when there is a Reactor Trip and Tavg is less than 554°F. The Main Feedwater Regulating Valves slowly close (20 seconds). Increasing Tavg does not reset this Feedwater Isolation condition. The Reactor Trip Breakers must be closed to remove the circuit seal-in and reset Feedwater Isolation.

C. Incorrect since restoration back to program temperature does not reset the Feedwater Isolation signal. Plausible because low Tave temperature is one of the part of the initiating slow closure signal for the Main Feedwater Regulating Valves.

D. Incorrect since the Feedwater Bypass Isolation Reset pushbuttons allow opening the Feedwater Bypass Valves not the Main Feedwater Regulating Valves. Plausible since the applicant uses these pushbuttons in restoration of Feedwater Flow. Also plausible since the novice applicant believes these buttons reset any Feedwater Isolation signal.

Technical Reference(s): 5610-T-L1 Sheet 14A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 37048 - Analyze automatic features and interlocks associated with the RPS (As available)

Question Source: Bank #

Modified Bank # 66866 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Modified from 2010 Diablo Canyon

3.4 Shutdown

OBJECTIVE 9d

As load on the turbine is reduced in preparation for unit shutdown, feedwater and condensate flow requirements will be reduced. At approximately 45 to 50% power on Feedwater Pump is secured. At 450 MWe one Heater Drain Pump is secured. At 300 MWe the remaining Heater Drain Pump is secured. At 275 MWe stop one Condensate pump.

As condensate flow decreases, recirculation valve CV-1400 should modulate open to circulate water back to the condenser.

As unit power decreases to 15%, feedwater flow control is transferred to the feedwater bypass valve. As the turbine generator is secured the main feedwater pump can be secured and feed to the steam generators is provided by the Standby Steam Generator Feed Pumps.

3.5 Safeguards Actuation

OBJECTIVE 9h

To prevent excessive RCS cooldown upon actuation of Safety Injection the feedwater system will respond by closing the Main and Bypass Feedwater Regulating Valves in 6.8 seconds (fast close) and trip the Main Feedwater Pumps which also closes the Main Feedwater Pump discharge valves.

In addition, actuation of Steam Generator Hi Hi level protection at 80% level will cause the Feedwater Main and Bypass Regulating Valves to fast close, the Feedwater Pumps to trip, and the Turbine Generator to trip.

The Feedwater Isolation signal for the Feedwater Regulating Bypass valves can be reset from pushbutton on VPB even if the SI or Hi Hi S/G level is in effect.

Feedwater Isolation also occurs when there is a Reactor Trip and Tavg is less than 554°F. This only affects the Main Feedwater Regulating Valves and causes them to close slowly (20 seconds). Tavg increasing above 554°F will not reset this Feedwater Isolation condition; closing the Reactor Trip Breakers will cause this to reset.

The motor operated isolation valve is controlled from the console by an OPEN/CLOSE switch.

OBJECTIVE 8e, 8h

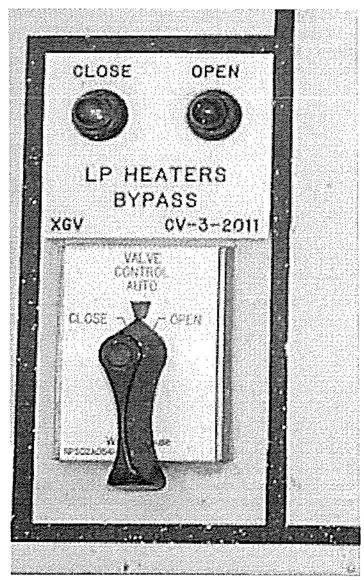
Feedwater Isolation from Safety Injection or High-High S/G level will close the main and bypass FCV's and trip the SGFP's valves. This isolation will fast close the Main Feedwater Regulating valves in 6.8 seconds. Resetting Feedwater Isolation will allow restarting the SGFP's and opening of the Bypass FCV's.

Reactor trip with Low Tavg of 554°F will also provide a Feedwater Isolation to slow close the main FCV's in 20 seconds.

Feedwater Heater Bypass

OBJECTIVE 8c

The Low Pressure Feedwater Heaters can be bypassed by flow through CV 2011 in the event of low suction pressure to the feedwater pumps. Refer to Figure 1. Control valve CV-2011 is actuated to open by pressure switch PS-2011 or PS-2014 when feedwater pump suction pressure drops to 220 psig. In the open direction, CV-2011 has a fast response time and goes to full open almost immediately. When the bypass valve is >1% open Annunciator D-7/4, LP Heaters Bypass Open, will alarm. The opening of CV-2011 allows bypassing the LP heaters and supplies condensate directly to the suction of the feedwater pumps. CV-2011 is also actuated by PS-1604 with a fast load reduction. Maintain CV-2011 open until the load reduction stops or until the feedwater pump suction pressure increases to above 226 psig at which time it is manually closed by means of the control switch. In the close direction, CV-2011 is adjusted to close slowly. It takes 60 seconds for the valve to go fully closed. The valve does not automatically close.

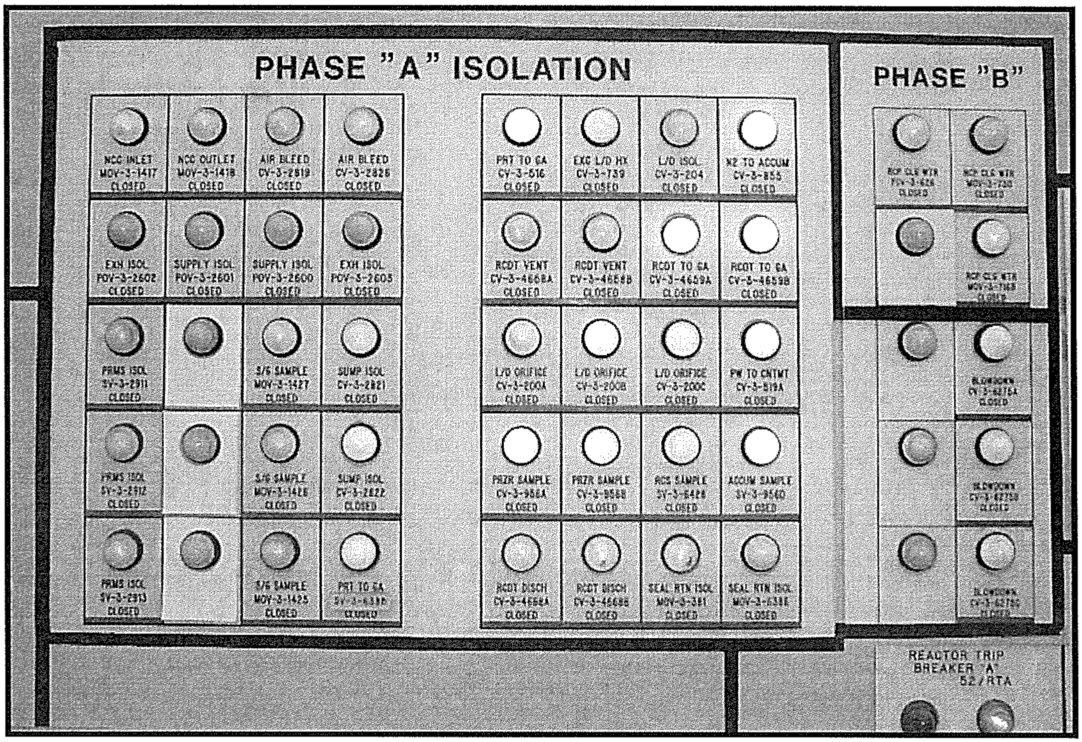


<u>**LP Heater Bypass**</u>

Exhaust Hood Sprays

OBJECTIVE 10f

Automatic sprays are provided to control high exhaust temperature. The regulators, one for each low pressure turbine, are sensitive to temperature and control the flow of water from the condensate header to spray nozzles in both exhaust ends of the low pressure turbines. The valves will start to open at 175°F.



Feedwater Isolation

The Main Feedwater Control Valves (FCV-478, 488, 498) are closed by any of the following signals:

1. The safety injection signal (fast close – 6.8 sec.)

2. The steam generator high-high level signal (≥ 80% level on 2/3 LTs on 1/3 S/G) (fast close – 6.8 sec.)

3. The reactor trip signal in coincidence with low T_{AVG} of 554°F↓ (slow close - 20 sec.)

The SI signal or the Hi-Hi SG level signal also result in a Turbine Trip, closure of the Feedwater Bypass Control Valves (FCV-479, 489, 499), and a trip of the Steam Generator Feedwater Pumps (SGFPs). These signals, 1 and 2 above, are also required to be operable by Technical Specifications 3.3.2, and sound annunciator E-2/6, HI-HI SG LVL TURB TRIP/FEEDWATER ISOLATION upon actuation.

The Feedwater isolation can be reset with the signal still present by depressing the individual "FEEDWATER BYPASS ISOLATION RESET" pushbuttons on VPB for each SG. The reset will allow the operation of the SGFPs, the FW Bypass Valves, and a relatch for the turbine trip.

INSTRUCTOR ACTIVITIES

d. Variable select switches (A or B)

   i. Six position switch that displays a four digit code

   ii. Requires manual to decipher

e. Toggle switches (A Train or B Train)

   i. Used to remove any of the instrumentation channel inputs and into the bypass position.

   ii. Bypass instructs the processor to ignore the signal

   iii. Bypassing both power signals or all three level signals to a processor will inhibit the firing of the processor

      a) Will cause a trouble lamp for the processor to light

f. System reset

   i. Resets AMSAC system and panel lights 3C391 (4C391).

   ii. Signal must clear before reset

Alarms	143

g. Alarms and Indications (AMSAC Panel)

   i. (A, B) tripped

      a) Comes on if 2/3 steam generator levels below the setpoint for 25 seconds or more.

      b) Comes on if actuating conditions are simulated when testing the logic.

      c) A/B Trouble, micro processor trouble

      d) A/B Bypass micro processor in bypass

   ii. Power Lamps (A or B) (turbine first stage pressure)

      a) Power below PT longer than 360 sec time delay

      b) Power channel in bypass

      c) Power channel fails in processor test

   iii. Test lamp, processor (A or B)

      a) Comes on during auto diagnostic or during manual test of processor

INSTRUCTOR ACTIVITIES

VPB indications	145-46

d. Signal Outputs (3C04) (3C04)

i. AMSAC Armed, single processor mode, green light

ii. AMSAC Armed, dual processor mode, green light

iii. AMSAC Trouble, Amber light and annunciator D-7/6, AMSAC Trouble/Actuated

a) Processor A or B trouble (can be caused if a steam generator level deviates more than 10% from the other steam generator levels)

b) AMSAC bypassed

c) AMSAC actuated

d) Loss of voltage to the safety circuit inputs or the vital/non-vital power supplies

iv. AMSAC actuated red light, D-7//6, AMSAC Trouble/Actuated

v. AMSAC reset pushbutton,

a) resets AMSAC selector in Control Room

b) does not reset AMSAC panel lights in Cable Spreading Room, must reset AMSAC panel locally

Alarm	147

e. Actuation Signal Outputs

i. Trips Control Rod MG Set Output Breakers

ii. Energize Aux FW Autostart Relays

a) Closes steam generator blowdown

b) Closes steam generator sample system

iii. Turbine Trip, energize Auto Stop Solenoids, 20AST and 20ASB

iv. "AMSAC Trouble/Activated" annunciator, D-7/6

f. DDPS plant computer

i. AMSAC actuated

ii. AMSAC processor A actuated

iii. AMSAC processor B actuated

g. Red Actuation Light on Control panel and AMSAC panel

INSTRUCTOR ACTIVITIES

e. Loss on Non-vital Power _P31

   i. Disables AMSAC

   ii. AMSAC Trouble Alarm D-7/6

      a) Local "Loss of Voltage" alarm at AMSAC cabinet

f. Reset pushbutton does not reset after actuation

   i. Failure of AMSAC to reset will not actuate AMSAC

   ii. Local control panel has a reset pushbutton that resets the local panel and indicating lights.

g. AMSAC normal/bypass switch, AMSAC does not go bypass

   i. Failure will not actuate AMSAC

   ii. Will give AMSAC trouble alarm on 3C04 (4C04) and local panel indication

h. AMSAC single/dual logic processor switch, does not go to single or dual processor mode

   i. Failure will not actuate AMSAC

   ii. Room and Local Panel of AMSAC armed in the dual or single processor mode.

i. Logic/Operation

   i. Review logic sheets with class

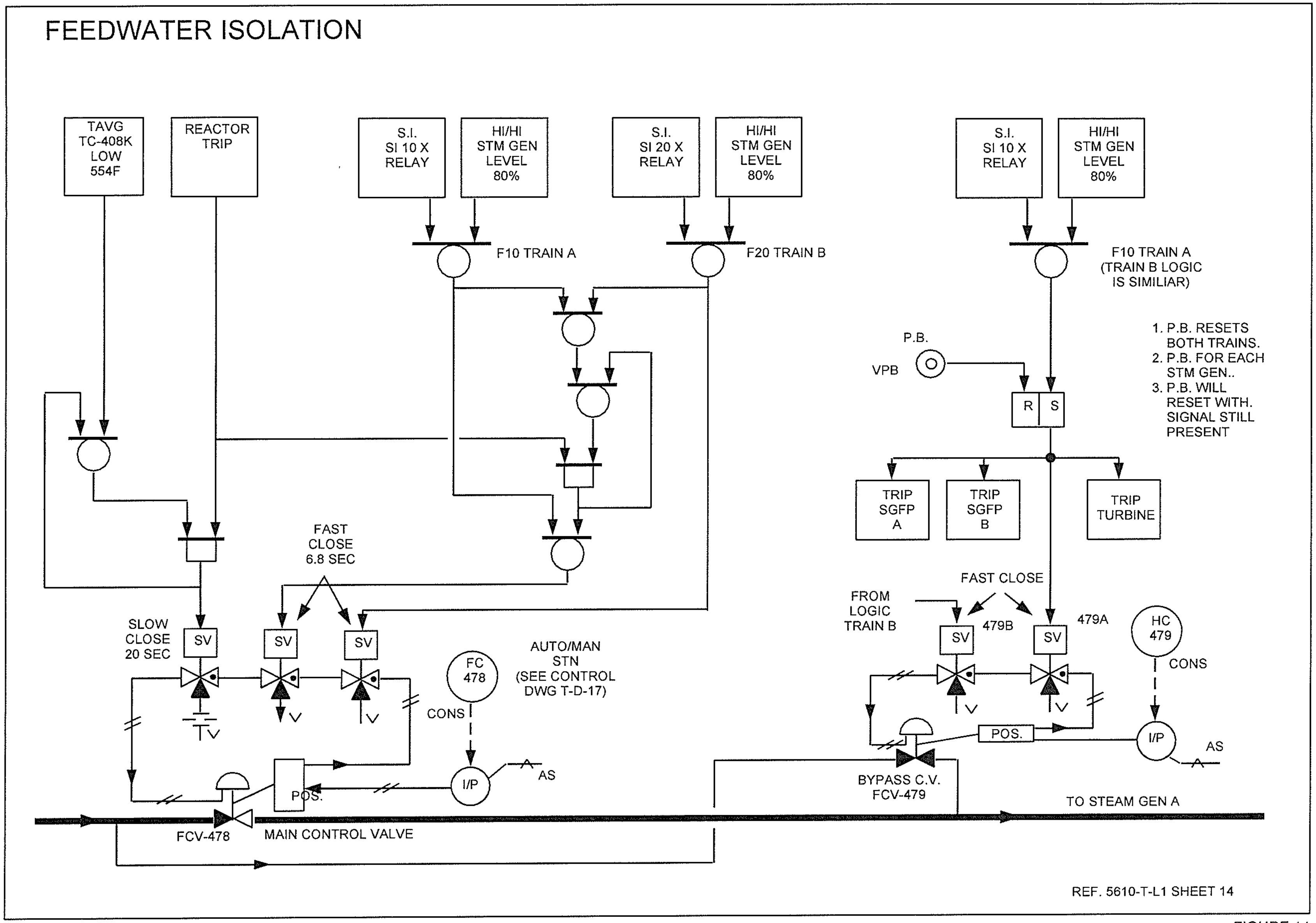
      a) Operation 5610-T-L1 Sh 33A

         - TP-56

      b) Alarms 5610-T-L1 Sh 33B

         - TP-57

INSTRUCTOR ACTIVITIES



SD 063

FIGURE 14
Rev.6:12/12/95

QUESTION #40

Facility: Diablo Canyon

Vendor: WEC

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	2
Group #	1	1
K/A #	059	A4.11
Importance Rating	3.1	3.3

Ability to manually operate and monitor in the control room: Recovery from automatic feedwater isolation

Proposed Question:

The plant experienced a reactor trip and Safety Injection (SI) from 100% power.

What actions must be taken before the Main Feedwater Regulating and Bypass valves can be opened?

A: Reset SI signal, cycle the Reactor trip breakers, reset Feedwater Isolation

B: Reset SI signal, cycle the Reactor trip breakers only

C: Cycle the Reactor trip breakers, reset Feedwater Isolation only

D: Reset Feedwater Isolation only

Proposed Answer: A

Explanation (Optional):

A: Correct. The sealin for P-4 and SI must be removed, then FWI reset

B: Incorrect. Feedwater Isolation must be reset

C: Incorrect. SI must be reset

D: Incorrect. Both the P-4 and SI seal-in must be removed

Technical Reference(s):	OIM, Feedwater Isolation Signals, Page B-6-12, Rev. 28	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective:	37048 - Analyze automatic features and interlocks associated with the RPS	(As available)

Question Source:	Bank #	X	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41
	55.43

Comments:

Comments:
Modified from 2010 Diablo Canyon

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	059	A4.11
Importance Rating	3.1	

Ability to manually operate and monitor in the control room: Recovery from automatic feedwater isolation

Proposed Question: RO Question # 40

Given the following:

- Unit 4 experienced a Reactor Trip from 100% power.
- A Feedwater Isolation Signal was generated.
- A Safety Injection occurred.
- S/G levels have been maintained less than 60% NR during the event.

Which ONE of the following describes the MINIMUM actions must be taken before the Main Feedwater Regulating Valves can be opened?

A. ONLY reset SI signal

B. Reset SI signal and close the Reactor Trip Breakers

C. Close the Reactor Trip Breakers and push Feedwater Bypass Isolation Reset

D. Reset SI signal, close the Reactor Trip Breakers, and push Feedwater Bypass Isolation Reset

Proposed Answer: B

Explanation (Optional):

A. Incorrect since this will not allow the *Main* FRVs to be opened, if Tave went below 554. Plausible because it will allow the *Bypass* FRVs to be opened. See Figure 14 of LP 6902163.

B. CORRECT. Per LP 6902122, Section 3.5: "Feedwater Isolation also occurs when there is a Reactor Trip and Tavg is less than 554°F. This only affects the Main Feedwater Regulating Valves and causes them to close slowly (20 seconds). Tavg increasing above 554°F will not reset this Feedwater Isolation condition; closing the Reactor Trip Breakers will cause this to reset."

C. Incorrect since the FW Isolation signal does not have to be reset. Plausible because the 1st part is correct. Also plausible because the 2nd part is correct if the Bypass FRVs are to be opened.

D. Incorrect since the FW Isolation signal does not have to be reset. Plausible because 2 of the 3 actions must be performed

Technical Reference(s): LP 6902163, RPS and ESFAS, Figure 14. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 37048 - Analyze automatic features and interlocks associated with the RPS (As available)

Question Source: Bank #

Modified Bank # 66866 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

QUESTION 41

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	061	K5.01
Importance Rating	3.6	

Knowledge of the operational implications of the following concepts as the apply to the AFW: Relationship between AFW flow and RCS heat transfer

Proposed Question: RO Question # 41

Given the following:

- Unit 4 was operating at 100% power.
- A Reactor Trip due to a Loss of Main Feedwater.
- All RCPs are running.
- Due to equipment malfunctions, ONLY 'A' AFW Pump is in service.
- The 'A' AFW Pump speed has begun to slowly LOWER due to a malfunctioning governor.

Which ONE of the following describes the INITIAL impact on Pressurizer level if the pump speed CONTINUES to lower?

Pressurizer level:

A. rises due to a bubble formation in the Rx Vessel Head

B. rises due to decreased primary to secondary heat transfer

C. lowers due to a density decrease in the Pressurizer level

D. lowers due to Pressurizer outsurge to RCS

Proposed Answer: B

Explanation (Optional):

A. Incorrect since a bubble should not form in the Reactor Vessel Head with RCPs running. Plausible because "rises" could be correct if such a bubble were to form.

B. CORRECT. As AFW Pump speed decreases due to the governor valve closing, less heat is removed from the RCS via less steam to the pump turbine and less feedwater flow generated. PZR will rise as the RCS fluid expands.

C. Incorrect since PZR level will rise, not lower. Plausible because density decrease in Pressurizer level is possible with an insurge of cooler water lowering the saturation temperature of the fluid. This effect will cause the water volume to contract.

D. Incorrect since PZR level will rise, not lower. Plausible because PZR pressure *will* increase, causing spray flow to increase; If an applicant believes PZR level will lower, it is logical to believe that the level change is due to an outsurge

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # 68080

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2004 Indian Point 2

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 14

55.43

Principles of heat transfer, thermodynamics and fluid mechanics.

Comments:

NO ADDITIONAL

REFERENCE

MATERIAL PROVIDED

QUESTION 42

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062	K2.01
	Importance Rating	3.3	

Knowledge of bus power supplies to the following: Major system loads

Proposed Question: RO Question # 42

Given the following:

- Unit 3 is operating at 100% power.
- 3D 4KV Bus is aligned to the 3A 4KV Bus.
- A Loss of Offsite Power occurs.
- A 3A 4KV Bus undervoltage condition occurs and clears after 15 seconds.
- During the transient, the Supply From 4KV Bus 3A, 3AD01, trips OPEN.

(Assume no operator action.)

Which ONE of the following sets of loads DOES NOT have 4KV Bus power available?

A. Component Cooling Water Pump 3C and Emergency Containment Filter Fan 3C

B. Intake Cooling Water Pump 3A and Emergency Containment Cooler Fan 3C

C. Intake Cooling Water Pump 3C and Component Cooling Water Pump 3C

D. Emergency Containment Filter Fan 3A and Emergency Containment Cooler Fan 3C

Proposed Answer: C

Explanation (Optional):

A. Incorrect since Emergency Filter Cooler Fan 3C still has power available. Plausible – With an undervoltage condition on the 3A 4KV Bus, the applicant assumes UV load stripping of the Emergency Filter Cooler Fan 3C. Also, plausible because Component Cooling Water Pump 3C will be without power available without operator manual action in 3-EOP-E-0.

B. Incorrect since Intake Cooling Water Pump 3A and Emergency Containment Cooler Fan 3C still have power available. Plausible – With an undervoltage condition on the 3A 4KV Bus, the applicant assumes UV load stripping of the Intake Cooling Water Pump 3A and Emergency Containment Cooler Fan 3C.

C. CORRECT. When breaker 3AD01 opens, power was lost to the 3D 4KV Bus. Intake Cooling Water Pump 3C and Component Cooling Water Pump 3C will have power restored with operator manual action in 3-EOP-E-0.

D. Incorrect since Intake Emergency Containment Filter Fan 3A and Emergency Containment Cooler Fan 3C still have power available. Plausible – With an undervoltage condition on the 3A 4KV Bus, the applicant assumes UV load stripping of the Emergency Containment Filter Fan 3A and Emergency Containment Cooler Fan 3C.

Technical Reference(s):

5610-T-E, Operating Diagram Electrical Distribution

3-OP-055, Emergency Containment Cooling and Filtering Systems, Attachment 2

SD-170, Emergency Load Sequencers

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902157, Obj. 6 (As available)

Question Source: Bank # WTSI 61491

Modified Bank # (Note changes or attach parent)

New

Question History:	Last NRC Exam:	2003	Palo Verde

Question Cognitive Level:	Memory or Fundamental Knowledge	(1F)
	Comprehension or Analysis	
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	8
	55.43	

Components, capacity, and functions of emergency systems.

Comments:

ATTACHMENT 3
(Page 2 of 2)

EMERGENCY CONTAINMENT FILTERING SYSTEM BREAKER ALIGNMENT

480V MCC 3B

Component No.	Component Description	Normal Position	Checked (Initials)	Verify Labeling (√)
30611	Emergency Containment Filter Fan 3A	ON		

480V MCC 3C

Component No.	Component Description	Normal Position	Checked (Initials)	Verify Labeling (√)
30719	Emergency Containment Filter Fan 3C	ON		

480V MCC 3D

Component No.	Component Description	Normal Position	Checked (Initials)	Verify Labeling (√)
30806	Emergency Containment Filter Fan 3B	ON		

120V Lighting Panel LP38

Component No.	Component Description	Normal Position	Checked (Initials)	Verify Labeling (√)
LP38-1	Emergency Containment Filter 3A Motor Heater	ON		
LP38-2	Emergency Containment Filter 3B Motor Heater	ON		
LP38-4	Emergency Containment Filter 3C Motor Heater	ON		

Check (√) Tag column if tag is missing, mislabeled, illegible or improperly secured.

FINAL PAGE

ATTACHMENT 2
(Page 2 of 2)

EMERGENCY CONTAINMENT COOLING SYSTEM BREAKER ALIGNMENT

480V MCC 3B

Component No.	Component Description	Normal Position	Checked (Initials)	Verify Labeling (√)
30650	Emergency Containment Cooler Fan 3A	ON		

480V MCC 3C

Component No.	Component Description	Normal Position	Checked (Initials)	Verify Labeling (√)
30729	Emergency Containment Cooler Fan 3C	ON		

480V MCC 3D

Component No.	Component Description	Normal Position	Checked (Initials)	Verify Labeling (√)
30820	Emergency Containment Cooler Fan 3B	ON		

120V Lighting Panel LP37

Component No.	Component Description	Normal Position	Checked (Initials)	Verify Labeling (√)
LP37-18	Emergency Containment Cooler 3A, 3B, 3C Heaters	ON		

Check (√) Tag column if tag is missing, mislabeled, illegible or improperly secured.

QUESTION 43

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063	K1.03
	Importance Rating	2.9	

Knowledge of the physical connections and/or cause-effect relationships between the dc electrical system and the following systems: Battery charger and battery

Proposed Question: RO Question # 43

In accordance with 0-NOP-003.01, 125V Vital DC System, which ONE of the choices below completes the following statements?

If two battery chargers are connected to the battery bank, each battery charger is required to have a minimum output of ___(1)___ amps or be demonstrated to be able to supply the DC bus load independent of its associated battery charger.

The 125 VDC Battery Charger normal output voltage is ___(2)___ VDC.

A. (1) 10
(2) 131 to 140 VDC

B. (1) 10
(2) 121 to 127 VDC

C. (1) 20
(2) 131 to 140 VDC

D. (1) 20
(2) 121 to 127 VDC

Proposed Answer: A

Explanation (Optional):

A. <u>CORRECT</u>

1. If two battery chargers are connected to the battery bank, each battery charger is required to have a minimum output of 10 amps or be demonstrated to be able to supply the DC bus load independent of its associated battery charger. (T.S. S.R. 4.8.2.1.a.3)

2. In accordance with 0-NOP-003.01, 125V VITAL DC SYSTEM, Normal Battery Voltage is between 131 and 140 VDC.

B. <u>Incorrect</u>

1. If two battery chargers are connected to the battery bank, each battery charger is required to have a minimum output of 10 amps or be demonstrated to be able to supply the DC bus load independent of its associated battery charger. (T.S. S.R. 4.8.2.1.a.3)

2. Battery terminal voltage must be greater than 129 VDC, not 121-127 VDC. Plausible because the 1st part is correct.

C. <u>Incorrect</u>

1. If two battery chargers are connected to the battery bank, each battery charger is required to have a minimum output of 10 amps or be demonstrated to be able to supply the DC bus load independent of its associated battery charger. (T.S. S.R. 4.8.2.1.a.3) Plausible because student could associate 10 amps per charger which added together is 20 amps.

2. In accordance with 0-NOP-003.01, 125V VITAL DC SYSTEM, Normal Battery Voltage is between 131 and 140 VDC. Plausible because the 2nd part is correct.

D. <u>Incorrect</u>

1. If two battery chargers are connected to the battery bank, each battery charger is required to have a minimum output of 10 amps or be demonstrated to be able to supply the DC bus load independent of its associated battery charger. (T.S. S.R. 4.8.2.1.a.3) Plausible because student could associate 10 amps per charger which added together is 20 amps.

2. Battery terminal voltage must be greater than 129 VDC, not 121-127 VDC. Plausible because the 1st part is correct.

i

Technical Reference(s): 0-NOP-003.1, 125V Vital DC System

T.S. S.R. 4.8.2.1.a.3

(Attach if not previously provided)

Proposed References to be provided to applicants during examination:			None
Learning Objective:	LP 6902139, Obj. 12		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:		Last NRC Exam:	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge	(1F)	
	Comprehension or Analysis		
Question Difficulty Level: B			
10 CFR Part 55 Content:	55.41	8	
	55.43		

Components, capacity, and functions of emergency systems.

Comments:

2.2 Limitations

1. Technical Specification limit for operability of each battery bank and associated chargers is battery terminal voltage is greater than or equal to 129 VDC. (SR 4.8.2.1.a.2) Distractors B+D

2. If two battery chargers are connected to the battery bank, each battery charger is required to have a minimum output of 10 amps or be demonstrated to be able to supply the DC bus load independent of its associated battery charger. (T.S. S.R. 4.8.2.1.a.3) CORRECT ANSWER

3. The maximum current limiter setting for the 3A1, 3A2, 4B1, 4B2, and D51 Battery Chargers is 460 amps.

4. The maximum current limiter setting for the 3B1, 3B2, 4A1, and 4A2 Battery Charger is 345 amps.

5. High voltage trip setpoint of all battery chargers is 141 -142 VDC.

6. Normal battery ambient temperature is 60 to 90°F.

3.0 PREREQUISITES

None

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

Distractors B+D

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).

QUESTION 44

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064	A2.07
	Importance Rating	2.5	

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of operating under/over-excited

Proposed Question: RO Question # 44

Given the following:

- Unit 3 is operating at 100%.
- 3A Emergency Diesel Generator (EDG) has been manually started from the local panel to run in parallel with offsite power in accordance with 3-OP-023, Emergency Diesel Generator.
- The operator adjusts the EDG Output Voltage to be lower than line voltage with the synchroscope running slowly in the FAST direction.
- After the operator parallels the 3A EDG with the bus, the following indications are observed:
    - Diesel amps = 175 amps
    - Diesel load = 1000 KW

Which ONE of the following statements describes the (1) condition of the 3A Emergency Diesel Generator and the consequence of operating the EDG without taking action and (2) the corrective action to return the condition to normal?

A. (1) The diesel has assumed too much reactive load in LEAD which could result in limiting additional real load to prevent overheating generator windings.
(2) Slowly LOWER the A Diesel Gen Volt Regulator until A Diesel amps increase.

B. (1) The diesel has assumed too much reactive load in LAG which could result in limiting additional real load to prevent overheating generator windings.
(2) Slowly LOWER the A Diesel Gen Volt Regulator until A Diesel amps increase.

C. (1) The diesel has assumed too much reactive load in LEAD which could result in a reverse power trip of the output breaker or disruption of the rotor/stator coupled magnetic field.
(2) Slowly RAISE the A Diesel Gen Volt Regulator until A Diesel amps increase.

D. (1) The diesel has assumed too much reactive load in LAG which could result in a reverse power trip of the output breaker or disruption of the rotor/stator coupled magnetic field.
(2) Slowly RAISE the A Diesel Gen Volt Regulator until A Diesel amps increase.

Proposed Answer: C

Explanation (Optional):

A. Incorrect since diesel voltage is lower than line voltage, the diesel will be underexcited, not overexcited. Plausible because the student may assume lowering voltage will correct the problem.

B. Incorrect since the diesel assumed too much reactive load in lead. Plausible because the student may assume raising voltage will correct the problem.

C. Correct.

D. Incorrect and plausible since the diesel assumed too much reactive load in lead. The action is correct in this case for too much reactive load in lead.

Technical Reference(s): LP 6902136, *Emergency Diesel Generators and Auxiliaries*

3-OP-023, *Emergency Diesel Generator* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902136, Obj. 11 (As available)

Question Source: Bank # 60411

Modified Bank # (Note changes or attach parent)

New

Question History:		Last NRC Exam: 2004	Catawba
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		(3PEO)
Question Difficulty Level: C			
10 CFR Part 55 Content:	55.41	8	
	55.43		

Components, capacity, and functions of emergency systems.

Comments:

INITIALS
CK'D VERIF

5.5.2.25 (Cont'd)

CAUTION

The following guidelines are required to be followed to reduce the probability of EDG overload conditions without main generator lockout protection:

- ***If the 3A 4KV bus is not powered via the auxiliary transformer (3AA02 open), special attention is required to be given to the 3A EDG operating parameters during parallel operation to the system and the EDG is required to be tripped upon indication of impending overload.***
- ***If the 3A 4KV bus is powered via the auxiliary transformer (3AA02 closed), no special precautions are required as protection is provided by the main generator lockout.***
- ***Starting any of the following pumps may cause an EDG paralleled to the affected 4160V bus to trip and could cause damage to the EDG:***
  - ***Reactor coolant pump***
  - ***Condensate pump***
  - ***Steam generator feed pump***
  - ***Heater drain pump***
  - ***Circulating water pump***
- ***The Diesel Generator is out-of-service while parallel with the grid.***

____ b. Place the EDG Bkr 3AA20 Synchronizing Switch to ON.

____ c. Verify the WHITE synchronizing lights are ON.

____ d. Using the voltage adjust control switch adjust the generator voltage on the Bus Voltage Incoming indicator to match the voltage on the Bus Voltage Running indicator.

____ e. Using the governor control switch, adjust engine speed so that the pointer on the synchroscope is rotating slowly in the FAST direction.

____ f. Using the voltage adjust control switch, adjust the generator voltage on the Bus Voltage Incoming indicator slightly higher than the voltage on the Bus Voltage Running indicator.

____ g. Using the voltmeter switch, check generator voltages across all three phases are approximately equal.

INITIALS
CK'D VERIF

5.5.2.25 (Cont'd)

____ h. Verify 3A Diesel Generator frequency is between 58.8 and 61.2 Hz on the Gen Frequency indicator.

____ i. **WHEN** the Synchroscope pointer is at 12 o'clock position, **THEN** close the diesel generator breaker by placing the EDG Bkr 3AA20 Control Switch to the CLOSE position (spring return to normal).

____ (1) Verify the Diesel Generator Breaker 3AA20 has closed (Breaker GREEN light is OFF and RED light is ON).

____ j. Place the EDG Bkr 3AA20 Synchronizing switch to OFF.

____ k. Turn the governor control switch in the RAISE direction **AND** slowly increase diesel generator load to approximately 1000 KW on the Gen Power indicator.

NOTE

The following voltage adjustment will place the generator reactive load in lag.

26. Adjust the reactive load by performing the following:

____ a. While monitoring the Gen Current indicator, momentarily position the voltage adjust control switch to RAISE.

(1) **IF** Gen Current amps increased, **THEN** perform the following:

____ (a) Slowly LOWER the voltage until amps stop decreasing and start to increase (lead).

____ (b) Slowly RAISE the voltage until Gen Current amps increase (slightly in lag).

OR

____ (2) **IF** Gen Current amps decreased, **THEN** slowly RAISE the voltage until Gen Current amps increase (slightly in lag).

INITIALS
CK'D VERIF

5.5.2 (Cont'd)

NOTE

The Cooling Water System contains chromates, and if any cooling system leakage is observed, the Shift Manager and Chemistry are required to be notified.

____ 27. Inspect the 3A Diesel Generator for leaks or abnormalities.

____ a. Inspect the bucket under the air box drain for any additional accumulation of fluids resulting from the start.

____ b. **IF** Cooling Water System leakage is observed, **THEN** plug floor drains under air skid and on south side of engine.

CAUTION

The Emergency Diesel Generator load shall not exceed 2750 KW and generator amperage shall not exceed 477 amps.

____ 28. Turn the governor control switch in the RAISE direction **AND** increase diesel generator load until it is between 2300 and 2500 KW.

NOTE

The following voltage adjustment will place the generator reactive load in lag.

29. Adjust the reactive load by performing the following:

____ a. While monitoring the Gen Current indicator, momentarily position the voltage adjust control switch to RAISE.

(1) **IF** Gen Current amps increased, **THEN** perform the following:

____ (a) Slowly LOWER the voltage until amps stop decreasing and start to increase (lead).

____ (b) Slowly RAISE the voltage until Gen Current amps increase (slightly in lag).

OR

____ (2) **IF** Gen Current amps decreased, **THEN** slowly RAISE the voltage until Gen Current amps increase (slightly in lag).

INITIALS
CK'D VERIF

5.5.2 (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.

30. 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)

____ 31. Verify 3A Diesel Generator operating parameters are within normal ranges per Enclosure 1 (except lube oil level).

____ a. **IF** diesel generator is to be run greater than 1 hour, **THEN** perform Attachments 9 and 10, Control Room and Local Data Sheets.

____ 32. Operate the 3A Diesel Generator for a minimum of 1 hour, **OR** as directed by the Shift Manager.

____ 33. Ensure all log entries specified in Subsection 2.2 are recorded.

____ 34. Complete the QA Record Page for this subsection.

____ 35. Enter Section 6.0 of this procedure to shut down 3A Diesel Generator.

6.3 Shutdown of the 3A Emergency Diesel Generator from the Diesel Building

INITIALS
CK'D VERIF

Date/Time Started:________________

6.3.1 Initial Conditions

____ 1. All applicable prerequisites as listed in Section 3.0 are satisfied.

____ 2. 3A Diesel Generator is operating per Section 5.0 of this procedure or has automatically started and conditions allow shutdown.

____ 3. Permission has been obtained from the Shift Manager to perform this section.

6.3.2 Procedure Steps

NOTE

Placing the Master Control switch in the OFF or LOCAL position will cause Control Room Annunciator F 8/5, EDG A MASTER CONTROL SW OFF-NORMAL to activate. The Unit 3 RCO should be informed.

____ 1. Place the master control switch to LOCAL at 3A EDG Electrical Control Panel 3C12A.

____ 2. **IF** EDG breaker is closed, **THEN** perform the following at 3A EDG Electrical Control Panel 3C12A:

CAUTION

Diesel generator load should not be reduced to less than 200 KW.

____ a. Turn the governor control switch in the LOWER direction **AND** decrease diesel generator load to approximately 200 KW indicated on the Gen Power indicator.

____ b. Open 3A Diesel Generator Breaker 3AA20, by placing the EDG Bkr 3AA20 Control Switch to the TRIP position (spring return to normal).

____ (1) Verify the 3A Diesel Generator Breaker, 3AA20, has opened. (Breaker control GREEN light is ON and RED light is OFF).

____ (2) **IF** the 3A Diesel Generator Breaker, 3AA20, was closed from the Control Room, **THEN** have the Unit 3 RCO position the EDG A to 3A 4KV Bus 3AA20 switch to the TRIP position (spring return to normal).

Refer to PPT slide	***194***

12. **HANDLING REACTIVE LOAD ON THE DIESEL GENERATOR**

A. Ensures there is a lagging power factor

B. Pick up electrical load

1) Real Load - With a constant voltage on the 4160 volt buses, there is for every value of real load (KW) supplied a minimum value of amps that must be present.

2) Reactive Load - The diesel generator ammeter on the console will always indicate that minimum value plus any amps due to reactive power (VARS).

3) Power Factor - When the power factor (p.f.) is one (1), the ammeter will indicate the minimum value.

4) The Operator can verify the diesel generator is operating in the lag mode at any time by just bumping the diesel generator Voltage Adjust Switch in the Raise direction.

a. If the diesel generator current decreases, the machine is operating in the Lead mode.

b. Operator should continue to bump in the Raise direction until current shows an increase.

11	EXPLAIN the operation and alignment of the Emergency Diesel Generators and Auxiliaries for the following conditions/ plant evolutions: • Normal Start Operation • Paralleling Operations • Normal Stop	X	X	X	X	X

13. **PARALLEL OPERATION**

A. Voltage regulation

1) Extreme caution should be used when adjusting grid voltage while the EDG is parallel to the grid. Main Generator uses MVAR while EDG uses KVARS.

QUESTION 45

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073	A1.01
	Importance Rating	3.2	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels

Proposed Question: RO Question # 45

Which ONE of the following describes (1) an initiating signal to cause a Control Room Ventilation Isolation and (2) the order in which the Emergency Air Supply Fans, SF-1A (V-29A) and SF-1B (V-29B) start?

A. (1) RAI-6642, Control Room HVAC Radiation Monitor
(2) SF-1A starts, SF-1B starts only on LOW flow

B. (1) R-14, Plant Vent Gas Monitor
(2) SF-1A starts, SF-1B starts only on LOW flow

C. (1) RAI-6642, Control Room HVAC Radiation Monitor
(2) SF-1B starts, SF-1A starts only on LOW flow

D. (1) R-14, Plant Vent Gas Monitor
(2) SF-1B starts, SF-1A starts only on LOW flow

Proposed Answer: C

Explanation (Optional):

A. Incorrect since SF-1B does start first. Plausible because the 1st part is correct. Also plausible because normally an “A” component will start before a “B” component. “A” will start if “B” is out of service.

B. Incorrect since SF-1B does start first and RAI-6642, Control Room HVAC Radiation Monitor is the initiating signal. Plausible because normally an “A” component will start

before a “B” component. Also, “A” will start if “B” is out of service.

C. CORRECT

D. Incorrect since RAI-6642, Control Room HVAC Radiation Monitor is the initiating signal. Plausible because 2nd part is right. Also, the applicant assumes the initiating signal to protect the Control Room is generated from R-14, Plant Vent Gas Monitor.

Technical Reference(s): LP 6902155, Ventilation System and Air Conditioning (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902155, Obj. 5 (As available)

Question Source: Bank # WTSI 52577

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2006 VC Summer

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests the ability to predict changes to parameters (CR ventilation lineup - start of fan on low flow) on HIGH radiation level (in the CR ventilation intake).

ATTACHMENT 1
(Page 2 of 2)

CONTAINMENT/CONTROL BUILDING VENTILATION SYSTEMS ISOLATION VERIFICATION

CONTROL BUILDING VENTILATION SYSTEMS

COMPONENT Number	COMPONENT DESCRIPTION	ISOLATION POSITION
D-1A	Ventilation Inlet Damper	CLOSE
D-1B	Ventilation Inlet Damper	CLOSE
EF-9	Toilet Exhaust Fan (V-28)	STOP
EF-20	Kitchen Exhaust Fan (V-56)	STOP
D-14	Toilet Exhaust Damper	CLOSE
D-22	Kitchen Exhaust Damper	CLOSE
SF-1B	Control Room – Emergency Ventilation Supply Fan (V-29B)	START
D-2	East Inlet Damper	OPEN
D-3	West Inlet Damper	OPEN
D-11A	Control Room Recirc Damper	OPEN
D-11B	Control Room Recirc Damper	OPEN

INSTRUCTOR ACTIVITY

OBJECTIVES **3.a., 4.a., 5.a., 6.a. and 7.a.**	NSO/LPRO/LPSO
Control Building Component Description	**PPT Slides 17 - 13**

1.3 ' Component Description

A. Supply Fans and Dampers

a. Emergency supply fans SF-1B and SF-1A, are centrifugal type fans located in the mechanical equipment room.

b. Supply fan SF-1A receives power from 4D MCC (bkr 40831)

c. SF-1B receives power from 3D MCC (bkr 30809).

d. normally used in conjunction with the charcoal filter unit to remove radioactive particulate from the Control Room atmosphere.

e. Upon an initiating signal, fan SF-1B will start.

f. If the fan fails to start, a low flow signal is generated by FS-6659A which starts the alternate fan SF-1A after 75 seconds. The Control Room operator is alerted to this condition by annunciator X-6/1 "CONTROL BLDG ELEV CAB TRBL/EMERG VENT LOW FLOW" which alarms at 300 CFM.

g. In the event that both fans are running, and a high flow signal (1260 CFM) is generated by FS-6659B, then after 30 seconds supply fan SF-1B will be stopped. Time delays permit damper operation and flow stability before the high or low flow logic is addressed.

h. The supply fans, SF-1A and SF-1B, and the other associated dampers for this system are operated from the control panel 4QR81 and 4QR82 in the control room

i. Dampers D-2 and D-3 provide a path for the outside air to the suction of the supply fans.

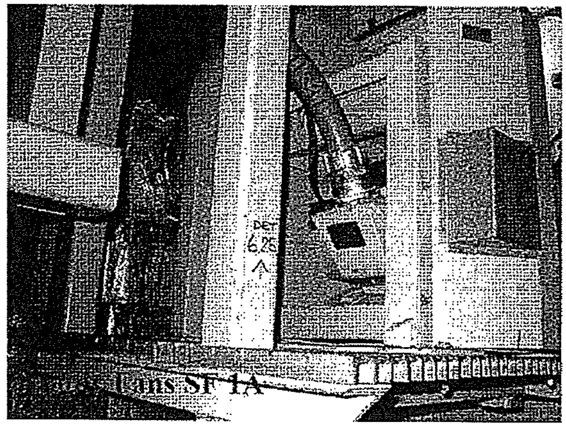
j. Actual system flow is regulated by two volume control dampers (D-20 and D-21) located in the mechanical equipment room.

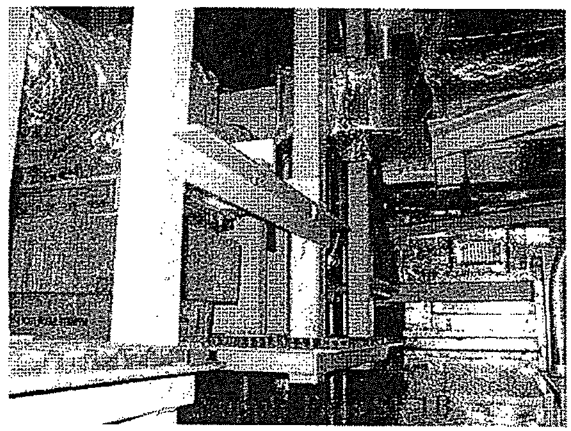
k. Damper D-2 is powered from 3P22-11 and D-3 is powered from 4P23-11.

OBJECTIVES **3.a., 4.a., 5.a., 6.a. and 7.a.** NSO/LPRO/LPSO

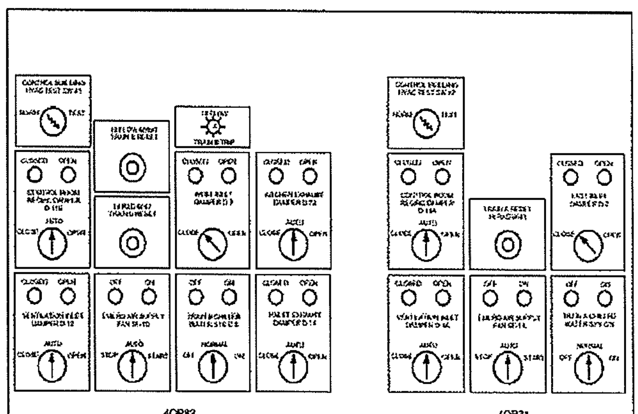
1.3 Component Descriptions

Supply Fans and Dampers





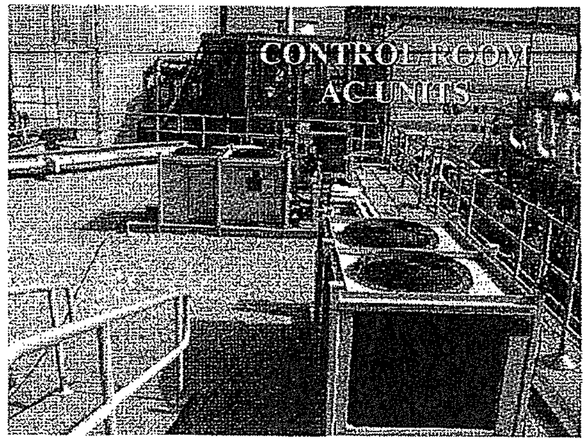
The Control Building emergency supply fans, SF-1B and SF-1A, are centrifugal type fans located in the mechanical equipment room. Supply fan SF-1A receives power from 4D MCC (bkr 40831) and SF-1B receives power from 3D MCC (bkr 30809). They are normally used in conjunction with the charcoal filter unit to remove radioactive particulate from the Control Room atmosphere. Upon an initiating signal, fan SF-1B will start. If the fan fails to start, a low flow signal is generated by FS-6659A which starts the alternate fan SF-1A after 75 seconds. The Control Room operator is alerted to this condition by annunciator X-6/1 "CONTROL BLDG ELEV CAB TRBL/EMERG VENT LOW FLOW" which alarms at 300 CFM. In the event that both fans are running, and a high flow signal (1260 CFM) is generated by FS-6659B, then after 30 seconds supply fan SF-1B will automatically stop. Time delays permit damper operation and flow stability before the high or low flow logic is addressed. The supply fans, SF-1A and SF-1B, and the other associated dampers for this system are operated from the control panel 4QR81 and 4QR82 in the control room. Refer to Figure 12.



Control & Test Panel located on Unit 4 only

Dampers D-2 and D-3 provide a path for the outside air to the suction of the supply fans. Actual system flow is regulated by two volume control dampers (D-20 and D-21) located in the mechanical equipment room. Damper D-2 is powered from 3P22-11 and D-3 is powered from 4P23-11.

Air Conditioning Units

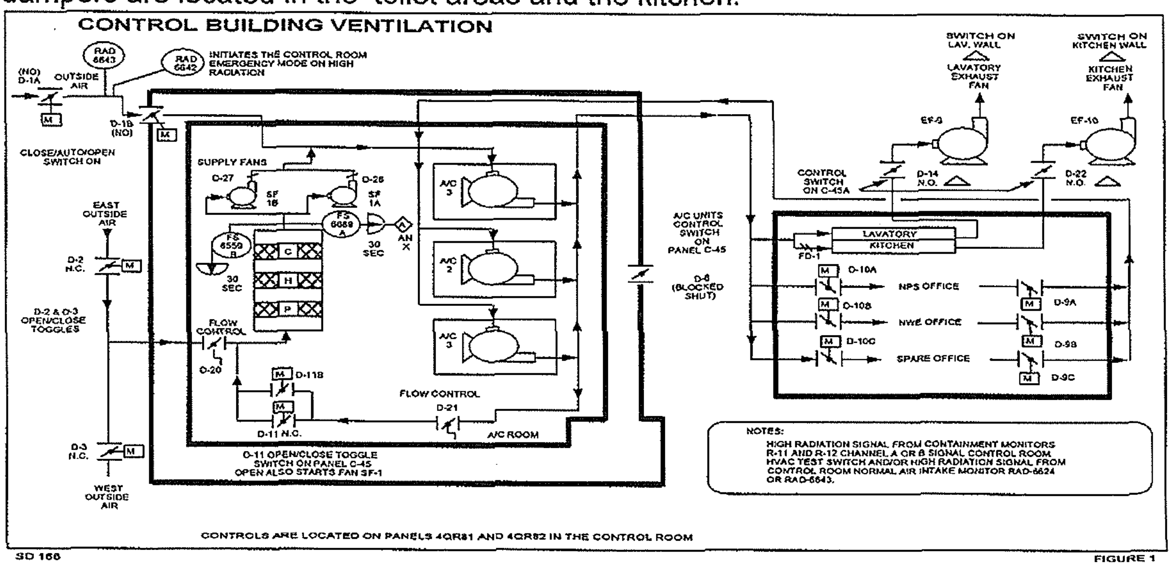


The air conditioning units are located on top of the Control Building roof. There are three separate but identical air conditioner units, each consisting of a compressor E-17A, E-17B and E-17C and an associated air handler (E-16A, E-16B and E-16C) which maintains temperature and humidity in the Control Building ventilation system. Refer to 5610-M-3025 Sheet 1. These units discharge cooled air to a common supply header which supplies air to various ductwork plenums for distribution throughout the control building. Each unit transfers up to 1.65 x 10-5 BTU/hr of cooling.

The air conditioning units are run continuously thus, have no control switches. However, each air handler is fused for circuit and motor overload protection. The fuse blocks are located in the Mechanical Equipment Room. Each unit operates independently with its own thermostat, seismically mounted in the Control Room. Unit A power is supplied from MCC 3B (breaker 30636) or from MCC-4B (breaker 40698) via transfer switch S-76 which is located in the unit four M/G set room. Unit B is powered from MCC 4D (breaker 40823), and unit C from MCC 3D (breaker 30823). Local knife switches for the a/c units are located on the roof.

Exhaust Fans and Dampers (Refer to Figure 1)

There are two exhaust fans with associated dampers in the Control Room ventilation system. They exhaust from the rooms to the outside atmosphere. These fans and dampers are located in the toilet areas and the kitchen.



SD 166

FIGURE 1
Rev. 2: 7/88

1.4 System Operations

Normal Operations

The Control Room atmosphere is filtered and conditioned by a separate ventilation system. The system circulates air from the Control Room and the Control Room offices through return air ducts to three air handling units located in the mechanical equipment room. The air is drawn into the air handling units through roughing filters and then is cooled. Conditioned air is then directed back to the rooms through a supply air duct system.

During normal operation, fresh makeup air is admitted to this system through an intake louver and two dampers in series (D-1A and D-1B) located in the east wall of the Control Building. This system maintains a positive pressure in the Control Room Envelope greater than that in the cable spreading room in order to prevent smoke from a potential fire in the cable spreading room from entering the Control Room. All Control Room penetrations, including doors, are designed for leak tightness standards. Since the Control Room is maintained at slightly more than atmospheric pressure, the infiltration of contaminated air into the Control Room is negligible. Two radiation monitors located in the air intake downstream of dampers D-1A and D-1B continuously monitor for radiation in the incoming air. Refer to Figure 1 and control drawing 5610-T-E-4535.

INSTRUCTOR ACTIVITY

Control Building Accident Conditions	PPT Slides 31 - 35

B. Accident Conditions

a. The Control Room ventilation will automatically be placed in a recirculation mode for a number of reasons.

b. In the recirculation mode the air is processed through a series of filters to maintain the control room environment acceptable during adverse radiological conditions. The normal outside air intake dampers D-1A and D-1B will close and emergency recirculation fan SF-1B will start. Outside air dampers D-2 and D-3 will open and recirculation dampers D-11A and D-11B will open. Kitchen and lavatory exhaust fans V-56 and V-28 will automatically turn off and their associated dampers will close.

c. The following signals will automatically activate the recirculation mode:

- Control Room HVAC test switches (two)
- High radiation from Control Room intake radiation monitors (6642 or 6643)
- High containment radiation from R11 or R12
- Automatic or manual initiation of safety injection
- Manual containment isolation phase 'A' (1/2 pushbuttons)
- Manual containment isolation phase 'B' (2/2 pushbuttons)

Immediate automatic actions that will occur:

C. Control room (HVAC) system shifts to recirculation mode.

a. Recirculation damper D-11A and D11B opens (MOV 6543 A & B) and the Emergency Supply fan SF-1B starts (V-29B)

Note: If the flow on the emergency supply fan decreases below 300 SCFM or the FS6659A setpoint, then the second fan will start after a 75 second time delay. (Control room ventilation lo flow is shown on common annunciator panel.)

b. Outside air dampers D-2 (East) and D-3 (West) open (MOV 6541 & 6542)

c. Inlet air damper D-1A and D-1B closes

d. Lavatory exhaust damper D-14 (MOV 6550) and kitchen exhaust damper D-22 (MOV 6549) close (See Figure 1.)

e. EF-20 Kitchen and EF-9 Lavatory Exhaust Fans STOP.

Accident Conditions

The Control Room ventilation system, which normally draws in fresh air from the outside, also has the capability to go into a recirculation mode. The Control Room ventilation will automatically be placed in a recirculation mode for a number of reasons. In the recirculation mode the air is processed through a series of filters to maintain the control room environment acceptable during adverse radiological conditions. The normal outside air intake dampers D-1A and D-1B will close and emergency recirculation fan SF-1B will start. Outside air dampers D-2 and D-3 will open and recirculation dampers D-11A and D-11B will open. Kitchen and lavatory exhaust fans V-56 and V-28 will automatically turn off and their associated dampers will close.

The following signals will automatically activate the recirculation mode:

1. Control Room HVAC test switches (two)
2. High radiation from Control Room intake radiation monitors (6642 or 6643)
3. High containment radiation from R11 or R12
4. Automatic or manual initiation of safety injection
5. Manual containment isolation phase 'A' (1/2 pushbuttons)
6. Manual containment isolation phase 'B' (2/2 pushbuttons)

Immediate automatic actions that will occur:

1. Control room (HVAC) system shifts to recirculation mode.
   a. Recirculation damper D-11A and D11B opens (MOV 6543 A & B) and the Emergency Supply fan SF-1B starts (V-29B)

Note: If the flow on the emergency supply fan decreases below 300 SCFM or the FS6659A setpoint, then the second fan will start after a 75 second time delay. (Control room ventilation lo flow is shown on common annunciator panel.)

   b. Outside air dampers D-2 (East) and D-3 (West) open (MOV 6541 & 6542)
   c. Inlet air damper D-1A and D-1B closes
   d. Lavatory exhaust damper D-14 (MOV 6550) and kitchen exhaust damper D-22 (MOV 6549) close (See Figure 1.)
   e. EF-20 Kitchen and EF-9 Lavatory Exhaust Fans STOP.

QUESTION 46

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	076	2.2.12
Importance Rating	3.7	

Equipment Control: Knowledge of surveillance procedures.

Proposed Question: RO Question # 46

Which ONE of the choices below completes the following statement regarding limitations placed on the Intake Cooling Water Pump?

In accordance with 3-NOP-019, Intake Cooling Water System, if an ICW Pump has flows greater than ___(1)___ gpm for more than twenty minutes, then once pump flow is less than that value, the crew contacts the IST Coordinator and performs 3-OSP-019.1, Intake Cooling Water Pump Inservice Test to ___(2)___.

A. (1) 10,000;
(2) prove no pump damage exists by measuring pump vibration and d/p

B. (1) 10,000;
(2) restore pump operability

C. (1) 18,500;
(2) prove no pump damage exists by measuring pump vibration and d/p

D. (1) 18,500;
(2) restore pump operability

Proposed Answer: C

Explanation (Optional):

A. Incorrect since the flowrate limit is 18,500, not 10,000 gpm. Plausible because the 2nd part is correct. Also plausible because 3-NOP-019, Section 2.2.4.2. states: "Maximum ICW flowrate to each CCW HX during normal operation should **NOT** exceed 10,000 gpm in order to minimize long term tube side erosion of the CCW HXs."

B. Incorrect since the flowrate limit is 18,500, not 10,000 gpm. Also incorrect since 3-OSP-019.1 is not performed to restore operability, it is performed to verify operability.

C. CORRECT. Per 3-NOP-019, Section 2.2.2.5: "Maximum ICW Pump flowrate is permitted up to 18,500 gpm. If an ICW Pump is operated in excess of 18,500 gpm, then flow should be reduced to less than 18,500 gpm as soon as possible. If an ICW pump has operated at flows greater than 18,500 gpm for more than twenty (20) minutes, then once pump flow has been reduced to 18,500 gpm or less, the IST Coordinator should be notified to perform vibration and pump DP testing per 3-OSP-019.1, Intake Cooling Water Pump Inservice Test to ensure integrity of the affected pump."

D. Incorrect since 3-OSP-019.1 is performed to prove no pump damage (inoperability). Plausible because first part is correct, and the ICW Pump remains operable until proven otherwise.

Technical Reference(s): 3-NOP-019, Intake Cooling Water System

3-OSP-019.1, Intake Cooling Water Pump Inservice Test

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests what portions of a surveillance procedure have to be performed when an ICW (Service Water) pump exceeds flow limits.

2.2 Limitations

2.2.1 General

1. When in MODE 1, 2, 3 and 4 the ICW System shall be OPERABLE with three ICW pumps and two ICW headers.

2. When in MODE 1, 2, 3 and 4, AND there are **NOT** two OPERABLE ICW pumps with independent power supplies, then a 72 hour LCO Action Statement is in effect to restore two pumps from independent power supplies to OPERABLE status.

3. When removing an ICW header from service in MODE 1, 2, 3 or 4, a 72 hour LCO Action Statement is in effect until header is returned to service. Because **NO** flow is being diverted away from the CCW HXs, as when a strainer is being backwashed, a continuous watch is **NOT** required to be posted when the header is out of service.

2.2.2 ICW Pumps

1. ICW pumps should **NOT** be operated with water level of Pump Well lower than 23 feet 2 inches below the centerline of the pump discharge, which is 20 feet 8 inches below the Intake floor grating.

2. If ICW pump motor heaters are OOS for more than five days, then Engineering is required to determine pump OPERABILITY.

3. The minimum flow for any ICW pump during continuous operation should be greater than or equal to 3200 gpm.

4. When starting ICW Pumps, the maximum flow rates for the inservice HXs may be exceeded. The high flow rate is acceptable provided the duration of high flow is minimized.

5. Maximum ICW Pump flowrate is permitted up to 18,500 gpm. If an ICW Pump is operated in excess of 18,500 gpm, then flow should be reduced to less than 18,500 gpm as soon as possible. If an ICW pump has operated at flows greater than 18,500 gpm for more than twenty (20) minutes, then once pump flow has been reduced to 18,500 gpm or less, the IST Coordinator should be notified to perform vibration and pump DP testing per 3-OSP-019.1, Intake Cooling Water Pump Inservice Test to ensure integrity of the affected pump.

2.2.3 Basket Strainers

1. DP across the ICW Basket Strainers at the CCW and TPCW HXs should **NOT** exceed 1.5 psid.

2. While isolating a ICW/CCW Strainer, ICW flows below the "Minimum Required ICW/CCW Flowrate" are permitted for up to 5 minutes, without declaring ICW System inoperable. If flow is below the "Minimum Required ICW/CCW Flowrate" for greater than 5 minutes, then the ICW System shall be declared inoperable at the time when flow initially went below the "Minimum Required ICW/CCW Flowrate."

2.2.4 CCW Heat Exchangers

1. ICW outlet temperature from CCW HXs should **NOT** exceed 120°F.

2. Maximum ICW flowrate to each CCW HX during normal operation should **NOT** exceed 10,000 gpm in order to minimize long term tube side erosion of the CCW HXs. The ICW flowrate for each CCW HX may be increased to 12,850 gpm for up to 72 hours period to accommodate HX or Basket Strainer cleanings.

3. Minimum ICW flowrate to CCW HXs during normal operation should be determined as follows:

   - If three CCW HXs are in service, then the minimum required ICW flow rate is 11,000 gpm TOTAL, AND at least 3500 gpm through each CCW HX.

   - If less than three CCW HXs are in service, OR flow through any HX is less than 3500 gpm, the Minimum Required ICW/CCW Flowrate will be determined by the STA or Shift Manager per Attachment 5, Instructions for Determining Minimum Flow Using Flow Rate Curves or Manual Calculation.

   - To decrease the Minimum Required ICW/CCW Flowrate, the CCW HX with highest tube resistance may be declared OOS at the discretion of the Shift Manager while the basket strainer is OOS. If a CCW HX is declared OOS, then the ICW flow through that CCW HX can **NOT** be used to calculate the Minimum Required ICW/CCW Flowrate.

QUESTION 47

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	078	A4.01
Importance Rating	3.1	

Ability to manually operate and/or monitor in the control room: Pressure gauges

Proposed Question: RO Question # 47

Given the following:

- Unit 4 is at 8% power.

With the above conditions, which ONE of the following Control Room (VPA) sustained indications would require a Reactor Trip?

A. PI-4-1629, Turbine Bearing Oil at 5.0 psig

B. PI-4-1444, Inst Air Pressure at 50 psig

C. PI-4-1406, Condenser Vacuum, at 19" Hg.

D. Pressurizer Pressure Channels:
PI-4-455 at 2385 psig, PI-4-456 at 2250 psig, and PI-4-457 at 2245 psig.

Proposed Answer: B

Explanation (Optional):

A. Incorrect since a Reactor Trip is not warranted for this condition. Plausible because, per 4-ARP-097.CR.E, 6/2, the Turbine should have tripped for these conditions. If power was above P-7 (10%), a Reactor Trip is required.

B. CORRECT. Per 0-ONOP-013, RNO Step 2: "IF pressure is less than 65 psig AND the available Instrument Air Compressor(s) is/are unable to restore pressure, THEN trip the affected unit(s) and enter 3/4-EOP-E-0 while continuing with this procedure."

C. Incorrect since a Reactor Trip is not warranted for this condition. Plausible because, per 4-ARP-097.CR.E, 6/3, CONDENSER LO VACUUM TRIP, the Turbine should have tripped for these conditions. If power was above P-7 (10%), a Reactor Trip is required.

D. Incorrect since a Reactor Trip is not warranted for this condition. Plausible because, per 4-ARP-097.CR.A, 8/1, PZR PROTECTION HI PRESS, it would be correct if either 456 or 457 read greater than or equal to 2385 psig.

Technical Reference(s): 4-ARP-097.CR.E, 6/3

4-ARP-097.CR.E, 6/2

4-ARP-097.CR.A, 8/1

0-ONOP-013, Loss of Instrument Air (Attach if not previously provided)

LP 6902286, Loss of Instrument Air

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902286, Obj. 3 (As available)

Question Source: Bank # 70365

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 Braidwood

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation,

signals, interlocks, failure modes, and automatic and manual features.

Comments:

CAUSES: Inability to maintain condenser vacuum

E24

CONDENSER LO VACUUM TRIP

DEVICE:	SETPOINT:	LOCATION:
PS-3614	20" HG	N/A

NOTE

- The turbine will also trip at 3 psig, but will **NOT** cause annunciator E 6/3 to alarm.
- Lo vacuum turbine trip is performed by the Turbine trip block (see Ref. 3)

ALARM CONFIRMATION

1. **CHECK** condenser vacuum gauges PI-1406 and PI-1612 on VPA.

OPERATOR ACTIONS

1. **ENSURE** turbine trip.
2. IF turbine trip has resulted in a reactor trip, THEN **GO TO** 3-EOP-E-0, Reactor Trip or Safety Injection.
3. **INVESTIGATE** cause of loss of vacuum using 3-ONOP-014, Main Condenser Loss of Vacuum.

REFERENCES:

1. FPL Logic Diagram 5610-T-L1, Sh 3
2. FPL EWD 5610-E-29, Sh 23
3. FPL only 5610-M-3-68, Sh 1

Turbine should trip but no reactor trip < 10% power

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** air dryer is excessively purging, **THEN** CLOSE the associated purge exhaust block valve: U-3 (3T9): 3-40-380 U-4 (4T9): 4-40-380 * **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.

REVISION NO.: 1	PROCEDURE TITLE:	PAGE: 37
PROCEDURE NO.: 3-ARP-097.CR.E	CONTROL ROOM RESPONSE - PANEL E TURKEY POINT UNIT 3	WINDOW: 6/2 (Page 1 of 1)

CAUSES:
1. Break or large leak in bearing lube oil header
2. Changing over to an unfilled lube oil cooler

E15

TURB BEARING OIL LO PRESS TRIP

DEVICE: PS-3612

SETPOINT: 5.5 psig

LOCATION: N/A

NOTE

Turbine trip is performed mechanically by the turbine trip block (see Ref. 3).

ALARM CONFIRMATION

1. **CHECK** turbine bearing oil pressure, PI-1629 indicating 5.5 psig or less on VPA.

OPERATOR ACTIONS

1. **ENSURE** the following automatic actions have occurred:
   - Turbine trip
   - Auxiliary oil pump start (10 psig)
   - Turning gear oil pump start (9 psig)
   - Emergency bearing oil pump start (7 psig)
2. IF turbine trip has resulted in a reactor trip, THEN **GO TO** 3-EOP-E-0, Reactor Trip or Safety Injection.
3. **DISPATCH** operator to check for leaking oil.
4. IF oil press is below 3 psig, THEN **BREAK** vacuum to reduce turbine roll down time.

REFERENCES:
1. FPL Logic Diagram 5610-T-L1, Sh 3, 4
2. FPL EWD 5610-E-29, Sh 23
3. FPL only 5610-M-3-68, Sh 1

Distractor A:

Power < 10% ; TURBINE TRIP but no reactor trip. DISTRACTOR C

CAUSES:
1. Failed instrument
2. Load rejection
3. Pressure control failure
4. Rod control malfunction

A8

PZR PROTECTION HI PRESS

DEVICE:	SETPOINT:	LOCATION:
• PC-455A	2385 psig	N/A
• PC-456A	2385 psig	N/A
• PC-457A	2385 psig	N/A

ALARM CONFIRMATION

1. **CHECK** the following greater than or equal to 2385 psig at VPA:
   - PI-3-455
   - PI-3-456
   - PI-3-457
2. **CHECK** the following bistables ON at VPB:
   - BS-3-455A
   - BS-3-456A
   - BS-3-467A

OPERATOR ACTIONS

DISTRACTOR 'D'

1. IF either of the following conditions exist,
   - Two or more press protection indicators are greater than 2385 psig.
   - Two or more bistables are ON.

   THEN:

   A. **TRIP** the reactor and turbine.

   B. **PERFORM** 3-EOP-E-0, Reactor Trip Or Safety Injection.
2. IF an instrument has failed, THEN **REFER TO** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
3. **REFER TO** 3-ONOP-041.5, Pressurizer Pressure Control Malfunction.

REFERENCES:
1. FPL Drawing 5610-T-L1, Sheet 18, PZR Caused RX Trips and SI
2. Tech Spec 3.3.2, RX Trip System Instrumentation

QUESTION 48

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103	K1.02
	Importance Rating	3.9	

Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems: Containment isolation/containment integrity

Proposed Question: RO Question # 48

Given the following:

- Unit 3 was at 100% power.
- A steam break occurred outside Containment.
- A manual Reactor Trip was initiated.
- A manual Safety Injection was initiated.
- ONLY one Containment Phase A pushbutton was depressed.

Which ONE of the following correctly describes the status of the Phase A and Phase B isolation valves BEFORE any additional operator action(s)?

A. NOT all Phase A valves are closed; all Phase B valves are closed.

B. NOT all Phase A valves are closed; all Phase B valves are open.

C. All Phase A valves are closed; all Phase B valves are closed.

D. All Phase A valves are closed; all Phase B valves are open.

Proposed Answer: D

Explanation (Optional):

A. Incorrect since Containment pressure is not rising for this event; therefore, a Phase B isolation signal will not be generated. Plausible because a novice applicant could believe that Phase B isolation was setup the same way as Phase A.

B. Incorrect. Both train of Phase A Valves will close. Plausible - Phase A isolation valves are setup where one Train of Phase A closes one of the valves (either inside or outside

Containment) and the other Train closes the other valve for that penetration. Since Containment pressure is not rising for this event, a Phase B isolation signal will not be generated.

C. Incorrect since Containment pressure is not rising for this event; therefore, a Phase B isolation signal will not be generated. The 1st part is plausible because it is correct. The 2nd part is plausible because it would be correct if the MSLB was inside Containment.

D. Correct since one Phase A pushbutton will cause all valves to close. Phase B Valves will remain open due to the break is outside Containment.

Technical Reference(s): LP 6902163, RPS and ESFAS (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902163, Obj. 8 (As available)

Question Source: Bank # 18706

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2003 Indian Point 3

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

QUESTION 49

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004	A2.23
	Importance Rating	2.6	

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High filter D/P

Proposed Question: RO Question # 49

Given the following:

- Unit 4 is operating at 100% power.
- Annunciator SEAL WATER INJ FILTER HI ΔP (A 6/6) actuates.
- Local investigation indicates that the ΔP is 23 psid.

Which ONE of the following completes the following statements:

If not corrected, ___(1)___. Perform 4-ARP-097.CR.A, Control Room Response - Panel A, to ___(2)___ to mitigate the consequence.

A. (1) Labyrinth Seal ΔP goes high
(2) close HCV-4-121, Charging Flow to Regen Hx

B. (1) Seal Injection flow goes low
(2) close HCV-4-121, Charging Flow to Regen Hx

C. (1) Labyrinth Seal ΔP goes high
(2) place a standby filter in service

D. (1) Seal Injection flow goes low
(2) place a standby filter in service

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible due to these components are unique in function/design, however there are similar system titles between Seal Inj. Filter ΔP and Labyrinth Seal ΔP. The action of closing HCV-4-121, Charging Flow to Regen Hx, does lower the Labyrinth Seal ΔP.

B. Incorrect. Plausible due to Seal Injection flow lowers as Seal Inj. Filter ΔP rises. The applicant decides to close HCV-4-121, Charging Flow to Regen Hx, to raise Seal Injection flow to RCPS. This is incorrect because the initiating problem is high Seal Inj. Filter ΔP.

C. Incorrect. Plausible due to these components are unique in function/design, however there are similar system titles between Seal Inj. Filter ΔP and Labyrinth Seal ΔP. The action of placing a place a standby filter in service is logical with an initiating annunciator of SEAL WATER INJ FILTER HI ΔP (A 6/6).

D. CORRECT

<u>4-ARP-097.CR.A, 6/6</u>
IF filter ΔP is high, THEN:
A. IF standby filter is available, THEN:
(1) PLACE standby filter in service using 4-OP-047, CVCS - Charging and Letdown.
(2) ADJUST seal injection flow as necessary to establish 6 to 13 gpm per RCP using the applicable procedure(s):
- 4-NOP-041.01A, 4A Reactor Coolant Pump Operations
- 4-NOP-041.01B, 4B Reactor Coolant Pump Operations
- 4-NOP-041.01C, 4C Reactor Coolant Pump Operations

Technical Reference(s): 4-ARP-097.CR.A, Control Room Response - Panel A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902113, Obj. 7 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	(2RI)
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A by testing use of procedures (4-ARP-097.CR.A) to mitigate the effects of a CVCS malfunction (high seal injection filter ΔP).

CAUSES:
1. Hi seal injection flow rate
2. Dirty in service seal water injection filter

A51

SEAL WATER INJ FILTER HI ΔP

DEVICE:	**SETPOINT:**	**LOCATION:**
PIC-4-157	20 psid	N/A

ALARM CONFIRMATION

1. **CHECK** the following locally:
   - DPI-4-154A for Seal Water Injection Filter 4A ΔP.
   - DPI-4-154B for Seal Water Injection Filter 4B ΔP.

OPERATOR ACTIONS

CAUTION

- RCPs are limited to a maximum of 24 hours operation without seal injection.
- To prevent RCP seal damage, do **NOT** bypass seal injection filters.

1. IF seal water injection flow is above 13 gpm, THEN:

CAUTION

Care must be exercised when throttling HCV-4-121 in the closed direction. Throttling this valve completely closed can cause the Charging Pump discharge relief valve to lift resulting in a possible loss of charging if the relief valve fails to reseat.

   A. **ADJUST** seal injection flow as necessary to establish 6 to 13 gpm per RCP using the applicable procedure(s):
      - 4-NOP-041.01A, 4A Reactor Coolant Pump Operations
      - 4-NOP-041.01B, 4B Reactor Coolant Pump Operations
      - 4-NOP-041.01C, 4C Reactor Coolant Pump Operations

   B. **REDUCE** total charging pump speed.

2. IF filter ΔP is high, THEN:

   A. IF standby filter is available, THEN:
      (1) **PLACE** standby filter in service using 4-OP-047, CVCS - Charging and Letdown.
      (2) **ADJUST** seal injection flow as necessary to establish 6 to 13 gpm per RCP using the applicable procedure(s):
         - 4-NOP-041.01A, 4A Reactor Coolant Pump Operations
         - 4-NOP-041.01B, 4B Reactor Coolant Pump Operations
         - 4-NOP-041.01C, 4C Reactor Coolant Pump Operations

QUESTION 50

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010	K6.03
	Importance Rating	3.2	

Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR sprays and heaters

Proposed Question: RO Question # 50

Given the following:

- Unit 3 is at 100% power.
- Both of the Pressurizer Backup Heater Groups were manually placed in the "ON" position one hour ago for RCS boron equalization.
- PCV-3-455A, Spray Control Valve, fails to 100% OPEN.

With NO operator actions, which ONE of the following proper SEQUENCE of events will occur?

A. PCV-3-455B Spray Valve closes, ALL Pressurizer Heaters energize, PORV Operation Permissive drops out, and Reactor trips

B. PCV-3-455B Spray Valve closes, PORV Operation Permissive drops out, ALL Pressurizer Heaters energize, and Reactor trips

C. PCV-3-455B Spray Valve position remains unchanged, ALL Pressurizer Heaters energize, Reactor trips, and PORV Operation Permissive drops out

D. PCV-3-455B Spray Valve position remains unchanged, PORV Operation Permissive drops out, Reactor trips, and ALL Pressurizer Heaters energize

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Both spray valves should initially be open. Due to the integral nature of the PZR pressure master controller, a short time after the backup heater group was

energized, the spray valves should have opened and will be the first pressure control component to respond (valve close) as PZR pressure decreases.

B. Incorrect since heaters energize before the PORV interlock is reached. Plausible because the setpoints for the two functions are close (heaters 2210, PORV Interlock 2000).

C. Incorrect since the spray valve 455B will close. Plausible because a novice applicant may overlook the fact that the spray valves modulated open after the backup heaters were energized. Given that, the applicant may assume that the spray valves were closed in the first place; therefore, would 455B will not be affected.

D. Incorrect since the spray valve 455B will close. Plausible because a novice applicant may overlook the fact that the spray valves modulated open after the backup heaters were energized. Given that, the applicant may assume that the spray valves were closed in the first place; therefore, would 455B will not be affected.

Technical Reference(s): LP 6902109A, Pressurizer Pressure Control (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902109A, Obj. 7 (As available)

Question Source: Bank # 66916

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010 Diablo Canyon

Question Cognitive Level: Memory or Fundamental Knowledge (1I)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

QUESTION 51

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012	K5.01
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB

Proposed Question: RO Question # 51

The ______(1)______ protects against Departure from Nucleate Boiling (DNB) and, the trip setpoint ______(2)______ with a decrease in Pressurizer pressure?

	(1)	(2)
A.	Overpower ΔT Trip	rises
B.	Overtemperature ΔT Trip	rises
C.	Overpower ΔT Trip	lowers
D.	Overtemperature ΔT Trip	lowers

Proposed Answer: D

Explanation (Optional):

A. Incorrect since OPDT protects from overpower in the fuel (kw/ft). Plausible because its calculation is similar to OTDT, with the exception of a pressure input.

B. Incorrect since this trip setpoint becomes lower as PZR pressure is reduced. Plausible because this trip *does* provide DNB protection as well as kw/ft.

C. Incorrect since this trip does not take pressure into consideration. Plausible because

this trip protects against power excursion.

D. CORRECT. TS Table 2.2.1, Reactor Trip System Instrumentation Trip Setpoints.

.

Technical Reference(s): TS Table 2.2.1, Reactor Trip System Instrumentation Trip Setpoints (Attach if not previously provided)

0-ADM-536, Technical Specification Bases Control Program

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902163, Obj. 7 (As available)

Question Source: Bank # 48221

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2004 Wolf Creek

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

2.2.1 (Cont'd)

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations for all power levels to mitigate the consequences of a reactivity excursion which may be too rapid for the temperature and pressure protective trips.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 105 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit is taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors and pressure is within the range between the Pressurizer High and Low Pressure trips. The setpoint is automatically varied with: (1) Coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) Pressurizer pressure, and (3) Axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \; \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \frac{1}{(1+\tau_6 S)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Instrumentation

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = Lead/Lag compensator on measured ΔT; $\tau_1 = 0s$, $\tau_2 = 0s$

$\frac{1}{1+\tau_3 S}$ = Lag compensator on measured ΔT; $\tau_3 = 0s$

ΔT_0 = Indicated ΔT at RATED THERMAL POWER

K_1 = 1.24;

K_2 = 0.017/°F;

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4 , τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 25s$, $\tau_5 = 3s$;

T = Average temperature, °F;

$\frac{1}{1+\tau_6 S}$ = Lag compensator on measured T_{avg}; $\tau_6 = 0s$

T′ ≤ 577.2 °F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.001/psig;

P = Pressurizer pressure, psig;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P' $\geq$ 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

And $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between – 50% and + 2%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds – 50%, the ΔT Trip Setpoint shall be automatically reduced by 0.0% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 2%, that ΔT Trip Setpoint shall be automatically reduced by 2.19% of its value at RATED THERMAL POWER.

NOTE 2: The channels maximum trip setpoint shall not exceed its computed setpoint by more than 0.84% of instrument span.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

2.2.1 (Cont'd)

Overpower ΔT

The Overpower ΔT trip prevents power density anywhere in the core from exceeding 118% of the design power density. This provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The setpoint is automatically varied with: (1) Coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) Rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine first stage pressure at approximately 10% of full power equivalent) and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer Water Level-High trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P–7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage pressure at approximately 10% of full power equivalent) and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 S}{1+\tau_7 S} \left(\frac{1}{1+\tau_6 S}\right) T - K_6 \left[T \frac{1}{1+\tau_6 S} - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT = As defined in Note 1,

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = As defined in Note 1,

$\frac{1}{1+\tau_3 S}$ = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 $\le$ 1.10,

K_5 $\ge$ 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 s}{1+\tau_7 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_7 \ge 10$ s,

$\frac{1}{1+\tau_6 S}$ = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.0016/°F for T > T”
	=	0 for T ≤ T”,
T	=	As defined in Note 1,
T”	≤	577.2°F (Nominal T_{avg} at RATED THERMAL POWER)
S	=	As defined in Note 1, and
f_2 (ΔI)	=	0 for all ΔI

NOTE 4: The channel’s maximum trip setpoint shall not exceed its computed trip setpoint by more than 0.96% of instrument span.

QUESTION 52

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	013	K2.01
Importance Rating	3.6	

Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

Proposed Question: RO Question # 52

In accordance with 3-ONOP-003.7, Loss of 120V Vital Instrument Panel 3P07, which ONE of the following describes the effect on the <u>sequencer output</u>, if any, when 3P07 is lost: (1) AFW Actuation and (2) Bus Stripping?

	AFW Actuation	**Bus Stripping**
A.	Lost	Lost
B.	NOT affected	Lost
C.	Lost	NOT affected
D.	NOT affected	NOT affected

Proposed Answer: A

which

Explanation (Optional):

A. CORRECT

3A bus sequencer is out of service, due to Vital Panel 3P07 deenergized, resulting in the following Tech Spec implications:

1) AFW actuation signals from bus stripping on 3A 4KV bus will NOT be generated, placing the unit in Tech Spec 3.0.3 (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 6d action 23 invokes Tech Spec 3.0.3.)
2) Loss of Power signals are lost via the 3A bus sequencer, placing the unit in Tech Spec 3.0.3 (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 7a, b, and c.)
3) Bus stripping will NOT automatically occur, 3A EDG will NOT automatically close in on the bus and is out of service (actions of Tech Spec 3.8.1.1 apply).

B. Incorrect since the AFW Actuation Signal will be Lost. Plausible because the 2nd part is correct.

C. Incorrect since the Bus Stripping Signal will be Lost. Plausible because the 1st part is correct.

D. Incorrect. Plausible since if 3P08 or 3P09 were lost, these would be true statements.

Technical Reference(s): 3-ONOP-003.7, Loss of 120V Vital Instrument Panel 3P07 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900139, Obj. 11 (As available)

Question Source: Bank #

Modified Bank # 59789 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Modified from Diablo Canyon 2008 exam

ENCLOSURE 1
(Page 1 of 3)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON FAILURE OF VITAL INSTRUMENT PANEL 3P07

FUNCTIONS, OPERATING

Loss of Auto Control of 3B Feedwater Control Valve, FCV-3-488
3B Charging Pump Controller Locks Up as is
Auto VCT makeup will occur due to LT-3-115 failure.
Failure Closed of Train 1 AFW Flow Control Valves: (CV-3-2816, 2817, 2818)
HCV-3-121 fails full open
Loss of Diesel 3A Load Sequencer, 3C23A-1 deenergized
QSPDS Channel A (If 3A inverter and CVT are lost)
Lose AMSAC Processor B
ANN B 9/2
ANN B 9/3
Loss of power to hand/auto station for CV-3-1606 which fails closed
Loss of primary power to Rod Deviation/Axial Flux Mon Rack, 3QR64. (Redundant power is supplied from Inverter 3Y111, Panel 3P31A, Ckt. 9.)

NOTES

3A bus sequencer is out of service, due to Vital Panel 3P07 deenergized, resulting in the following Tech Spec implications:

1) *AFW actuation signals from bus stripping on 3A 4KV bus will **NOT** be generated, placing the unit in Tech Spec 3.0.3 (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 6d action 23 invokes Tech Spec 3.0.3.)*

2) *Loss of Power signals are lost via the 3A bus sequencer, placing the unit in Tech Spec 3.0.3 (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 7a, b, and c.)*

3) *Bus stripping will **NOT** automatically occur, 3A EDG will **NOT** automatically close in on the bus and is out of service (actions of Tech Spec 3.8.1.1 apply).*

INDICATORS

LI-3-115	VCT Level
LI-3-106	A Boric Acid Tank Level
PI-3-155	B RCP Seal ΔP
FR-3-154A	B RCP Seal Leakoff Wide Range Recorder (Blue Pen)
PI-3-125A	C RCP Thermal Barrier ΔP
FR-3-154B	B RCP Seal Leakoff Narrow Range Recorder (Blue Pen)
TI-3-453	Pzr Liquid Temp
TI-3-450	Pzr Surge Line Temp
TI-3-454	Pzr Vapor Temp

QUESTION #52

Facility: Diablo Canyon

Vendor: WEC

Exam Date: 2008

Exam Type: R

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	2
Group #	1	1
K/A #	013	K2.01
Importance Rating	3.6	3.8

Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

Proposed Question:

Unit 1 is operating at steady-state, 100% power with all equipment operable and aligned for normal full power conditions.

What is the effect on the status and operation of engineered safeguards features (ESF) equipment if vital 120 VAC instrument bus 1-1 (PY-11) de-energizes?

A: Most SSPS channel 1 input bay relays for both trains will trip.

B: Most SSPS channel 1 input bay relays for train "A" only will trip.

C: None of the SSPS input bay relays on either train will trip.

D: Most SSPS channel 1 input bay relays for train "B" only will trip.

Proposed Answer: A

Explanation (Optional):

A: Answer A is correct. PY-11 maintains most of the input bay relays for both trains in the energized, non-trip condition.

B: Answer B is incorrect. Input relays on both trains will actuate and SSPS train A will lose slave relay power (the logic cards have redundant power).

C: Answer C is incorrect. Input relays on both trains will actuate, but the train A equipment will not auto start/actuate without slave relay power.

D: Answer D is incorrect. Input relays on both trains will actuate and SSPS train A will lose slave relay power (the logic cards have redundant power).

Technical Reference(s):	LB-6B, Eagle 21 & Solid State Protection System. Page 10. Rev. 10. OIM Page B-6-1b, SSPS Power Supplies, Rev. 19.	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective:	3291 - State the power supplies to Eagle 21 and Solid State Protection System components	(As available)

Question Source:	Bank #	X	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41

55.43

Comments:

QUESTION 53

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	026	A3.02
Importance Rating	3.9	

Ability to monitor automatic operation of the CSS, including: Verification that cooling water is supplied to the containment spray heat exchanger

Proposed Question: RO Question # 53

Given the following:

- A Large Break Loss of Coolant Accident (LBLOCA) has occurred on Unit 4 when operating at 100% power.
- Containment pressure has reached 40 psig.

Which ONE of the following conditions describes (1) the status of CCW FLOW through the Containment Spray Pump Seal Water Heat Exchanger immediately after a Containment Spray Actuation signal and (2) the TEMPERATURE change of the Containment Spray Pump Seal Water Heat Exchanger CCW Outlet after the Containment Spray Pump starts?

(Assume constant Containment Spray Pump suction temperature.)

	(1) Flow	**(2) Temperature**
A.	Higher	Lower
B.	Higher	Higher
C.	Lower	Lower
D.	Lower	Higher

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because flow will be higher. Also, the candidate may assume a cooling effect since flowrate will be higher.

B. Correct. CCW flow will be increased to the CSP Seal Water Heat Exchanger when Containment Phase B occurs due to the isolation of CCW to RCPs. Also, with the pump running, the seals will heat up and cause temperature to rise as more heat is transferred to CCW.

C. Incorrect. Plausible because candidate assumes flow will be lower due to an increased demand of CCW loads. Also, the candidate may assume less heat is removed with a lower flowrate.

D. Incorrect. Plausible because candidate may assume flow will be lower due to an increased demand of CCW loads. The candidate correctly assumes with the pump running, the seals will heat up and cause temperature to rise as more heat is transferred to CCW.

Technical Reference(s):

LP 6902140, Component Cooling Water System

LP 6902129, Containment Vent & Heat Removal

4-EOP-E-0, Reactor Trip or Safety Injection

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902140, Obj. 8
LP 6902129, Obj. 7 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New WTSI 66667

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	(2DR)
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 3
(Page 2 of 7)

PROMPT ACTION VERIFICATIONS

4.	**Verify Proper ICW System Operation**	
	a. Verify ICW pumps - AT LEAST TWO RUNNING	a. Start ICW pump(s) to establish at least two running.
	b. Verify ICW to TPCW Heat Exchanger – ISOLATED • POV-3-4882 – CLOSED • POV-3-4883 – CLOSED	b. Manually close valve(s). **IF** valve(s) can **NOT** be closed, **THEN** locally close the following valves: • 3-50-319 for POV-3-4882 • 3-50-339 for POV-3-4883
	c. Check ICW headers - TIED TOGETHER	c. **IF** both ICW headers are intact, **THEN** direct operator to tie headers together.
5.	**Verify Proper CCW System Operation**	
	a. CCW Heat Exchangers – THREE IN SERVICE	a. Perform the following: 1) Start or stop CCW pumps as necessary to establish ONLY ONE RUNNING CCW PUMP. 2) Verify Emergency Containment Coolers - ONLY TWO RUNNING 3) Go to Step 5c.
	b. CCW pumps - ONLY TWO RUNNING	b. Start or stop CCW pumps as necessary to establish ONLY TWO RUNNING CCW PUMPS.
	c. CCW headers - TIED TOGETHER	c. **IF** both CCW headers are intact, **THEN** direct a field operator to tie the headers together.
	d. RCP Thermal Barrier CCW Outlet, MOV-3-626 – OPEN	d. **IF** containment isolation phase B **NOT** actuated **AND** CCW radiation levels are normal, **AND** RCP number one seal leak-off temperature is less than 235°F, **THEN** manually open MOV-3-626. **IF** MOV-3-626 can **NOT** be manually opened, **THEN** direct operator to open MOV-3-626 locally.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 3
(Page 3 of 7)

PROMPT ACTION VERIFICATIONS

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**Verify Containment Cooling**	
	a. Check emergency containment coolers - ONLY TWO RUNNING	a. Manually start or stop emergency containment coolers to establish - ONLY TWO RUNNING.
	b. Verify emergency containment filter fans - AT LEAST TWO RUNNING	b. Manually start emergency containment filter fans.
7.	**Verify Pump Operation**	
	a. At least two high head SI pumps running	a. Manually start high-head pump(s).
	b. Both RHR pumps running	b. Manually start RHR pump(s).
8.	**Verify SI Flow**	
	a. RCS pressure - LESS THAN 1600 PSIG [2000 PSIG]	a. Go to Step 9.
	b. High-head SI pump flow indicator – CHECK FOR FLOW	b. Manually start pumps **AND** align valves to establish an injection flowpath.
	c. RCS pressure - LESS THAN 250 PSIG [650 PSIG]	c. Go to Step 9.
	d. RHR pump flow indicator - CHECK FOR FLOW	d. Manually start pumps **AND** align valves to establish an injection flowpath.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 3
(Page 4 of 7)

PROMPT ACTION VERIFICATIONS

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	**Realign SI System**	
	a. Verify Unit 3 high-head SI pumps - TWO RUNNING	a. Perform the following: 1) Operate Unit 3 and Unit 4 high-head SI pumps to establish injection to Unit 3 from two high-head SI pumps. 2) Direct Unit 4 Reactor Operator to align Unit 4 high-head SI pump suction to Unit 3 RWST using ATTACHMENT 1 of this procedure. 3) Go to Step 10.
	b. Stop both Unit 4 high-head SI pumps **AND** place in standby	
10.	**Verify Containment Isolation Phase A Valve White Lights On VPB – ALL BRIGHT**	Perform the following: a. Manually actuate Containment Isolation Phase A. b. **IF** any Containment Isolation Phase A valve is **NOT** closed, **THEN** manually close valve. **IF** valve(s) can **NOT** be manually closed, **THEN** manually or locally isolate affected containment penetration.
11.	**Verify SI Valve Amber Lights On VPB - ALL BRIGHT**	Manually align valves to establish proper SI alignment for an injection flowpath.
12.	**Verify SI – RESET**	Reset SI
13.	**Verify Containment Phase A – RESET**	Reset Phase A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 3
(Page 5 of 7)

PROMPT ACTION VERIFICATIONS

14. Reestablish RCP Cooling

a. Check RCPs – AT LEAST ONE RUNNING
   a. Go to step 15.

b. Open CCW to normal containment cooler valves
   - MOV-3-1417
   - MOV-3-1418   CLOSED phase Ø

   b. Stop all RCPs

c. Reset and start normal containment coolers
   c. Stop all RCPs

15. Monitor Containment Pressure To Verify Containment Spray NOT Required

a. Containment pressure - HAS REMAINED LESS THAN 20 PSIG
   - PR-3-6306A

   AND

   - PR-3-6306B

   a. Perform the following:

   1) **IF** containment spray **NOT** initiated, **THEN** manually initiate containment spray.

   2) Verify Containment Isolation Phase B - ACTUATED.

   3) Verify Containment Isolation Phase B valve white lights on VPB – ALL BRIGHT.

   4) **IF** any Containment Isolation Phase B valve did **NOT** close, **THEN** manually or locally isolate affected containment penetration.

   5) Stop all RCPs.

QUESTION 54

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062	2.2.37
	Importance Rating	3.6	

Equipment Control: Ability to determine operability and / or availability of safety related equipment.

Proposed Question: RO Question # 54

Given the following condition:

- Unit 3 is in MODE 1
- A failure of 3Y05, 3C INVERTER, which was powering 3P06 has occurred.

Which ONE of the choices below completes the following statement?

3P06 may be supplied from the ____(1)_______ as an operable source of power to vital instrumentation. In accordance with Technical Specifications, this lineup is allowed ______(2)________ .

REFERENCE PROVIDED

A. (1) 3Y06, CS Spare Inverter;
(2) for ONLY 2 hrs

B. (1) 3Y06, CS Spare Inverter;
(2) Indefinitely

C. (1) Constant Voltage Transformer (CVT);
(2) Indefinitely

D. (1) Constant Voltage Transformer (CVT);
(2) for ONLY 2 hrs

Proposed Answer: B

Explanation (Optional):

A. Incorrect. 3Y06, CS Spare Inverter is an operable 3P06 power source for vital instrumentation. There is plausibility for the 2 hour time frame which is the action time to restore 3P06 with a loss of power.

B. CORRECT.

C. Incorrect. The Constant Voltage Transformer (CVT) is an operable 3P06 power source for vital instrumentation with limitations. There is plausibility for indefinitely which applies to 3Y06, CS Spare Inverter.

D. Incorrect. The Constant Voltage Transformer (CVT) is an operable 3P06 power source for vital instrumentation. There is plausibility for the 2 hour time frame which is the action time to restore 3P06 with a loss of power. The time limit for the CVT is 24 hours.

Technical Reference(s): 3-ONOP-003.6, *Loss of 120V Vital Instrument Panel 3P06*
3-OP-003, *120V Vital Instrument AC System*
TS 3.8.3.1
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.8.3.1

Learning Objective: LP 6900139, Obj. 11, 12 (As available)

Question Source: Bank # WTSI 66040
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Q#54

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses* shall be energized in the specified manner with the tie breakers open between redundant busses within the unit** and between the busses of Units 3 and 4.

a. One train of A.C. Busses consisting of:

1) 4160-Volt Bus A,
2) 480-Volt Load Center Busses A, C and H***, and
3) 480-Volt Motor Control Center Busses A (Unit 4 only),C and D***,

b. One train of A.C. Busses consisting of:

1) 4160-Volt Bus B
2) 480-Volt Load Center Busses B, D and H***, and
3) 480-Volt Motor Control Center Busses B and D***

c. One opposite unit train of AC busses consisting of either:

1) 4160-Volt Bus A, 480-Volt Load Center Busses A, C and H***, and 480-Volt Motor Control Center Busses A (Unit 4 only), C and D***, or
2) 4160-Volt Bus B, 480-Volt Load Center Busses B, D and H***, and 480-Volt Motor Control Center Busses B and D***.

d. 120 Volt AC Vital Panel 3P06 and 3P21 energized from its associated inverter connected to D.C. Bus 3B.****

e. 120 Volt AC Vital Panel 4P06 and 4P21 energized from its associated inverter connected to D.C. Bus 3B.****

f. 120 Volt AC Vital Panel 3P07 and 3P22 energized from its associated inverter connected to D.C. Bus 3A.****

g. 120 Volt AC Vital Panel 4P07 and 4P22 energized from its associated inverter connected to D.C. Bus 3A.****

h. 120 Volt AC Vital Panel 3P08 and 3P23 energized from its associated inverter connected to D.C. Bus 4B.****

i. 120 Volt AC Vital Panel 4P08 and 4P23 energized from its associated inverter connected to D.C. Bus 4B.****

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* For Motor Control Center busses, vital sections only.

** With the opposite unit in MODE 5 or 6, its 480-Volt Load Center can be cross-tied under conditions specified in Specification 3.8.3.2.a.

*** Electrical bus can be energized from either train of its unit and swing function to opposite train must be OPERABLE for the Unit(s) in MODES 1, 2, 3, and 4.

****A back-up inverter may be used to replace the normal inverter provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

LIMITING CONDITION FOR OPERATION (Continued)

j. 120 Volt AC Vital Panel 3P09 and 3P24 energized from its associated inverter connected to D.C. Bus 4A.****

k. 120 Volt AC Vital Panel 4P09 and 4P24 energized from its associated inverter connected to D.C. Bus 4A.****

l. 125 Volt D.C. Bus 3D01 energized from an associated battery charger and from Battery Bank 3A or spare battery bank D-52,

m. 125 Volt D.C. Bus 3D23 energized from an associated battery charger and from Battery Bank 3B or spare battery bank D-52,

n. 125 Volt D.C. Bus 4D01 energized from an associated battery charger and from Battery Bank 4B or spare battery bank D-52, and

o. 125 Volt D.C. Bus 4D23 energized from an associated battery charger and from Battery Bank 4A or spare battery bank D-52

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required trains (3.8.3.1a., b., and c) of A.C. emergency busses not fully energized (except for the required LC's and MCC's associated with the opposite unit), reenergize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any of the required LC's and/or MCC's associated with the opposite unit inoperable, restore the inoperable LC or MCC to OPERABLE status in accordance with Table 3.8-1 or Table 3.8-2 as applicable or place the unit in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) Reenergize the A.C. vital panel within 2 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panel from an inverter connected to its associated D.C. bus

****A back-up inverter may be used to replace the normal inverter, provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

within 24 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.

d. With one D.C. bus not energized from its associated battery bank or associated charger, reenergize the D.C. bus from its associated battery bank within 2 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

4.8.3.1 The specified busses shall be determined energized and aligned in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

* 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).

TABLE 3.8-1

APPLICABLE TO UNIT 3 BASED ON UNIT 4 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 4 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 3 – MODES 1, 2, 3 and 4		
	With AC Trains 3A, 3B, 4A, & 4B OPERABLE	With AC Trains 3A, 3B, & 4A OPERABLE	With AC Trains 3A, 3B, & 4B OPERABLE
LC 4A	N/A	72	N/A
MCC 4A	N/A	N/A	N/A
LC 4C and/or MCC 4C	2*	2*	N/A
LC 4H and/or MCC 4D	2**	2**	2**
LC 4B and/or MCC 4B	2*	N/A	2*
LC 4D	N/A	N/A	72

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours.

TABLE 3.8-2

APPLICABLE TO UNIT 4 BASED ON UNIT 3 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 3 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 4 – MODES 1, 2, 3 and 4		
	With AC Trains 4A, 4B, 3A, & 3B OPERABLE	With AC Trains 4A, 4B, & 3A OPERABLE	With AC Trains 4A, 4B, & 3B OPERABLE
LC 3A	N/A	72	N/A
LC 3C and/or MCC 3C	2*	2*	N/A
LC 3H and/or MCC 3D	2**	2**	2**
LC 3B and/or MCC 3B	2*	N/A	2*
LC 3D	N/A	N/A	72

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours.

Q #54

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses* shall be energized in the specified manner with the tie breakers open between redundant busses within the unit** and between the busses of Units 3 and 4.

a. One train of A.C. Busses consisting of:

1) 4160-Volt Bus A,
2) 480-Volt Load Center Busses A, C and H***, and
3) 480-Volt Motor Control Center Busses A (Unit 4 only),C and D***,

b. One train of A.C. Busses consisting of:

1) 4160-Volt Bus B
2) 480-Volt Load Center Busses B, D and H***, and
3) 480-Volt Motor Control Center Busses B and D***

c. One opposite unit train of AC busses consisting of either:

1) 4160-Volt Bus A, 480-Volt Load Center Busses A, C and H***, and 480-Volt Motor Control Center Busses A (Unit 4 only), C and D***, or
2) 4160-Volt Bus B, 480-Volt Load Center Busses B, D and H***, and 480-Volt Motor Control Center Busses B and D***.

d. 120 Volt AC Vital Panel 3P06 and 3P21 energized from its associated inverter connected to D.C. Bus 3B.****

e. 120 Volt AC Vital Panel 4P06 and 4P21 energized from its associated inverter connected to D.C. Bus 3B.****

f. 120 Volt AC Vital Panel 3P07 and 3P22 energized from its associated inverter connected to D.C. Bus 3A.****

g. 120 Volt AC Vital Panel 4P07 and 4P22 energized from its associated inverter connected to D.C. Bus 3A.****

h. 120 Volt AC Vital Panel 3P08 and 3P23 energized from its associated inverter connected to D.C. Bus 4B.****

i. 120 Volt AC Vital Panel 4P08 and 4P23 energized from its associated inverter connected to D.C. Bus 4B.****

* For Motor Control Center busses, vital sections only.

** With the opposite unit in MODE 5 or 6, its 480-Volt Load Center can be cross-tied under conditions specified in Specification 3.8.3.2.a.

*** Electrical bus can be energized from either train of its unit and swing function to opposite train must be OPERABLE for the Unit(s) in MODES 1, 2, 3, and 4.

****A back-up inverter may be used to replace the normal inverter provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

LIMITING CONDITION FOR OPERATION (Continued)

j. 120 Volt AC Vital Panel 3P09 and 3P24 energized from its associated inverter connected to D.C. Bus 4A.****

k. 120 Volt AC Vital Panel 4P09 and 4P24 energized from its associated inverter connected to D.C. Bus 4A.****

l. 125 Volt D.C. Bus 3D01 energized from an associated battery charger and from Battery Bank 3A or spare battery bank D-52,

m. 125 Volt D.C. Bus 3D23 energized from an associated battery charger and from Battery Bank 3B or spare battery bank D-52,

n. 125 Volt D.C. Bus 4D01 energized from an associated battery charger and from Battery Bank 4B or spare battery bank D-52, and

o. 125 Volt D.C. Bus 4D23 energized from an associated battery charger and from Battery Bank 4A or spare battery bank D-52

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required trains (3.8.3.1a., b., and c) of A.C. emergency busses not fully energized (except for the required LC's and MCC's associated with the opposite unit), reenergize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any of the required LC's and/or MCC's associated with the opposite unit inoperable, restore the inoperable LC or MCC to OPERABLE status in accordance with Table 3.8-1 or Table 3.8-2 as applicable or place the unit in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) Reenergize the A.C. vital panel within 2 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panel from an inverter connected to its associated D.C. bus

DISTRACTOR C & D

****A back-up inverter may be used to replace the normal inverter, provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

within 24 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.

d. With one D.C. bus not energized from its associated battery bank or associated charger, reenergize the D.C. bus from its associated battery bank within 2 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

4.8.3.1 The specified busses shall be determined energized and aligned in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

* 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).

TABLE 3.8-1

APPLICABLE TO UNIT 3 BASED ON UNIT 4 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 4 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 3 – MODES 1, 2, 3 and 4		
	With AC Trains 3A, 3B, 4A, & 4B OPERABLE	With AC Trains 3A, 3B, & 4A OPERABLE	With AC Trains 3A, 3B, & 4B OPERABLE
LC 4A	N/A	72	N/A
MCC 4A	N/A	N/A	N/A
LC 4C and/or MCC 4C	2*	2*	N/A
LC 4H and/or MCC 4D	2**	2**	2**
LC 4B and/or MCC 4B	2*	N/A	2*
LC 4D	N/A	N/A	72

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours.

TABLE 3.8-2

APPLICABLE TO UNIT 4 BASED ON UNIT 3 LOAD
CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 3 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 4 – MODES 1, 2, 3 and 4		
	With AC Trains 4A, 4B, 3A, & 3B OPERABLE	With AC Trains 4A, 4B, & 3A OPERABLE	With AC Trains 4A, 4B, & 3B OPERABLE
LC 3A	N/A	72	N/A
LC 3C and/or MCC 3C	2*	2*	N/A
LC 3H and/or MCC 3D	2**	2**	2**
LC 3B and/or MCC 3B	2*	N/A	2*
LC 3D	N/A	N/A	72

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours.

4.12 During energization of large loads or QSPDS circuits, an automatic transfer to CVT may occur due to static switch sensitivity and characteristics. If the inverter voltage, frequency, and current are in the normal ranges following this transfer, the inverter output shall be transferred back to the normal supply. Any parameter outside of the normal range should be reported to the Shift Manager immediately.

4.13 Technical Specification Subsection 3.8.3.1 provides the minimum margin of safety for the Vital AC Distribution System. When operating on a CVT, the associated vital inverter or the spare inverter shall be placed in service within 24 hours.

4.14 After an inverter failure has occurred which requires the use of a CVT, all construction efforts in the area must be halted until an inverter is restored to service. This is to preclude construction activities from disabling a redundant inverter.

4.15 Unit startup shall not commence if a CVT is in service powering an instrument bus.

4.16 The Alternate Source Transfer Switches are under administrative control and shall be locked at all times except during switching.

4.17 The internal Sync Reference Selector Switch (SW-2) for the 4 spare inverters shall be in the EXTERNAL (UP) position except when the spare inverter is placed in service and the applicable CVT has been transferred to the spare, at which time the switch is placed in the NORMAL (DOWN) position.

4.18 When a spare inverter is in service to power a bus, the Alternate Source Transfer Switch will be locked in the Backup to Spare Inverter position.

4.19 The Manual Bypass Switch is a make-before-break type. Caution should be exercised while making a load transfer with this switch. CVT and Inverter must be verified in sync prior to the load transfer.

4.20 Both channels of ICCS (QSPDS) are required to be operable in accordance with Technical Specification Section 3/4.3.3.5. If required to remove one channel of ICCS (QSPDS) from service, the other channel shall be verified to be operable prior to removing a channel from service.

4.21 Alternate Shutdown System 120V Vital Instrument AC Panel, 4P93 shall be energized from Inverter 4B (4Y02), Normal Supply or from Inverter BS (4Y04), Alternate Supply (Breaker CB7).

4.22 When the test pushbutton is depressed on Panel 4P93A or 3P93A, the power supply will transfer to the standby source. If the normal source is available, the power supply will transfer back automatically in 25 seconds.

QUESTION 55

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063	K3.02
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the dc electrical system will have on the following: Components using dc control power

Proposed Question: RO Question # 55

Given the following:

- Unit 4 is operating at 40% power.
- 125 VDC control power is lost to ONLY 4A RCP Breaker (4AA01).
- Normal/Isolate Switch for 4A RCP Breaker (4AA01) is in NORMAL.

Which ONE of the following choices identifies (1) the status of the local breaker indicating lights for 4A RCP Breaker (4AA01) and (2) the effect on the 4A RCP Breaker (4AA01) operation?

A. (1) Local indicating lights are NOT LIT for 4A RCP Breaker (4AA01).
(2) The 4A RCP Breaker (4AA01) will only open by depressing the Manual Trip Latch on the local breaker.

B. (1) Local indicating lights are NOT LIT for 4A RCP Breaker (4AA01).
(2) The 4A RCP Breaker (4AA01) can be opened locally by Test Switch for the breaker.

C. (1) Local indicating lights are LIT for 4A RCP Breaker (4AA01).
(2) The 4A RCP Breaker (4AA01) will only open by depressing the Manual Trip Latch on the local breaker.

D. (1) Local indicating lights are LIT for 4A RCP Breaker (4AA01).
(2) The 4A RCP Breaker (4AA01) can be opened locally by Test Switch for the breaker.

Proposed Answer: A

Explanation (Optional):

A. CORRECT.

B. Incorrect. Plausibility – The candidate understands control power is required for automatic operation of the breaker. They assume power for the local Test Switch to trip the breaker is not lost and that indicating lights have a different power supply. If a fuse originally blew, the Normal/Isolate Handswitch taken to the isolate position puts another set of fuses in the circuit and allows local Handswitch operation.

C. Incorrect. Plausibility – The candidate recalls local light indication exists and believes this power is derived from the 4A KV Bus Control Power. The candidate correctly understands the breaker can be operated locally.

D. Incorrect. The candidate recalls local light indication and connects the logic that this power is derived from the 4A KV Bus Control Power. They assume power for the local Test Switch to trip the breaker is not lost and that indicating lights have a different power supply.

Technical Reference(s): 5614-E-25, Sheet 1A
0-NOP-003.01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Loss DC Control Power discussed in EPU SD-140, Main Power Distribution. No LP available (As available)

Question Source: Bank # WTSI 66667
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation,

signals, interlocks, failure modes, and automatic and manual features.

Comments:

QUESTION 56

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	002	K5.11
Importance Rating	4.0	

Knowledge of the operational implications of the following concepts as they apply to the RCS: Relationship between effects of the primary coolant system and the secondary coolant system

Proposed Question: RO Question # 56

In accordance with 0-ADM-200, Conduct of Operations, which ONE of the following choices below completes the statements?

Unit 3 is at 100% power when 3B S/G Steam Dump To Atmosphere Valve, CV-3-1607, fails open. The RCS Loop ΔT(s) will __(1)__. The operator action to maintain 100% power is to __(2)__.

(Assume constant Turbine load with time for steam flow to equalize.)

A. (1) rise
(2) insert Control Rods

B. (1) lower
(2) insert Control Rods

C. (1) rise
(2) reduce Turbine load

D. (1) lower
(2) reduce Turbine load

Proposed Answer: C

Explanation (Optional):

A. Incorrect since in accordance with 0-ADM-200, secondary power excursions are mitigated by Turbine load reduction. Plausible because the 1st part is correct. Also plausible because inserting rods would initially reduce reactor power.

B. Incorrect since in accordance with 0-ADM-200, secondary power excursions are

mitigated by Turbine load reduction. Plausible because inserting rods would initially reduce reactor power. The applicant believes there exists an overcooling effect on the S/Gs which will drive the RCS Loop ΔT mismatch lower.

C. CORRECT. 0-ADM-200, Conduct of Operations, Enclosure 4

D. Incorrect since the applicant believes with one S/G Steam Dump To Atmosphere Valve opening there exists an overcooling effect on the S/Gs which will drive the RCS Loop ΔT mismatch lower. Plausible since in accordance with 0-ADM-200, secondary power excursions are mitigated by Turbine load reduction.

Technical Reference(s): 0-ADM-200, Conduct of Operations (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902337, Obj. 2, 4 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

Question Selection Methodology:

4.4.3 Reactivity Control (continued)

6. (continued)

C. IF the DCS hourly heat rate report exceeds 100%, THEN perform the following:

(1) Ensure the 8-hour average Reactor Power will remain below 2299.9 MWth.

(2) Notify the AOM.

(3) Generate a Condition Report to document the event.

D. If a planned evolution (blowdown flow change, AFW pump run, etc.) is expected to cause a transient increase in reactor power that could exceed the licensed power limit (100%), then action should be taken to reduce power prior to the evolution. Additional guidance is provided in 3/4-GOP-301.

7. Increasing Reactor Power which would cause the DCS Hourly Heat Rate Report to exceed 100.00% or instantaneous power level to exceed 102%, caused by plant secondary transients (i.e., CV-2011 opening, large steam leak, turbine control problem, AFW actuation, etc.) shall be turned and reduced below 100 percent by a reduction in steam demand/turbine load.

8. Increasing Reactor Power which would cause the DCS Hourly Heat Rate Report to exceed 100.00% or instantaneous power level to exceed 102% caused by a reduction in boron concentration shall be turned and reduced below 100 percent by control rod insertion.

9. All calculations including CVCS blend calculations should be independently verified. Calculations may be performed using electronic spreadsheets but should be verified by an alternate means.

10. For procedures noted as **Reactivity Management Procedure** on the procedure cover sheet:

A. The US shall determine if the procedure section being performed has the potential to impact reactivity.

B. If it is determined to have a potential reactivity impact, then a pre-job brief shall be conducted with the Shift Manager present, or with SRO designee during Shift Manager absence from Control Room.

QUESTION 57

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	014	A4.01
	Importance Rating	3.3	

Ability to manually operate and/or monitor in the control room: Rod selection control

Proposed Question: RO Question # 57

Given the following:

- The plant is operating at 88% power.
- Rod Control is in MANUAL.
- Control Bank D rods are at 200 steps.
- All Tavg channels are approximately 4.0°F higher than Tref.

Which ONE of the following describes the effect of placing the Rod Control Bank Selector Switch in AUTO prior to matching Tave and Tref?

A. Rods will step in at 68 SPM initially and will stop when Tavg and Tref are within 1.0°F.

B. Rods will step in at 68 SPM initially and will stop when Tavg and Tref are matched.

C. Rods will step in at 40 SPM initially and will stop when Tavg and Tref are within 1.0°F.

D. Rods will step in at 40 SPM initially and will stop when Tavg and Tref are matched.

Proposed Answer: C

Explanation (Optional):

A. Incorrect since, with a 4.0 degree mismatch, rod control will be at 40 SPM. Plausible because 68 SPM is the rod speed when rods are operated in the manual mode.

B. Incorrect because rods will move in at 40 SPM. Plausible because a novice may assume rod control is designed to maintain Tave and Tref matched.

C. CORRECT. Since Tavg is greater than Tref the Reactor Control Unit will call for inward rod motion. Speed is 8 SPM at 3°F mismatch linearly up to 72 SPM at 5°F mismatch. (72-8 = 64 SPM; 5-3=2°F) = 32 SPM/°F. At a 4.0°F mismatch, (4 - 3)°F x 32 SPM/°F = 32 SPM. Add + 8 SPM (initial speed at 3°F mismatch) to the 32 SPM for a total speed of 40 SPM.

D. Incorrect. Correct speed. Plausible because a novice may assume rod speed lowers until Tave and Tref are matched.

Technical Reference(s): LP, 6902105, Full Length Rod Control System

SD-005, Full Length Rod Control System (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902105, Obj. 9 (As available)

Question Source: Bank # 18786

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 6

55.43

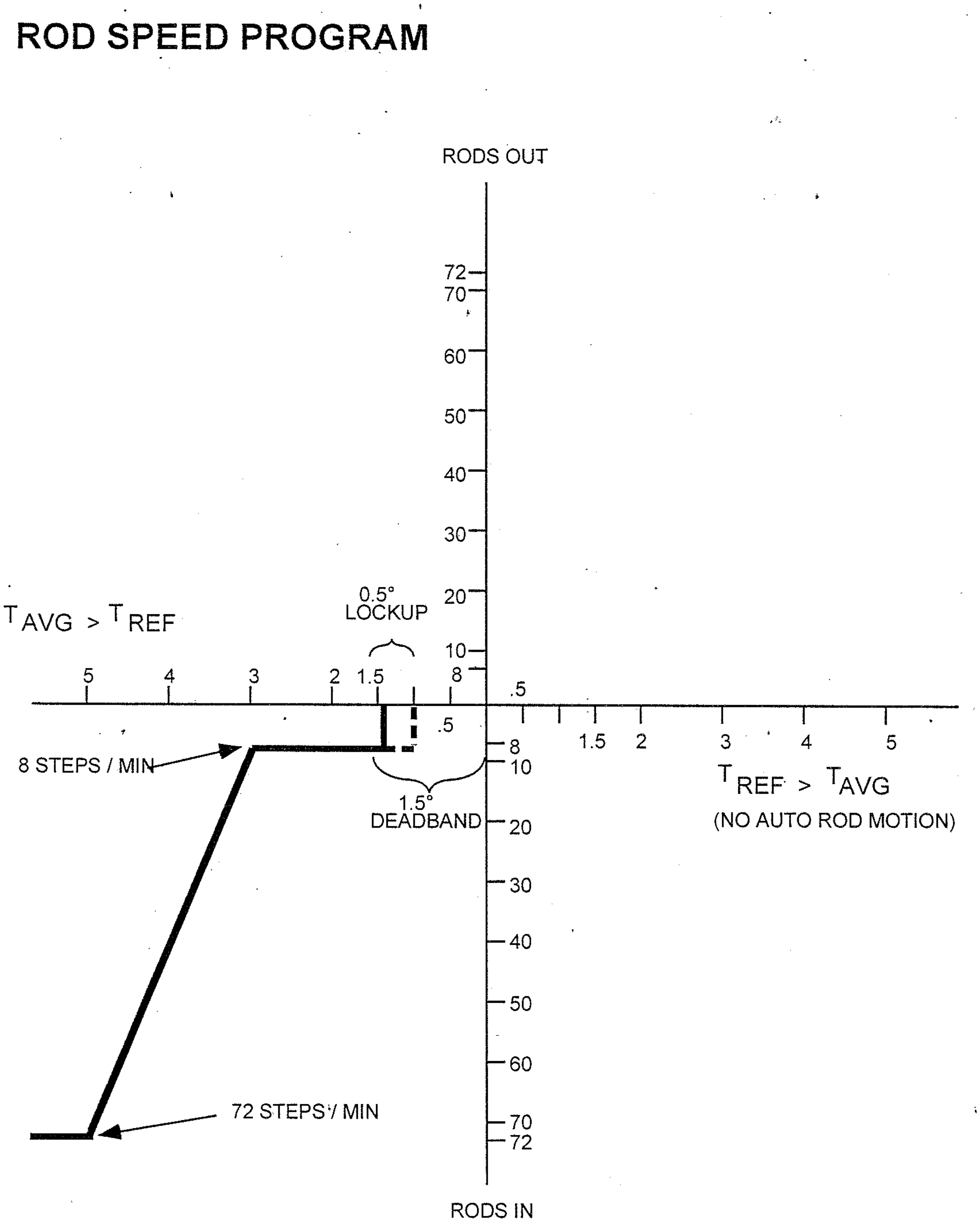
Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

#57

Figures for Rod Control

Figure 14



(SPEED VS COMPENSATED TEMPERATURE ERROR)

SD 005 FIGURE 14

QUESTION 58

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	017	K4.01
	Importance Rating	3.4	

Knowledge of ITM system design feature(s) and/or interlock(s) which provide for the following: Input to subcooling monitors

Proposed Question: RO Question # 58

The Subcooling Margin Monitor uses ___(1)___ to calculate subcooling. After a Reactor Trip, RCS Subcooling is ___(2)___ Core Exit Thermocouple Subcooling.

A. (1) Wide Range RCS Pressure
(2) less than

B. (1) Narrow Range Pressurizer Pressure
(2) less than

C. (1) Wide Range RCS Pressure
(2) greater than

D. (1) Narrow Range Pressurizer Pressure
(2) greater than

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausibility – The candidate understands correctly that Wide Range RCS Pressure is used. Also, they believe the mixing from the RCPs provide for a better measure of Subcooling there by using RCS Subcooling.

B. Incorrect. Plausibility – The candidate understands Narrow Range Pressurizer Pressure is more accurate. Also, they believe the mixing from the RCPs provide for a better measure of Subcooling there by using RCS Subcooling.

C. CORRECT.

D. Incorrect. Plausibility – The candidate understands Narrow Range Pressurizer Pressure is more accurate. Also, they correctly understand the hottest part of the RCS will be the most limiting on Subcooling.

LP 6900171, *QSPDS*

Technical Reference(s): LP 6902103, *Incore Instrumentation System* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900171, Obj. 4 (As available)

Question Source: Bank # 65245

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 TMI

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS
1. Containment Pressure (Wide Range)	2	1	1, 2, 3	31, 32
2. Containment Pressure (Narrow Range)	2	1	1, 2, 3	36
3. Reactor Coolant Outlet Temperature T_{HOT} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
4. Reactor Coolant Inlet Temperature T_{COLD} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
5. Reactor Coolant Pressure – Wide Range	2	1	1, 2, 3	31, 32
6. Pressurizer Water Level	2	1	1, 2, 3	31, 32
7. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator	1, 2, 3	31, 32
8. Reactor Coolant System Subcooling Margin Monitor	2(2)	1(2)	1, 2, 3	31, 32
9. PORV Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	33
10. PORV Block Valve Position Indicator	1/valve	1/valve	1, 2, 3	33
11. Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	32
12. Containment Water Level (Narrow Range)	2	1	1, 2, 3	36
13. Containment Water Level (Wide Range)	2	1	1, 2, 3	31, 32

ERDADS/QSPDS
(PART II – QSPDS)

INTRODUCTION

The Qualified Safety Parameter Display System (QSPDS) is a two channel, seismically and environmentally qualified Class IE system that is used to determine if inadequate core cooling conditions exist or are likely. It is also referred to as the Inadequate Core Cooling System (ICCS).

The following parameters are displayed on each of the two independent channels.

- Water level in the reactor vessel above the upper core plate
- Core Exit Thermocouple (CET) temperatures
- Hot and cold leg temperatures
- RCS pressure
- Subcooling/Saturation margins (calculated from measured parameters above)

QSPDS is a post TMI system supplied by Combustion Engineering. Each of the channels consist of sensors in the reactor vessel, a computer (unrelated to ERDADS) in the computer room, and a display and keypad in the control room. Each channel is supplied with power directly from a 120 volt vital A.C. inverter.

FUNCTION

The primary purpose for QSPDS is to aid operators in determining if inadequate core cooling conditions exist. It was installed as a result of the lessons learned at TMI.

DESIGN BASIS

QSPDS is a Class IE safety grade system. It is both environmentally (EQ) and seismically qualified as it is required to function under post accident conditions. There are two independent and redundant channels with independent 120 volt vital A.C. power supplies.

DETAILED DESCRIPTION

Each channel uses sensors to monitor the required temperatures, pressures, and water levels. Each channel also includes a central processing unit (in the computer room) and an operator interface (in the control room).

SENSORS

Various sensors are used to provide the data required. Sensors include:

- Thermocouples for monitoring core exit temperatures (CET's)
- RCS pressure transmitters
- RTD's for RCS T_{hot} and T_{cold} temperatures
- Heated Junction Thermocouples for monitoring water level above the upper core plate

Core Exit Thermocouples (CET)

51 thermocouples are arranged over the mixing columns on the upper core support plate. They are used for direct temperature indication of core exit temperatures and for calculating subcooling. SD-003/(Sys. 059B), Incore Instrumentation, provides a detailed description of these thermocouples.

Heated Junction Thermocouples (HJTC)

A vertical column of heated junction thermocouples is used by each QSPDS channel to determine the reactor vessel water level above the upper core plate (Fig. 17 and 18). Each HJT consists of a pair of thermocouples in close proximity. One of the two thermocouples is heated by an electric heater. If the pair of thermocouples is immersed in water there is approximately a 90°F temperature difference between them. The temperature difference will be about 200°F if they are uncovered due to the lower thermal conductivity of steam or non-condensable gasses.

Each column contains eight HJTC's; two in the reactor vessel head and six in the outlet plenum above the upper core plate. This allows water level to be determined in eight discreet steps. If RCP's

CETS hottest temp in core – thus lowest subcooling.
Thus RCS subcooling is greater than CET subcooling.

QUESTION 59

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	029	K3.02
Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on the following: Containment entry

Proposed Question: RO Question # 59

Given the following:

- Unit 3 is at 100% power.
- The Shift Manager determines a Containment Purge must be started and run during a Containment entry.
- A Unit 3 Containment Purge is ongoing in accordance with 3-NOP-053, Containment Purge System with the following fans running:
  - 3V9, U-3 Cntmt Purge Supply Fan
  - 4V20, U-4 Cntmt Purge Exhaust Fan
- 5 minutes after Containment is entered, R-3-12, Gaseous Containment Radiation Monitor, fails high.

What effect will the R-3-12, Gaseous Containment Radiation Monitor, failure have on Containment entry and the fans purging Containment?

(Assume no operator actions.)

A. Containment entry is suspended due to loss of Unit 3 Containment Purge with:
- 3V9, U-3 Cntmt Purge Supply Fan tripped
- 4V20, U-4 Cntmt Purge Exhaust Fan tripped

B. Containment entry is suspended due to loss of Unit 3 Containment Purge with:
- 3V9, U-3 Cntmt Purge Supply Fan tripped
- 4V20, U-4 Cntmt Purge Exhaust Fan is running

C. Containment entry is suspended due to loss of Unit 3 Containment Purge with:
- 3V9, U-3 Cntmt Purge Supply Fan is running
- 4V20, U-4 Cntmt Purge Exhaust Fan tripped

D. Containment entry may proceed due to Unit 3 Containment Purge is unaffected with:
- 3V9, U-3 Cntmt Purge Supply Fan is running
- 4V20, U-4 Cntmt Purge Exhaust Fan is running

Proposed Answer: B

Explanation (Optional):

A. Incorrect. The U-4 Cntmt Purge Exhaust Fan does not receive a trip signal. Plausibility - The candidate assumes both fans receive a trip signal from a signal which causes a Containment Ventilation signal. Also, plausible because the 1st part is correct.

B. CORRECT. 3-NOP-053, Step 4.1.1 CAUTION: Unit 4 Containment Purge Exhaust Fan 4V20 may be used for purging operations. Section 2.1 Step 3 lists automatic fan trip signals that are lost when using 4V20. U-4 Cntmt Purge Exhaust Fan remains running.

C. Incorrect. The U-4 Cntmt Purge Exhaust Fan does not receive a trip signal. Plausibility - The candidate assumes only the U-4 Cntmt Purge Exhaust Fan receives a trip signal from a signal which causes a Containment Ventilation signal. Tripping the Purge Exhaust Fan will eliminate discharge to the atmosphere. Also, plausible because the 1st part is correct.

D. Incorrect. Containment is evacuated. Plausibility - The candidate assumes neither fan receive a trip signal from a failed instrument. Therefore, they do not evacuate Containment.

Technical Reference(s): 3-NOP-053, Containment Purge System

0-ADM-009, Containment Entries When Containment Integrity is Established

0-ADM-713, Confined Space Entry Procedure

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 690004, Obj. 5 (As available)

Question Source:	Bank #		
	Modified Bank #	PTN - Item: 1.1.24.29.5.8	(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	Turkey Point

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	(2RI)

Question Difficulty Level: C

10 CFR Part 55 Content:	55.41	10
	55.43	

Radiological safety principles and procedures.

Comments:

4.1.1 Containment Purge Initiation (continued)

CAUTION

Unit 4 Containment Purge Exhaust Fan 4V20 may be used for purging operations. Section 2.1 Step 3 lists automatic fan trip signals that are lost when using 4V20.

15. **START** one Containment Purge Exhaust Fan:

- 3V20, U-3 CNTMT PURGE EXHAUST FAN
- 4V20, U-4 CNTMT PURGE EXHAUST FAN

CAUTION

To prevent an uncontrolled release, a Containment Purge Supply Fan shall **NOT** be started unless the containment equipment hatch, emergency hatch, and at least one personnel air lock door is closed.

16. IF POV-3-2600, CONTAINMENT PURGE SUPPLY ISOLATION (O.C.) and POV-3-2601, CONTAINMENT PURGE SUPPLY ISOLATION (I.C.) are OPEN, THEN **START** 3V9, U-3 CNTMT PURGE SUPPLY FAN.

17. IF a containment purge is started in MODE 6 and any of the following are met:

    A. Within 100 hours of starting core alterations

    B. During core alterations

    C. During the movement of irradiated fuel in containment

    THEN **PERFORM** 3-OSP-067.1, Process Radiation Monitoring Operability Test, Channel R-3-11 and Channel R-3-12 Functional Test Sections.

1.0 PURPOSE

This procedure provides the instructional guidance for operation of the Unit 3 Containment Purge System.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

1. A Radioactive Gas Release Permit is **NOT** required for CV-3-2819 and CV-3-2826, CNTMT INSTRUMENT AIR BLEED ISOL valves. These valves shall be normally open to prevent buildup of pressure in the containment.

2. If 3-12-031, SFP TRANSFER CANAL GATE VALVE, is open, the refueling cavity is filled with water, and refueling integrity is established; then a slow increase in containment pressure will occur and cause the SFP level to increase when the Containment Purge System is secured. Use of air operated equipment inside containment will result in a more rapid increase in containment pressure and will require monitoring SFP level more closely. (OE 3989)

3. The Containment Purge Exhaust Fan from either unit may be used for purging operation. If 4V20, U-4 CNTMT PURGE EXHAUST FAN, from the opposite unit is used, it will **NOT** automatically trip from any of the following conditions:

   - High containment activity on R-3-11, PARTICULATE Containment Radiation Monitor
   - High containment gaseous activity on R-3-12, GASEOUS Containment Radiation Monitor
   - Automatic or manual Safety Injection signal
   - Containment Isolation Phase A signal
   - Containment Isolation Phase B signal

2.1 Precautions (continued)

4. The use of 4V20, U-4 CNTMT PURGE EXHAUST FAN, from the opposite unit will **NOT** affect operability of the Unit 3 containment isolation valves or operability of the containment radiation monitors that initiate containment and Control Room ventilation isolation. A manual trip of the 4V20 exhaust fan will be required following containment isolation to prevent damage to the fan.

2.2 Limitations

1. When CV-3-2819, CNTMT INSTRUMENT AIR BLEED ISOL. (I.C.) AND CV-3-2826, CNTMT INSTRUMENT AIR BLEED ISOL. (O.C.) are open, then R-3-11 PARTICULATE or R-3-12 GASEOUS Containment Radiation Monitors shall be OPERABLE to provide automatic isolation upon high airborne activity inside containment. This requirement does **NOT** apply when Unit 3 is in MODES 5, 6, or DEFUELED (except during core alterations) with Technical Specification required compensatory actions taken.

Item: 1.1.24.29.5.8

69021290508

- A Containment purge is in progress on Unit 3.
- The Unit 3 Purge Supply Fan is running.
- The Unit 4 Purge Exhaust Fan is running
- Unit 3 receives an SI signal

What effect will the SI signal have on the Containment Purge Fans?

A) The Unit 3 Purge Supply Fan will trip.
The Unit 4 Purge Exhaust Fan will trip.

B) The Unit 3 Purge Supply Fan will trip.
The Unit 4 Purge Exhaust Fan will NOT trip.

C) The Unit 3 Purge Supply Fan will NOT trip.
The Unit 4 Purge Exhaust Fan will trip.

D) The Unit 3 Purge Supply Fan will NOT trip.
The Unit 4 Purge Exhaust Fan will NOT trip.

CORRECT or INCORRECT feedback for item: 1.1.24.29.5.8
0-OP-53 4.15 & 4.17. Opposite unit's exhaust fan will have to be manually tripped.

Item Classification: Comprehension
Item difficulty: 0.00
Keywords: 103 A3.01
Item weight: 10
Points required for mastery: 1
Correct alternative(s): B
Judging values of alternatives:
A=-1 B=1 C=-1 D=-1

QUESTION 60

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	033	A3.02
	Importance Rating	2.9	

Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including: Spent fuel leak or rupture

Proposed Question: RO Question # 60

Initial Conditions:

- Unit 3 is operating at 100% power.
- 3A Spent Fuel Pool Cooling Pump is running.
- Spent Fuel moves are being performed in the Spent Fuel Pit.

When the following occurs:

- Fuel movement is stopped after a Spent Fuel Assembly is dropped.
- Bubbles are coming to the surface of the Spent Fuel Pit.
- 3A Spent Fuel Pool Cooling Pump Motor is running and its breaker automatically trips.
- 3A Spent Fuel Pool Cooling Pump Motor shows no sign of apparent damage.
- Spent Fuel Pool Level remains stable in band.
- Spent Fuel Pool Temperatures are rising.
- Spent Fuel Building Area Radiation Monitors are alarming.
- RP confirms that a radioactive release is occurring from the Spent Fuel Building.

Which ONE of the following sets of actions are required (1) FIRST in accordance with 3-ONOP-033.3, Accidents Involving New or Spent Fuel and (2) in accordance with 3-ONOP-033.1, Spent Fuel Pit (SFP) Cooling System Malfunction, with a SFP high temperature condition?

A. (1) Evacuate all personnel from the Spent Fuel Building.

(2) Reset 3A Spent Fuel Pool Cooling Pump Motor Breaker and restart the pump

B. (1) Evacuate all personnel from the Spent Fuel Building.

(2) Start 3B Spent Fuel Pool Cooling Pump from its alternate power alignment (Unit 4).

C. (1) Place the Control Room HVAC in recirculation mode

(2) Start 3B Spent Fuel Pool Cooling Pump from its alternate power alignment (Unit 4).

D. (1) Place the Control Room HVAC in recirculation mode

(2) Reset 3A Spent Fuel Pool Cooling Pump Motor Breaker and restart the pump

Proposed Answer: A

Explanation (Optional):

A. Correct. 3-ONOP-033.3 - All personnel are evacuated from the Fuel Building first. The 3A Spent Fuel Pool Cooling Pump Motor shows no sign of apparent damage and the pump is restarted.

B. Incorrect. 3-ONOP-033.3 - All personnel are evacuated from the Fuel Building first. The start of 3B Spent Fuel Pool Cooling Pump is not aligned from its alternate power alignment (Unit 4) during Mode 1 operations. Plausible – Correct first action. Also, the 3B SFP Cooling Pump is aligned to its alternate supply during refueling.

C. Incorrect. 3-ONOP-033.3 - All personnel are evacuated from the Fuel Building first, not placing the Control Room HVAC in recirculation mode. The start of 3B Spent Fuel Pool Cooling Pump is not aligned from its alternate power alignment (Unit 4) during Mode 1 operations. Plausible – first action is required in 3-ONOP-033.3. Also, the 3B SFP Cooling Pump is aligned to its alternate supply during refueling.

D. Incorrect. 3-ONOP-033.3 - All personnel are evacuated from the Fuel Building first, not placing the Control Room HVAC in recirculation mode. The 3A Spent Fuel Pool Cooling Pump Motor shows no sign of apparent damage and the pump is restarted. Plausible –first action is required in 3-ONOP-033.3. Also, correct second action.

Technical Reference(s): 3-ONOP-033.3, Accidents Involving New or Spent Fuel (Attach if not previously provided)

3-ONOP-033.1, Spent Fuel Pit Cooling Malfunction

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902283, Obj. 4 (As available)

Question Source: Bank #

Modified Bank #		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	(1P)
	Comprehension or Analysis	
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

CAUSES:
1. High radiation in one of systems monitored by PRMS
2. PRMS system component failure

H1/4

PRMS HI RADIATION

DEVICE:
- R-11
- R-12
- R-14
- R-15
- R-17A
- R-17B
- R-18
- R-19
- R-20

SETPOINT:
Variable with each PRMS channel

LOCATION:
N/A

ALARM CONFIRMATION

1. **CHECK** the following:
   - Countrate meter on each PRMS drawer in Rack QR-66
   - Alarm indicators on each drawer in Rack QR-66

OPERATOR ACTIONS

1. IF 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 for expected automatic actions.
2. IF alarm is on R-15 or R-19, THEN **REFER TO** 3-ONOP-071.2, Steam Generator Tube Leakage for expected automatic actions.
3. IF alarm is on R-14, R-17A, R-17B, R-18, or R-19, THEN **CHECK** alarm valid as follows:
   A. **CHECK** FAIL/TEST light **NOT** LIT.
   B. **PUSH** FAIL/TEST light (meter reading of 288 or 289K)
   C. **PUSH** SOURCE CHECK light (should get meter increase).
   D. **PUSH** HIGH ALARM light to determine if meter level is above high alarm setpoint.
4. **ENSURE** required automatic actions.
5. IF 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.
6. IF alarm is on R-15 OR R-19, THEN **REFER TO** 3-ONOP-071.2, Steam Generator Tube Leakage.
7. **REFER TO** TS 3.3.3, 3.4.6, and 3.9.13 for additional required actions.

REFERENCES: Tech Spec Sections 3.3.3, 3.4.6, and 3.9.13
PC/M 07-055, R-15 Steam Jet Air Ejector Monitor Replacement

CAUSES:
1. Local area high radiation
2. Failed ARMS detector

X4/1

ARMS HI RADIATION

DEVICE: Area Rad Monitors

SETPOINT: N/A

LOCATION: Channels on ARMS rack

ALARM CONFIRMATION

IDENTIFY alarming channel(s) by noting individual channel(s) HIGH alarm light(s) illuminated on Area Radiation Monitoring System Control Panel R-30.

OPERATOR ACTIONS

1. **CHECK** valid alarm by performing 0-ONOP-066, High Area Radiation Monitoring System Alarm.
2. IF any ARM channels 1 through 6 is determined to have failed while in Mode 5 or 6, THEN **NOTIFY** Radiation Protection to install a portable area radiation monitor, with alarm.
3. IF any ARM channels 21 through 24 is determined to have failed, THEN **NOTIFY** Radiation Protection to install a portable area radiation monitor capable of meeting requirement of 10 CFR 50.68(b).

REFERENCES:
1. FPL Dwg 5610-J-822
2. Victoreen Tech Manual V000452

1.0 PURPOSE

1.1 This procedure provides instructions to be followed in the event of a new or spent fuel element is dropped or damaged during fuel handling operations.

2.0 SYMPTOMS

2.1 Annunciators

2.1.1 Annunciator H 5/2, CNTMT ISOLATION ACTIVATED, alarm during fuel handling operations.

2.2 Indications

2.2.1 Containment atmosphere process radiation monitors R-3-11 or R-3-12 alarm during fuel handling operations.

2.2.2 SPING 4 Unit 3 spent fuel pit alarm.

2.2.3 Containment or Spent Fuel Building area radiation monitors R-2, R-7, R-19, or R-21 alarm during fuel handling operations.

2.2.4 Gas bubbles originating from the damaged element.

2.2.5 Notification that a fuel assembly has been dropped or damaged.

3.0 AUTOMATIC ACTIONS

3.1 **IF** Containment Atmosphere Process Radiation Monitors R-3-11 or R-3-12 alarm, **THEN** refer to Attachment 1 for equipment lineup checklist.

3.2 **IF** the Plant Vent Exhaust Radiation Monitor R-3-14 increases to the alarm setpoint, **THEN** Waste Disposal System Gaseous/Discharge Valve RCV-014 CLOSES.

4.0 IMMEDIATE ACTIONS

4.1 None

5.0 SUBSEQUENT ACTIONS

5.1 Inform the Control Room of the accident.

5.1.1 Evacuate all personnel from the area in which the accident occurred.

5.1.2 **IF** the accident involves spent fuel inside Containment, **THEN** perform the following:

1. Announce over the plant PA System:

   Attention all personnel in Unit 3 Containment, evacuate Unit 3 Containment.

2. Sound the Containment evacuation alarm.

3. Announce over the plant PA System:

   Attention all personnel in Unit 3 Containment, evacuate Unit 3 Containment.

4. Stop the Contmt Purge Air Supply Fan 3V-9.

5. Stop the Contmt Purge Exhaust Fan 3V-20.

6. Close the Contmt Purge Supply Isol. Valves POV-3-2600 and 2601.

7. Close the Contmt Purge Exhaust Isol. Valves POV-3-2602 and 2603.

8. Close the Contmt Inst Air Bleed Valves CV-3-2826 and CV-3-2819.

5.2 Accident Involving Spent Fuel

5.2.1 Accident Occurring in the Containment

1. Verify or place the Control Room HVAC in the recirculation mode using Attachment 1.

2. Concurrently perform 3-ONOP-067, Radioactive Effluent Release.

3. Inform the Shift Manager to refer to 0-EPIP-20101, Duties of Emergency Coordinator, **AND** take any actions that may be required.

4. Notify the Radiation Protection Shift Supervisor to monitor radiation levels **AND** to evaluate airborne particulate and gaseous samples.

5. Monitor the Containment Atmosphere Process Radiation Monitors R-3-11 **AND** R-3-12 for indication of increasing particulate or gaseous activity.

5.2.1 (Cont'd)

6. Verify that the Containment has been evacuated **AND** the Personnel Hatch, Equipment Hatch, and Escape Hatch are closed.

7. **IF** radiation levels or sampling results indicate the damaged spent fuel element cladding has been breached, **THEN** proceed as follows:

   a. **IF** available, **THEN** start the Emergency Containment Filter Fans.

   b. **WHEN** Radiation Protection sampling results indicate further recirculation of the Containment atmosphere through the Emergency Containment Filter Fans will **NOT** substantially reduce the I-131 concentration, **THEN** perform the following:

      (1) Stop the Emergency Containment Filter Fans using 3-NOP-055, Emergency Containment Cooling and Filtering System.

      (2) Commence purging using 3-NOP-053, Containment Purge System.

8. **IF** there are **NO** indications the spent fuel element cladding has been breached, **THEN** re-establish the Containment purge using 3-NOP-053, Containment Purge System.

9. **WHEN** the atmospheric radiological conditions have been verified to be normal, **THEN** return the Control Room HVAC to normal operation using 0-NOP-025, Control Room Ventilation System.

10. Obtain permission from the Radiation Protection Shift Supervisor **AND** then enter Containment.

11. Perform visual inspection on the fuel element for damage.

12. **WHEN** the visual inspection is complete, **THEN** transfer the fuel element to a storage location designated by Reactor Engineering.

13. Disposition the damaged fuel element using instructions from Reactor Engineering.

5.2.2 Accident Occurring in the Spent Fuel Pit

1. Place the Control Room HVAC in the recirculation mode using Attachment 1.
2. Concurrently perform 3-ONOP-067, Radioactive Effluent Release.
3. Inform the Shift Manager to refer to 0-EPIP-20101, Duties of Emergency Coordinator, and take any actions that may be required.
4. Notify the Radiation Protection Shift Supervisor to monitor radiation levels **AND** to evaluate airborne particulate and gaseous samples.
5. Verify the Spent Fuel Building has been evacuated.
6. **WHEN** the atmospheric radiological conditions have been verified to be normal, **THEN** return the Control Room HVAC to normal operation using 0-NOP-025, Control Room Ventilation System.
7. Obtain permission from the Radiation Protection Shift Supervisor **AND** then enter the Spent Fuel Building.
8. Perform visual inspection on fuel element for damage.
9. **WHEN** the visual inspection is complete, **THEN** transfer the fuel element to a storage location designated by Reactor Engineering.
10. Disposition the damaged fuel element using instructions from Reactor Engineering.

5.3 Accident Involving New Fuel

5.3.1 Notify the Radiation Protection Shift Supervisor to survey the affected area to determine if radioactive release has occurred.

5.3.2 Refer to 0-ADM-115, Notification of Plant Events.

5.3.3 Obtain permission from Radiation Protection Shift Supervisor **AND** then visually inspect the fuel element for damage.

5.3.4 Transfer the fuel element to a storage location designated by Reactor Engineering.

5.3.5 Disposition the damaged fuel element taking instructions from Reactor Engineering.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Step 8 is NOT applicable to a channel failure of R-11, R-12, R-15, or R-19.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	**Check For PRMS Channel Failure** • Check Fail indicator - OFF • Display and recorder reading – **NOT** FAILED LOW	Perform the following: a. **IF** R-14 fails low **AND** a gas decay tank release is in progress, **THEN** stop the release. b. **IF** R-18 fails low **AND** a liquid release is in progress, **THEN** stop the release. c. Notify the Shift Manager to refer to Tech Specs **AND** take all required actions for the failed channel(s). d. Notify I&C of the PRMS failure.
9	**Check R11/12 RM-80 Green Monitor Light - ON**	Identify and correct the cause of failure using applicable Steps 18 through 28.
10	**Check If Effluent Radiation Monitors ALARMS - OFF**	
	a. Check the following radiation monitor alarms - OFF • RAD-3-6417 (SJAE SPING) • RAD-3-6426 (DAM-1 Monitor)	a. Go to 3-ONOP-071.2, STEAM GENERATOR TUBE LEAKAGE.
	b. Check RAD-6304 (Plant Vent SPING) alarm - OFF	b. Perform the following as applicable: 1) 4-ONOP-033.1, SFP COOLING SYSTEM MALFUNCTION 2) **IF** Steps 42 through 45 **NOT** previously performed, **THEN** go to Step 42.
	c. Check RAD-6418 (SFP Vent SPING) alarm - OFF	c. Perform 3-ONOP-033.3, ACCIDENT INVOLVING NEW OR SPENT FUEL.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16	**Check For High Containment Airborne Activity**	
	a. Check for R-11/12 HIGH ALARMS	a. Go to Step 18.
	* Check R-11 Red HIGH LED - ON	
	* Check R-11 PART alarm monitor pushbutton - FLASHING	
	* Check R-12 Red HIGH LED - ON	
	* Check R-12 GAS alarm monitor pushbutton - FLASHING	
	* R-11/12 display reading - GREATER THAN OR EQUAL TO ALARM SETPOINT	
	b. Verify Containment And Control Building Ventilation Systems aligned using Attachment 1	
	c. Dispatch an operator to the RM-80 skid to perform the following	
	• Silence the local alarm	
	• Check for any abnormal indications	
	d. Direct Radiation Protection and Chemistry Departments to verify actual activity inside containment	
	e. Perform the following to evaluate plant status	
	* 3-ONOP-041.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE	
	* 3-ONOP-033.3, ACCIDENTS INVOLVING NEW OR SPENT FUEL	
	f. Determine if alarm is valid by checking against current plant status	f. Go to Step 19.
	g. Evacuate non-essential personnel from containment	
17	**Return To Step 1**	

QUESTION 61

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035	K6.01
	Importance Rating	3.2	

Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: MSIVs

Proposed Question: RO Question # 61

Given the following conditions:

- Unit 4 is at 21% power with all systems in normal alignments.
- The Main Generator is synchronized to the grid.
- A single Main Steam Isolation Valve closes on a spurious signal.

Assuming the reactor does NOT trip, which ONE of the following correctly describes the INITIAL response of S/G Pressure and Level in the affected loop?

	S/G Pressure	S/G Level
A.	rises	rises
B.	lowers	rises
C.	lowers	lowers
D.	rises	lowers

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausibility – 1st part is correct. The applicant believes S/G level will rise due to the loss of steam flow while maintaining feedwater flow. This is an incorrect initial response which does not take into account the S/G shrink & swell effect.

B. Incorrect. Plausibility – The applicant believes heat is still removed from the S/G or S/G pressure spikes causing a Steam Dump To Atmosphere to open to lower pressure. This is NOT the initial response. Also, the applicant believes S/G level will rise do to the loss of steam flow while maintaining feedwater flow. This is an incorrect initial response

which does not take into account the S/G shrink & swell effect.

C. Incorrect. Plausibility – The applicant believes heat is still remove from the S/G or S/G pressure spikes causing a Steam Dump To Atmosphere to open to lower pressure. This is NOT the initial response. Also, the second part is correct.

D. CORRECT.
S/G pressure initial rises due to heat no longer being removed from the S/G. With this pressure increase, the S/G level will shrink or lower due to the saturation pressure rising.

Technical Reference(s): LP 6900912, *Abnormal Transient Accident Analysis* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902117, Obj. 10
LP 6900912, Obj. 3 (As available)

Question Source: Bank # 70214
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2007 Comanche Peak

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis (2DR)

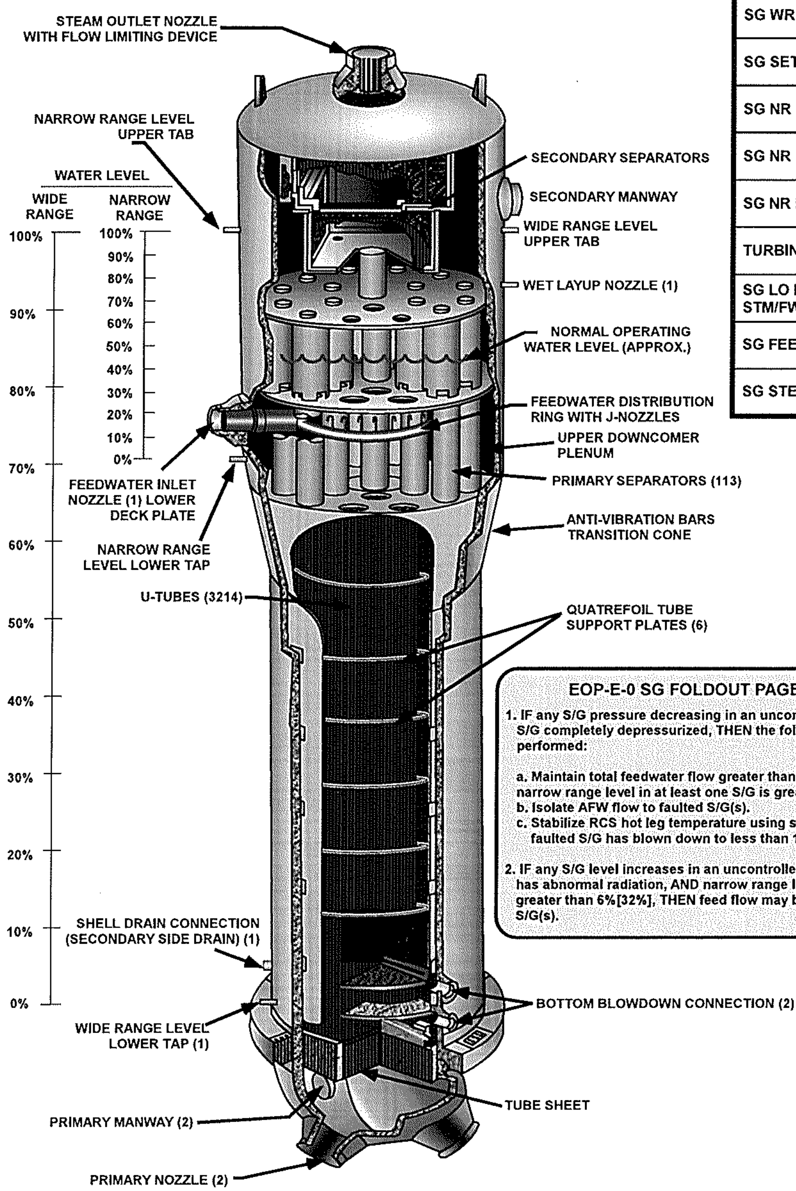
Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 14
55.43

Principles of heat transfer, thermodynamics and fluid mechanics.

Comments:

STEAM GENERATOR



ALARM	SETPOINT	COINCIDENCE	ANNUNCIATOR
SG WR HIGH/LOW	93% / 40%	1/1	C-3/1(/2,/3)
SG SETPOINT DEVIATION	+5%	1/1	C-6/1(/2,/3)
SG NR LO-LO LEVEL ALARM	35%	1/3	C-1/1(/2,/3)
SG NR LO-LO LEVEL TRIP	10%	2/3	C-1/4
SG NR HIGH LEVEL	68%	1/3	C-2/1(/2,/3)
TURBINE TRIP - SG HI-HI LVL	80%	2/3	E-2/6
SG LO LVL and STM/FWF MISMATCH	10% LVL and 0.64×10^6 lb/hr	1/2	C-5/4(/5,/6)
SG FEED > STEAM	FF > SF 0.64×10^6 lb/hr	1/2	C-4/1(/2,/3)
SG STEAM > FEED	SF > FF 0.5×10^6 lb/hr	1/2	C-5/1(/2,/3)

TECH SPECS

3/4.4.5, STEAM GENERATORS
- Each S/G shall be operable > 200°F
- Modes: 1, 2, 3, 4

3/4.4.6.2.c, OPERATIONAL LEAKAGE
- 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator
- Modes: 1, 2, 3, 4

3/4.7.1.4, SECONDARY SPECIFIC ACTIVITY
- Specific Activity ≤ 0.1 μ cu/gr DOSE EQUIVALENT I-131
- Modes: 1, 2, 3, 4

6.8.4.c Secondary Water Chemistry
- Program for monitoring SG water chemistry to inhibit SG tube degradation

3-ONOP-071.2 FOLDOUT PAGE ITEMS

1. 3-EOP-E-0 Transition Criteria
2. Standard Briefing
3. Plant Announcement
4. Blowdown Release Path Isolation
5. AFW Steam Supply Release Path Isolation

SG DESIGN/OPERATING DATA

Secondary Side Pressure (Design/Operating)	1085/830 PSIG
Secondary Side Design Temperature	565°F
Steam Temperature, Full Power	516°F
Secondary Full Power Pressure Drop	20-21 PSID
RCS DP Across Steam Generator	32.4 PSID
Heat Transfer Area	43,467 FT^2
Steam Flow Rate, Full Power	3.38×10^6 lb/hr
Secondary Fluid Volume, Full Power	4683 FT^3
Steam Quality (minimum)	99.75%
Blowdown Flow Rate (maximum)	200,000 lb/hr

EOP-E-0 SG FOLDOUT PAGE CRITERIA

1. IF any S/G pressure decreasing in an uncontrolled manner OR any S/G completely depressurized, THEN the following may be performed:

   a. Maintain total feedwater flow greater than 345 gpm until narrow range level in at least one S/G is greater than 6%[32%].
   b. Isolate AFW flow to faulted S/G(s).
   c. Stabilize RCS hot leg temperature using steam dumps when faulted S/G has blown down to less than 10% wide range.

2. IF any S/G level increases in an uncontrolled manner OR any S/G has abnormal radiation, AND narrow range level in affected S/G(s) is greater than 6%[32%], THEN feed flow may be stopped to affected S/G(s).

SECONDARY CHEMISTRY ACTION LEVELS

ACTION LEVEL 1

Action Level 1 is entered whenever any parameter's value exceeds its normal value. The plant can be operated for up to 1 week with water chemistry parameters at action level 1 values. If the parameter cannot be returned to its normal value within 1 week, Action Level 2 must be entered.

ACTION LEVEL 2

Action Level 2 is entered whenever any parameter's value could result in some degree of steam generator corrosion during extended full power operation. Within 8 hours after the Action Level 2 value is detected, power must be reduced to a level (typically 30 percent of full power or less) that will permit sufficient flow to maintain automatic secondary system operation while simultaneously reducing heat flux (i.e., impurity concentration rate) in the steam generator. If the parameter cannot be returned to its normal value (no Action Level) within 100 hours after exceeding Action Level 2 values, Action Level 3 must be entered.

ACTION LEVEL 3

Action Level 3 is entered whenever any parameter's value will result in rapid steam generator corrosion. Its objective is to avoid the ingress and concentration of harmful impurities in the steam generator while the cause of the Action Level 3 value is eliminated. The plant must be shut down as quickly as safe plant operations permit after the Action Level 3 value is detected, or the impurities will cause rapid steam generator corrosion. The plant should be cleaned up by feed and bleed or drain and refill as appropriate, until normal values are obtained.

SHRINK AND SWELL PHENOMENON

OVERVIEW

During steady-state conditions, steam flow matches feed flow and SG level is relatively constant. During transient conditions, fluid inventory on SG secondary side is not constant. A SGWLC system must be used to attempt to maintain level as constant as possible.

SHRINK

During a rapid downpower excursion (such as a turbine load rejection caused by a reactor trip), secondary temperature and pressure increase causing less boiling in the secondary side of the steam generator and compression of the steam bubbles occurs resulting in a drop in SG level. The Westinghouse SG resembles a large manometer. At high power levels, high velocity flow exists down the annular downcomer and up through the tube bundle. On a sudden load reduction, the momentum of this flow causes a sharp level drop and oscillation, and the level instrumentation reads SG level in this area.

SWELL

Swell occurs when the steam demand is increased with a resultant pressure reduction, causing an increase in the level due to the increased boiling and expansion of the steam bubbles. When power is increased, steam generator pressure decreases. Since the steam generator is operated under saturated conditions, this results in a larger void fraction in the tube bundle region and thus a higher indicated water level. The second effect during a power increase is increased mass flow rate through the moisture separators. This results in more water in the downcomer region and again a higher indicated water level.

CORRECTIVE ACTIONS/MITIGATION STRATEGY

To minimize SHRINK and SWELL, the water level must be programmed high enough so that the SHRINK that occurs during a Large Step Load Decrease will not result in the low steam generator level reactor trip setpoint being reached, and the SWELL that occurs during a Large Step Load Increase will not result in SG carryover into the main turbine.

Steam generator levels should be maintained at normal operating range to ensure the margin between the level setpoint and the level trips are maximized.

SG RECIRCULATION RATIO

Moisture separation is essential to ensure high quality dry steam leaves the steam generator for efficient secondary plant operations. Equally important are the effects of water recirculated back to the steam generator by the moisture separators. The recirculated water provides preheating for feedwater flow entering the steam generator. Because of the benefits provided by the recirculation flow, a substantial quantity is desirable.

Mathematically, Recirculation Ratio is equal to: recirculation flow / feed flow (total to the steam generator)
More commonly it is discussed as: Circulation Ratio, which is equal to: steam & water flow (leaving the tube region) / steam flow (dry steam leaving the steam generator).

This recirculation ratio is a major consideration in steam generator design. Empirical data has been obtained to show that as power, or steaming rate, is increased, the recirculation ratio will decrease. Typically the ratio is approximately 25 at 50% power and approximately 5 at 100% power.

References:
5610 TD-17, Steam Gen. Level Control and Protection
5610 TE-4061, Sh. 1, Main Steam System
3-ARP-097.CR, Control Room Annunciator Response
SD-011, Steam Generator System Description
SD-187, Secondary System Chem Inj, Sampling, Chem Limits

QUESTION 62

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041	A4.04
	Importance Rating	2.7	

Ability to manually operate and/or monitor in the control room: Pressure mode

Proposed Question: RO Question # 62

Given the following:

- Unit 4 is in STARTUP at 7% power.
- The Steam Dump Control MODE SELECTOR Switch on the Control Room Console is in the MAN position.
- The Steam Pressure Controller is in automatic.
- The Steam Pressure Controller demand is at 30%.

Which ONE of the following listed below describes (1) the Condenser Steam Dump(s) that are armed and (2) the given Condenser Steam Dump's position?

	ARMED	**POSITION**
A.	ONLY CV-2827	PARTIALLY OPEN
B.	ONLY CV-2827	FULLY OPEN
C.	BOTH CV-2827 & CV-2828	CV-2827 IS FULLY OPEN CV-2828 IS PARTIALLY OPEN
D.	BOTH CV-2827 & CV-2828	BOTH CV-2827 & CV-2828 ARE PARTIALLY OPEN

Proposed Answer: C

Explanation (Optional):

A. Incorrect. In pressure control mode, steam dumps are armed and the signal to the steam dumps would have one valve fully open at 8 ma, and the second valve modulating open at 8 ma. 30% demand is approximately 9 ma

B. Incorrect. In pressure control mode, steam dumps are armed and the signal to the steam dumps would have one valve fully open at 8 ma, and the second valve modulating open at 8 ma. 30% demand is approximately 9 ma

C. CORRECT. In pressure control mode, steam dumps are armed and the signal to the steam dumps would have one valve (2827) fully open at 8 ma, and the second valve (2828) modulating open at 8 ma. 30% demand is approximately 9 ma

D. Incorrect. This is plausible because in trip open mode, steam dumps do not open sequentially. The applicant may confuse pressure control mode operation for trip open mode. Valves open sequentially as demand is raised

Technical Reference(s): SD-105, *Steam Dump System* (Attach if not previously provided)
LP-6902118, *Steam Dump System*

Proposed References to be provided to applicants during examination:

Learning Objective: LP 6902118, Obj. 7 (As available)

Question Source: Bank #
Modified Bank # X (Note changes or attach parent)
New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests monitoring the condenser dumps in the Control Room, via the output from the steam pressure controller.

Question Selection Methodology:

Question derived from the bank question below; Found only two multiple choice questions in the PTN exam bank that matched the K/A. Used a random number generator to select this one.

69021180634
Given the following: Unit at Hot Zero Power Tavg = 549°F Steam Dump in STM PRESS mode with 1005 psig set into steam pressure controller in automatic. The Steam Dump System response is ________________________.

a. Arm Only
b. Arm and Actuate
c. Disarm
d. none/no effect

#62

OBJECTIVE 4b, 5, 6, 7

The Steam Dump Control Selector Switch and the reactor coolant system temperature instruments provide the second arm/block signal. The temperature instruments provide a block signal when the protection TAVG signal from two out of 2 three reactor coolant loops decreases to 543°F and the Steam Dump Control Switch is in the ON position. Refer to Figure 4A. If the Steam Dump Control Switch is placed in the BYPASS position with the Low TAVG signal present the Low TAVG signal will be removed and a status light will be lit on VPA indicating that the Low TAVG block has been bypassed. This would allow a cooldown of the reactor coolant system. Returning the Steam Dump Control Selector Switch to the ON position will not remove the bypass signal if the Low TAVG signal is present. In order to provide another Low TAVG signal the bypass signal must be reset by removing the Low TAVG signal or placing the Steam Dump Control Selector Switch in the RESET/OFF position first. These actions would allow another Low TAVG signal to block the condenser steam dump activation or allow a still present low TAVG condition to again block the condenser steam dump activation and de-energize the status light on VPA. The Steam Dump Control Selector Switch can block steam dump activation by placing the switch in the RESET/OFF position. To summarize, any of the signals listed below will provide a blocking signal to prevent the arming of the steam dumps.

1. High condenser pressure of 20" Hg (low or no condenser vacuum). (Can NOT be Bypassed.)
2. Low protection TAVG of 543°F with the Steam Dump Control Selector Switch in the ON position. (Can be Bypassed.)
3. Steam Dump Control Switch in the RESET/OFF position.

The steam dump system has two automatic modes of operation, Steam Pressure mode and TAVG mode. The operational mode is selected by the operator by the Steam Dump Control Mode Selector Switch on the console. Refer to Figure 4B & 4C. When placed in the MANUAL position a steam pressure signal provides an operational signal to modulate the condenser steam dump valves. This steam pressure signal maintains the steam pressure setpoint set by the operator, using the steam pressure controller on the console. In the AUTO position, two steam dump controllers are available. Refer to Figure 4B & 4C. The turbine trip controller operates the condenser steam dump valves to restore no load TAVG following a turbine trip. The load rejection (turbine runback) controller operates the valves to restore TAVG to a program (TREF) value.

The control signals to the condenser steam dump valve positioners are sent through individual analog gates. These signals modulate the valves open in a sequence beginning with CV-2827 and going through CV-2830.

INSTRUCTOR ACTIVITY

OBJECTIVES 4b, 5, 6, 7	NSO/LPRO/LPSO

Program	Slide(s) 57-58

Question #8:

Where does the signal for T_{REF} come from?

Answer #8:

T_{REF} determined by PT-446.

T_{AVG} is median selected signal. Turbine trip signal from 2/3 auto stop oil press. switches or 2/2 Turbine Stop Valves closed.

c. Conditions to energize solenoid valves for bank
   1) Steam Dump Control Mode Selector Switch in AUTO and either:
      a) T_{AVG} 9.5°F above T_{REF} without turbine trip.
      b) Turbine trip in coincidence with high T_{AVG} of 554.5°F
   2) Steam dump alarm on C-8/3
d. Conditions to energize solenoid valves for bank 2.
   1) Steam Dump Control Mode Selector Switch in AUTO and either;
      a) T_{AVG} 14.5°F greater than T_{REF} without turbine trip
      b) Turbine trip coincidence with T_{AVG} of 561.5°F.
   2) Steam dump alarm on C-8/3

4. Valve positioner
   a. Receives signal from one of three analog gates depending on Steam Dump Control Mode Selector Switch position
      1) MANUAL
         a) Steam pressure controller output connected
         b) Other two controllers disconnected
      2) AUTO
         a) Either turbine trip or load rejection controller connected
         b) Depends upon if a turbine trip occurred
   b. Controller determine output of analog gate which provides a 4-20 ma output
      1) CV-2827 opens at approximately 4-8 ma
      2) CV-2828 opens at approximately 8-12 ma
      3) CV-2829 opens at approximately 12-16 ma
      4) CV-2830 opens at approximately 16-20 ma

INSTRUCTOR ACTIVITY

OBJECTIVES 4b, 5, 6, 7	NSO/LPRO/LPSO

Program	Slide(s) 57-58

c. Load rejection controller, TM-408J.
   1) 5°F deadband to allow rod control to control temperature error
   2) Temperature error; $T_{AVG} - T_{REF}$
   3) Conditions for operation
      a) No turbine trip
      b) Steam Dump Control Mode Selector Switch in AUTO
   4) Modulates all four dump valves

d. Turbine trip controller, TM-408L
   1) No deadband
   2) Output when T_{AVG} greater than 547°F (no load T_{AVG}).
   3) Conditions for operation
      a) Turbine trip
      b) Steam Dump Control Mode Selector Switch in AUTO
   4) Modulates all four dump valves

e. Steam pressure controller, PC-464B
   1) Output connected to all four steam dump valves when Steam Dump Control Mode Selector Switch in MANUAL.
   2) Controller is auto/manual setpoint station on console.
   3) Normal setpoint
      a) 1005 PSIG
      b) Setpoint adjusted downward for RCS cooldown.
   4) PT-464 provides main steam pressure signal.
   5) Above controller on console are steam dump valve open and closed indicating lights.

QUESTION #62

Q#62 Possibilities

@Q:1.1.24.18.6.34,M
@QNAME:Arming
@QKEYW:NLOCT
@QKEYW:SNPO
@QKEYW:NPO
@QKEYW:RCO
@QKEYW:STA
@QKEYW:041 A1.01
@QIWT:10
@QC:Comprehension
@QMEMO:REFERENCE: SD-105, Page 9 & 10
DL: 300
@QS:69021180634Given the following: Unit at Hot Zero Power Tavg = 549°F Steam Dump in STM PRESS mode with 1005 psig set into steam pressure controller in automatic.The Steam Dump System response is ______________________.
@QA:Arm Only
@QA+:Arm and Actuate
@QA:Disarm
@QA:none/no effect

@Q:1.1.24.18.6.57,M
@QNAME:PT-464 lo/ability c/
@QKEYW:LOP
@QKEYW:039 K1.06
@QKEYW:RCO Audit Exam
@QIWT:10
@QC:Comprehension
@QS:69021180657;The following conditions exist on Unit 3. Mode 3 at 547°F:The Steam Dump to Condenser (SDTC) 'control 'switch is in "ON."The SDTC 'mode' switch is in "MAN."The SDTC 'controller' is in "AUTO."PT-464, Steam Header Pressure transmitter, fails low.Which ONE of the following identifies the effect this failure will have on the SDTC system's ability to cool down to Mode 5?
@QIFB*:RCO Group 19 Audit Exam 5610-T-L1, Sheet 22A
@QA+:Steam pressure control mode: Unavailable Tavg control mode: Unavailable Manual-Manual mode: Available
@QA:Steam pressure control mode: UnavailableTavg control mode: AvailableManual-Manual mode: Available
@QA:Steam pressure control mode: AvailableTavg control mode: UnavailableManual-Manual mode: Unavailable
@QA:Steam pressure control mode: AvailableTavg control mode: AvailableManual-Manual mode: Unavailable

Only two multiple choice bank questions available that test STM PRESS Mode. Random Number Generator selected

QUESTION 63

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	045	A1.05
Importance Rating	3.8	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of primary plant parameters (temperature and pressure) following T/G trip

Proposed Question: RO Question # 63

Which ONE of the following describes the INITIAL response of (1) RCS temperature and (2) RCS pressure, due to a manual trip of the Unit 3 Turbine at 100% power without the Turbine directly initiating a Reactor Trip?

(Assume NO other operator action with all controls operating automatically.)

A. (1) Tavg drops and stabilizes at 547°F
(2) RCS Pressure lowers once the PZR PORVs open

B. (1) Tavg drops and stabilizes at 547°F
(2) RCS Pressure lowers once the Pressurizer Safety Valves open

C. (1) Tavg increases until Condenser Steam Dumps open
(2) RCS Pressure increases until the PZR PORVs open

D. (1) Tavg increases until Condenser Steam Dumps open
(2) RCS Pressure increases until the Pressurizer Safety Valves open

Proposed Answer: C

Explanation (Optional):

A. Incorrect since RCS temperature and pressure will normally rise immediately after the Turbine Trip. Plausible because after a unit trip this is an expected response. Also, 2nd part is a system response when a Turbine Trip without the Reactor Trip occurs.

B. Incorrect since RCS temperature and pressure will normally rise immediately after the Turbine Trip. Plausible because after an ATWS with no operator action will lead to the Pressurizer Safety Valves opening.

C. CORRECT. RCS temperature will rise, causing condenser dumps to open to return temperature to the no-load value. PZR pressure will follow suite, increasing until the PZR PORVs open to return pressure to less than 2335 psig.

D. Incorrect since RCS temperature will rise, causing condenser dumps to open to return temperature to the no-load value. Also, 2nd part is an incorrect due to the high decay heat load initially after the trip which will cause a Pressurizer insurge and pressure increase and PORVs will prevent a pressure increase to the Pressurizer Safety Valve Setpoint.

Technical Reference(s): LP 6902915, Transient and Accident Analysis - Decreased Heat Removal (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902915, Obj. 2 (As available)

Question Source: Bank #

Modified Bank # 64519 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

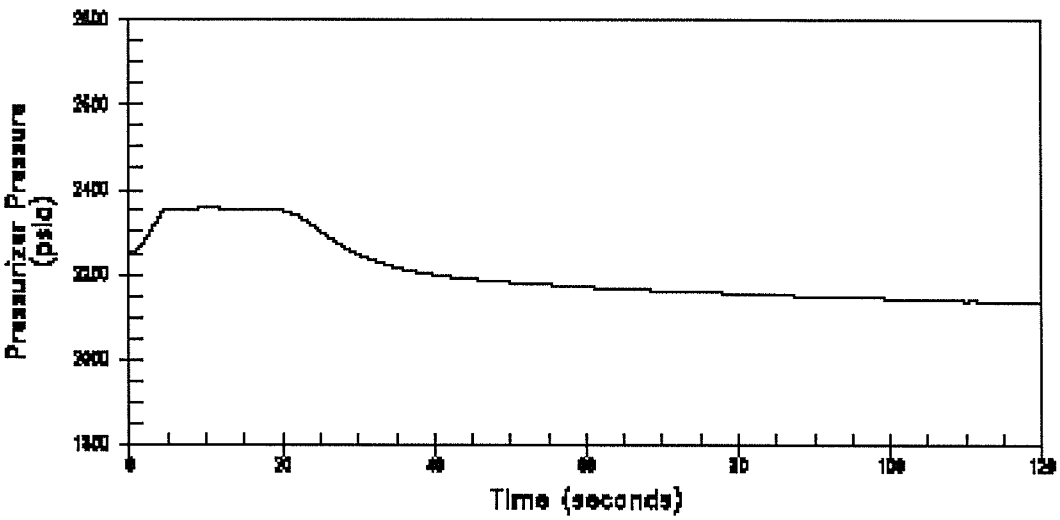
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Modified from VC Summer 2006

The pressurizer relief valves and sprays maintain primary pressure below 110% of the design value. The pressurizer pressure remains below the safety valve setpoint during the transient. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.



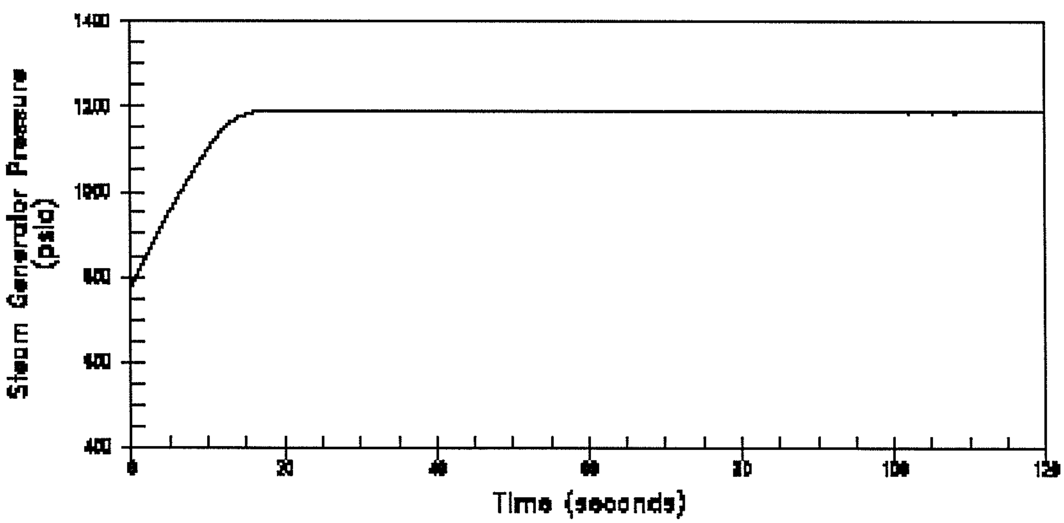


Figure 6- TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD WITH PRESSURE CONTROL MAXIMUM REACTIVITY FEEDBACK (14.1.10-6)

Facility:	VC Summer		Original Item for Question 63
Vendor:	WEC		
Exam Date:	2006		
Exam Type:	R		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
	Group #	2	2
	K/A #	045	A1.05
	Importance Rating	3.8	4.1

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of primary plant parameters (temperature and pressure) following T/G trip

Proposed Question:

Given the following plant conditions:
- An ATWS has occurred from 100% power.
- The CRS has transitioned to *EOP-13.0, Response to Abnormal Nuclear Power Generation*.
- The RO determines that the following occurs in rapid succession:
- RCS temperature and pressure increasing
- PZR PORVs indicate OPEN
- PRT temperature, level, and pressure increasing

Which ONE (1) of the following has occurred and which procedural action is required?

A: The Turbine has tripped. Return to EOP-1.0.

B: The Turbine has tripped. Continue in EOP-13.0

C: The Turbine Driven EFW Pump has tripped. Return to EOP-1.0

D: The Turbine Driven EFW Pump has tripped. Continue in EOP-13.0.

Proposed Answer: B

Explanation (Optional):

A: Incorrect. Indications support turbine trip response, but EOP-13 is still in effect due to rising RCS T and P indicating that the reactor is still acting as a heat

source

B: Correct. Large load rejection

C: Incorrect. TDEFW trip would have effect but MDEFW still available, so not severe enough to cause indications listed. Transition to EOP-1.0 is incorrect.

D: Incorrect. TDEFW trip would have effect but MDEFW still available, so not severe enough to cause indications listed. EOP-13.0 is still in effect.

Technical Reference(s): EOP-13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: 2043 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

NRC Comment: Appears to match K/A. From what power level. If the load on the turbine could be assumed by the steam dumps the loss of the turbine and the TDEFW pump may look about the same. Need to be at a power that is higher than steam dump capacity. BANK

QUESTION 64

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	055	2.2.44
Importance Rating	4.2	

Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: RO Question # 64

Given the following:

- Main Condenser vacuum is at 23" Hg and degrading.
- E 5/3, CONDENSER LO VACUUM, Annunciator is LIT.
- Main Turbine load at 300 MW.

Which ONE of the following identifies (1) any IMMEDIATE operator action ~~to take~~ in accordance with 3-ONOP-014, Main Condenser Loss of Vacuum, and (2) the follow-up action(s) to take if Condenser vacuum continues to degrade?

A. (1) Place the standby set of Air Ejectors in service
(2) Trip the Turbine ONLY

B. (1) Place the SJAE Hogging Jet in service
(2) Trip the Turbine ONLY

C. (1) Place the standby set of Air Ejectors in service
(2) Trip the Reactor AND the Turbine

D. (1) Place the SJAE Hogging Jet in service
(2) Trip the Reactor AND the Turbine

Proposed Answer: D

Explanation (Optional):

A. Incorrect since placing the standby set of Air Ejectors in service is not an IOA. Plausible due to placing an additional set of Air Ejectors in service will improve vacuum. Also, incorrect the Turbine and the Reactor must be tripped when above 10% power. Also plausible because the 2nd part would be correct if power was less than 10%.

B. Incorrect the Turbine and the Reactor must be tripped when above 10% power. Plausible because the 1st part is correct. Also plausible because the 2nd part would be correct if power was less than 10%.

C. Incorrect since placing the standby set of Air Ejectors in service is not an IOA. Plausible due to placing an additional set of Air Ejectors in service will improve vacuum. Plausible because the 2nd part is correct. Also plausible because closing the drain valve is the 1st Subsequent Action and 2nd action overall.

D. CORRECT.

<u>3-ONOP-014:</u>
4.0 IMMEDIATE ACTIONS
4.1 Place the SJAE hogging jet in service as follows:
4.1.1 Open the Steam Supply to Hogging Jet Valve, 3-30-043.
4.1.2 Slowly open Steam Supply to Hogging Jet Valve, 3-30-44, to obtain 250 to 260 psig (3-PI-1597) hogging jet supply pressure.
4.1.3 Open the Condenser Air Removal to Hogging Jet Valve, 3-30-010.

5.5 IF reactor power is greater than 10% (At Power Trips enabled) AND vacuum can NOT be maintained greater than required by Enclosure 1, THEN perform the following:
5.5.1 Trip the Reactor.
5.5.2 Trip the Turbine.
5.5.3 Go to 3-EOP-E-0, Reactor Trip and Safety Injection.

Technical Reference(s): 3-ONOP-014, Main Condenser Loss of Vacuum (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902131, Obj. 10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests interpretation of CR indications (condenser vacuum) and understanding of directives (3-ONOP-014) and their affect on the plant (Turbine and Reactor tripped).

1.0 **PURPOSE**

1.1 This procedure provides instructions to be followed when main condenser vacuum is low.

2.0 **SYMPTOMS**

2.1 Indications

NOTE

Diverse indications of condenser vacuum should be used to validate the loss of vacuum including DCS back pressure.

2.1.1 Main condenser vacuum decreasing.

2.1.2 Main generator load decreasing.

2.1.3 Main condenser air-inleakage greater than 30 SCFM.

2.2 Alarms

2.2.1 E5/3, CONDENSER LO VACUUM.

2.2.2 E5/5, TURBINE GLAND SEAL LO PRESSURE

2.2.3 I3/5, CONDENSER WATER BOX LOW VACUUM

3.0 **AUTOMATIC ACTIONS**

3.1 Turbine trip at 20 inches Hg main condenser vacuum.

4.0 **IMMEDIATE ACTIONS**

CAUTION

Hot water may be emitted from the silencer causing the potential for personnel injury.

4.1 Place the SJAE hogging jet in service as follows:

4.1.1 Open the Steam Supply to Hogging Jet Valve, 3-30-043.

4.1.2 Slowly open Steam Supply to Hogging Jet Valve, 3-30-44, to obtain 250 to 260 psig (3-PI-1597) hogging jet supply pressure.

4.1.3 Open the Condenser Air Removal to Hogging Jet Valve, 3-30-010.

5.0 SUBSEQUENT ACTIONS

NOTES

- *For the remainder of this procedure, the most conservative of the following three indications of Main Condenser vacuum should be used to determine the appropriate actions:*
  PI-3-1612 and PI-3-1406 on VPA
  P1612X_A on DCS
- *P1612X_A on DCS indicates Main Condenser backpressure. To determine vacuum from backpressure:*
  Main Condenser vacuum = (30 in Hg) – (P1612X_A)

5.1 Close Hogging Jet Drain, 3-30-045.

5.2 Notify Chemistry the SJAE Hugging Jet has just been placed in service and this renders the SJAE SPING and PRMS Channel R-15 radiation monitors inoperable. Chemistry shall commence compensatory actions to ensure compliance with Technical Specifications 3.3.3.3, Table 3.3-5, Action 34 and ODCM, Table 3.1-1, Action 3.1.3.

5.3 **IF** only one set of SJAEs is in service, **THEN** place the standby set in service using Attachment 1.

5.4 **IF** vacuum can NOT be maintained by the SJAE hogging jet, **THEN** reduce turbine load as necessary using 3-GOP-103, Power Operation to Hot Standby, **OR** 3-ONOP-100, Fast Load Reduction, to maintain condenser vacuum greater than required by Enclosure 1.

5.5 **IF** reactor power is greater than 10% (At Power Trips enabled) **AND** vacuum can **NOT** be maintained greater than required by Enclosure 1, **THEN** perform the following:

5.5.1 Trip the Reactor.

5.5.2 Trip the Turbine.

5.5.3 Go to 3-EOP-E-0, Reactor Trip and Safety Injection.

5.6 **IF** reactor power is less than or equal to 10% (At Power Trips blocked) **AND** vacuum is less than or equal to 24.5 inches Hg, **THEN** trip the Turbine **AND** stabilize the plant.

5.7 Attempt to identify the cause of the decreasing vacuum by checking the following:

5.7.1 Verify proper operation of the Circulating Water Pumps.

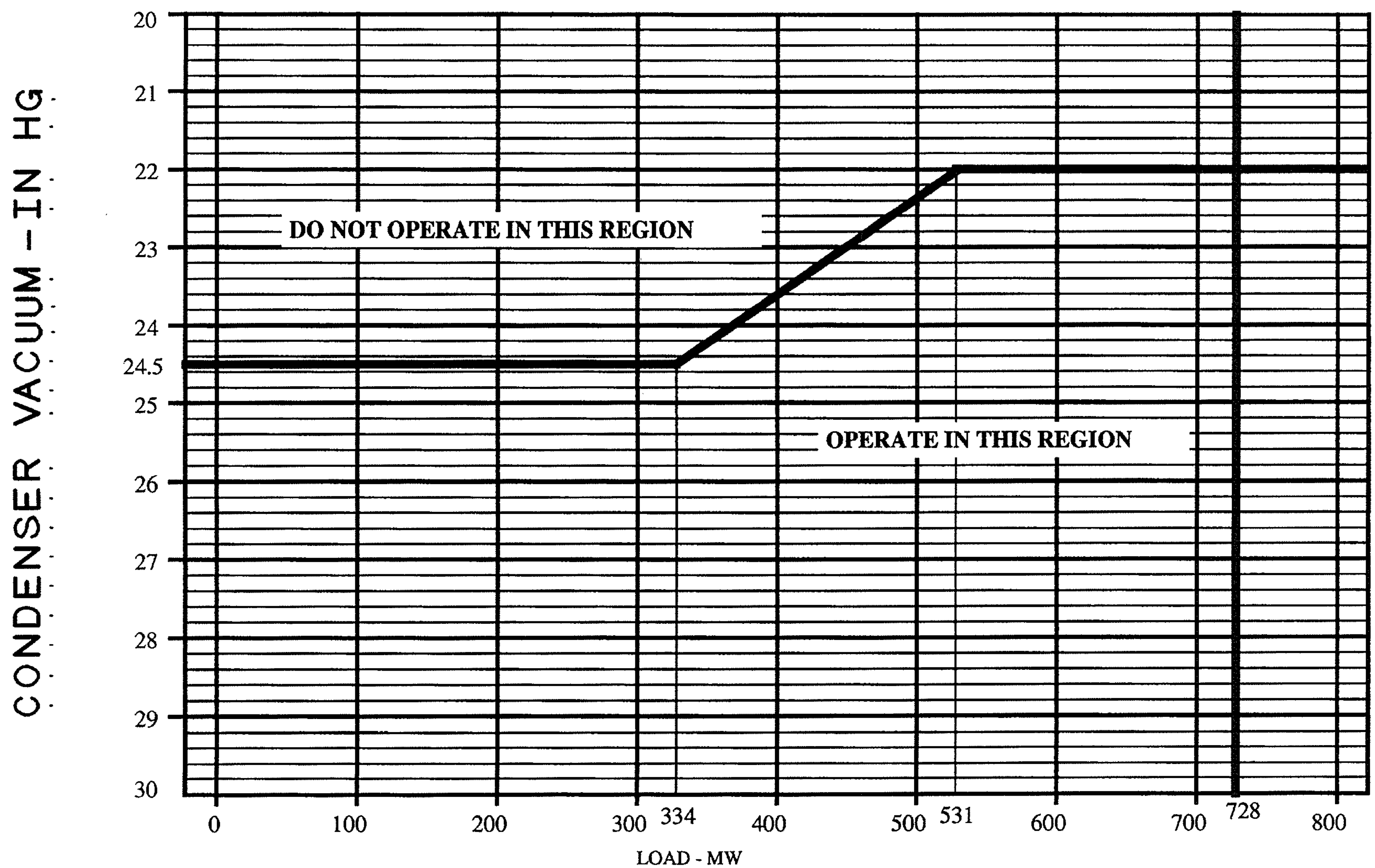
5.7.2 Verify that steam is being supplied and regulated at 260 to 275 psig to both sets of SJAEs.

5.7.3 Verify that the SJAE loop seal is full of water.

a. **IF** the SJAE loop seal is NOT full, **THEN** open Loop Seal Fill Line Isol, 3-30-081, as necessary to fill the loop seal.

ENCLOSURE 1
(Page 1 of 1)

CONDENSER VACUUM LIMITATIONS



QUESTION 65

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068	A2.02
	Importance Rating	2.7	

Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Lack of tank recirculation prior to release

Proposed Question: RO Question # 65

Which ONE of the choices below correctly completes the following statements regarding preparing for a liquid release?

0-NCOP-003, Preparation of Liquid Waste Release Permits, recircs a Waste Monitor Tank for a MINIMUM of ____(1)____ hour(s) on **mini recirc** prior to sample and release.

Without adequate recirc time, the chemist draws a LOW non-representative sample which could cause ____(2)____.

A. (1) one
(2) a release to exceed the radioactive discharge permit to the environment

B. (1) two
(2) the discharge flowrate on the radioactive discharge permit to be lower than required

C. (1) two
(2) a release to exceed the radioactive discharge permit to the environment

D. (1) one
(2) the discharge flowrate on the radioactive discharge permit to be lower than required

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausibility – To potentially exceed the radioactive discharge permit to the environment is a valid concern, however the mini recirc time does not meet the minimum time of 2 hours required by 0-NCOP-003, Preparation of Liquid Waste Release Permits.

B. Incorrect. Plausibility – The discharge flowrate on the radioactive discharge permit to be lower than required is incorrect, but the calculated flowrate is affected and will be higher. Also, the correct mini recirc time of 2 hours is stated in accordance with 0-NCOP-003, Preparation of Liquid Waste Release Permits.

C. Correct.

D. Incorrect. Plausibility – The discharge flowrate on the radioactive discharge permit to be lower than required is incorrect, but the calculated flowrate is affected and will be higher. Also, the mini recirc time does not meet the minimum time of 2 hours required by 0-NCOP-003, Preparation of Liquid Waste Release Permits.

Technical Reference(s): 0-NOP-061.11A - E Controlled Release from Recycle and Waste Monitor Tanks

0-NCOP-003, Preparation of Liquid Waste Release Permits

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902149, Obj. 11 (As available)

Question Source: Bank # 18686

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2003 Indian Point 3

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 13

55.43

Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Comments:

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	068	A2.02
Importance Rating	2.7	

Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Lack of tank recirculation prior to release

Proposed Question: RO Question # 65

Which ONE of the choices below correctly completes the following statements regarding preparing for a liquid waste discharge?

0-NCOP-003, Preparation of Liquid Waste Release Permits, recircs a Waste Monitor Tank for a MINIMUM of ____(1)____ hour(s) on **mini recirc** prior to sample and release.

Without adequate recirc time, the chemist draws a LOW non-representative sample which could cause ____(2)____.

A. (1) one
(2) a release to exceed the radioactive discharge permit to the environment

B. (1) two
(2) the discharge flowrate on the radioactive discharge permit to be lower than required

C. (1) two
(2) a release to exceed the radioactive discharge permit to the environment

D. (1) one
(2) the discharge flowrate on the radioactive discharge permit to be lower than required

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausibility – To potentially exceed the radioactive discharge permit to the environment is a valid concern, however the mini recirc time does not meet the minimum time of 2 hours required by 0-NCOP-003, Preparation of Liquid Waste Release Permits.

B. Incorrect. Plausibility – The discharge flowrate on the radioactive discharge permit to be lower than required is incorrect, but the calculated flowrate is affected and will be higher. Also, the correct mini recirc time of 2 hours is stated in accordance with 0-NCOP-003, Preparation of Liquid Waste Release Permits.

C. Correct.

D. Incorrect. Plausibility – The discharge flowrate on the radioactive discharge permit to be lower than required is incorrect, but the calculated flowrate is affected and will be higher. Also, the mini recirc time does not meet the minimum time of 2 hours required by 0-NCOP-003, Preparation of Liquid Waste Release Permits.

Technical Reference(s): 0-NOP-061.11A - E Controlled Release from Recycle and Waste Monitor Tanks

0-NCOP-003, Preparation of Liquid Waste Release Permits

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902149, Obj. 11 (As available)

Question Source: Bank # 18686

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2003 Indian Point 3

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 13

55.43

Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Comments:

ATTACHMENT 1
(Page 1 of 1)

RADIOACTIVE LIQUID RELEASE PERMIT

FLORIDA POWER AND LIGHT CO **TURKEY POINT PLANT** **RADIOACTIVE LIQUID RELEASE PERMIT**	LRP No. DATE:

Monitor Tank ☐ A ☐B   Waste Monitor Tank ☐ A ☐B ☐ C   Volume to be Released ________ Gals.

Part I - Pre-Release Data and Calculations

Radiochemical Analysis - Specific Activity (Liquid)	μCi/ml
Calculated Activity to be Released	μCi
Estimated Dose for this Release	mR
Month-to-date dose prior to this Release	mR
Total Estimated Dose after this Release	mR
Administrative Release Limit	0.25 mR/month
Σ C/EC Σ C/EC ≤ 1.0	
Dissolved Gas Activity after dilution <2 X 10^{-4} μCi/ml	μCi/ml
Expected R-18/19 Countrate	CPM

Part II - Limits

R-18/19 Background =CPM ( ) R-18/19 Warning =CPM ( )	R-18/19 Setpoint = CPM ( )
Max. Release Flow Rate 100 GPM	Min. No. of CW Pumps
Recirc. Start Time	Sample Time
Recirc. Pump Disch Press ________ psig	Recirculation Pump Flowrate:________ GPM
Notes: Minimum 2 hr recirc. time when using 1 inch mini-recirc. on WMTs.	

Part III - Authorization and Approvals:

The approval of the analysis by the Radiochemist (or designee) shall be obtained if the Specific Activity in Part 1 is greater than or equal to 1 x 10^{-4} μCi/ml. The Shift Manager shall review and sign Attachment 5 ensuring that the tank recirculated was the same tank that was sampled and that the permit was generated for the correct tank.	
Permit Prepared by	Technician
Analysis Approved by	Radiochemistry Supervisor
Release Approved by	SM

Part IV - Release Data

Release Performed By		
No. of Circ. Water pumps in service	Units 1 and 2	Units 3 and 4
Release Date:	Release Start Time:________	Release Stop Time:________
R-18 (or R-19) Readings every 15 min from the start of the release		
Recorder/Meter Readings (CPM)	Maximum	Average
Flow Rate (Estimate) GPM	Level before %	Level after %

ATTACHMENT 5
(Page 1 of 1)

TANK RECIRCULATION AND SAMPLING VERIFICATION SHEET

I have verified that the ________________ tank was placed on recirculation at ______________ on
TIME
__________ with a flowrate of ______________ gpm [for WMT] OR a recirculation pump
DATE
discharge pressure of _____________ psig [for MT].

NOTE

If the WMT is on mini recirc, a minimum 2 hr recirc is required.

This was verified by **all** of the following methods:

a. Valve lineup

b. Logbook entries

c. Review of applicable procedure

Senior Nuclear Plant Operator: ____________________

Date/Time: __________ / __________

I have sampled the __________ tank in accordance with (circle one):

CY-TP-104-0045 0-NCZP-061.2

This tank was sampled at __________ on _________ and was verified to be recirculated
TIME DATE
One tank volume prior to sampling.

Chemistry Technician: ____________________________

Shift Manager or designee: ____

FINAL PAGE

3.0 PREREQUISITES

3.1 Multi channel Analysis System shall be available.

4.0 PRECAUTIONS/LIMITATIONS

4.1 Approval of the Radiochemist (or designee) shall be obtained for a controlled discharge of waste tanks whose contents contain a total non-gas isotopic activity of greater than or equal to 1.0 E-4 microCuries/ml.

4.2 Radionuclides are designated hazardous substances by 40 CFR 302. They require notification of various local, state, and federal agencies if released in a quantity greater than or equal to the levels in the revised Reportable Quantities (RQ) table without a federally issued permit.

4.3 The computer program provides the data for performing Subsection 7.2 when the OpenEMS job stream computer program is available.

4.4 The sum of the ratios between isotopic concentration (C) in a liquid waste release and the effluent concentration (C/EC) shall be less than or equal to one. The effluent concentration (EC) shall be limited to ten times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, and Column 2 for radionuclides other than dissolved or entrained noble gas.

4.5 The Radiochemist, or designee, should be notified if the sum of the ratios between isotopic concentration in a liquid waste release and the effluent concentration (C/EC) is greater than or equal to 0.1.

4.6 All liquid waste tanks potentially contain liquid of moderate to high activity level. Therefore, adhere to the requirements of the appropriate radiation work permit when sampling and analyzing samples.

4.7 All samples shall be analyzed to meet the ODCM Liquid Radiation Effluent Lower Limit of Detection (LLD) requirements.

4.8 For release to unrestricted areas dissolved or entrained noble gas activity shall be less than or equal to 2.0 E- 4 microCuries per milliliter.

4.9 The Radiochemist or designee should be notified when a Waste Monitor Tank or Monitor Tank isotopic analysis exceeds 5.9 E-5 microCuries per milliliter.

4.10 Prior to sampling, the Waste Monitor Tank or Monitor Tank <u>shall</u> be recirculated for a minimum of one tank volume as determined by the appropriate Chemistry Sampling procedure.

QUESTION 66

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	G1	2.1.15
Importance Rating	2.7	

Conduct of Operations: Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Proposed Question: RO Question # 66

In accordance with 0-ADM-200, Conduct of Operations, and 0-ADM-202, Shift Relief and Turnover, which ONE of the following describes (1) the MINIMUM requirement to review the Special Instructions Book and (2) the individual who must APPROVE Special Instructions?

A. (1) prior to assuming EACH shift watch;
(2) Operations Manager

B. (1) ONLY prior to assuming the shift watch on first day back;
(2) Shift Manager

C. (1) ONLY prior to assuming the shift watch on first day back;
(2) Operations Manager

D. (1) prior to assuming EACH shift watch;
(2) Shift Manager

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Per 0-ADM-200, Page 41: "The Special Instruction Book contains instructions that may be issued as needed to identify temporary changes in plant conditions and provides guidance for handling situations that have short-term applicability." Step 4.4 states: "Special Instructions shall be approved by the Operations Manager." Required to be reviewed prior to assuming each shift as part of turnover

B. Incorrect since the Shift Manager (SM) does not approve SIs. Plausible because the 1st part is partially correct, but not totally correct. It must be reviewed on 1st day back but not ONLY. Also plausible because the SM does approve many other aspects of Operations' activities that are addressed in 0-ADM-200. Examples include: 1) two-handed operations (4.4.20.1, Page 40), 2) Core Alterations SRO (Att 4, Page 55), 3) waivers for fire brigade (Page 23), and 4) surveillance tests (Page 43).

C. Incorrect. Plausible because second part is correct and because the 1st part is partially correct, but not totally correct. It must be reviewed on 1st day back but not ONLY.

D. Incorrect since the Shift Manager (SM) does not approve SIs. Plausible because the SM does approve many other aspects of Operations' activities that are addressed in 0-ADM-200. Examples include: 1) two-handed operations (4.4.20.1, Page 40), 2) Core Alterations SRO (Att 4, Page 55), 3) waivers for fire brigade (Page 23), and 4) surveillance tests (Page 43). Also plausible because first part is correct

Technical Reference(s): 0-ADM-200, *Conduct of Operations* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900025, Obj. 5 (As available)

Question Source: Bank # 66395

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 Seabrook

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

5.0 **PROCEDURE**

5.1 Shift Turnover Requirements

5.1.1 Operations Department personnel on the off-going shift should complete and sign the applicable Shift Relief Checklists on a form similar to Attachments 1 through 7 or complete applicable section of the online turnover report.

5.1.2 Under normal conditions, shift turnovers should be made at the operator stations.

5.1.3 The off-going shift should make checks and remarks on the required Shift Relief checklists or online turnover report in such a manner as to inform the oncoming shift of the following (as a minimum):

1. Status of safety-related systems.
2. Running equipment and safety train alignments.
3. Significant or important inoperable equipment, including instrumentation, and limiting conditions for operations, including surveillance requirements.
4. Reasons for new annunciator alarms.
5. Surveillance/equipment work in progress at the time of the shift relief.
6. Unusual events that have occurred during the last 24 hours.
7. If the Plant is in Mode 5 or 6 or is defueled, provide shutdown risk status, current protected/operating train, and expected changes during the next shift.

5.1.4 Prior to assuming the shift, the oncoming operators should review pertinent documents such as logbooks, logsheets, and special instructions as specified in the following substeps and discuss with the shift's supervision watchstanding standards and expectations if not normally assigned to that shift. This discussion need only take place prior to assuming the first watch with a new shift if the operator will be working with that shift continuously for some period of time (e.g., if working 5 consecutive shifts to maintain license active with the same crew, discussion is only required prior to first shift). This discussion is not required for the SM:

1. The oncoming Shift Manager (SM) and Unit Supervisor (US) shall perform the following:
   a. Review the RO Logbooks back to the last shift worked, or, as a minimum, for the previous 24 hours, whichever is shorter. For those cases where individuals have been off shift for a considerable period of time, as far back as necessary to ensure that they are knowledgeable of plant conditions.
   b. Review the Special Instruction Book to ensure that all applicable instructions have been read.
   c. Review the Night Orders to ensure that pertinent information is reviewed with appropriate shift personnel as soon as is practical after shift turnover.

5.1.4.1 (Cont'd)

d. Review the Watchstander Out-of-Service Book to ensure that the individuals assigned to stand shift are qualified to assume the shift. [Commitment Step 2.3.1]

e. Review the Equipment Out-of-Service Books.

f. Ensure required re-qualification is up to date. Requirements can be found in 0-ADM-311, Licensed Operator Training Program.

g. Verify all LMS requirements are up to date.

h. Ensure any required Essential Job Functions due to medical issues have been completed and turned in to HR Department. SM concurrence is mandatory prior to assuming any operations functions.

i. Verify all respirator requirements are current, including Respirator Fit Test, Annual Respirator JPM, and Annual Medical Physical.

j. If any item(s) could not be completed, notify the Shift Manager (SM).

2. Oncoming Field Supervisor shall perform the following:

a. Review the RO Logbooks back to the last shift worked, or as a minimum, for the previous 24 hours, whichever is shorter. For those cases where an individual has been off shift for a considerable period of time, as far back as necessary to ensure that they are knowledgeable of plant conditions.

b. Review the Special Instruction Book to ensure that all applicable instructions have been read.

c. Review the Watchstander Out-of-Service Book to ensure that the individuals assigned to stand shift are qualified to assume the shift. [Commitment Step 2.3.1]

d. Review the Equipment Out-of-Service Books.

e. Ensure required re-qualification is up to date. Requirements can be found in 0-ADM-312, Nuclear System Operator (NSO) Initial Training and Continued Training Program.

f. Verify all LMS requirements are up to date.

g. Ensure any required Essential Job Functions due to medical issues have been completed and turned in to HR Department. SM concurrence is mandatory prior to assuming any operations functions.

h. Verify all respirator requirements are current, including Respirator Fit Test, Annual Respirator JPM, and Annual Medical Physical.

i. If any item(s) could not be completed, notify the Shift Manager (SM).

4.5 Shift Records (continued)

2. Operations Narrative Logbooks - Operations Narrative Logbooks shall be maintained in accordance with 0-ADM-204, Operations Narrative Logbooks.

4.6 <u>Special Instruction Book</u>

1. The Special Instruction Book is maintained to improve communications between the plant management staff and shift personnel in the control Room.

> **CAUTION**
>
> Special Instructions shall **NOT** be used to circumvent established procedural requirements, nor shall Special Instructions be construed to supersede procedural requirements.

2. The Special Instruction Book contains instructions that may be issued as needed to identify temporary changes in plant conditions and provides guidance for handling situations that have short-term applicability.

3. The SM is responsible for informing members of their crew as appropriate.

4. Special Instructions shall be approved by the Operations Manager.

5. The AOM or designee shall review the Special Instruction Book at least once per quarter to verify the validity and currency of all active instructions.

6. Copies of each new instruction placed in the Special Instruction Book shall be forwarded to the QC Supervisor and to the Plant General Manager.

4.7 <u>Night Orders Book</u>

> **CAUTION**
>
> Night Orders shall **NOT** provide Plant Operating Procedures.

1. The Night Order Book is used to provide the following types of information:

   A. Notification of important procedure changes, such as changes to Immediate Operator Actions

   B. Operations or tests to be performed (using approved procedures)

   C. Communication between management and the operating shifts

   D. Upcoming events, notification of policy changes

4.7 Night Orders Book (continued)

1. (continued)

- **E.** Problems encountered during operations
- **F.** Short-term information

2. Night Orders that are expected to last longer than 30 days should be incorporated into plant programs such as log readings, schedules, procedures, department instructions, etc.

3. Night Order entries over 30 days old are expired.

4. Night Order Book entries may be made by the Assistant Operations Manager (AOM), Shift Manager (SM), Unit Supervisor (US), Field Supervisor, or as designated by the AOM.

4.8 Operations Testing

The Operations Manager is responsible for the overall control and coordination of all surveillance testing.

4.8.1 Operations Surveillance Tests

1. The SM's approval shall be obtained prior to the commencement of any surveillance testing. If a step is **NOT** completed for any reason, then the reason must be recorded on the procedure. When a test is performed and does **NOT** meet the specified acceptance criteria, the SM, US, or Field Supervisor shall be notified and corrective action shall be initiated.

2. The SM or his/her SRO designee should review all Operations Department surveillance tests performed on his/her shift for completeness and accuracy and shall so indicate by signing and dating the procedure in the appropriate space. In addition, he/she shall ensure that completion of the test(s) is documented in accordance with the appropriate test scheduling procedures.

4.9 Licensed Operator Medical Status Reporting

1. Each licensed operator is required to report all changes to their medical status that may affect the condition of their license in accordance with 0-ADM-342, Tracking of Licensed Operators and Licensed Candidates.

2. The change in medical status report shall be made for changes that may be an improved condition, i.e., laser surgery and corrective lenses are **NO** longer required, or a degenerative condition, i.e., heart problem.

4.10 Watchstander Out-of-Service Book

1. The Watchstander Out-Of-Service Book is maintained to provide a list of operators who are **NOT** able to assume the watchstander duties for which they are normally qualified.

2. The Watchstander Out-Of-Service Book is to be maintained on a form similar to Attachment 5, Watchstander Out-of-Service Book.

3. Entries to The Watchstander Out-Of-Service Book are made for the following:

   A. Failure to meet a training evaluation (written exam, simulator evaluation, task performance).

   B. Insufficient proficiency hours.

ATTACHMENT 4
Core Alterations SRO and Refueling Supervisor Record
(Page 1 of 1)

CORE ALTERATIONS SRO RECORD

SRO NAME ______________________ ______________
DATE

SRO NAME ______________________________

ACTIVE PTN SRO	YES ☐	NO ☐	
8-HOURS OF INSTRUCTION REQUIRED	YES ☐	NO ☐	NOTE: REQUIRED IF SRO HAS NOT HAD PREVIOUS REFUELING EXPERIENCE
APPROVED AS CORE ALTERATIONS SRO	YES ☐	NO ☐	______________ SHIFT MANAGER

REFUELING SUPERVISOR RECORD

SUPERVISOR NAME ______________________ ______________
DATE

ACTIVE PTN SRO	YES ☐	NO ☐	
SRO SIMILAR FACILITY	YES ☐	NO ☐	______________ FACILTY NAME
SPECIALIZED EXPERIENCE	YES ☐	NO ☐	______________ OPERATIONS MANAGER
APPROVED AS REFULEING SUPERVISOR – AT LEAST ONE OF THE ABOVE SATISIFED	YES ☐	NO ☐	______________ SHIFT MANAGER

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G1	2.1.27
	Importance Rating	3.9	

Conduct of Operations: Knowledge of system purpose and / or function.

Proposed Question: RO Question # 67

Given the following plant conditions:

- The Unit 3 is at 46% power and increasing in accordance with 3-GOP-301, Hot Standby To Power Operation
- A loss of Feedwater occurs.
- All three S/G NR levels are at 5% and lowering.

After 60 seconds, which ONE of the following identifies some of the expected responses of the ATWS Mitigation System Actuation Circuitry (AMSAC)?

A. AMSAC will NOT actuate because it is not armed.

B. AMSAC will directly TRIP the Reactor Trip Breakers and START the AFW Pumps.

C. AMSAC will NOT actuate because arming circuit has not timed out.

D. AMSAC will actuate and TRIP the Main Turbine and START the AFW Pumps.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Arms at approx 40%.

B. Incorrect. Above 40% power equivalent on First Stage pressure, AMSAC will actuate, but won't directly TRIP the Reactor and START the AFW Pumps.

C. Incorrect. Time delay for power is 360 seconds after going BELOW 40% to ensure transients

do not preclude initiation.

D. Correct. AMSAC will actuate and TRIP the Main Turbine and START the AFW Pumps.

Technical Reference(s): SD-013, CVCS
LP 6902113, CVCS (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902113, Obj. 5 (As available)

Question Source: Bank # 47874
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2003 Indian Point 3

Question Cognitive Level: Memory or Fundamental Knowledge (1B)
Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 3
55.43

Mechanical components and design features of reactor primary system.

Comments:

REACTOR PROTECTION AND SAFEGUARDS ACTUATION SYSTEM

Design Basis

AMSAC meets 10 CFR 50.62 criteria to initiate a turbine trip, actuate auxiliary feedwater, isolate blowdown and blowdown sampling isolation upon sensing of an ATWS. In addition to these 10 CFR 50.62 requirements, AMSAC also initiates a reactor trip by opening the output breakers from the Control Rod Drive Motor Generator sets which removes power to the rod control system and allows the RCCAs to fall into the reactor. AMSAC is required to protect against a loss of feedwater ATWS (the most limiting case), and may not actuate for other less significant transients.

AMSAC is also designed to preclude inadvertent actuation at power. In most failure modes, failure of inputs or power disables the system. AMSAC energizes to actuate. AMSAC is designed as a dual processor system which requires both processors to actuate (in normal lineup) to yield an actuation signal. However, one processor can be removed from service without disabling AMSAC. AMSAC can be tested at power.

General Description

As discussed above, AMSAC acts to back up the RPS in case of an ATWS. AMSAC senses whether the reactor is (or has been within last 360 seconds) at power (via 2/2 turbine first stage pressure ≥ 35.98% power) and also senses whether steam generator levels have dropped significantly below the reactor trip setpoint (2/3 S/G levels ≤ 8.65%). If these criteria are made up, after a time delay (25 seconds to allow RPS actuation) AMSAC actuates and initiates AFW (which also isolates blowdown and blowdown sampling), and opens the CRDM MG set output breakers to trip the reactor.

Rod Drop Indication

This function no longer initiates a runback or affects rod motion (since auto rod withdrawal is deactivated) on either unit (now indication only). Rod drop is detected by any 1/4 power range nuclear instruments sensing a power reduction of 5% power within 5 seconds or the rod position indication system indicating a single rod on the bottom (within 20 steps of full insertion).

Note: Continued operation with one dropped rod may or may not be permitted due to detrimental effects on core radial power distribution. The potential exists for exceeding core hot channel factor limits due to local power peaking in fuel elements adjacent to the element with the dropped rod. A flux map and determination of rod worth are required. Operation with two or more inoperable rods is not permitted. Refer to section 3.1.3 of Technical Specification.

OTΔT and OPΔT Rod Stops

Both of these protection functions are in a 2/3 coincidence. When the setpoints are reached, automatic and manual rod withdrawal are blocked.

Main Feedwater Pump Trip Turbine Runback

The main feedwater pump trip turbine runback is actuated by a loss of one main feedwater pump with turbine load greater than 45% (sensed by turbine first stage pressure). No rod stops are associated with this runback. The turbine governor will be runback at 200% per minute until the turbine load is below 45%. This runback can be manually defeated with a selector switch on the console.

ATWS MITIGATING SYSTEM

Analysis & Set Points

STEAM GENERATOR LEVEL: AMSAC conceptual design specifies that AMSAC will actuate on a low-low S/G level of 5% of narrow range, provided that all permissives are met. Considering

instrument loop uncertainties and to insure the set point never is less than 5% the actual setpoint for AMSAC actuation is 8.65%. EOP-FR-S.1 is entered when a reactor trip should have occurred and a manual trip is not effective. Procedure directs the operator to maintain 6% S/G level, therefore an AMSAC trip of 8.65% is consistent with EOP-FR-S.1.

STEAM GENERATOR TIME DELAYS: A time delay allows the RPS to actuate prior to the AMSAC system while also permitting temporary unstable steam generator levels to stabilize without causing AMSAC actuation.

TURBINE FIRST STAGE PRESSURE: AMSAC armed at 40% turbine load has been established as the setpoint to ensure spurious AMSAC actuations do not occur at low power levels or during startups or shutdowns. Westinghouse analyses have confirmed that the peak RCS pressure resulting from an ATWS at >70% load will not exceed the ASME level C service limit of 3200 PSIG, as used by the NRC. However, as the pressure decreases, there will be bulk boiling of the reactor coolant system inventory after the ATWS peak pressure even with operator intervention. Therefore, considering instrument loop uncertainties, the setpoint for First Stage Pressure input arming signal to AMSAC is set at 35.98%, below this value the system is unarmed.

AMSAC ARMED TIME DELAY: The time delay of 360 seconds ensures that the AMSAC system can actuate following a turbine trip if an anticipated transient has occurred.

Locations

The AMSAC system panels (3C391 and 4C391) are located in the cable spreading room, directly west of 4QR35 and 4QR36, and are seismically supported, to prevent interaction with other equipment. AMSAC system status is displayed by way of a see-through panel on the front of the cabinet. AMSAC is a backup system for the reactor protection system during an ATWS event. This system is an enhancement to the existing RPS system and provides additional assurance of shutting down the reactor if the RPS fails. The AMSAC system receives safety related signals. Therefore, the cabinet is divided into safety and non-safety related sections and maintains wiring separation between class 1E and non-class 1E circuits.

REACTOR PROTECTION AND SAFEGUARDS ACTUATION SYSTEM

Alarms And Indications (AMSAC Panel)

1. (A, B) Tripped

   The indication comes on when the associated processor sees 2/3 steam generator levels below the set point for 25 seconds or more. It also comes on if actuating conditions are simulated when testing the logic.

2. Power Lamps (A or B)

3. Test Lamp, processor (A or B)

   Comes on during auto diagnostic or during manual test of processor.

4. Time Triggered Lamp on processor (A or B)

   The lamp lights when the 25 second timer is counting down, and goes out when reset or when timer has timed out.

5. Variable tripped lamp (A or B)

   The lamp lights when one or more of the level variables in bypass and lights when level variable in bypass.

6. (A or B) armed

   Both lights will be on if in the dual processor mode. This is the only applicable light, if in single processor mode.

Power Supplies

120 V uninterruptable power from 3P08-8 (4P08-8) is supplied to the input isolators (see signal inputs below). These are the only components in the cabinets to receive this power source.

120 V non-vital uninterruptable power is supplied from 3P31-13 (4P31-22) is supplied to all other components in the AMSAC cabinet.

120 V non-vital power for panel lights only is supplied from LP50-28. This source provides power for both AMSAC panels 3C391 and 4C391.

Signal Inputs

Each AMSAC input is isolated from its respective Hagan loop input by an electrical isolator. Each level or pressure input is provided with a visible indication that the Hagan loop bistable device has tripped. Also each individual processor unit has lights to indicate processor trouble.

Steam Generator Level Signals

1. LT-3-474 (channel 1)
2. LT-3-484 (channel 1)
3. LT-3-494 (channel 1)
   and
4. LT-3-475 (channel 2)
5. LT-3-485 (channel 2)
6. LT-3-495 (channel 2)

These signals are fed to the isolator which isolate the safety related signal from the steam generators to the non-safety related AMSAC circuitry.
First Stage Turbine Pressure

1. PT-3-446 (channel 3)
2. PT-3-447 (channel 4)

First Stage Turbine Pressure is also fed to isolators.

Control Board Indication

AMSAC Armed, single processor mode, green, 3C04 (4C04)
AMSAC Armed, dual processor mode, green, 3C04 (4C04)
AMSAC Trouble, Amber 3C04(4C04), AMSAC TROUBLE/ACTUATED Annunciator (D-7/6)
AMSAC Actuated, Red 3C04(4C04), AMSAC Trouble/Actuated Annunciator (D-7/6)

AMSAC Reset Pushbutton, resets AMSAC. Pushbutton in Control Room does not reset AMSAC panel light in Cable Spreading Room. Must reset AMSAC locally to reset Panel lights on 3C391 and 4C391.

Actuation Signal Outputs

Trips Control Rod M. G. Set Output Breakers 3S6 (4S6) and 3S7 (4S6) in panel 3C31/4C31.

Energize AFW Autostart Relays

AMSAC is connected to the AFW in racks:

1. 3QR50 & 3QR51 for Unit 3
2. 4QR50 & 4QR51 for Unit 4

The AFW initiation closes steam generator blowdown isolation CVs and also closes steam generator sample MOVs.

Energize Turbine Primary and Backup Auto Stop Solenoids

AMSAC connected in parallel to circuit on 3C02 and 4C02 to trip 20AST and 20ASB.
Alarms "AMSAC TROUBLE/ACTIVATED" Annunciator D-7/6, 3C04 (4C04), panel "D", "CONDENSATE AND FEEDWATER"

This alarm sounds under the following conditions:

1. Processor A or B trouble (can be caused if a steam generator level deviates more than 10% from the other steam generator levels)
2. AMSAC bypassed
3. AMSAC actuated
4. Loss of voltage to the safety circuit inputs or the vital/non-vital power supplies

QUESTION 68

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	G1	2.1.28
Importance Rating	4.1	

Conduct of Operations: Knowledge of the purpose and function of major system components and controls.

Proposed Question: RO Question # 68

Given the following:

- A loss of instrument air is in progress on Unit 4.
- Instrument Air header pressure is 70 psig and lowering slowly as read on PI-4-1444.

Which ONE of the following describes the STATUS of (1) CV-4-1605, Distribution Header Pressure Control Valve and (2) Annunciator - INSTR AIR SYSTEM HI TEMP/PRESS LOW (I 6/1)?

A. (1) CLOSED to ensure all stored air is available to the Instrument Air System on Unit 3;
(2) Alarm is NOT lit.

B. (1) OPEN to supply the Instrument Air Header from Service Air;
(2) Alarm is NOT lit.

C. (1) CLOSED to ensure all stored air is available to the Instrument Air System on Unit 3.
(2) Alarm is lit.

D. (1) OPEN to supply the Instrument Air Header from Service Air.
(2) Alarm is lit.

Proposed Answer: C

Explanation (Optional):

A. Incorrect because the annunciator actuates at 95 psig, so it will be lit at 70 psig. Plausible because the 1st part is correct. Also plausible because Step 2 of 0-ONOP-013 requires the crew to maintain IA pressure greater than 65 psig. Step 13 requires the same for maintaining Aux Building pressure. If applicant only remembered this "setpoint," the applicant may assume that the annunciator will not actuate until pressure drops an additional 5 psig.

B. Incorrect since CV-4-1605 will close, not open when pressure is below 75 psig. Plausible because, according to RNO Step 4 of 0-ONOP-013, the Service Air System *can* be aligned to backup the IA System. Also plausible because Step 2 of 0-ONOP-013 requires the crew to maintain IA pressure greater than 65 psig. Step 13 requires the same for maintaining Aux Building pressure. If applicant only remembered this "setpoint," the applicant may assume that the annunciator will not actuate until pressure drops an additional 5 psig.

C. CORRECT. The Unit 3 and 4 headers are cross-connected by two control valves and two check valves. These control valves (CV-3/4-1605) operate to isolate a faulted instrument air system, throttling closed between 88 and 75 PSIG, decreasing. If one unit suffers a major line break, its CV-3/4-1605 valve will go closed. The non-faulted unit is now supplying air both to its loads and to the affected unit via the check valve. As the non-faulted units instrument air pressure drops, its CV-3/4-1605 valve will also throttle closed until closure allows pressure to be maintained greater than or equal to 75 PSIG.

   0-ONOP-013, Symptom/Entry Condition 2.5, states: "Distribution Header Pressure Control Valve, CV-*-1605 throttles closed (occurs at 88 psig) to protect the unaffected unit. Per 4-ARP-097.CR.I, 6/1, the INSTR AIR HI TEMP/PRESS LOW actuates at 95 psig.

D. Incorrect since CV-4-1605 will close, not open when pressure is below 75 psig. Plausible because the 2nd part is correct. Also plausible because, according to RNO Step 4 of 0-ONOP-013, the Service Air System *can* be aligned to backup the IA System.

Technical Reference(s): 0-ONOP-013, *Loss of Instrument Air*
LP 6900145, *Instrument Air System* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900145, Obj. 7 (As available)

Question Source:	Bank #	63614	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2007 Harris

Question Cognitive Level:	Memory or Fundamental Knowledge	(1I)
	Comprehension or Analysis	
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

#68

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *Both Control Valves CV-3/4-1605 will start to close when pressure decreases to less than 88 psig. The affected unit's control valve will close, while the unaffected unit's control valve will be throttled.*
- *An abnormal pressure difference equal to or greater than 10 psig between header manifold pressure readings may establish the affected unit and general area if Instrument Air press is greater than 88 psig.*
- *Steps 11 and 12 should be used to assist in limiting the search area and not for isolation of Air Headers.*

11 **Identify The Affected Unit Locally Checking The Following At The Instrument Air Automatic Cross-Tie Valve Manifold**

- PI-3-1615
- CV-3-1605
- PI-4-1615
- CV-4-1605

12 **Identify The Affected Area As Indicated By Locally Checking The Instrument Air Header Manifold**

- PI-*-1516, Turbine Area
- PI-*-1517, Containment Area
- PI-*-1518, Auxiliary Building, Intake Area and Control Room

1.0 PURPOSE

1.1 This procedure provides for operator actions in the event of a loss of the Instrument Air System to a single unit or both units.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 Annunciator I 6/1, INSTR AIR HI TEMP/LO PRESS, is in alarm.

2.2 Instrument air pressure of less than 95 psig as indicated on PI-3-1444 (3-VPA) or PI-4-1444 (4-VPA).

2.3 Failure of components supplied by instrument air (e.g., unexplained letdown isolation valve closure, a loss of feedwater flow with full demand on the controllers, charging pump(s) failed to high speed.)

2.4 Failure of an Instrument Air line (leak or rupture).

2.5 Distribution Header Pressure Control Valve, CV-*-1605 throttles closed (occurs at 88 psig) to protect the unaffected unit.

2.6 Backup Compressor(s) started automatically.

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 FSAR

1. Section 1.9, Quality Assurance Program
2. Section 7.7, Operating Control Stations

3.1.2 Technical Specifications

1. Section 3.0, Applicability
2. Section 3.7.1.5, Main Steam Isolation Valves

3.1.3 Operating Diagrams

1. 5613-M-3013, Sheet 1, Instrument Air System - Compressors
2. 5613-M-3013, Sheet 5, Turbine Building
3. 5614-M-3013, Sheet 1, Instrument Air System - Compressors
4. 5614-M-3013, Sheet 5, Turbine Building

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Attachment 1 provides instructions on how to manually start a Temporary Diesel Instrument Air Compressor.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Start Any Available Instrument Air Compressor**	IF an Instrument Air Compressor cannot be started, **THEN** perform the following: 1. Verify the Service Air System is in service. 2. Verify both Service Air Compressors are running. 3. Open the four inch Service Air Supply to Unit 3/Unit 4 Tie Valve, 40-2059 on the mezzanine SW of 3A Heater Drain Tank. OR 4. Open the two inch Service Air Supply to Unit 3/Unit 4 Tie Valve, 40-215 on the mezzanine South of 3A Heater Drain Tank. 5. Notify Maintenance to restore an Instrument Air Compressor to service.
5	IF any Instrument Air Compressor is suspected to be the source of the Instrument Air loss, **THEN** perform the following: * For 3CM, Close IAC 3CM Discharge Isol Vlv IAS-3-058 * For 3CD, Close IAC 3CD Discharge Isol Vlv IAS-3-059 * For 4CM, Close IAC 4CM Discharge Isol Vlv IAS-4-058 * For 4CD, Close IAC 4CD Discharge Isol Vlv IAS-4-059	

CAUSES:
1. Instrument air low pressure
2. Instrument air high temperature
3. Loss of power to AN-4-IA (Bkr 4D31-4)

I6/1

INST AIR SYSTEM HI TEMP/ LO PRESS

DEVICE:	SETPOINT:	LOCATION:
• PS-4-2004	95 psig	At air filters
• PS-4-2035	95 psig	At air dryers
• TS-4-2104	110°F	At air receiver
• AN-4-IA	Loss of Power	

ALARM CONFIRMATION

1. **CHECK** the following:
   - Instrument air pressure indication, PI-4-1444 on VPA
   - Local Annunciator Panel AN-4-IA at South end Unit 4, 18 foot elevation for specific cause of alarm AND **ACKNOWLEDGE** alarm

OPERATOR ACTIONS

1. IF a loss of instrument air exist, THEN **GO TO** 0-ONOP-013, Loss of Instrument Air.
2. **DISPATCH** operator to locally check operation of instrument air compressors
3. **START** standby air compressors AND **STOP** faulty compressors as required to maintain system operability.
4. IF high temperature cause of alarm, THEN **CHECK** operation of after cooler(s).

REFERENCES:
1. FPL EWD 5614-E-27, Sh 26D
2. FPL DWG 5614-M-3013, Sh 1
3. PC/M 93-109, Instrument Air System Compressor Upgrade

INSTRUCTOR ACTIVITY

Each units instrument air header is divided into branch lines supplying the areas listed below:

a. Steam dump valve accumulators

b. Turbine area

c. Intake structure and water treatment plant

d. Auxiliary building and control room

e. Containment and blowdown area

B. The air supplied to the intake/WTP and the control room/auxiliary building is supplied from both units to preclude losing air to those areas if instrument air is lost to a single unit. Check valves are provided to prevent the opposite units instrument air system from being supplied by from these headers. Normally open manual isolation valves are also installed on the supply lines to these headers.

C. Figure 1 illustrates a simplified diagram of the instrument air system. The Unit 3 and 4 headers are cross-connected by two control valves and two check valves. These control valves (CV-3/4-1605) operate to isolate a faulted instrument air system, throttling closed between 88 and 75 PSIG, decreasing. If one unit suffers a major line break, its CV-3/4-1605 valve will go closed. The non-faulted unit is now supplying air both to its loads and to the affected unit via the check valve. As the non-faulted units instrument air pressure drops, its CV-3/4-1605 valve will also throttle closed until closure allows pressure to be maintained greater than or equal to 75 PSIG.

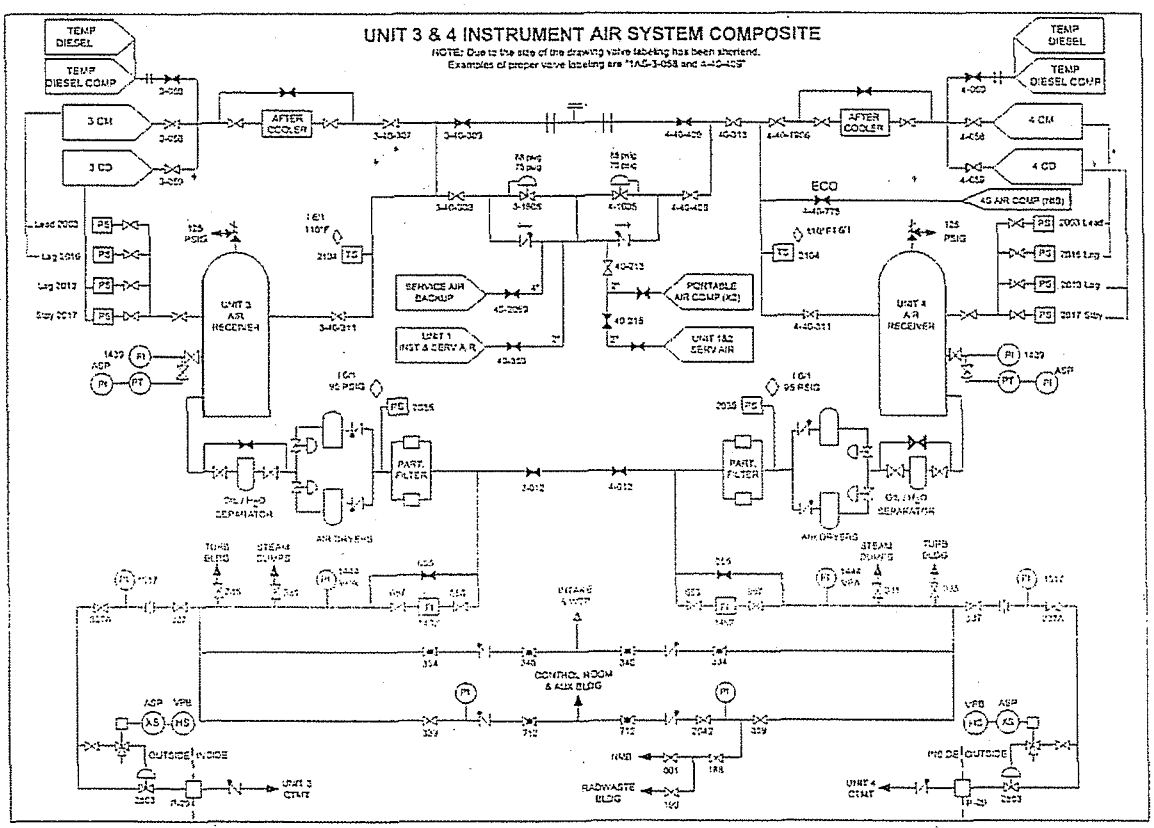
D. Normal system pressure is maintained above 95 PSIG on the header, with the receiver at 104-110 PSIG.

Each unit's instrument air header is divided into branch lines supplying the areas listed below:

1. Steam dump valve accumulators
2. Turbine area
3. Intake structure and water treatment plant
4. Auxiliary building and control room
5. Containment and blowdown area

The air supplied to the intake/WTP and the control room/auxiliary building is supplied from both units to preclude losing air to those areas if instrument air is lost to a single unit. Check valves are provided to prevent the opposite units instrument air system from being supplied by from these headers. Normally open manual isolation valves are also installed on the supply lines to these headers.

Figure 1 illustrates a simplified diagram of the instrument air system.



UNIT 3 & 4 INSTRUMENT AIR SYSTEM COMPOSITE

NOTE: Due to the size of the drawing valve labeling has been shortend.
Examples of proper valve labeling are "IAS-3-058 and 4-40-409"

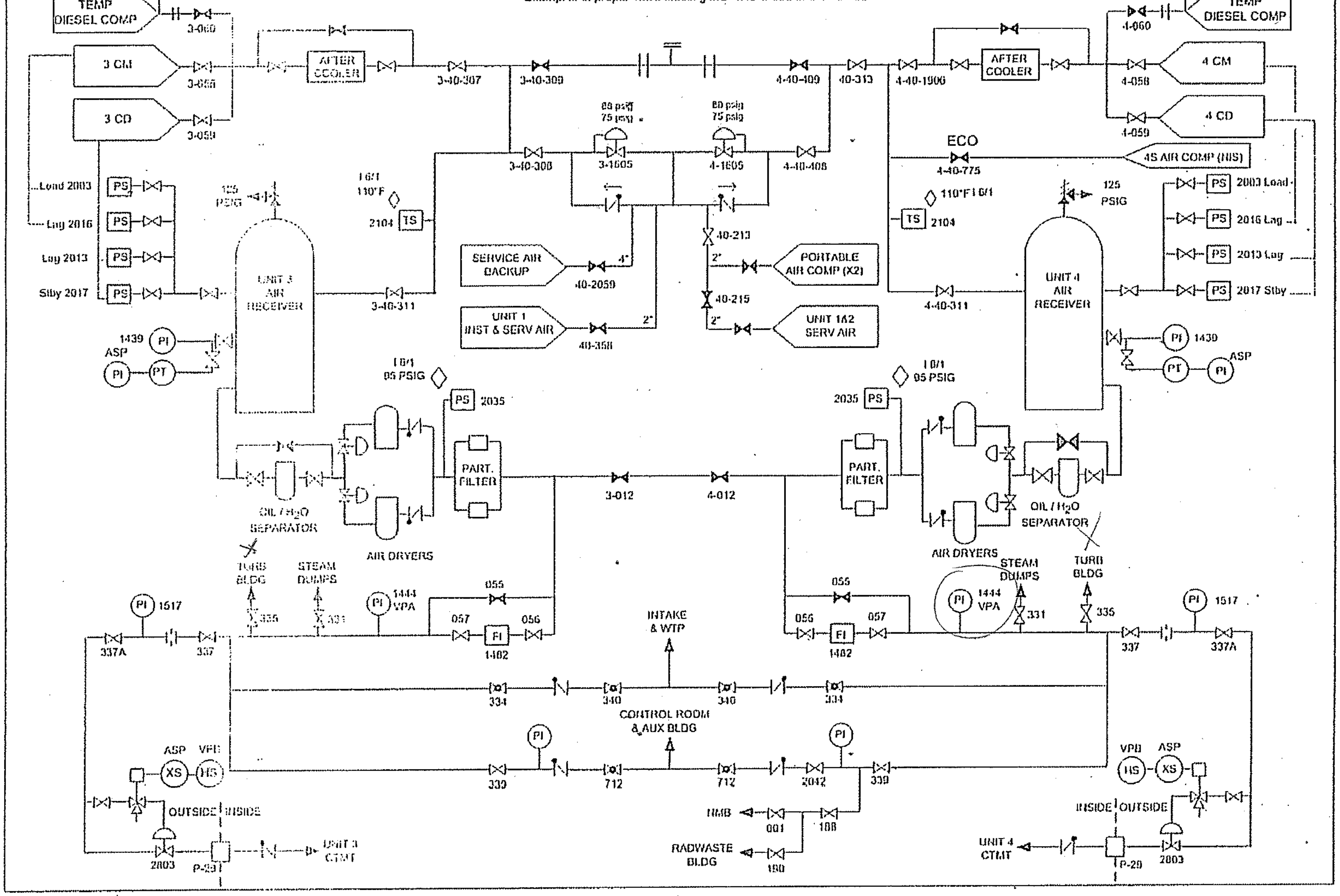


Figure 1

QUESTION 69

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	2	
K/A #	G2	2.2.22
Importance Rating	4.0	

Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Proposed Question: RO Question # 69

Given the following conditions:

- Unit 3 is shutdown.
- RCS temperature is 400F.
- A compounded system malfunction occurred where RCS Pressure Control is unavailable.
- All Pressurizer PORVs and Safeties are NOT opening.
- RCS Pressure is 2785 psig and stable.

Which ONE of the following completes the following statement?

The Reactor Coolant System Pressure Safety Limit of ___(1)___ is exceeded. With the current MODE of operation, the required ACTION is to restore within limits in ___(2)___.

	(1)	(2)
A.	2485 psig;	5 minutes
B.	2485 psig;	1 hour
C.	2735 psig;	5 minutes
D.	2735 psig;	1 hour

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The candidate recalls the correct Action time associated with MODE 3, 4, and 5 operation. Incorrectly, they believe the RCS Pressure Safety Limit is 2485 psig which is the same as the Pressurizer Safety Valves.

B. Incorrect. Plausibility – The candidate believes the RCS Pressure Safety Limit is 2485 psig which is the same as the Pressurizer Safety Valves. Also, they use the Action time associated with MODE 1 & 2 operation.

C. CORRECT.

D. Incorrect. Plausibility – The candidate correctly recalls the correct RCS Pressure Safety Limit. However, they use the Action time associated with MODE 1 & 2 operation.

Technical Reference(s): Technical Specifications, 2.1 Safety Limits (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902521, Obj. 1 (As available)

Question Source: Bank # 62494

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Farley

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis 2DR

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating limitations in the technical specifications and their bases.

Comments:

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.

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.

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.

ATTACHMENT 1
(Page 3 of 112)

TECHNICAL SPECIFICATION BASES

2.1.1 (Cont'd)

These curves are based on an enthalpy hot channel factor, $F_{\Delta H}^{N}$, and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F_{\Delta H}^{N}$ at reduced power based on the expression:

$$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP}[1+ PF_{\Delta H}(1\text{-}P)]$$

Where P is the fraction of RATED THERMAL POWER

$F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT.

$PF_{\Delta H}$ = Power Factor multiplier for $F_{\Delta H}$ as specified in the CORE OPERATING LIMITS REPORT.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit assuming the axial power imbalance is within the limits of the f (ΔT) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core Safety Limits.

Fuel rod bowing reduces the values of DNB ratio (DNBR). The penalties are calculated pursuant to Fuel Rod Bow Evaluation, WCAP-8691-P-A Revision 1 (Proprietary) and WCAP-8692 Revision 1 (Non-Proprietary). The restrictions of the Core Thermal Hydraulic Safety Limits assure that an amount of DNBR margin greater than or equal to the above penalties is retained to offset the rod bow DNBR penalty.

2.1.2 Reactor Coolant System Pressure

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The RCS piping, valves and fittings are designed to ANSI B31.1, which permits a maximum transient pressure of 120% of design pressure of 2485 psig. The Safety Limit of 2735 psig is therefore more conservative than the ANSI B31.1 design criteria and consistent with associated ASME Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure to demonstrate integrity prior to initial operation.

QUESTION 70

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2	2.2.43
	Importance Rating	3.0	

Equipment Control: Knowledge of the process used to track inoperable alarms.

Proposed Question: RO Question # 70

Given the following:

- Unit 4 is in a refueling outage.
- A Control Room annunciator is to be defeated for a clearance order.

In accordance with 0-ADM-219, Annunciator Response Procedure Usage, which ONE of the choices below completes the following:

____(1)____ , then ____(2)____, the applicable portions of 0-OSP-200.5, Miscellaneous Tests, and Operating Evolutions, for Defeated/Out-Of-Service Annunciators must be completed.

A. (1) When defeating any annunciator associated with any system
(2) within ONE hour

B. (1) When defeating any annunciator associated with any system
(2) PRIOR to defeating the annunciator

C. (1) When defeating an annunciator with systems that are required to remain operable
(2) within ONE hour

D. (1) When defeating an annunciator with systems that are required to remain operable
(2) PRIOR to defeating the annunciator

Proposed Answer: D

Explanation (Optional):

A. Incorrect since the 0-ADM-219 actions only apply to those systems that are required to be operable in Mode 6. Plausible because the 2nd part is correct for an annunciator associated with a nuisance alarm.

B. Incorrect since the 0-ADM-219 actions only apply to those systems that are required to be operable in Mode 6. Plausible because 2nd part is correct.

C. Incorrect since the within ONE hour requirement is only applicable for an annunciator associated with a nuisance alarm. Plausible because the 1st part is correct.

D. CORRECT per the requirements of 0-ADM-219, Annunciator Response Procedure Usage. Additionally, for a clearance order, the requirements are to complete 0-OSP-200.5, Miscellaneous Tests, and Operating Evolutions, for Defeated/Out-Of-Service Annunciators prior to defeating the annunciator.

Technical Reference(s): 0-ADM-219, Annunciator Response Procedure Usage (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900041, Obj. 2 (As available)

Question Source: Bank # 59738

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2008 Diablo Canyon

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

#70

4.0 INSTRUCTIONS (continued)

7. Technical Specifications Limits

   A. ARPs which have possible Technical Specification impact have the applicable Technical Specification listed in the reference section on the individual attachment page.

   B. If a Technical Specification is referenced for a particular annunciator, the Technical Specification should be referenced for appropriate actions.

8. Annunciator Window Upgrade

   A. The characteristics of the annunciator panel windows have been changed due to human factors review and implemented by PC/M.

   B. Attachment 1, Annunciator Window Characteristics provides a graphic description of the annunciator window characteristics.

9. Nuisance Alarms Or Alarms Defeated On A Clearance Order

NOTE

During outage periods, these actions are only required for annunciators associated with systems that are required to remain operable. Determination will be made by the Shift Manager or designee which annunciators need to be tracked in accordance with this procedure and which are being tracked as part of the outage work control process, or are **NOT** necessary for plant conditions.

   A. If the annunciator is creating a nuisance alarm, <u>or</u> is being defeated on a clearance order, then Shift Manager/ Unit Supervisor is responsible for ensuring the following:

      (1) ARP is reviewed for the potential cause of the nuisance alarm and for what other alarm inputs will be lost if the annunciator is defeated.

      (2) If unit is **NOT** in the EOPs or ONOPs, <u>and</u> annunciator is **NOT** alarming at an interval that the Shift Manager/ Unit Supervisor determines that nuisance annunciator is impacting the ability of the Operators to effectively manipulate/monitor the unit, then PRIOR to defeating the annunciator, complete applicable portions of 0-OSP-200.5, Miscellaneous Tests, and Operating Evolutions, for Defeated /Out-Of-Service Annunciators.

DIST D

4.0 INSTRUCTIONS (continued)

9. A. (continued)

(3) If the unit is in the EOPs, ONOPs, or the annunciator is alarming at such an interval that the Shift Manager/Unit Supervisor determines that the nuisance annunciator is impacting the ability of the Operators to effectively manipulate/monitor the unit, then:

a. Immediately establishing interim compensatory actions to take.

Dist n/c

b. Within one hour, completing applicable portions of 0-OSP-200.5, Miscellaneous Tests, and Operating Evolutions, for Defeated / Out-Of-Service Annunciators.

(4) If required by Operations Department Instructions, then recording the annunciator in the Annunciator Status Log.

(5) Ensuring that applicable sections of 0-OSP-200.5, Miscellaneous Tests, and Operating Evolutions, for Annunciator Status Review, are completed daily for defeated annunciators.

QUESTION 71

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G3	2.3.11
	Importance Rating	3.8	

Radiation Control: Ability to control radiation releases.

Proposed Question: RO Question # 71

Given the following:

- A Steam Generator Tube Rupture has occurred on Unit 3.
- The operating crew has implemented 3-EOP-E-3, Steam Generator Tube Rupture and has prepared for a cooldown using Steam Dumps To Condenser.
- The crew desires to stop Auxiliary Feedwater Pumps.

According to 3-EOP-E-3, which ONE of the following is the PREFERRED method of secondary makeup during cooldown and the reason why?

A. Use the Standby Feedwater System to limit the amount of secondary water that will have to be released post Tube Rupture.

B. Use the Normal Feedwater System to limit the amount of secondary water that will have to be released post Tube Rupture.

C. Use the Normal Feedwater System to lower the amount of activity released through unmonitored release paths.

D. Use the Standby Feedwater System to lower the amount of activity released through unmonitored release paths.

Proposed Answer: B

Explanation (Optional):

A. Incorrect because the normal feedwater train, not the Standby Feedwater System is the preferred method if the condenser dumps will be used. Plausible because the 2nd part

is correct. Also plausible because the Standby Feedwater System will be used if atmospheric dumps are used (per 3-EOP-E-3, NOTES Step 49).

B. CORRECT. Per 3-EOP-E-3, NOTES Step 49: "If the condenser steam dumps are being used for RCS cooldown, the normal feedwater train is preferred to limit the amount of secondary water that will have to be RELEASED post Tube Rupture."

C. Incorrect since, if dumping to the condenser, the release pathways will be monitored, not unmonitored. Plausible because the 1st part is correct. Also plausible because the second note for 3-EOP-E-3, Step 49, contains verbiage related to unmonitored releases.

D. Incorrect because the normal feedwater train, not the Standby Feedwater System is the preferred method if the condenser dumps will be used. Also incorrect since, if dumping to the condenser, the release pathways will be monitored, not unmonitored. Plausible because this would be correct if the steam dumps to atmosphere were used for the cooldown.

Technical Reference(s): 3-EOP-E-3, *Steam Generator Tube Rupture* (Attach if not previously provided)

LP 6902339, *SGTR*

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902339, Obj. 5 (As available)

Question Source: Bank #

Modified Bank # 58817 (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Waterford 3

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *If the condenser steam dumps are being used for RCS cooldown, the normal feedwater train is preferred to limit the amount of secondary water that will have to be RELEASED post Tube Rupture.*
- *If S/G Steam Dump to Atmosphere is used for cooldown, the Standby Feedwater System will lower the amount of activity released through ~~unmonitored release~~ paths.*

49 **Check If Auxiliary Feedwater Pumps Should Be Stopped**

a. Verify narrow range level in all S/Gs - GREATER THAN 15%

RNO: a. Go to Step 50.

b. Verify AFW actuation signal reset – BOTH AFW AUTO START WHITE LIGHTS OUT (3QR50 AND 3QR51)

RNO: b. Perform the following:

1) Verify AMSAC – RESET
2) Verify SI – RESET
3) Verify main feedwater pump switch flags and indicating lights – MATCHED
4) **IF** AFW Auto Start Lights (3QR50 AND 3QR51) are OUT, **THEN** go to Step 49c.
5) **IF** either AFW Auto Start Light (3QR50 AND 3QR51) is LIT, **THEN** go to Step 50.

c. Establish feedwater flow from one of the following

* One main feedwater pump using 3-NOP-074, STEAM GENERATOR FEEDWATER SYSTEM

OR

* Standby feedwater using 0-NOP-074.01, STANDBY STEAM GENERATOR FEEDWATER SYSTEM

RNO: c. Go to Step 50.

d. Stop AFW pumps using 3-NOP-075, AUXILIARY FEEDWATER SYSTEM

BASIS DOCUMENT

WOG Procedure Step N/A **PTN Procedure Step** 49 – NOTE 1

If the condenser steam dumps are being used for RCS cooldown, the normal feedwater train is preferred to limit the amount of secondary water that will have to be RELEASED post Tube Rupture.

BASIS:

If the condenser steam dumps are being used for RCS cooldown, the normal feedwater train is preferred to limit the amount of secondary water that will have to be RELEASED post Tube Rupture.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

9 The WOG guide does not address limiting the amount of contaminated waste produced during a tube rupture event. This note was added to guide the operation staff in methods to minimize contaminated waste.

PLANT SPECIFIC SETPOINTS:

N/A

BASIS DOCUMENT

WOG Procedure Step N/A **PTN Procedure Step** 49 – NOTE 2

If S/G Steam Dump to Atmosphere is used for cooldown, the Standby Feedwater System will lower the amount of activity released through unmonitored release paths.

BASIS:

If S/G Steam Dump to Atmosphere is used for cooldown, the standby Feedwater System will lower the amount of activity released through unmonitored release paths.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

9 The WOG guide does not provide directions for minimizing on-going, unmonitored releases. This note added to inform the operations staff of a potential method to minimize the unmonitored release activity, if the atmospheric dumps are in use.

PLANT SPECIFIC SETPOINTS:

N/A

QUESTION #71

Facility:	Waterford 3
Vendor:	CE
Exam Date:	2007
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		3
	Group #	3		3
	K/A #	G3	2.3.11	
	Importance Rating	3.8		4.3

Radiation Control: Ability to control radiation releases.

Proposed Question:

Given the following conditions:

- A Steam generator Tube Rupture has occurred on SG #1.
- Actions of 902-007, Steam Generator Tube Rupture Recovery, are in progress.
- RCPs are running.
- The crew is performing cooldown of the RCS to Shutdown Cooling Entry conditions.

Which ONE (1) of the following is the preferred method of cooling down the RCS for these conditions?

A: Dump Steam to Condenser using SBCS from both SG #1 and SG #2 to minimize radiological releases

B: Dump Steam to Condenser using SBCS from SG #2 only to minimize radiological releases

C: Dump Steam to Atmosphere using the SG #1 and SG #2 Atmospheric Relief Valves to minimize secondary system contamination for ALARA concerns

D: Dump Steam to Atmosphere using the SG #2 Atmospheric Relief Valve only to minimize secondary system contamination for ALARA concerns

Proposed Answer: B

Explanation (Optional):

A: incorrect. Do not use the affected SG if not required

B: correct. Steam to condenser using unaffected SG. (Affected SG is isolated at this point.)

C: incorrect. Only use ADVs if condenser is not available, and do not use affected SG

D: incorrect. ADVs are a backup to condenser

Technical Reference(s): OP-902-007 step 30 basis (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

SONGS 2006

QUESTION 72

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	3	
K/A #	G3	2.3.12
Importance Rating	3.2	

Radiation Control: Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: RO Question # 72

During a core offload, which ONE of the following conditions will require personnel to evacuate Unit 3 Containment?

A. Containment Integrity is lost during fuel movement activities

B. 3V9, U-3 Cntmt Purge Supply Fan, trips

C. Source Range N-31 fails low.

D. A red light is lit on Containment Air Monitor, R-3-11 for current conditions.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausibility – With Containment Integrity not established, fuel movement activities are halted. The candidate draws an incorrect conclusion that Containment must be evacuated. Also, the candidate may possibly connect Containment Integrity with Containment Habitability which is incorrect.

B. Incorrect. Plausibility – The candidate understands Containment Habitability is required. Incorrectly, they believe continuous Containment Purge is required for Containment occupancy.

C. Incorrect. Plausibility – With 1 Source Range Instrument available, the audio count rate in Containment is required to move fuel. The student incorrectly believes if N-31 and audio count rate is lost, then Containment must be evacuated as a precaution.

D. CORRECT. 3-ARP-097.CR.B, 4/1, Operator Actions #1, states: "IF a startup is NOT in progress, THEN ENSURE actuation of Containment Evacuation alarm." Since a startup is not stipulated in the given information, then containment evacuation is required.

Technical Reference(s): 3-ONOP-067, Radioactive Effluent Release
3-ONOP-059.5, Source Range Nuclear Instrumentation Malfunction
3-ONOP-053, Loss of Containment Integrity
3-NOP-053, Containment Purge
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # 67762 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	(2DR)
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	11
	55.43	

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Modified from SONGS 2009

3.0 AUTOMATIC ACTIONS

3.1 None

4.0 IMMEDIATE ACTIONS

4.1 None

5.0 SUBSEQUENT ACTIONS

5.1 **IF** in Modes 1, 2, 3, or 4, **THEN** restore containment integrity within one hour.

5.2 **IF** in Modes 5 or 6 in a reduced inventory condition, **THEN** stop all operations which could lower RCS level. [Commitment - Step 6.3.1]

5.3 **IF** in Mode 6, **THEN** stop all core alterations and fuel movement within Containment.

5.4 Refer to Technical Specification 3.6 or 3.9 for the applicable Limiting Conditions for Operation.

5.5 Notify the Shift Manager of any condition which may be reportable per PI-AA-204, Condition Identification and Screening Process.

5.6 Take actions as necessary to restore Containment integrity required by the existing plant conditions.

4.1.1 Containment Purge Initiation (continued)

22. **MONITOR** R-3-11, PARTICULATE, R-3-12, GASEOUS, and RAD-6304, PLANT VENT SPING Radiation Monitors during purge operation.

    A. IF the R-3-14 PLANT VENT or RAD-6304 PLANT VENT SPING limits given in Part II of the Radioactive Gas Release Permit are exceeded AND CONTAINMENT INTEGRITY is established, THEN:

    (1) **SHUTDOWN** the Containment Purge System.

    (2) IF plant personnel are inside containment, THEN **EVACUATE** containment, as appropriate.

    (3) **INITIATE** actions to identify and correct the problem.

    B. IF R-3-14 PLANT VENT or RAD-6304 PLANT VENT SPING limits given in Part II of the Radioactive Gas Release Permit are exceeded AND CONTAINMENT INTEGRITY is **NOT** established, THEN:

    (1) **MAINTAIN** the Containment Purge System in operation to enable personnel to monitor the release.

    (2) IF plant personnel are inside containment, THEN **EVACUATE** containment, as appropriate.

    (3) **INITIATE** actions to identify and correct the problem.

    (4) IF cause of high activity can **NOT** be identified OR isolated, THEN **INITIATE** actions to re-establish CONTAINMENT INTEGRITY.

23. IF containment purge operation will extend beyond 4 hours, THEN **COMPLETE** Part C, Extended Purge Operation, on Attachment 4, Containment Purge Data, at the time intervals specified.

4.4 Modes 3, 4 and 5 - with reactor trip breakers in the open position/control rod drive system NOT capable of rod withdrawal.

4.4.1 Malfunction of ONE channel:

1. **IF** applicable, **THEN** notify plant personnel of erroneous Containment Evacuation Alarm.

4.4.2 Malfunction of BOTH channels:

1. **IF** applicable, **THEN** notify plant personnel of erroneous Containment Evacuation Alarm.

4.5 Mode 6 - Refueling

4.5.1 Malfunction of ONE channel:

1. Switch the AUDIO COUNT RATE CHANNEL SELECTOR to the operable source range.
2. Verify at least 2 out of 4 Source Range and Backup NIS (Gamma Metrics) channels are operable, with one Source Range having audible count rate in the Control Room and Containment.
   a. **IF** the above requirement is not met, **THEN** suspend all operations involving core alterations **OR** positive reactivity changes.
3. **IF** applicable, **THEN** notify plant personnel of erroneous Containment Evacuation Alarm.

4.5.2 Malfunction of BOTH channels:

1. Suspend all operations involving Core Alterations.
2. Suspend all operations involving positive reactivity changes.
3. **IF** applicable, **THEN** notify plant personnel of erroneous Containment Evacuation Alarm.

5.5 Mode 6 - Refueling

5.5.1 Malfunction of ONE channel:

1. Place LEVEL TRIP switch on failed channel in BYPASS position.
2. Place HIGH FLUX AT SHUTDOWN switch on failed channel in BLOCK position.
3. Switch an NIS RECORDER to the operable source range.
4. **IF** one Source Range having audible count rate in the Control Room and Containment, **AND** 2 out of 4 NIS (NSSS Source Range and Gamma Metrics) Channels are not operable, **THEN** verify RCS boron concentration is greater than or equal to the required boron concentration at least once per 12 hours.
5. Notify I&C.
6. Monitor Backup NIS (Gamma Metric) Source Range Count Rate.

5.5.2 Malfunction of BOTH channels:

1. Place LEVEL TRIP switches on failed channels in BYPASS position.
2. Place HIGH FLUX AT SHUTDOWN switches on failed channels in BLOCK position.
3. Switch NIS RECORDERS to the intermediate ranges.
4. Verify RCS boron concentration is greater than or equal to required concentration at least once per 12 hours.
5. Notify I&C.
6. Monitor Backup NIS (Gamma Metrics) Source Range Count Rate.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16	**Check For High Containment Airborne Activity**	
	a. Check for R-11/12 HIGH ALARMS * Check R-11 Red HIGH LED - ON * Check R-11 PART alarm monitor pushbutton - FLASHING * Check R-12 Red HIGH LED - ON * Check R-12 GAS alarm monitor pushbutton - FLASHING * R-11/12 display reading - GREATER THAN OR EQUAL TO ALARM SETPOINT	a. Go to Step 18.
	b. Verify Containment And Control Building Ventilation Systems aligned using Attachment 1	
	c. Dispatch an operator to the RM-80 skid to perform the following • Silence the local alarm • Check for any abnormal indications	
	d. Direct Radiation Protection and Chemistry Departments to verify actual activity inside containment	
	e. Perform the following to evaluate plant status * 3-ONOP-041.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE * 3-ONOP-033.3, ACCIDENTS INVOLVING NEW OR SPENT FUEL	
	f. Determine if alarm is valid by checking against current plant status	f. Go to Step 19.
	g. Evacuate non-essential personnel from containment	
17	**Return To Step 1**	

QUESTION #72

Facility: Songs

Vendor: CE

Exam Date: 2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
	Group #	3	3
	K/A #	G3	2.3.12
	Importance Rating	3.2	3.7

Radiation Control: Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question:

Given the following conditions on Unit 3:

• Unit 3 is in a Refueling shutdown with a Reduced Inventory Condition.

Which ONE (1) of the following conditions requires that the Unit 3 Containment be evacuated?

A: Notification from Security that a Direct Armed Attack (Code RED) is in progress.

B: Annunciator CONTAINMENT SUMP HI LEVEL alarms in the Control Room.

C: An inadvertent Reactor Coolant System dilution of 100 gallons.

D: Receipt of a seismic alarm at less than Operating Basis Earthquake.

Proposed Answer: A

Explanation (Optional):

A: Correct. This is the required action per SO23-13-25, Operator Actions During Security Events.

B: Incorrect. Plausible because it could be thought that receipt of this alarm would require a
Containment evacuation, however, it is not required for this condition.

C: Incorrect. Plausible because it could be thought that a dilution, especially in Reduced Inventory Condition would result in increasing count rate, however, AOI SO23-13-11 Emergency Boration of the RCS/Inadvertent Dilution or Boration does not require Containment Evacuation. An increase in radiation levels could require evacuation but this condition is not specified.

D: Incorrect. Plausible because it could be thought that evacuating Containment would be prudent for a seismic event but is not an action specified in SO23-13-3, Earthquake.

Technical Reference(s): SO23-13-25, Step 12a RNO, p6
SO23-13-11, Step 5a
SO23-13-3, Attachment 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: 54470
Given a SONGS Security Event, DESCRIBE basis for each step, caution or
note in the Abnormal Operating Instruction and the expected plant or
operator response in accordance with SO23-13-25. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

Comments:

QUESTION 73

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G4	2.4.22
	Importance Rating	3.6	

Emergency Procedures / Plan: Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Proposed Question: RO Question # 73

Which ONE of the following identifies the HIGHEST RED PATH priority among those listed for Critical Safety Functions (CSF) and the bases for this function?

A. Heat Sink;
to protect the RCS boundary

B. Heat Sink;
to protect the fuel clad/matrix

C. Integrity;
to protect the RCS boundary

D. Integrity;
to protect the fuel clad/matrix

Proposed Answer: B

Explanation (Optional):

A. Incorrect since the Heat Sink CSF is designed to protect the fuel matrix/clad, not the RCS boundary. Plausible because the 1st part is correct. Also plausible because the RCS boundary is correct for the other CSF, Integrity, given in Choices C & D.

B. CORRECT. Per BD-EOP-F-0, Page 23: "A loss of secondary heat sink occurs if decay heat removal is needed through the S/Gs and all feed flow capability is lost. Feed flow must be reestablished or an alternative heat removal mode, such as bleed and feed, must be established to prevent core uncovery and eventually an inadequate core cooling condition. Since this is an extreme challenge to the fuel clad/matrix barrier to radioactivity release, immediate operator action is required and a Red priority is warranted. The loss of secondary heat sink condition is the only Red priority included on this tree."

C. Incorrect since Integrity is the 4th highest priority, not the 3rd. Plausible because the 2nd part is correct for the Integrity CSF. Also plausible because a novice applicant could postulate that the RCS boundary should be a higher priority because loss of the RCS boundary could lead to adverse impacts on Core Cooling, which has a higher priority than Heat Sink.

D. Incorrect since Integrity is the 4th highest priority, not the 3rd. Also incorrect because the Integrity CSF is based on protecting the RCS boundary, not the fuel matrix/clad. Plausible because the 2nd part is part of the correct answer, although not for the given CSF. Also plausible because a novice applicant could postulate that the RCS boundary should be a higher priority because loss of the RCS boundary could lead to adverse impacts on Core Cooling, which has a higher priority than Heat Sink.

Technical Reference(s): BD-EOP-F-0, *Critical Safety Function Status Trees* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900353, Obj. 1 & 3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:	Last NRC Exam:	Turkey Point

Question Cognitive Level:	Memory or Fundamental Knowledge	(1P)
	Comprehension or Analysis	
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

BASIS DOCUMENT

WOG Procedure Step F-0.3 PTN Procedure Step N/A

ENCLOSURE 3

BASIS:

The Status Tree F-0.3, HEAT SINK, provides a systematic method to explicitly determine the status of the HEAT SINK Critical Safety Function. This tree requires no operator action other than monitoring a limited set of plant parameters and comparing them to reference values within the tree.

This tree represents the third highest priority Critical Safety Function and, as such, is always entered directly after the CORE COOLING tree. The tree can direct the operator to any of five separate Function Restoration Guidelines.

This tree monitors the state of secondary heat sink and integrity based on SG level, feed flow and SG pressure. The Critical Safety Function is considered to be satisfied if all SG levels and pressures are within the normal range.

Correct Answer (B)

Distractor D

The most serious challenge to the Critical Safety Function is an indication of loss of secondary heat sink. A loss of secondary heat sink occurs if decay heat removal is needed through the S/Gs and all feed flow capability is lost. Feed flow must be reestablished or an alternative heat removal mode, such as bleed and feed, must be established to prevent core uncovery and eventually an inadequate core cooling condition. Since this is an extreme challenge to the fuel clad/matrix barrier to radioactivity release, immediate operator action is required and a Red priority is warranted. The loss of secondary heat sink condition is the only Red priority included on this tree. There are no Orange priority conditions on this tree. A not satisfied condition, Yellow, on this tree can be reached if 1) any S/G pressure is above the highest S/G safety valve setpoint; 2) any S/G level is higher than S/G high-high feedwater isolation setpoint; 3) any S/G pressure is above the lowest S/G safety valve setpoint; and 4) any S/G level is below the narrow range.

(Continued on next page)

BASIS DOCUMENT

WOG Procedure Step F-0.4 PTN Procedure Step N/A

ENCLOSURE 4

BASIS:

The Status Tree F-0.4, INTEGRITY, provides a systematic method to explicitly determine the status of the Integrity Critical Safety Function. This tree requires no operator action other than monitoring a limited set of plant parameters and comparing them to reference values within the tree.

This tree represents the fourth highest priority Critical Safety Function and, as such, is always entered directly after the HEAT SINK tree. The tree can direct operators to either of two Function Restoration Guidelines.

Correct Answer (B)

Distractors C & D

This tree is unique among all the Critical Safety Function Status Trees in that all the reference values against which current plant parameters are compared do not appear explicitly at the branch points.

The first concern of this Status Tree is for a transient which produces large cooldown rates, and thus large thermal stresses. The cooldown at the vessel wall could be caused by a secondary break cooling down the entire RCS, and/or the addition of cold injection water from the RWST into the cold legs and downcomer region of the vessel under relatively stagnant RCS flow conditions. The degree of the cooldown determines the severity of the challenge to the vessel wall, if any. A more detailed description of the thermal shock and pressurized thermal shock concerns is presented in the documents PRESSURIZED THERMAL SHOCK and STAGNANT REACTOR COOLANT LOOPS in the Generic Issues section of the Executive Volume.

(Continued on next page)

BASIS DOCUMENT

WOG Procedure Step F-0.4 PTN Procedure Step N/A

(Continued)

The intent of the Integrity Critical Safety Function Status Tree is to define symptoms that indicate a challenge is occurring to the Integrity Critical Safety Function, and to prioritize operator action required to address this challenge. Challenges are defined for two types of plant transients. The first concern is ~~transients that result in rapid~~ and severe RCS cooldown that could lead to challenging vessel integrity, which is called pressurized thermal ~~shock. The other concern is transients~~ that occur while the RCS is relatively cold and a rapid pressure increase occurs, which is called cold overpressure condition.

Distractors A & C

HAS ANY RCS COLD LEG TEMPERATURE DECREASED GREATER THAN 100°F IN THE LAST 60 MINUTES?	YES
	NO

If the temperature decrease in any cold leg has exceeded 100°F in the previous 60 minutes, then there is a potential concern for thermal shock. If not, then no other checks on rate-dependent limits are necessary. The only concern remaining is cold overpressure which will be checked in subsequent blocks. If the temperature decrease has exceeded 100°F in the previous 60 minutes, the degree of cooldown must be assessed before a thermal shock concern can be identified. This is checked in subsequent blocks.

HAS ANY RCS COLD LEG TEMPERATURE BEEN LESS THAN 290°F?	YES
	NO

The objective of Limit A setpoint is to provide a limit that when exceeded indicates a potential thermal thermal condition. For category III plants, the limit A curve is a straight line with little change in temperature and may be expressed as a single temperature value. The basis of this limit is to prevent growth of a flaw that could conservatively be present in the vessel wall. If Limit A has been exceeded, then operator action is necessary to limit further RCS temperature decreases or RCS pressure increases. A Red priority is warranted since an extreme challenge to the function is occurring and FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, is the appropriate procedure for functional response. If Limit A setpoint has not been exceeded, additional checks are made in subsequent blocks to determine if a less severe thermal shock condition exists.

(Continued on next page)

QUESTION 74

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	4	
K/A #	G4	2.4.3
Importance Rating	3.7	

Emergency Procedures / Plan: Ability to identify post-accident instrumentation.

Proposed Question: RO Question # 74

Which ONE of the choices below identifies (1) a post-accident instrument in accordance with TS 3.3.3.3, Accident Monitoring Instrumentation from Table 3.3-5 and (2) the color of its tag in accordance with 0-ADM-209, Equipment Tagging and Labeling?

A. (1) Pressurizer Pressure;

(2) blue

B. (1) Pressurizer Pressure;

(2) purple

C. (1) Reactor Coolant Inlet Temperature Tcold (Wide Range);

(2) blue

D. (1) Reactor Coolant Inlet Temperature Tcold (Wide Range);

(2) purple

Proposed Answer: D

Explanation (Optional):

A. Incorrect since WR, not PZR, RCS pressure instrumentation is required by TS Table 3.3-5. Also incorrect since WR Tcold *is* an Accident Monitoring instrument and 0-ADM-201, *Equipment Tagging and Labeling*, Definition 4.6, requires a purple label, not a blue label. Plausible because TS Table 3.3-1, Reactor Trip System Instrumentation requires the Functional Units 7 & 8, Pressurizer Pressure.

Also plausible since TS Table 3.3-2, ESF Actuation System Instrumentation requires the Functional Unit 1d, Pressurizer Pressure. Also plausible because many safety-related instruments in the Control Room have blue labels.

B. Incorrect since WR, not PZR, RCS pressure instrumentation is required by TS Table 3.3-5. Plausible because the 2nd part is correct. Also plausible because TS Table 3.3-1, Reactor Trip System Instrumentation, requires the Functional Units 7 & 8, Pressurizer Pressure. Also plausible since TS Table 3.3-2, ESF Actuation System Instrumentation, requires the Functional Unit 1d, Pressurizer Pressure.

C. Incorrect since WR Tcold *is* an Accident Monitoring instrument and 0-ADM-201, *Equipment Tagging and Labeling*, Definition 4.6, requires a purple label, not a blue label. Plausible because the 1st part is correct. Also plausible because many safety-related instruments in the Control Room have blue labels.

D. CORRECT. TS Table 3.3-5, Accident Monitoring Instrumentation, Instrument 4, requires WR Tcold. Per 0-ADM-201, *Equipment Tagging and Labeling*, Definition 4.6, Reg Guide 1.97, Common Markings - A fade resistant vinyl type tape colored purple which will enable Control Room Operators to identify instruments/indicators which may be relied upon in a Post Accident Condition.

Technical Reference(s): TS Tables 3.3.1, 3.3-2, 3.3-5
0-ADM-201, *Equipment Tagging and Labeling* (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900523, Obj. 3 (As available)

Question Source: Bank #

Modified Bank # 62401 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Modified from VC Summer 2008

#74

4.0 DEFINITIONS

4.1 ETI Tag: Equipment Temporary Identification Tag

4.2 Operator Aids - Information including sketches, notes, graphs, instructions, drawings, and other documents used to assist operators in performing assigned duties. For the purposes of administrative control, a Temporary Information Tag is not considered to be an Operator Aid.

4.3 Temporary Information Tags - Preprinted tags filled in with information of the status of equipment and precautions or instructions for its operation. These tags may be white, red, or green, depending on the application:

4.3.1 White/clear information tags (of any format) are normally used to provide general information.

4.3.2 Red or Green information tags are normally used on control switches for equipment with the breaker deenergized (indicating lights are off) to indicate the open/closed, on/off status of the equipment.

4.3.3 Tags should be attached to the switches or equipment to which they refer.

4.4 Operator Aid and Temporary Information Tag Log - A notebook maintained in the Control Room, which contains two indexes; Operator Aid Index and Temporary Information Tag Index (a form similar to Attachment 2).

4.5 Permanent Information - Information that appears on a medium not suitable to change and determined by the Assistant Operations Manager to be applicable indefinitely. An example would be notes or cautions produced on a Bakelite plate. Information posted in this format is not considered to be an operator aid in this procedure and does not provide direction to operate the plant.

4.6 Reg Guide 1.97, Common Markings - A fade resistant vinyl type tape colored purple which will enable Control Room Operators to identify instruments/indicators which may be relied upon in a Post Accident Condition.

Correct Answer (D)

TECHNICAL SPECIFICATIONS INSTRUMENTATION (3/4.3)

TERMINAL OBJECTIVE:

Given a scenario involving plant operations, **implement technical specifications to ensure that the unit is in a safe operating condition**. The required implementation shall include: (1) evaluation of the conditions and events presented against applicable technical specification requirements, (2) identification of limiting conditions for operation, (3) determination of required actions and (4) identification of the procedural and administrative completions required.

ENABLING OBJECTIVES:

		LEVEL	EVAL
1.	Given a plant operating mode, a description of the system status, and without references, **relate the system status to Tech Spec operability requirements**. The Tech Spec requirements for the system must be identified and compared to the given status to determine if the Tech Spec requirements are satisfied. (RCO, SRO, LOCT)	C-3	W
2.	Given a technical specification requirement and without references, **describe the basis of the requirement** as stated in the Tech Specs basis document. (RCO, SRO, LOCT)	C-3	W
3.	Given a unit in any mode, a set of plant conditions including evolutions in progress, materials normally available at the job location and notification of a condition/event which may involve Tech. Specs., **evaluate plant conditions against the Tech. Spec. requirements**. Plant conditions must be compared to Tech. Spec. requirements and all required actions identified. (RCO, SRO, LOCT)	C-6	W

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

Distractors A & B

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 2	1 1	2 2	1, 2 3*, 4*, 5*	1 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

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The accident monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

b. The provisions of Specification 3.0.4 are not applicable to ACTIONS in Table 3.3-5 that require a shutdown.

c. Separate Action entry is allowed for each Instrument.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS
1. Containment Pressure (Wide Range)	2	1	1, 2, 3	31, 32
2. Containment Pressure (Narrow Range)	2	1	1, 2, 3	36
3. Reactor Coolant Outlet Temperature T_{HOT} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
4. Reactor Coolant Inlet Temperature T_{COLD} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
5. Reactor Coolant Pressure – Wide Range	2	1	1, 2, 3	31, 32
6. Pressurizer Water Level	2	1	1, 2, 3	31, 32
7. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator	1, 2, 3	31, 32
8. Reactor Coolant System Subcooling Margin Monitor	2(2)	1(2)	1, 2, 3	31, 32
9. PORV Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	33
10. PORV Block Valve Position Indicator	1/valve	1/valve	1, 2, 3	33
11. Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	32
12. Containment Water Level (Narrow Range)	2	1	1, 2, 3	36
13. Containment Water Level (Wide Range)	2	1	1, 2, 3	31, 32

Correct Answer (D)
& Distractor C

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermo-couples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
d. Main Steam Lines	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

QUESTION #74

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G4	2.4.3
	Importance Rating	3.7	

Emergency Procedures / Plan: Ability to identify post-accident instrumentation.

Proposed Question: RO Question # 74

Which ONE (1) of the choices below completes the following statement?

Post-Accident Monitoring instrumentation/components are identified by a ________ label which includes ___________.

A. green;

either a 'P', 'S', or 'T'

B. yellow;

either a 'P', 'S', or 'T'

C. green;

the instrument/component designator or nomenclature

D. yellow;

the instrument/component designator or nomenclature

Proposed Answer: D

Explanation (Optional):

A. Plausible because there are green labels on the MCB and the 'P', 'S', or 'T' indicate that this equipment will actuate during an accident

B. Plausible because the 1st part is correct and the 'P', 'S', or 'T' indicate that this equipment will actuate during an accident.

C. Plausible because the 2nd part is correct and because there are green labels on the MCB

D. CORRECT. Yellow labels with the instrument/component designator or nomenclature are PAMS grade instruments (see OAP-103.4, Section 6.6).

Technical Reference(s): TS Table 3.3.10 (Page 3-58) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: (As available)

Question Source: Bank # WTSI 62401

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2008 VC Summer

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

QUESTION 75

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G4	2.4.9
	Importance Rating	3.8	

Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: RO Question # 75

Given the following:

- Unit 3 is in Mode 5 for a refueling outage.
- 3A RHR train is in operation for shutdown cooling.
- Time to boil in the Reactor vessel is 2 hrs.

The following occurs:

- The crew enters 3-ONOP-030, Component Cooling Water Malfunction.
- Component Cooling Water Surge Tank Level, LI-3-613A is at 10% and lowering.
- Component Cooling Water Surge Tank Makeup, MOV-3-832 has been opened for makeup.
- Letdown and Excess Letdown have been isolated.
- A system walkdown FAILS to identify the leak location.
- ALL running CCW Pumps are showing signs of cavitation.

In accordance with 3-ONOP-030, which ONE of the following describes (1) the actions to take for the Component Cooling Water Pumps and (2) the Containment Closure requirement with ALL RHR lost in accordance with 3-ONOP-050, Loss of RHR?

A. (1) Stop and PULL-TO-LOCK all, but one CCW Pump
(2) Ensure Containment Closure is set within 30 minutes

B. (1) Stop and PULL-TO-LOCK all, but one CCW Pump
(2) Ensure Containment Closure is set within 2 hrs

C. (1) Stop and PULL-TO-LOCK all CCW Pumps
(2) Ensure Containment Closure is set within 2 hrs

D. (1) Stop and PULL-TO-LOCK <u>all</u> CCW Pumps
(2) Ensure Containment Closure is set within 30 minutes

Proposed Answer: D

Explanation (Optional):

A. Incorrect since all CCW Pumps are placed in PTL if there are signs of cavitation. Plausible - the candidate incorrectly believes one CCW Pump is left running. Also plausible because Containment closure time is correct.

B. Incorrect since all CCW Pumps are placed in PTL if there are signs of cavitation. Plausible - the candidate incorrectly believes one CCW Pump is left running. Also plausible because 2 hrs is the required amount of time to ensure Containment closure is set which is equivalent to the time to boil.

C. Incorrect since Containment closure is required within 30 minutes per 3-ONOP-050. Plausible - the candidate incorrectly believes 2 hrs is the required amount of time to ensure Containment closure is set which is equivalent to the time to boil.

D. CORRECT. All CCW is lost due to placing CCW Pumps in PTL (pump cavitation). A loss of CCW leads to a loss of RHR. Therefore, Containment closure is set within 30 minutes.

Technical Reference(s): 3-ONOP-030, Component Cooling Water Malfunction; 3-ONOP-050, Loss of RHR (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900229, Obj. 4 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3SPK)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

FOLDOUT FOR 3-ONOP-030

1. **TOTAL LOSS OF CCW FLOW**

   A. Manually trip the reactor, verify reactor trip using the EOP network, **THEN** stop the RCPs.

   B. Isolate letdown and excess letdown.

   C. Establish one charging pump running at maximum speed **AND** dispatch operator to establish emergency cooling water to one of the remaining two charging pumps using Attachment 1. Monitor RCS pressure closely while running charging pump at maximum speed.

   D. **WHEN** Attachment 1 is complete, **THEN** operate charging pump supplied with emergency cooling as necessary to maintain RCP seal cooling.

2. **LOSS OF CCW TO ANY COMPONENT**

   IF component cooling water flow to any component cooled by CCW is lost, **THEN** shut down the affected component.

3. **CHARGING PUMP EMERGENCY COOLING CRITERIA**

   IF Cooling Water is **NOT** available to charging pumps, **THEN** charging pump operation shall be at maximum speed until cooling is restored from CCW System or using Attachment 1.

4. **CCW PUMP STOPPING CRITERIA**

   IF any Component Cooling Water Pump is cavitating, **THEN** stop the affected Component Cooling Water Pumps and place in Pull-To-Lock.

5. **REACTOR TRIP CRITERIA**

   IF tripping a RCP is required, **THEN** manually trip the reactor prior to stopping the RCP.

6. **RCP STOPPING CRITERIA**

   IF any RCP bearing temperature annunciator alarm actuates **AND** its associated motor bearing temperature is greater than 195°F, **THEN** trip reactor and stop the affected RCPs.

7. CCW System operation once CCW System Hdr has been restored shall be within the operating restrictions of 3-NOP-030 summarized as follows: [Commitment - Step 3.3.2]

   CCW Pumps, Heat Exchangers, and Flows/Loads.

   - N-1 CCW Pumps (where N = number of CCW Hxs aligned to CCW)
   - All CCW Hxs in service when RHR in service **OR** with only 2 CCW Hxs in service, place 2 CCW Pumps in Pull-To-Lock.
   - Maximum of 5 out of 6 CCW Heat Loads.

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, trip the reactor and stop the affected RCPs.

NOTE

Foldout page should be monitored throughout this procedure.

1 Verify Power To 4KV Bus 3D

a. Maintain 4KV Bus 3D energized - ALIGNED TO AN ENERGIZED 4KV BUS

Response Not Obtained:

a. **IF** lockout of 4KV Bus 3D **NOT** present, **THEN** perform the following:

1. Verify 3C CCW Pump - BREAKER OPEN
2. Verify 3C ICW Pump - BREAKER OPEN
3. Operate bus supply breakers to energize bus.

2 Verify Component Cooling Water Pumps In Service

Response Not Obtained:

a. **IF** starting an idle CCW pump will **NOT** overload an EDG, **THEN** start CCW pumps as necessary to establish flow in both headers.

3 Verify Flow In Both Component Cooling Water Headers - NORMAL

- FT-3-613A for header A
- FT-3-613B for header B

Response Not Obtained:

Perform the following:

a. **IF** CCW flow to RCPs can **NOT** be established, **THEN** manually trip the reactor **AND** verify reactor trip using the EOP Network, **AND** then stop all RCPs **AND** perform the following:

1. Isolate Letdown and Excess Letdown
2. **IF** any charging pump is running, **THEN** operate at maximum speed until Attachment 1 is completed.
3. 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).*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**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	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 **AND** perform 3-EOP-E-0, Reactor Trip or Safety Injection, while continuing with this procedure. b) **WHEN** rector verified tripped, **THEN** stop all RCPs. 3. Observe NOTES prior to Step 8 and go to Step 8.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *Steps 8 through 33 are only applicable when leakage from the Component Cooling Water System is in progress.*
- *When leakage has been located and isolated, recovery actions shall continue starting with Step 27.*

8 **Establish Stable Plant Parameters**

a. Station a watch at Header B local surge tank level indicator, LI-3-615

b. Isolate Letdown and Excess Letdown

c. Dispatch an operator to establish emergency cooling water to desired charging pump using Attachment 1

d. Dispatch an operator to locate and isolate CCW System leakage by performing a System Walkdown.

RESPONSE NOT OBTAINED:

d. **IF** the system walkdown fails to identify the leak and CCW Surge Tank Level is still decreasing **OR** makeup still required, **THEN** perform the following:

1. Stop and PULL-TO-LOCK all but one CCW Pump.

2. Attempt to determine if the leak is in a Heat Exchanger by removing ONLY one CCW Heat Exchanger from service at a time using the following CCW isolation valves:

HX	Inlet	Outlet
3A	3-712A	3-713A
3B	3-712B	3-713B
3C	3-712C	3-713C

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	**Check If Additional Makeup Is Required**	
	a. Component Cooling Water Surge Tank Level, LI-3-613A - DECREASING OR EMPTY	a. Perform the following: 1) **IF** surge tank level can **NOT** be maintained during subsequent recovery actions, **THEN** do Steps 9b, 9c and 9d. 2) Go to Step 10.
	b. Locally open Component Cooling Water Header A Makeup Water Isolation, 3-711A	
	c. Throttle closed DWDS-001 to provide Demin Water Flow to the CCW Surge Tank	
	d. Locally throttle Demin Water Supply Valve, 3-724D, as necessary to maintain surge tank level stable or increasing	
10	**Verify CCW Pumps - NOT CAVITATING**	Stop any cavitating CCW pumps and place all CCW pumps in Pull-To-Lock.
	a. Running CCW pump amps - STABLE	
	b. Component Cooling Water Header Flows - STABLE • FI-3-613A • FI-3-613B	

FOLDOUT PAGE

1. **CONTAINMENT CLOSURE CRITERIA**

When at reduced inventory operations, containment closure shall be initiated within 5 minutes of the loss of RHR and shall be completed within the time to core boiling, or within 30 minutes of the loss of RHR, whichever is less.

When not in reduced inventory operations, containment closure shall be completed within the time to core boiling, or within 30 minutes of the loss of RHR, whichever is less, unless the containment closure time limit has been extended as allowed by 0-ADM-051, Outage Risk Assessment and Control, Enclosure 13, Containment Closure Time Limits.

FINAL PAGE