ANSWER KEY CPNPP 2021 NRC SRO Exam

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

RO	Overall	SRO
A – 19 B – 16 C – 18 D – 22	A – 23 B – 23 C – 25 D – 29	A – 4 B – 7 C – 7 D – 7

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	003.G.2.2.39		
Level of Difficulty: 3	Importance Rating	3.9		

Reactor Coolant Pump: Knowledge of less than or equal to one hour Technical Specification action statements for systems.	
Question # 1	

Given the following conditions:

- Unit 1 is in MODE 3, with the RCS temperature at 400ºF
- Reactor Trip Breakers are closed
- Rod Drive MG sets are operating
- An RCS dilution is in progress

In accordance with TS 3.4.5, RCS Loops – MODE3, a MINIMUM of __(1)__ RCPs must be in operation.

If one of the minimum number of operating RCPs were to trip, within 1 hour __(2)__.

A. (1) two
(2) commence Emergency Boration

B. (1) two
(2) make the Rod Control System incapable of withdrawal

C. (1) four
(2) commence Emergency Boration

D. (1) four
(2) make the Rod Control System incapable of withdrawal

Answer: B	

K/A Match: K/A match due to requiring knowledge of a one-hour TS associated with the operation of the RCPs.

Explanation:

A. Incorrect. First part is correct (see B). Second part is incorrect, but plausible since various TS require boration be commenced in response to actions not being met.

B. Correct. First part is correct. Under the given conditions, with a dilution in progress, two RCS loops are required to be operating. Second part is correct. If one of the two operating loops stops operating, within 1 hour restore it to operating or make the rod control system incapable of withdrawal.

C. Incorrect. First part is incorrect, but plausible (see D). Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible since MODES 1 and 2, TS 3.4.4, require all four loops operating. Second part is correct (see B).

Technical Reference(s)	TS 3.4.5	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given RCS parameter indications and plant conditions, **ASSESS** from memory any required TS/TR entries, including any actions which must be completed within one hour in accordance with Technical Specifications or TRM. (SYS.RC1.OB06)

Question Source: Bank # ______
Modified Bank # 62575 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: Bank 62575	Revision:

- Unit 1 is in MODE 3 following a Refueling outage.
- All four Reactor Coolant System (RCS) Loops are OPERABLE, with all four Reactor Coolant Pumps (RCP) in operation.
- Both Control Rod Drive Motor Generators are energized and Reactor Trip Breakers are CLOSED.

If RCP 1-02 trips, which of the following identifies the MINIMUM RCP requirements in accordance with LCO 3.4.5, RCS Loops -- MODE 3 and IPO-001A, Plant Heatup from Cold Shutdown to Hot Standby, Attachment 5, Checklist Required Prior to Closing Reactor Trip Breakers?

A. LCO 3.4.5 is satisfied with two RCPs OPERABLE, with one RCP in operation.
IPO-001A is satisfied with two RCPs in operation.

B. LCO 3.4.5 is satisfied with two RCPs OPERABLE, with two RCPs in operation.
IPO-001A is NOT satisfied with only three RCPs in operation.

C. LCO 3.4.5 is satisfied with two RCPs OPERABLE, with one RCP in operation.
IPO-001A is NOT satisfied with only three RCPs in operation.

D. LCO 3.4.5 is satisfied with two RCPs OPERABLE, with two RCPs in operation.
IPO-001A is satisfied with two RCPs in operation.

Answer: B

Answer Explanation

Comments / Reference: Bank 62575	Revision:

E. Incorrect. Plausible because in MODE 3 with the Rod Control System capable of rod withdrawal two RCS loops are required operable with two in operation, however, only one is required in operation if the Rod Control System is not capable of rod withdrawal. IPO-001A only requires two RCS loops remain in operation in MODE 3 if the Rod Control System is not capable of rod withdrawal. This answer would be correct if the Rod Control System were not capable of rod withdrawal in the current conditions.

F. Correct. In MODE 3 with the Rod Control System capable of rod withdrawal, two RCS loops shall be OPERABLE with 2 loops in operation IAW TS 3.4.5 LCO. IPO-001A requires that four RCS loops be in operation with the Rod Control System capable of rod withdrawal and thus IPO-001A is not satisfied.

G. Incorrect. Plausible because in MODE 3 with the Rod Control System capable of rod withdrawal two RCS loops are required operable with two in operation, however, only one is required in operation if the Rod Control System is not capable of rod withdrawal. IPO-001A requires that four RCS loops be in operation with the Rod Control System capable of rod withdrawal and thus IPO-001A is not satisfied.

H. Incorrect. Plausible because in MODE 3 with the Rod Control System capable of rod withdrawal, two RCS loops shall be OPERABLE with 2 loops in operation IAW TS 3.4.5 LCO.. IPO-001A only requires two RCS loops remain in operation in MODE 3 if the Rod Control System is not capable of rod withdrawal. This answer would be correct if the Rod Control System were not capable of rod withdrawal in the current conditions.

Question 186 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	4.00
System ID:	62575
User-Defined ID:	ILOT8719
Cross Reference Number:	SYS.RC1.OB06.017
Topic:	Unit 1 is in MODE 3 following a Refueling outage. All four Reactor Coolant System (RCS) Loops are
K/A:	015/017.G.2.2.22
Question Reference:	
SRO:	
Comments:	LC22 NRC REF: TS 3.4.5

Comments / Reference: TS 3.4.5	Revision: 156

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops -- MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and either:

a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

---NOTE---

All reactor coolant pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

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Comments / Reference: TS 3.4.5	Revision: 156

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One required RCS loop not in operation, with Rod Control System capable of rod withdrawal.	C.1 Restore required RCS loop to operation. OR C.2 Place the Rod Control System in a condition incapable of rod withdrawal.	1 hour 1 hour
D. Four RCS loops inoperable. OR No RCS loop in operation.	D.1 Place the Rod Control System in a condition incapable of rod withdrawal. AND D.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND D.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	2		
	Group	1		
	K/A	004.K3.04		
Level of Difficulty: 3	Importance Rating	3.7		

Chemical and Volume Control: Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: RCPS

Question # 2

Given the following conditions:

- Unit 1 Reactor power is 3%
- ANY RCP SEAL WTR INJ FLO LO (5A-1.6) is in alarm
- RCP SEAL WTR INJ FILT 1 ΔP HI (5A-2.6) is in alarm
- Seal injection flow is 0 gpm
- ABN-101, REACTOR COOLANT PUMP TRIP/MALFUNCTION, has been initiated

RCPs are required to be tripped if CCW Thermal Barrier heat exchanger flow is not a MINIMUM of __(1)__.

Seal injection is isolated in order to prevent thermal shock to the seals which could directly result in __(2)__.

A. (1) 35 gpm
(2) excessive RCS leakage

B. (1) 35 gpm
(2) voiding the CCW system

C. (1) 64 gpm
(2) excessive RCS leakage

D. (1) 64 gpm
(2) voiding the CCW system

Answer: A

K/A Match: K/A match due to requiring knowledge of the effect of a loss of seal injection flow from CVCS to the RCPs

Explanation:

A. Correct. First part is correct. Per ABN-101, Section 7 (Loss of Seal Injection), if CCW is less than 35 gpm, you are directed to section 9 which will direct tripping the reactor, tripping RCPs and isolating seal injection and CCW thermal barrier return from the RCPs. Second part is correct. Thermal shocking the seals could result in seal damage (not seating correctly) which could result in excessive leakage.

B. Incorrect. First part is correct (See A). Second part is incorrect, but plausible because the CCW return valve from the RCPs, HV-4709, is manually closed on a loss of seal injection and CCW to the RCPs to prevent voiding in the CCW system.

C. Incorrect. First part is incorrect, but plausible because CCW flow at 64 gpm will result in isolating the CCW thermal barrier HX. Second part is correct (See A).

D. Incorrect. First part is incorrect but plausible (see C). Second part is incorrect but plausible (see B).

Technical Reference(s)	ABN-105	Attached w/ Revision # See Comments / Reference
	ABN-101	
	RCS Study Guide	

Proposed references to be provided during examination: ______________________

Learning Objective: **DISCUSS** the response for a Loss of Seal Injection in accordance with ABN-101, Reactor Coolant Pump Trip/Malfunction. (ABN.101.OB07)

Question Source: Bank # ________
Modified Bank # 58153 (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments / Reference: Bank 58153	Revision:

- Unit 1 Reactor power is 100%
- ANY RCP SEAL WTR INJ FLO LO (5A-1.6) is in alarm
- RCP SEAL WTR INJ FILT 1 ΔP HI (5A-2.6) is in alarm
- Seal injection flow is 0 gpm
- ABN-101, REACTOR COOLANT PUMP TRIP/MALFUNCTION has been initiated
- CCW Thermal Barrier heat exchanger flow indicates 30 gpm

Based on the above plant conditions, complete the following statements.

1. RCPs ____(1)____ required to be tripped.
2. Seal injection is isolated in order to prevent thermal shock to the seals which could directly result in ____(2)____.

A. (1) are
(2) excessive RCS leakage

B. (1) are
(2) voiding the CCW system

C. (1) are NOT
(2) excessive RCS leakage

D. (1) are NOT
(2) voiding the CCW system

Answer: A

Answer Explanation

Comments / Reference: Bank 58153	Revision:

A 1st part is correct. Per ABN-101, Section 7 (Loss of Seal Injection), if CCW is less than 35 gpm, you are directed to section 9 which will direct tripping the reactor, tripping RCPs and isolating seal injection and CCW thermal barrier return from the RCPs. 2nd part is correct. Thermal shocking the seals could result in seal damage (not seating correctly) which could result in excessive leakage. Plausible because voiding CCW is a concern but is addressed by isolating CCW return from the thermal barrier heat exchanger.

B 1st part is correct (See A). 2nd part is incorrect because the concern is with thermal shocking the RCP seals, ultimately resulting in excessive leakage not CCW voiding. (See A).

C 1st part is incorrect because the RCPs are required to be tripped if CCW flow is < 35 gpm with a loss of seal injection. It is plausible because if flow were 5 gpm higher, it would be correct. 2nd part is correct (See A).

D 1st part is incorrect but plausible (see C). 2nd part is incorrect but plausible (see B).

Question 42 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	58153
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	Unit 1 Reactor power is 100% ANY RCP SEAL WTR INJ FLO LO (5A-1.6) is in alarm RCP SEAL WTR INJ FIL
K/A:	15/17 AA1.07
Question Reference:	ABN-101
SRO:	
Comments:	LC22 RO Retake NRC

Comments / Reference: ABN-105	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-105
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 7 OF 40

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION:
- With NO Seal Injection flow AND NO Thermal Barrier cooling the affected RCP must be secured PROMPTLY (Within approximately FOUR minutes) to prevent damage to the RCP Seal.
- Consideration should be given to ensure gas binding not a factor before starting a charging pump. Indications of potential gas binding are:
  - PDP SUCT STAB LVL HI-HI (6A-1.8)
  - CHRG FLO HI/LO (6A-3.4)
  - VCT LVL LO-LO (6A-4.5)
  - Fluctuating charging header pressure/flow prior to pump trip.
- Section 7.0 provides for recovery from gas binding of a charging pump.

NOTE: Diamond step 1 denotes Initial Operator Action. Step 1 RNO actions may be performed concurrently.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ ◇1 START a Centrifugal Charging Pump	PERFORM the following: a. ENSURE Letdown Flow - ISOLATED • 1/u-8149A, LTDN ORIFICE ISOL VLV (45 GPM) - CLOSED • 1/u-8149B, LTDN ORIFICE ISOL VLV (75 GPM) - CLOSED • 1/u-8149C, LTDN ORIFICE ISOL VLV (75 GPM) - CLOSED • 1/u-LCV-459, LTDN ISOL VLV - CLOSED • 1/u-LCV-460, LTDN ISOL VLV - CLOSED

"Step continued next page"

Section 3.3

Comments / Reference: ABN-105	Revision: 8

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3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 Continued	b. VERIFY Component Cooling Water flow to RCP Thermal Barrier HX(s)-GREATER THAN 35 GPM: • u-FI-4678, RCP 1 THBR CLR CCW RET FLO • u-FI-4682, RCP 2 THBR CLR CCW RET FLO • u-FI-4686, RCP 3 THBR CLR CCW RET FLO • u-FI-4690, RCP 4 THBR CLR CCW RET FLO IF Thermal Barrier Flow ≤ 35 gpm, THEN PERFORM ABN-101 while continuing this procedure.

NOTE: IF NO Charging Pump available, THEN Plant Management should be notified prior to shutdown due to NO boration path.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2 VERIFY at least one Charging Pump - RUNNING	START a PD Pump per SOP-103A/B. IF Charging pump will NOT start, THEN GO TO ABN-101.

Section 3.3

-

Comments / Reference: ABN-101	Revision: 13

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 13	PAGE 46 OF 54

9.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: With NO Seal Injection flow AND NO Thermal Barrier cooling the affected RCP must be secured PROMPTLY (Within approximately FOUR minutes) to prevent damage to the RCP Seal.

☐ 1 TRIP the Reactor AND GO TO EOP-0.0A/B while other operators continue this procedure.

NOTE: IF all RCPs are stopped during the performance of this procedure, THEN Attachment 3 should be PERFORMED to isolate dilution paths when time permits.

[C]

☐ 2 STOP affected RCP(s).

- 1/u-PCPX1, RCP 1
- 1/u-PCPX2, RCP 2
- 1/u-PCPX3, RCP 3
- 1/u-PCPX4, RCP 4

☐ 3 VERIFY the number 1 Seal Leakoff Valve for affected RCP(s) - OPEN

- 1/u-8141A, RCP 1 SEAL 1 LKOFF VLV
- 1/u-8141B, RCP 2 SEAL 1 LKOFF VLV
- 1/u-8141C, RCP 3 SEAL 1 LKOFF VLV
- 1/u-8141D, RCP 4 SEAL 1 LKOFF VLV

Section 9.3

Comments / Reference: ABN-101	Revision: 13

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 13	PAGE 47 OF 54

9.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 4 CLOSE the Seal Injection Isolation Valve to affected RCP(s). [C] • 1/u-8351A, RCP 1 SEAL WTR INJ VLV • 1/u-8351B, RCP 2 SEAL WTR INJ VLV • 1/u-8351C, RCP 3 SEAL WTR INJ VLV • 1/u-8351D, RCP 4 SEAL WTR INJ VLV	
☐ 5 VERIFY RCP Thermal Barrier Cooler CCW Return Valves from affected RCP(s) - CLOSED: • u-HS-4691, RCP 1 THBR CLR CCW RET VLV • u-HS-4692, RCP 2 THBR CLR CCW RET VLV • u-HS-4693, RCP 3 THBR CLR CCW RET VLV • u-HS-4694, RCP 4 THBR CLR CCW RET VLV	Manually CLOSE valves as necessary.
☐ 6 VERIFY u-HS-4709, THBR CLR CCW RET ISOL VLV - CLOSED	Manually CLOSE the CCW Thermal Barrier Return Isolation Valve.

Section 9.3

Comments / Reference: RCS Study Guide	Revision: 00-0000

OP51.SYS.RC1

Hi/Low lube oil level - operator should monitor pump parameters. If the bearing temperatures reach 195°F, then the reactor and the affected RCP should be tripped.

#1 seal failure - This section is divided up into four different scenarios. If #1 seal leakoff flow on affected RCP is greater than 6.0 gpm with temperatures increasing or total seal leakoff flow exceeds 8 gpm, then the reactor and the affected RCP should be tripped immediately, and the leakoff valve closed when the RCP has stopped rotating (3-5 minutes). If greater than 6.0 gpm and temperatures are stable, then the unit must be shutdown within 8 hours. If the #1seal leakoff is less than .8 gpm, then the unit must be shutdown within 8 hours. If the RCP radial bearing or seal inlet temperature increases, then the reactor and the pump should be tripped immediately, and the #1 seal leakoff valve closed after the RCP has stopped rotating.

#2 or #3 seal failure - Pump operation is allowed to continue as long as other pump parameters (vibration, temperatures, etc.) stay within normal bands.

Excessive RCP vibration - A vibration of 20 mil shaft or 5 mil frame, or increase of ≥ 1 mil/hr on the shaft when ≥ 15 mils, or increase ≥ 0.2 mil/hr on the frame when ≥ 3 mils requires immediate tripping of the reactor and the RCP. Vibration ≥15 mil shaft or 3 mil frame requires consulting management and engineering to determine if the unit is to be shutdown and the affected pump stopped.

Loss of seal injection - The operator verifies CCW flow to the thermal barrier heat exchanger, since this is required to cool the pump radial bearing and seal package. Other actions require a check of other parameters on the affected pump. If the pump radial bearing temperature increases to 225°F or seal inlet temperature increases to 235°F, then the reactor and the affected RCP should be tripped.

RCP high temperature or loss of CCW to any RCP - The operators verify that satisfactory seal injection flow is being supplied to the RCP. Temperatures for the pump and motor bearing and motor windings are monitored. If these temperatures exceed maximum limits, then the reactor is tripped and the pump is stopped.

RCP temperature limits are as follows:

Motor stator winding temperature - 300°F
Motor upper radial bearing temperature - 195°F
Motor upper thrust bearing temperature - 195°F
Motor lower radial bearing temperature - 195°F
Motor lower thrust bearing temperature - 195°F
Lower seal water (pump radial) bearing temperature - 225°F

Loss of seal injection and thermal barrier cooling water - With no seal injection flow and no thermal barrier cooling the affected RCP must be secured within approximately FOUR minutes. The reactor is tripped and the affected pump(s) stopped. Seal injection and thermal barrier cooling return valves are closed. Isolating the RCP seal package from seal injection prevents the thermal shock that would be encountered by restoring seal injection to an abnormally hot RCP seal. RCS leakage is monitored and a cooldown to MODE 5 is initiated.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	005.A1.02		
Level of Difficulty: 2	Importance Rating	3.3		

Residual Heat Removal: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: RHR flow rate

Question # 3

Given the following conditions:

- RHR is in service in Mode 5
- Instrument Air is lost to 2-FCV-618, RHR HX 1 BYP FLO
- ABN-104, Residual Heat Removal System Malfunction, Section 4.0, Mode 4, 5 or 6 Loss of RCS Temperature/flow Control RCS Filled, is in progress

2-FCV-618 fails __(1)__ when air is lost.

ABN-104 directs manual control of the RHR HX Bypass and Outlet valves to maintain a MINIMUM design RHR flow of __(2)__ gpm.

A. (1) OPEN
   (2) 2900

B. (1) OPEN
   (2) 3800

C. (1) CLOSED
   (2) 2900

D. (1) CLOSED
   (2) 3800

Answer: D

K/A Match: K/A match due to requiring knowledge of flow requirements for the RHR system.

Explanation:

A. Incorrect. First part is incorrect because the valve fails closed. It is plausible because if it were the RHR HX Outlet Valve, it would be correct. Second part is incorrect because the minimum design flow to be established is 3800 gpm. It is plausible because it were asking about the minimum CCW flow through the HX, it would be correct.

B. Incorrect. First part is incorrect but plausible (see A). Second part is correct. ABN-104 directs establishing between 3800 and 4000 gpm RHR flow.

C. Incorrect. First part is correct. The RHR HX Bypass Valve fails closed upon a loss of air. Second part is incorrect but plausible (see A).

D. Correct. First part is correct (see C). Second part is correct (see B).

Technical Reference(s)	ABN-301	Attached w/ Revision # See Comments / Reference
	ABN-104	
	DBD ME-260	

Proposed references to be provided during examination: ____

Learning Objective: **ANALYZE** the response to a Loss of RCS Temperature/Flow Control in accordance with ABN-104, Residual Heat Removal System Malfunctions. (ABN.104.OB04)

Question Source: Bank # 81933
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments / Reference: ABN-301	Revision: 14

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 14	PAGE 48 OF 130

ATTACHMENT 1
PAGE 4 OF 15

CONTROL BOARD AIR OPERATED VALVE FAILURE POSITIONS

LOCATION	COMPONENT	NOMENCLATURE	FAILURE POSITION
CB-04	1/u-8879C	RHR TO CL 3 TEST VLV	F.C.
CB-04	1/u-8879D	RHR TO CL 4 TEST VLV	F.C.
CB-04	1/u-8880	SI/PORV ACCUM N2 ISOL VLV	F.C.
CB-04	1/u-8882	CCP SI TEST VLV	F.C.
CB-04	1/u-8890A	RHR TO CL 1 & 2 TEST VLV	F.C.
CB-04	1/u-8890B	RHR TO CL 3 & 4 TEST VLV	F.C.
CB-04	u-FK-618	RHR HX 1 BYP FLO CTRL	F.C.
CB-04	u-FK-619	RHR HX 2 BYP FLO CTRL	F.C.
CB-04	u-HC-606	RHR HX 1 FLO CTRL	F.O.
CB-04	u-HC-607	RHR HX 2 FLO CTRL	F.O.
CB-04	u-HC-943	ACCUM 1-4 VENT CTRL	F.C.
CB-04	u-HS-6719	SRG TK DEMIN WTR SPLY VLV	F.C.
CB-04	u-HS-6720	SRG TK RMUW SPLY VLV	F.C.
CB-04	u-HS-6712	SRG TK MU VLV	F.C.
CB-04	u-HS-6713	SRG TK MU VLV	F.C.
CB-05	1/u-7126	RCDT VENT ISOL VLV	F.C.
CB-05	1/u-7136	RCDT DRN ISOL VLV	F.C.
CB-05	1/u-7150	RCDT VENT ISOL VLV	F.C.
CB-05	1/u-8026	PRT VENT ISOL VLV	F.C.
CB-05	1/u-8027	PRT VENT ISOL VLV	F.C.
CB-05	1/u-8031	PRT DRN VLV	F.C.
CB-05	1/u-8032	RV SEAL LKOFF VLV	F.O.
CB-05	1/u-8045	RMUW TO PRT SPLY VLV	F.C.
CB-05	1/u-8047	RMUW TO PRT/CNTMT SPLY ISOL VLV	F.C.
CB-05	1/u-8141A	RCP 1 SEAL 1 LKOFF VLV	F.O.

Attachment 1

Comments / Reference: ABN-104	Revision:

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 24 OF 134

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5 Verify RHR flow - BETWEEN 3,800 GPM and 4,000 GPM AND STABLE: • u-FI-618, RHR TO CL 1 & 2 INJ FLO • u-FI-619, RHR TO CL 3 & 4 INJ FLO	Perform the following: a. Manually control the RHR heat exchanger bypass valve AND the RHR heat exchanger outlet valve to maintain between 3,800 gpm and 4,000 gpm AND RCS at desired temperature 1) RHR HX bypass valve: • u-FK-618, RHR HX 1 BYP FLO CTRL • u-FK-619, RHR HX 2 BYP FLO CTRL 2) RHR HX outlet valve: • u-HC-606, RHR HX 1 FLO CTRL • u-HC-607, RHR HX 2 FLO CTRL b. IF RHR flow control is lost due to loss of Instrument Air to flow control valve(s), THEN align emergency air supply to affected valve(s) as follows while continuing this section at Step 6: • Unit 1 Train A, Attachment 12 • Unit 1 Train B, Attachment 13 • Unit 2 Train A, Attachment 14 • Unit 2 Train B, Attachment 15 c. IF required RHR flow can NOT be restored, THEN GO TO Section 2.0, this procedure.

Section 4.3

Comments / Reference: DBD ME-260	Revision: 30

Attachments 1 and 2 contain the RHR pump shop test curves and acceptance curve, respectively.

The design capacity requirement for the RHR pump is the following:

- The two RHR pumps operating together must support normal cooldown as stated in the power generation functional requirements Section 4.3-B. To meet this requirement each pump is required to circulate one-half of the required flow of 7600 gpm from a RCS hot leg, through a RHR heat exchanger (HX), back to two RCS cold legs.

The runout capacity requirement for the RHR pump is the following:

- Each of the pumps must provide the minimum LHSI flow of 4900 gpm, which is required to mitigate a large LOCA. (Refer to the SIS DBD-ME-261 for additional details.)

Other RHR pumps capacity requirements that are not limiting are the following:

- Each of the pumps is required to circulate the flow at 3800 gpm for emergency cooldown. (See Section 4.3-A)

- Each of the pumps is required to circulate the flow at 3800 gpm to maintain the RCS temperature at 140°F during a refueling or cold shutdown while the other pump is being maintained. (See Section 4.3-B)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	2		
	Group	1		
	K/A	006.K1.02		
Level of Difficulty: 3	Importance Rating	4.3		

Emergency Core Cooling: Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: ESFAS

Question # 4

Given the following conditions:

- Unit 1 is operating at 100% power
- Train A equipment is running
- A slow transfer of both safeguards busses to their alternate power supply occurs

__(1)__ will be started as a result of the transfer.

The PDP cannot be started following actuation of the Blackout Sequencers until the BOS __(2)__ clears.

A. (1) Both CCPs
(2) Operator Lockout

B. (1) Both CCPs
(2) Automatic Lockout

C. (1) ONLY CCP 1-01
(2) Operator Lockout

D. (1) ONLY CCP 1-01
(2) Automatic Lockout

Answer: A

K/A Match: K/A match due to requiring knowledge of the relationship between the ESFAS blackout sequencer and the CCPs.

Explanation:

A. Correct. First part is correct. Both CCPs are started by the sequencers. Second part is correct. The PDP is locked out from starting until the operator lockout clears.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible because the PDP is prevented from starting by the operator lockout, not the automatic lockout. The operator lockout clears when the sequence completes timing and the automatic lockout clears when the operator resets the BOS.

C. Incorrect. First part is incorrect, but plausible since only CCP 1-01 was running prior to the transfer and it could be thought that only previously operating equipment would be started. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-602	Attached w/ Revision # See Comments / Reference
	SI and BO Sequencer Study Guide	
	CVCS Study Guide	
	ABN-601	

Proposed references to be provided during examination: ______

Learning Objective: **PREDICT** the response of the instrumentation and controls of the Safety Injection and Blackout Sequencers in accordance with CP-0575-001, Solid State Safeguards Sequencers and associated Automation Industries drawings. (SYS. ES3.OB05)

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 X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ABN-602	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 77 OF 107

ATTACHMENT 2
PAGE 1 OF 4

EQUIPMENT ACTUATED BY BLACKOUT SEQUENCER

The following is a list of components or systems actuated by Blackout Sequencer. Items are ordered in approximate actuation sequence by unit and train.

1. Train A Blackout Sequencer

   a. Starts: CCP 1 (CB-06)
      RECIRC PMP 5 (CB-04)

   b. Switches Control Room Ventilation to Emergency Recirculation

   c. Starts: BATT RM A EXH FN 7 (CV-01)
      BATT RM A EXH FN 8 (CV-01)
      BATT RM C EXH FN 11 (CV-01)

   d. Closes: u-HS-2484, CST DISCH VLV (CB-09)
      u-HS-2397, SG 1 BLDN ISOL VLV (CB-08)
      u-HS-2398, SG 2 BLDN ISOL VLV (CB-08)
      u-HS-2399, SG 3 BLDN ISOL VLV (CB-08)
      u-HS-2400, SG 4 BLDN ISOL VLV (CB-08)
      u-HS-2401A, SG 1 DRUM ISOL VLV (LV-08)
      u-HS-2401B, SG 1 BLDN SMPL ISOL VLV (LV-08)
      u-HS-2402A, SG 2 DRUM ISOL VLV (LV-08)
      u-HS-2402B, SG 2 BLDN SMPL ISOL VLV (LV-08)
      u-HS-2403A, SG 3 DRUM ISOL VLV (LV-08)
      u-HS-2403B, SG 3 BLDN SMPL ISOL VLV (LV-08)
      u-HS-2404A, SG 4 DRUM ISOL VLV (LV-08)
      u-HS-2404B, SG 4 BLDN SMPL ISOL VLV (LV-08)

NOTE: SG 1-4 DRUM and BLDN SMPL VLV may be closed using u-HS-2401C, SG SMPL ISOL VLV (CB-08) and verified using the associated (ZL) lights on CB-08.

Attachment 2

Comments / Reference: SI and BO Sequencer Study Guide	Revision: 6-10-2011

SI and Blackout Sequencers

Attachment 3 Equipment Affected by the Blackout Sequencer Step Time Output Relays				
Step	Relay	Equipment Affected * Bus Under voltage relays would have previously tripped open the breaker **- Loss of voltage to the 480 V AC bus will have resulted in a loss of the stepped down 120 V AC control power energizing the motor contactor. With the motor contactor de-energized, the seal-in circuit for the contactor is broken and a new start signal must be received to restart the equipment. Electrical drawing for Unit 2 would be E2-_____ unless otherwise noted, and for common equipment which will be E1.	Type of Relay- *** Reset at Step 11 (Type 1) ****Reset when BOS is Reset (Type 2)	Time-seconds
1	K035	CCP-01(02)- starts * E1-0031 sht. 53 (55)	Self-resetting***	0.0
1	K035	Chilled Water Recirc Pump 1-05 (1-06) – starts ** E1-0054 sht. 20 (21)	Self-resetting***	0.0
1	K037	HX-5877A(B), HX-5877A1(B1), HX-5877A4(B4) These relays place their Train's Control Room Ventilation Equipment in Emergency Recirculation Mode. Either Units BOS will place CR Vent. in Emergency Recirc Mode. E1-0035 sht. 76 (77)	Operator reset****	0.0

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Comments / Reference: SI and BO Sequencer Study Guide	Revision: 6-10-2011

SI and Blackout Sequencers

Attachment 2 Equipment Affected by Blackout Sequencer Operator Lockout Relays	
Relay	Equipment Affected by the BOS Operator Lockout Relay - *On a Unit's OL, this equipment is just prevented from being started by an Operator using a control switch, unless otherwise noted* *- Bus Under voltage relays would have previously tripped open the breaker, so a start signal will be needed to reclose the breaker **- Loss of voltage to the 480 V AC bus will have resulted in a loss of the stepped down 120 V AC control power energizing the motor contactor. With the motor contactor de-energized, the seal-in circuit for the contactor is broken and a new start signal must be received to restart the equipment. On equipment powered from a common bus (XEB bus) power may not be lost and the equipment may not have stopped. Equipment actuated by a Train B sequencer relay is shown in "()"
K091	HX-5878A1, HX-5878A2 – these CR HVAC relays are de-energized when the sequencer relay K091 A (B) energizes, and the HVAC relays N.O. contacts open to stop Emergency Ventilation mode if it was in progress. E1-0035 sht. 74 (75). Specifically, **CR Makeup Supply Fan 37**-stops, unless the switch is held in the start position. E1-0035 sht. 6 (8) **CR Exhaust Fan 01**-stops, unless the switch is held in the start position. E1-0035 sht. 25 (26). **X-HV-5826 (5289),** the damper won't open in automatic if the fan is running. The damper should close because the fan is not running (the damper would open if its control switch was held in the open position). E1-0035 sht. 3 (10) **Kitchen and Toilet Exhaust Fan 03 (04)**- stops. E1-0035 sht. 27 (28) - these relays basically prevent Emergency Ventilation while the OL is in. (Emergency Ventilation mode is stopped until the OL clears)
K093	**Chilled Water Recirc Pump 05 (6)**- ** E1-0054 sht. 20 (21). See E2-0054 sht. 20 (21) for Unit 2.
K093	**PRZR Heater Group A (D)** * E1-0033 sht. 35 (39). See E2-0033 sht. 35 (39) for Unit 2.
K093	**PRZR Heater Group C (B)*** E1-0033 sht. 33 (37). See E2-0033 sht. 33 (37) for Unit 2.
K093	**P.D. Pump** * E1-0033 sht. 13. See E2-0033 sht. 13 for Unit 2.

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Comments / Reference: CVCS Study Guide	Revision: 00-0000

OP51.SYS.CS1

SYSTEM RESPONSE TO A BLACKOUT SIGNAL

If either train's safeguards bus deenergizes, bus undervoltage signals are sent to that train's blackout sequencer. When the bus is reenergized, either automatically or manually, the affected blackout sequencer functions to automatically start designated loads. These sequencers, one for Train A and one for Train B, are intended to ensure vital loads automatically restart in a manner that will not overload an emergency diesel generator. The designated loads include the centrifugal charging pumps, component cooling water pumps, station service water pumps and the instrument air compressors.

A loss of offsite power when operating at power will result in a reactor trip and the loss of all 6.9kv and 480v buses. Most loads will be shed from the buses by breaker undervoltage trips. The safeguards buses will reenergize almost immediately from the emergency diesel generators. Vital station loads will begin to sequence onto the buses over approximately the next 90 seconds. The system response related to the chemical and volume control system is that both centrifugal charging pumps will start and cooling flow will be reestablished to the system heat exchangers.

If, instead, a single safeguards bus loses and then immediately regains power, only that train of equipment will be affected. For example, assume the Train A CCP is in operation at power. Breaker uEA2-1, the normal feeder breaker for the Train B 6.9kv Safeguards Bus, trips spuriously. As a result, bus uEA2 slow transfers to alternate power. The Train B Blackout Sequencer actuates because the bus deenergized and then reenergized. All equipment actuated by the Train B Blackout Sequencer starts, even if it was not previously running. In the chemical and volume control system, the Train B CCP starts. Following the sequencer actuation, one of the CCPs can be shutdown to return to the previous configuration of one CCP in operation.

When operating in a solid plant configuration, a reactor coolant system pressure excursion can be experienced if the safeguards bus which is powering the running CCP and RHR pump deenergizes and then reenergizes. The pressure excursion can occur because the blackout sequencer restarts the CCP but does not provide a start signal to the RHR pump. As described earlier, the loss of the RHR pump during solid plant operations can cause a pressure excursion because charging continues after letdown flow is lost.

Comments / Reference: ABN-601	Revision: 16

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-601
RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION	REVISION NO. 16	PAGE 5 OF 256

2.1 b. Plant Indications

XST1/XST1A/138 KV FEEDER

- Possible loss of Unit 1 safeguard buses
- Slow transfer of Unit 2 safeguard buses to their alternate supply

XST2/1ST/XST2A/345 KV FEEDER

- XST2 OR XST2A low side breakers OPEN
- 1ST low side breakers OPEN
- Slow transfer of Unit 1 safeguard buses to their alternate supply
- Possible loss of Unit 1 non-safeguard buses
- Possible loss of Unit 2 safeguard buses
- Possible start of Diesel Fire Pumps

2ST

- 2ST low side breakers OPEN
- Possible loss of Unit 2 non-safeguard buses

2.2 Automatic Actions

XST1/XST1A/138 KV FEEDER

- High speed ground switch 8083 (GXST1) CLOSED due to fault
- MOAS 8085 (DXST1) OR MOAS 8095 (DXST1A) opens due to fault
- 6.9 KV breakers 1EA1-2, 1EA2-2, 2EA1-1 and 2EA2-1 OPEN
- 138 KV switchyard breakers 7030 and 7040 OPEN

Section 2.1

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	2		
	Group	1		
	K/A	007.K5.02		
Level of Difficulty: 2	Importance Rating	3.1		

Pressurizer Relief/Quench Tank: Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR

Question # 5

Given the following conditions:

- After being used to draw a vacuum in the RCS, the PRT has been isolated from the PRZR
- RCS pressure is 5 psia
- RCS and PRZR temperatures are equalized at 130ºF
- Actual PRZR level is 50%

Which of the following describes the PREFERRED method of establishing a steam bubble in the PRZR in accordance with SOP-101A, Reactor Coolant System?

Adjust Charging and Letdown to __(1)__.

Energize PRZR heaters to heat up the PRZR, establishing a steam bubble at approximately __(2)__ in the PRZR.

A. (1) raise PRZR level to 100%
(2) 212ºF

B. (1) raise PRZR level to 100%
(2) 162ºF

C. (1) maintain PRZR level constant
(2) 212ºF

D. (1) maintain PRZR level constant
(2) 162ºF

Answer: D

K/A Match: K/A match due to requiring knowledge of the method used to establish a bubble in the pressurizer.

Explanation:

A. Incorrect. First part is incorrect, but plausible since this is similar to an alternate method of establishing a bubble while solid. Second part is incorrect, but plausible since 212°F is saturation temperature for normal atmospheric pressure.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A).

D. Correct. First part is correct. Per SOP-101, during vacuum fill of the RCS a bubble in the PRZR is established while the RCS is at a vacuum. Charging and Letdown are adjusted as needed to maintain level and pressure constant in preparation for establishing a bubble. Second part is correct. The bubble is formed at 162°F which is saturation for the 5 psia condition established.

Technical Reference(s)	SOP-101	Attached w/ Revision # See Comments / Reference
	Steam Tables	

Proposed references to be provided during examination: Steam Tables

Learning Objective: **DISCUSS** the steps required to fill the RCS and establish a pressurizer bubble in accordance with SOP-101A, Reactor Coolant System. (IPO.001.OB01)

Question Source: Bank # 23551
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: SOP-101A	Revision: 18

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-101A
REACTOR COOLANT SYSTEM	REVISION NO. 18 CONTINUOUS USE	PAGE 39 OF 86

5.5.2 As 1-LI-462, PRZR LVL COLD CAL approaches 50%, PERFORM the following to ensure the level increase can be stopped.

A. IF the RWST is being used to fill the RCS,
THEN
TRANSFER charging pump suction to VCT.

1) OPEN the VCT to charging pump suction valves:

- ☐ 1/1-LCV-112B, VCT TO CHRG PMP SUCT VLV
- ☐ 1/1-LCV-112C, VCT TO CHRG PMP SUCT VLV

2) VERIFY OPEN the charging pump suction high point vent valves:

- ☐ 1-ZL-8220, CHARGING PMP SUCT HI POINT VENT VLV
- ☐ 1-ZL-8221, CHARGING PMP SUCT HI POINT VENT VLV

3) CLOSE the RWST to charging pump suction valves:

- ☐ 1/1-LCV-112D, RWST TO CHRG PMP SUCT VLV
- ☐ 1/1-LCV-112E, RWST TO CHRG PMP SUCT VLV

☐ B. Slowly OPEN 1-PK-131 LTDN HX OUT PRESS CTRL to establish letdown from RHR.

☐ C. ADJUST charging flow as necessary to maintain Pressurizer level constant AND RCP seal injection flow to each RCP between 6 AND 13 gpm.

Comments / Reference: SOP-101A	Revision: 18

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-101A
REACTOR COOLANT SYSTEM	REVISION NO. 18 CONTINUOUS USE	PAGE 42 OF 86

5.5.11 DRAW a Pressurizer bubble by performing the following steps:

A. ENSURE Pressurizer spray valve controllers are in MANUAL AND at 0% demand.

- ☐ 1-PK-455B, RC LOOP 1 PRZR SPR VLV CTRL
- ☐ 1-PK-455C, RC LOOP 4 PRZR SPR VLV CTRL

☐ B. VERIFY 1/1-8145, RCS AUX SPR VLV is CLOSED.

☐ C. ADJUST charging as necessary to MAINTAIN Przr level at 50%. (Actual Level)

☐ D. ENSURE OPT-407 is being performed (TS SR 3.4.3.C).

E. REMOVE Standard Clearance #00797 AND ENSURE the following breakers are RACKED IN to CONNECT:

- ☐ 1PCPR, PRZR 1-01 HEATER BACKUP GROUP C ISOLATION XFMR 1-07 FEEDER BREAKER
- ☐ 1PCPR1, PRZR 1-01 HEATER BACKUP GROUP A ISOLATION XFMR 1-05 FEEDER BREAKER
- ☐ 1PCPR2, PRESSURIZER 1-01 HTR GROUP B ISOL TRANSFORMER 1-06 FEEDER BREAKER
- ☐ 1PCPR3, PRESSURIZER 1-01 HTR GROUP D ISOL TRANSFORMER 1-08 FEEDER BREAKER

-

Comments / Reference: SOP-101A	Revision: 18

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-101A
REACTOR COOLANT SYSTEM	REVISION NO. 18 CONTINUOUS USE	PAGE 43 OF 86

CAUTION: The pressurizer heatup rate is limited to ≤100°F in one hour. A maximum temperature differential of 320°F should NOT be exceeded between the RCS AND the Pressurizer liquid temperature.

NOTE:
- Pressurizer heatup should be conducted at a uniform rate of change vice permitting step wise temperature increases.
- As RCS pressure increases, Pressurizer level may decrease if any voids remain in the S/G tubes. Experience has shown this decrease to be <5%.

5.5.11 F. TURN Pressurizer heaters ON, as required, to initiate a Pressurizer heatup ≤100°F in one hour.

- ☐ 1/1-PCPR1, PRZR BACKUP HTR GROUP A
- ☐ 1/1-PCPR2, PRZR BACKUP HTR GROUP B
- ☐ 1/1-PCPR3, PRZR BACKUP HTR GROUP D
- ☐ 1/1-PCPR, PRZR CTRL HTR GROUP C

☐ G. VERIFY 1-TI-453, PRZR LIQ TEMP is at saturation temperature for present RCS pressure. (Saturation temperature for 5 psia is ~162°F)

5.5.12 WHEN RCS pressure is > 100 psig AND Pressurizer level is > 35% as indicated on 1-LI-462, PRZR LVL COLD CAL,
THEN
PERFORM the following:

☐ A. NOTIFY the OCC that "Loops Not Filled" may be exited.

B. OPEN the RCP Seal Water Return Valves:

- ☐ 1/1-8112, RCP SEAL WTR RET ISOL VLV
- ☐ 1/1-8100, RCP SEAL WTR RET ISOL VLV
- ☐ 1/1-8141A, RCP 1 SEAL 1 LKOFF VLV
- ☐ 1/1-8141B, RCP 2 SEAL 1 LKOFF VLV
- ☐ 1/1-8141C, RCP 3 SEAL 1 LKOFF VLV
- ☐ 1/1-8141D, RCP 4 SEAL 1 LKOFF VLV

☐ C. OPEN 1/1-8142, RCP SEAL 1 BYP VLV.

☐ D. After approximately 5 minutes, CLOSE 1/1-8142, RCP SEAL 1 BYP VLV.

Comments / Reference: Steam Tables	Revision:

Input Data

Units Imperial

Steam Pressure 5 psi abs

Show Advanced Options

Calculate

Clear

Saturated Steam Temperature	162.186	°F
Latent Heat of Steam	1000.75	BTU/l

Input Data

Units Imperial

Steam Pressure 50 psig

Show Advanced Options

Calculate

Clear

Saturated Steam Temperature	297.653	°F
Latent Heat of Steam	912.106	BTU/l

Comments / Reference: SOP-101	Revision: 18

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-101A
REACTOR COOLANT SYSTEM	REVISION NO. 18 CONTINUOUS USE	PAGE 51 OF 86

5.6 Taking the Pressurizer Solid prior to Establishing a Bubble

This section describes the steps to take the Pressurizer to a solid condition, start the RCPs AND establish a bubble. This is NOT the preferred method for establishing a bubble when an RCS vacuum fill is performed.

5.6.1 As 1-LI-462, PRZR LVL COLD CAL approaches 100%, PERFORM the following to ensure the level increase can be STOPPED.

A. IF the RWST is being used to fill the RCS,
THEN
TRANSFER charging pump suction to VCT.

1) OPEN the VCT to charging pump suction valves:

- ☐ 1/1-LCV-112B, VCT TO CHRG PMP SUCT VLV
- ☐ 1/1-LCV-112C, VCT TO CHRG PMP SUCT VLV

2) VERIFY the charging pump suction high point vent valves are OPEN:

- ☐ 1-ZL-8220, CHARGING PMP SUCT HI POINT VENT VLV
- ☐ 1-ZL-8221, CHARGING PMP SUCT HI POINT VENT VLV

3) CLOSE the RWST to charging pump suction valves:

- ☐ 1/1-LCV-112D, RWST TO CHRG PMP SUCT VLV
- ☐ 1/1-LCV-112E, RWST TO CHRG PMP SUCT VLV

☐ B. Slowly OPEN 1-PK-131 LTDN HX OUT PRESS CTRL to establish letdown from RHR.

☐ C. ADJUST charging flow as necessary to maintain Pressurizer level constant AND RCP seal injection flow to each RCP between 6 AND 13 gpm.

5.6.2 CONTINUE to increase Pressurizer level to 100% as indicated on 1-LI-462, PRZR LVL COLD CAL.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	2		
	Group	1		
	K/A	008.A2.03		
Level of Difficulty: 2	Importance Rating	3.0		

Component Cooling Water: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low CCW temperature

Question # 6

Given the following conditions:

- Reactor power = 100%
- 1-ALB-1, 1.7, SSW TRN A/B HDR PRESS LO, alarms
- ABN-501, Station Service Water System Malfunction, is entered for the low SSW header pressure condition
- CCW 1-01 HX outlet temperature is slowly rising

In accordance with ABN-501, when CCW HX Outlet Temperature FIRST reaches __(1)__, CCW Pump 1-01 is placed in PULL-OUT.

A subsequent trip of SSW Pump 1-02 would require __(2)__.

A. (1) 122ºF
(2) a Reactor Trip ONLY

B. (1) 122ºF
(2) a Reactor Trip and Trip of ALL RCPs

C. (1) 140ºF
(2) a Reactor Trip ONLY

D. (1) 140ºF
(2) a Reactor Trip and Trip of ALL RCPs

Answer: B

K/A Match: K/A match due to requiring knowledge of the CCW temperature limits and the actions to be taken in the event the limits are exceeded.

Explanation:

A. Incorrect. First part is correct (see B). Second part is incorrect, but plausible since CCW Pump 1-02 remains running supplying the RCPs but with no SSW flow on the same train the plant is required to be tripped and the RCPs secured.

B. Correct. First part is correct. If CCW HX 1-01 outlet temperature reaches 122ºF during these conditions, CCW pump 1-01 is to be stopped. Second part is correct. If both CCW pump 1-01 and SSW pump 1-02, opposite trains, are not operating, cooling is lost to CCW loads and the Reactor and RCPs are to be tripped.

C. Incorrect. First part is incorrect, but plausible, because 140ºF is from the design of the system (to lower RCS temperature from 350ºF to 140ºF in 24 hours). Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	ALM-0011A	Attached w/ Revision # See Comments / Reference
	ABN-501	
	ABN-502	

Proposed references to be provided during examination: ______

Learning Objective: **DISCUSS** the response to Station Service Water Header Pressure Low in accordance with ABN-501, Station Service Water System Malfunction. (ABN.501.OB102)

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 ___X___

10 CFR Part 55 Content:
- 55.41 ___5___
- 55.43 ______

Comments / Reference: ALM-0011A	Revision: 10

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0011A
ALARM PROCEDURE 1-ALB-1	REVISION NO. 10	PAGE 20 OF 143

ANNUNCIATOR NOM./NO.: **SSW TRN A/B HDR PRESS LO** **1.7**

PROBABLE CAUSES:

Operating SSW pump malfunction
10" Safeguard loop out of service
System startup

AUTOMATIC ACTIONS:

The standby SSW pump AND associated CCW pump starts.

OPERATOR ACTIONS:

1. DETERMINE affected SSW pump:
   - 1-HS-4250A, SSWP 1
   - 1-HS-4251A, SSWP 2

   A. IF an SSW pump tripped,
      THEN
      REFER to ABN-501 for Station Service Water Pump Trip.

   B. IF Train A/B header pressure is low,
      THEN
      REFER to ABN-501 for Station Service Water Header Pressure Low.

2. WITH an SSW pump in service,
   THEN
   VENT the 10 inch safeguard header per SOP-501A for Filling AND Venting to clear alarm condition.

3. CORRECT the condition OR INITIATE a CR per STA-421, as applicable.

Comments / Reference: ABN-501	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 9 OF 50

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: The CCW Pump on the affected train may be left operating at the discretion of the Shift Manager. However, with this pump operating, the affected SSW Pump will have an Auto Start Signal to it.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 4 Verify equipment in the affected Train - NOT REQUIRED FOR OPERATION: • CCP • Diesel Generator • CCW Pump • SI Pump • Containment Spray Pumps	Start equipment in the unaffected Train as required to support Plant Operations: • CCP • Diesel Generator • CCW Pump • SI Pump • Containment Spray Pumps

NOTE:
- The diesel generator can be operated, with load, for approximately one minute without SSW flow and not affect diesel performance.
- When a fault exists on the 6.9KV safeguard bus, the SSW pump will not be running to supply cooling water to the DG. The time this condition exists should be minimized (approximately 15 minutes) to prevent damage to the DG.

☐ 5 Shutdown equipment in the affected Train as follows:

- ☐ CCP - PULL OUT
- ☐ Diesel Generator - place CS-uDGuE (emergency stop/start) in PULLOUT.
- ☐ SI Pump - PULL OUT
- ☐ Containment Spray Pumps - PULL OUT
- ☐ SSW Pump - PULL OUT

Section 3.3

Comments / Reference: ABN-501	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 12 OF 50

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8 Check status of affected CCW System:	
☐ a. Verify CCW Pump - RUNNING	a. Perform ABN-502, if required, THEN continue with this procedure at Step 9.

NOTE: Step b. is a continuous action step.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ b. Periodically, during the performance of this procedure, verify CCW Heat Exchanger Outlet Temperature on affected Train - LESS THAN 122°F: • u-TI-4530, CCW HX 1 OUT TEMP • u-TI-4534, CCW HX 2 OUT TEMP	b. Perform the following: 1) Stop the affected CCW Pump -AND- Place handswitch in PULL OUT 2) Perform ABN-502, THEN continue with this procedure at Step 9.

☐ 9 Refer to EPP-201.

☐ 10 Refer to TS listed in Section 6.1.

☐ 11 Complete OPT-215 verification within one hour, if required.

☐ 12 Initiate a Work Request per STA-606 as required.

END OF SECTION

Section 3.3

Comments / Reference: ABN-502	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION 11	PAGE 4 OF 75

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 VERIFY unaffected train CCW Pump - RUNNING	Manually START the CCW Pump in the unaffected train. IF the pump fails to start, THEN GO TO Section 6.0 of this procedure.

NOTE: Opposite train's SSW Pump and CCW Pump DO NOT provide cooling to CCW loads from the Ultimate Heat Sink.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2 VERIFY unaffected train SSW Pump - RUNNING	Perform the following: a. Manually start the SSW pump in the unaffected train. b. IF the SSW pump in the unaffected train will not start, THEN perform the following: 1) TRIP the Reactor 2) GO TO EOP-0.0A/B while other qualified operators continue this procedure. 3) TRIP ALL RCPs. 4) GO TO ABN-501, Section 5.
☐ 3 VERIFY unaffected train Safety Chiller Recirc Pump - RUNNING	Manually START the unaffected Safety Chiller Recirc Pump.

Section 2.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	2		
	Group	1		
	K/A	010.K5.01		
Level of Difficulty: 3	Importance Rating	3.5		

Pressurizer Pressure Control: Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Determination of condition of fluid in PZR, using steam tables

Question # 7

Given the following conditions:

- PRZR Pressure indicates 335 psig
- A malfunction in CVCS causes an insurge to the PRZR, resulting in level rising 4%
- PRZR liquid space temperature indicates 400ºF during the insurge

During the insurge, the PRZR liquid space is subcooled by approximately __(1)__.

Any pressure rise during the insurge into the PRZR is limited by __(2)__.

A. (1) 23ºF
(2) steam condensing into a liquid

B. (1) 23ºF
(2) liquid flashing to steam

C. (1) 32ºF
(2) steam condensing into a liquid

D. (1) 32ºF
(2) liquid flashing to steam

Answer: C

K/A Match: K/A match due to requiring ability to determine the amount of subcooling in the pressurizer using steam tables.

Explanation:

A. Incorrect. First part is incorrect, but plausible since this value would be obtained if 335 psig is converted incorrectly to a value of 320 psia which has a saturation temperature of 423°F. Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since this is the dynamics that occur on an outsurge and not an insurge.

C. Correct. First part is correct. 335 psig is equivalent to approximately 350 psia and saturation temperature for this pressure is approximately 432°F. With temperature indication of 400°F, it is subcooled by 32°F. Second part is correct. A pressure rise from an insurge into the Pressurizer is limited by steam condensing into liquid.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	Steam Tables	Attached w/ Revision # See Comments / Reference
	Pressurizer Pressure/Level Study Guide	

Proposed references to be provided during examination: Steam Tables

Learning Objective: **DESCRIBE** the instrumentation and controls of the Pressurizer Pressure Control System including the system response in accordance with the CPNPP FSAR and DBD-ME-250, Reactor Coolant System. (SYS.PP1.OB04)

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 X

10 CFR Part 55 Content: 55.41 5
55.43 ____

Comments / Reference: Steam Tables	Revision:

Input Data Units Imperial

Steam Pressure 335 psig

Show Advanced Optior

Calculate Clear

Saturated Steam Temperature 431.672 °F

Input Data Units Imperial

Steam Pressure 350 psi abs

Show Advanced Optio

Calculate Clear

Saturated Steam Temperature 431.754 °F

Input Data Units Imperial

Steam Pressure 320 psi abs

Show Advanced Opti

Calculate Clear

Saturated Steam Temperature 423.331 °F

Comments / Reference: Pressurizer Pressure/Level Study Guide	Revision: 00-0000

OP51.SYS.PP1

During normal plant operations, the pressurizer is filled with boiling water and steam. The temperature of the boiling (or saturated) water determines the pressure inside the entire RCS. Pressurizer temperature is controlled to regulate RCS pressure by energizing electric heaters in the bottom of the pressurizer to raise pressure, and by spraying the steam space (or steam bubble) in the top of the pressurizer with cooler water to reduce pressure.

Under normal operating conditions, the Pressurizer Pressure Control System will automatically maintain the plant at 2235 psig. Heaters maintain a saturated condition in the pressurizer and spray valves throttle open to hold pressure at the 2235 psig setpoint. Backup banks of heaters energize on decreasing RCS pressure. On increasing pressure, spray valves open automatically to cause partial steam bubble condensation. If pressure continues to increase, pneumatic Power Operated Relief Valves (PORVs) and code safety valves open to relieve steam from the pressurizer and ensure that the integrity of the RCS is not lost due to high pressure conditions.

A constant pressurizer level indicates that a balance exists between the charging flow into the RCS and the letdown flow into the CVCS. During transients, pressurizer level will change because the reactor coolant will expand and contract as the plant temperature changes. The expansion and compression of the steam bubble in the pressurizer limits RCS pressure changes.

On an outsurge, or drop in pressurizer level, the expansion of the steam bubble causes a drop in pressure. As pressure decreases, some of the pressurizer liquid, which is at saturation (boiling) temperature, flashes to steam and limits the pressure drop. Conversely, on an insurge, or increase in pressurizer level, the compression of the steam bubble causes an increase in pressure, which is limited by the condensation of some of the steam.

Average RCS temperature (TAVG) increases from 557°F at 0% reactor power to 585.4°F (589.2°F) at 100% reactor power. Pressurizer level is programmed to change as a function of the TAVG change. This allows the water in the RCS to expand as temperature increases from 0 - 100% power, raising pressurizer level from 25% to 60% without having to drain water from the RCS. In the same manner, pressurizer level is allowed to decrease during power reduction as the RCS water cools without the need to add water to make up for the contraction. The RCS volume is allowed to change as a result of temperature changes, while the mass of the RCS water remains constant. This reduces transient response time and the amount of water required to be processed during normal operations.

PRESSURIZER PRESSURE CONTROL

PRESSURE CONTROL COMPONENTS

Pressure Measuring Instruments

Pressure is a force exerted by some medium, usually a fluid, over a unit area (e.g. pounds per square inch). Pressurizer pressure instruments measure the difference between pressure in the pressurizer and in the containment building atmosphere. This measurement is referred to as gauge pressure and is expressed as pounds per square inch gauge (psig).

Five pressure detectors measure the pressure in the steam space at the top of the pressurizer. CPNPP uses bourdon tube instruments to provide pressurizer pressure signals. The bourdon tube elastic

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	012.K6.06		
Level of Difficulty: 2	Importance Rating	2.7		

Reactor Protection: Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Sensors and detectors

Question # 8

Given the following conditions:

- PRZR level transmitter LT-459 failed high (100%)
- 70 hours later the channel is being tripped
- The NCT switch LS/0459 BS-1 is in NORMAL

When the I&C Technician takes the Master Test Card (NMT) switch to CLOSED for LT-459:

LI-459, PRZR Level Channel I will indicate __(1)__.

The TSLB for PRZR LVL HI LB-459 will __(2)__.

A. (1) 0%
(2) remain LIT

B. (1) 0%
(2) change from DARK to LIT

C. (1) 100%
(2) remain LIT

D. (1) 100%
(2) change from DARK to LIT

Answer: A

K/A Match: K/A match due to requiring knowledge of how a failed transmitter will affect the reactor protection system indication.

Explanation:

A. Correct. First part is correct. Per ABN-706, indication will go to 0% as the transmitter is disconnected from the circuitry. Second part is correct. TSLB would light if dark, but TSLB PRZR LVL HI LB-459 was already lit from the failure high >92%.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible because TSLB PRZR LVL HI LB-459 would light if dark, but it is already lit from the high failure.

C. Incorrect. First part is incorrect, but plausible because the indication was at 100% already, but will go to 0% when NMT is closed. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-706	Attached w/ Revision # See Comments / Reference
	7300 Process Lesson Plan	

Proposed references to be provided during examination: ______

Learning Objective: **EXPLAIN** the instrumentation and controls of the 7300 Process Control system and **PREDICT** the system response in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems and Westinghouse Drawings 7247D05. (SYS.IC3.OB04)

Question Source: Bank # 19173
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ABN-706	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-706
PRESSURIZER LEVEL INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 10 OF 14

ATTACHMENT 3
PAGE 1 OF 1

PRESSURIZER LEVEL CHANNEL
BISTABLE TRIP SWITCH IDENTIFICATION

PROT SET I, CH 0459

1. PLACE the following NMT card test switch in CLOSED.

	CARD TAG #	CAB-FRAM-CARD	SWITCH	POSITION
☐	u-UY/0761S	01-08-73	SW7	CLOSED

2. ENSURE the following NCT card switches in NORM.

	CARD TAG #	CAB-FRAM-CARD	SWITCH	POSITION
☐	LS/0459	01-08-47	BS-1	NORM

PROT SET II, CH 0460

1. PLACE the following NMT card test switch in CLOSED.

	CARD TAG #	CAB-FRAM-CARD	SWITCH	POSITION
☐	u-UY/0762S	02-08-73	SW7	CLOSED

2. ENSURE the following NCT card switches in NORM.

	CARD TAG #	CAB-FRAM-CARD	SWITCH	POSITION
☐	LS/0460	02-08-47	BS-1	NORM

PROT SET III, CH 0461

1. PLACE the following NMT card test switch in CLOSED.

	CARD TAG #	CAB-FRAM-CARD	SWITCH	POSITION
☐	u-UY/0763S	03-08-73	SW7	CLOSED

2. ENSURE the following NCT card switches in NORM.

	CARD TAG #	CAB-FRAM-CARD	SWITCH	POSITION
☐	LS/0461	03-08-44	BS-1	NORM

Attachment 3

Comments / Reference: ABN-706	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-706
PRESSURIZER LEVEL INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 11 OF 14

ATTACHMENT 4
PAGE 1 OF 1

ANNUNCIATOR ALARMS AND TRIP STATUS LIGHTS

PROT SET I, CH 0459

	ALARM	ANN. WINDOW	PANEL
☐	PRZR 1 OF 3 LVL HI	4.2	ALB-5C
	TRIP STATUS	STATUS INDICATOR	TRIP STATUS PANEL
☐	PRZR LVL HI LB-459A	1.1	TSLB-5

PROT SET II, CH 0460

	ALARM	ANN. WINDOW	PANEL
☐	PRZR 1 OF 3 LVL HI	4.2	ALB-5C
	TRIP STATUS	STATUS INDICATOR	TRIP STATUS PANEL
☐	PRZR LVL HI LB-460A	2.1	TSLB-5

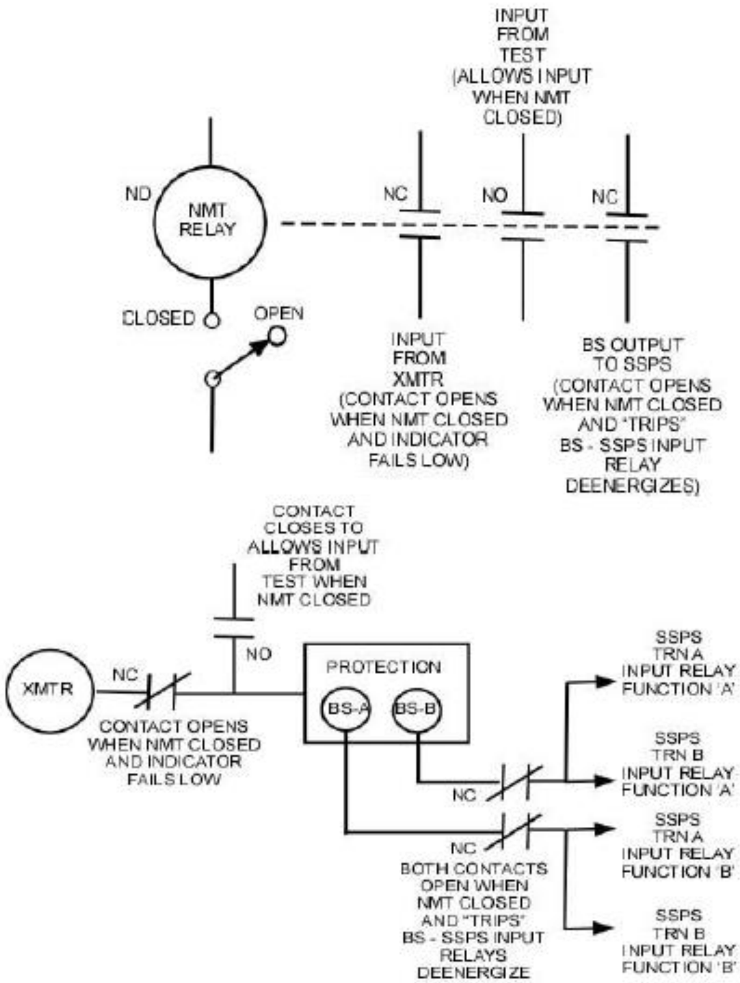
PROT SET III, CH 0461

	ALARM	ANN. WINDOW	PANEL
☐	PRZR 1 OF 3 LVL HI	4.2	ALB-5C
	TRIP STATUS	STATUS INDICATOR	TRIP STATUS PANEL
☐	PRZR LVL HI LB-461A	3.1	TSLB-5

Attachment 4

Comments / Reference: 7300 Process Lesson Plan	Revision: 00-0000

NOTES	LESSON OUTLINE
	3. NMT (Master Test Card)



Comments / Reference: 7300 Process Lesson Plan	Revision: 00-0000

NOTES	LESSON OUTLINE

a. The NMT card Open/Closed switch is used to swap between test jack and transmitter inputs and connect/disconnect bistable outputs to SSPS

b. OPEN position on the switch

1) Test Jack is not in service and transmitter is connected

2) Bistable card is connected to the SSPS input relays

3) SSPS input relays will be energized if Bistable card (NAL) is not in a tripped condition

c. CLOSED position on the switch

1) Test Jack is in service and transmitter is disconnected (Indication is failed low unless a test signal is being input via test jack)

2) Bistable card is disconnected from the SSPS input relays

3) SSPS input relays will be deenergized regardless of Bistable card (NAL) being in a tripped or non-tripped condition, unless the NCT switch is closed (NCT switch is discussed later)

d. A single NMT switch is used for each channel/loop; for example, Loop 455, Pressurizer Pressure Channel I, uses a single NMT switch – placing switch in CLOSED performs the following:

1) Disconnects PT-455 from Protection Cabinet, resulting in input failing to 0 mA, converted to a 0 VDC signal inside the cabinet

2) Connects the test jack to the Protection Cabinet input, allowing I&C to input a test signal for testing and troubleshooting

3) Disconnects PB-455A output from both trains of SSPS, causing the Channel I Input Relay in each train of SSPS associated with Pressurizer High Pressure Rx Trip to deenergize, indicating to SSPS that this setpoint has been exceeded unless the associated NCT switch is in BYPASS (NCT switch is discussed later)

4) Disconnects PB-455B output from both trains of SSPS, causing the Channel I Input Relay in each train of SSPS associated with Pressurizer P-11 to deenergize, indicating to SSPS that this setpoint has been exceeded (unless the associated NCT switch is in BYPASS (NCT switch is discussed later)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	013.K2.01		
Level of Difficulty: 3	Importance Rating	3.6		

Engineered Safety Features Actuation: Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

Question # 9

What effect does a loss of 2EC1 have on the Emergency Diesel Generators?

A. Train A EDG will not start following a Safety Injection.

B. Train B EDG will not start following a Safety Injection.

C. If Train A EDG is running due to a loss of offsite power, an 86-2 lockout will actuate, causing the DG to trip.

D. If Train B EDG is running due to a loss of offsite power, an 86-2 lockout will actuate, causing the DG to trip.

Answer: C

K/A Match: K/A match due to requiring knowledge of the power supplies to the controls for the EDGs and the effect of a loss of power.

Explanation:

A. Incorrect. Plausible as the DG will not start on a bus UV signal, but will start following a safety injection.

B. Incorrect. Plausible (see A) and it is thought that the 2 in 2EC1 indicates a Train B power supply.

C. Correct. The loss of 2EC1 will cause a loss of power to the EDG UV start relay and an 86-2 lockout will actuate due to loss of power to the generator outboard bearing high temperature relay and the DG will trip.

D. Incorrect. Plausible (see C) and if it is thought that the 2 in 2EC1 indicates a Train B power supply.

Technical Reference(s)	ABN-603	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to a Loss of Instrument Bus in accordance with ABN-603, Loss of Protection or Instrument Bus. (ABN.603.OB02)

Question Source: Bank # 21915
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ABN-603	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-603
LOSS OF PROTECTION OR INSTRUMENT BUS	REVISION NO. 8	PAGE 16 OF 34

3.0 LOSS OF INSTRUMENT BUS

3.1 Symptoms

a. The affected inverter trouble alarm:

- 118V INV IVuEC1 TRBL (10B-1.15)
- 118V INV IVuEC2 TRBL (10B-2.15)
- 118V INV IVuEC3 TRBL (10B-2.18)
- 118V INV IVuEC4 TRBL (10B-3.18)
- 118V INV IVuEC1/3 TRBL (10B-1.18)
- 118V INV IVuEC2/4 TRBL (10B-4.18)

b. The associated bus instruments alarming or failing (see Attachments 3 and 4):

- uEC1 from IVuEC1
- uEC2 from IVuEC2
- uEC5 from IVuEC3
- uEC6 from IVuEC4

NOTE: **(Unit 2 only)** On loss of 2EC1 or 2EC2, the FWIVs close due to loss of water hammer interlocks and the FPBVs open. During low power operations, this could cause overheating of the containment penetrations. 2-ALB-8A, 1.5, 2.5, 3.5 and 4.5 contain actions should this occur.

c. **(Unit 2 only)** A feed isolation will occur, FWIVs close (loss of 2EC1 or 2EC2).

d. A DG 86-2 lockout relay (loss of uEC1 or uEC2) will prevent diesel start on loss of power.
The DG will not emergency start due to loss of power to its emergency start relay. The diesel generators can be manually started in the emergency mode if needed.

e. If diesel running due to a loss of offsite power, a loss of uEC1 or uEC2 will restore normal trips and stop the diesel due to a DG 86-2 lockout relay.

f. Thermal barrier return isolation (u-HV-4696) closes (loss of uEC1)

g. During the period uEC1 or uEC2 is powered from bypass power, the Black Out Sequencer is inoperable per TS 3.8.1

3.2 Automatic Actions

None

Section 3.0

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	2		
	Group	1		
	K/A	022.A4.05		
Level of Difficulty: 4	Importance Rating	3.8		

Containment Cooling: Ability to manually operate and/or monitor in the control room: Containment readings of temperature, pressure, and humidity system

Question # 10

(1) When determining if Adverse Containment Conditions exist, which of the following Containment Pressure indications on the Control Board are to be used?

(2) When determining Containment Temperature for TS 3.6.5, Containment Air Temperature, the meter reading used on the Control Board is the __(2)__ of:

- TE-5400, CNTMT TEMP DOME EL 1001'-9"
- TE-5401, CNTMT TEMP EL 1001'-2 1/2"
- TE-5403, CNTMT TEMP EL 905'-9"
- TE-5404, CNTMT TEMP EL 863'-6"

A. (1) Intermediate Range (PT-934, 935, 936 and 937)
(2) average

B. (1) Narrow Range (PT-5470A and 5470B)
(2) average

C. (1) Intermediate Range (PT-934, 935, 936 and 937)
(2) highest

D. (1) Narrow Range (PT-5470A and 5470B)
(2) highest

Answer: A

K/A Match: K/A match due to requiring knowledge of the indications of containment temperature and pressure on the control board.

Explanation:

A. Correct. First part is correct. Intermediate range channels are used for determining adverse containment conditions as narrow range indications have a maximum value of 2.5 psig. Second part is correct. Containment temperature indicator is TI-5400A, CNTMT AVE TEMP, using average temperature.

B. Incorrect. First part is incorrect, but plausible since narrow range are typically read to ensure compliance with TS limits on pressure. Second part is correct (see A).

C. Incorrect. First part is correct (see A). Second part is incorrect, but plausible as a high failure of any of the four containment temperature inputs will result in a high Containment temperature alarm.

D. Incorrect. First part is incorrect, but plausible (see B). Second part is incorrect, but plausible (see C).

Technical Reference(s)	ALM-0031A	Attached w/ Revision # See Comments / Reference
	ODA-407	
	Containment Systems Study Guide	

Proposed references to be provided during examination: ____________________

Learning Objective: **DESCRIBE** the components of the Containment system including interrelations with other systems to include interlocks and control loops as described in DBD-ME-008 Containment Analysis. (SYS.CY1.OB02)

Question Source: Bank # ________
Modified Bank # 21668 (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments / Reference: Bank 21668	Revision:

Which range of containment pressure transmitters are used as inputs to the ESF protection scheme?

A. wide range (PT-938 and 939) ONLY

B. narrow range (PT-5470A and 5470B) ONLY

C. intermediate range (PT-934, 935, 936 and 937) ONLY

D. both narrow range (PT-5470A and 5470B) AND wide range (PT-938, 939)

Answer: C

Answer Explanation
A. Incorrect - Plausible as these are containment pressure channels but they do not input to the ESF protection system B. Incorrect - Plausible as these are containment pressure channels but they do not input to the ESF protection system C. Correct - 7247D05 sheet 8 D. Incorrect - Plausible as these are containment pressure channels but they do not input to the ESF protection system

Question 6 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	21668
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	Which range of containment pressure transmitters are used as inputs to the ESF protection scheme?
K/A:	013.A3.01
Question Reference:	7247D05
SRO:	
Comments:	R/S18E06

Comments / Reference: ALM-0031A	Revision: 8

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0031A
ALARM PROCEDURE 1-ALB-3A	REVISION NO. 8	PAGE 7 OF 113

ANNUNCIATOR NOM./NO.: **CNTMT TEMP HI** **1.1**

PROBABLE CAUSE:

Steamline break inside Containment
Feedline break inside Containment
Reactor coolant leakage
Inadequate Containment cooling

NOTE: 1-TE-5400, 5401, 5402 AND 5403 are averaged to provide indication on 1-TI-5400A, CNTMT AVE TEMP. A single instrument malfunction will invalidate these indications.

AUTOMATIC ACTIONS: None

NOTE: 1-HV-6082, 1-HV-6083 AND 1-HV-6084 close on Phase A Isolation.

OPERATOR ACTIONS:

1. MONITOR Containment Pressure.
   - 1-PI-934, CNTMT PRESS (IR) CHAN IV
   - 1-PI-935, CNTMT PRESS (IR) CHAN III
   - 1-PI-936, CNTMT PRESS (IR) CHAN II
   - 1-PI-937, CNTMT PRESS (IR) CHAN I
   - 1-PI-5470A, CNTMT PRESS (NR)
   - 1-PI-5470B, CNTMT PRESS (NR)

   A. IF all channels are approximately 3 psig AND increasing,
      THEN
      GO to EOP-0.0A.

   B. IF either narrow range channel is >1 psig,
      THEN
      REFER to TS 3.6.4.

2. DETERMINE affected temperature instrument from the Plant Computer.

NOTE: Due to instrument inaccuracies, containment average temperature should be assumed to be 10°F higher than indicated on the main control board. This value may then be used to determine if temperature is within Technical Specification limits.

3. MONITOR 1-TI-5400A, CNTMT AVE TEMP.

CONTINUED. . .

Comments / Reference: ODA-407	Revision: 17

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 17 INFORMATION USE	PAGE 39 OF 63

ATTACHMENT 8.A
PAGE 21 OF 25

ERG RULES OF USAGE

10. Adverse Containment parameters determine when a harsh environment begins to affect instrumentation located inside containment. The following indications identify that the ADVERSE CONTAINMENT values should be used in the ERGs.

- Containment pressure - Greater than 5 psig, or
- Containment radiation - Greater than 10^5 R/hr, or
- Integrated containment radiation dose - Greater than 10^6 RADS (to be determined by Plant Staff.

If either containment pressure exceeds 5 psig or containment radiation exceeds 10^5 R/hr, the ERGs are implemented using the ADVERSE CONTAINMENT (post-accident) process parameter values. If containment pressure decreases below 5 psig after it has been exceeded, the normal parameter values should be used. Once the radiation has exceeded 10^5 R/hr, the ADVERSE CONTAINMENT values are used until the integrated radiation dose is verified to be less than 10^6 Rads.

11. When the ERG procedure instructs a step to be performed where the desired condition already exists, the SRO may evaluate the step to determine if it needs to be performed. For example, the ERG step to "Reset Containment Isolation Phase A and Phase B" groups both Phase A and Phase B together for convenience. If Phase B has NOT initiated, then reset is not required, however it may be reset if desired.

12. ERGs direct the operating staff to "Start", "Stop", and "Secure" various plant equipment without specifically identifying those actions be taken per the System Operating Procedure. When the words "per SOP" are omitted, the intent is that the action may be performed from the control room using operator training and knowledge to accomplish the task. The SOP's should be used either before or after the action is taken, as time and circumstances allow, to ensure all aspects of proper system operation are covered. In a few specific cases certain actions should be taken by the operator before the opportunity to reference the appropriate SOP can be accomplished. An "emergency start" of a Centrifugal Charging Pump is one example of a procedure step requiring specific operator action based on assumed knowledge.

EXAMPLE - An emergency start of a Centrifugal Charging Pump is defined in SOP-103A/B as a start of the pump without first starting the Aux Lube Oil Pump. This emergency start is designated in ERGs by directing the start of the CCP without a "per SOP-103A/B" reference. The emergency start authorization only permits a pump start without prelubrication.

Comments / Reference: Containment Systems Study Guide	Revision: 5-2-2011

Containment Systems

The sensors will be automatically recalibrated using known calibration gases 2% and 6% hydrogen. The calibration cycle can be automatically initiated at regular intervals by the microprocessor system, and manual initiation is also possible.

Hydrogen Mixing -- All subcompartments are provided with vents to provide hydrogen mixing. Connection paths through these compartments maintain the subcompartments at the same hydrogen concentration as the rest of the containment. Containment spray (if used) also promotes air circulation.

INSTRUMENTATION & CONTROL

PRESSURE

Three different ranges of containment pressure indicate on the MCB.

- Narrow range (2 channels) CNTMT PRESS (NR); PI 5470A (5470B)

Digital readouts on CB-03 Range -2.5 psig to 2.5 psig

Alarm CNTMT NR PRESS HI/LO, window 4.6 on ALB-3A, has a setpoint High of □1.2263 psig and a setpoint Low of □ - 0.2263 psig.

- Wide range (2 channels), CNTMT PRESS (WR); PI-938 (939) on CB-03, has scale of 0-150 psig and provides indication only.
- Intermediate range (4 channels), CNTMT PRESS (IR) CHAN I (II, III, IV); PI-937 (936,935,934) on CB-03, provides indication, alarms, and protection.

Indication Scale -5 to 60 psig

Alarms: CNTMT PRESS 1 of 3 HI 1, yellow window 1.10 on ALB-2B, has a setpoint of >3.2 psig on 1/3 channels: HI 1 (2/3) generates SI. Uses channels II, III, IV, (936, 935, 934)

CNTMT PRESS 1 of 3 HI 2, yellow window 2.10 on ALB-2B, has a setpoint > 6.2 psig on 1/3 channels: HI 2 (2/3) generates Main Steam Line Isolation. Uses channels II, III, IV (936,935,934.)

CNTMT PRESS 1 of 4 HI 3, yellow window 3/10 on ALB-2B, has a setpoint of > 18.2 psig on 1/4 channels: HI 3 (2/4) generates CS and Phase B Containment Isolation. Uses all four channels.

CNTMT PRESS HI SI ACT, "First Out" annunciator on ALB-6C, has a setpoint > 3.2 psig on 2/3 channels and uses channels II, III, IV. Actuates Reactor Trip, SI, and Phase A Isolation.

CNTMT ISOL PHASE B ACT, red window 4.11 on ALB-2B, has a setpoint > 18.2 psig on 2/4 channels and alarms on manual actuation as well as automatic initiation. Uses all four channels and generates CS Actuation and Phase B Isolation.

Manual Actuation/Reset Controls

CNTMT ISOL PHASE A/CNTMT VENT ISOL MAN ACT hand switches

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	026.A1.06		
Level of Difficulty: 3	Importance Rating	2.7		

Containment Spray: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment spray pump cooling

Question # 11

Given the following conditions:

- Both trains of Containment Spray have just actuated on HI-3
- Train B CCW Pump has tripped

Train B Containment Spray Pumps must be stopped due to a loss of cooling to the __(1)__.

Prior to stopping any equipment, Train B Containment Spray Pump room temperatures will __(2)__.

A. (1) Seal Coolers
(2) be unaffected

B. (1) Seal Coolers
(2) approach design limits

C. (1) Bearing Coolers
(2) be unaffected

D. (1) Bearing Coolers
(2) approach design limits

Answer: B

K/A Match: K/A match due to requiring knowledge of the operation of the containment spray pump controls to prevent exceeding design room temperature limits.

Explanation:

A. Incorrect. First part is correct (see B). Second part is incorrect, but plausible since chilled water would be supplied by CCW if a Hi-3 condition had not occurred, which splits the two CCW trains. In this case Train B CT Pump room temperatures would be unaffected.

B. Correct. First part is correct. CCW supplies cooling to the pump seal coolers. Second part is correct. Upon a loss of CCW, due to the trip of the Train B CCW Pump and the split of the CCW trains, the chilled water system loses cooling and the room temperatures will approach limits due to the pump running.

C. Incorrect. First part is incorrect, but plausible since bearing coolers require cooling, but they are cooled by SSW, not CCW. Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	SOP-204A	Attached w/ Revision # See Comments / Reference
	ABN-503	
	ABN-502	

Proposed references to be provided during examination: __________

Learning Objective: **DESCRIBE** the components of the Containment Spray system including interrelations with other systems to include interlocks and control loops in accordance with the FSAR and DBD-ME-0232. (SYS.CT1.OB03)

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 X

10 CFR Part 55 Content:
- 55.41 7
- 55.43 __________

Comments / Reference: SOP-204A	Revision: 15

CPNPP SYSTEM OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. SOP-204A
CONTAINMENT SPRAY SYSTEM	REVISION NO. 15 CONTINUOUS USE	PAGE 3 OF 55

1.0 APPLICABILITY

This procedure provides instructions for operating the Containment Spray System.

2.0 PREREQUISITES

2.1 Placing the System in Standby

NOTE: CCW flow to the Containment Spray Pumps and Heat Exchanger is not required provided system temperature is maintained ≤150°F and the affected train is declared inoperable per TS 3.6.6.

- ☐ CCW is available and aligned to the pump seal coolers.
- ☐ CCW is available to the heat exchangers.
- ☐ SSW is available and aligned to the pump bearing coolers.
- ☐ Nitrogen is available to the Chemical Additive Tank.
- ☐ The Chemical Additive Tank is available for chemical addition.
- ☐ Both spray trains have been filled and vented and the respective Containment Spray Risers are above the low level alarm.
- ☐ The RWST is filled and aligned to the SI header.
- The following valve lineups are complete:
  - ☐ SOP-204A-CT-V01, Train A Valve Lineup
  - ☐ SOP-204A-CT-V02, RWST Valve Lineup
  - ☐ SOP-204A-CT-V03, Chem Add Tank Valve Lineup
  - ☐ SOP-204A-CT-V04, Train B Valve Lineup
- The following control switch lineups are complete:
  - ☐ SOP-204A-CT-C01, Train A Control Switch Lineup
  - ☐ SOP-204A-CT-C02, Train B Control Switch Lineup
- The following electrical lineups are complete:
  - ☐ SOP-204A-CT-E01, Train A Electrical Lineup
  - ☐ SOP-204A-CT-E02, Train B Electrical Lineup

Comments / Reference: ABN-503	Revision: 2

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-503
SAFETY CHILLED WATER SYSTEM MALFUNCTION	REVISION NO. 2	PAGE 11 OF 27

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: WHEN room cooling is lost AND the ESF Pump (motor) running, THEN equipment room temperatures will rise.

The time to reach Pump Room EQ temperature limits (TS 13.7.36, Area Temperature Monitoring) varies by room. Most limiting times are for RHR, Containment Spray and MD AFW Pump rooms (i.e., less than ten (10) minutes). (Reference Attachment 1)

Steps 1, 2 and 3 may be performed in parallel.

NOTE: Annunciator (u-ALB-4A-1.7) is common to Train A and Train B chiller status. Individual chiller status may be identified using either:

- local verification by dispatching a NEO OR
- plant computer points Y2281D and Y2282D.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 VERIFY Restoration of a Safety Chilled Water train is expected within 1 hour.	Reduce power to < 50% in 1 hour AND Be in MODE 3 in next 2 hours (Refer to TS 3.0.3)

Section 3.3

Comments / Reference: ABN-502	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION 11	PAGE 6 OF 75

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5 VERIFY required equipment, for existing conditions, supplied by unaffected train - IN OPERATION: • Control Room A/C Units • Containment Spray System • UPS HVAC Unit • Excess Letdown • RHR System	ALIGN AND START required equipment as necessary.
6 Shutdown equipment on the affected Train as necessary: a. To prevent auto operation without necessary support, shutdown the following on the affected train: ☐ • Containment Spray Pumps - PULL OUT ☐ • RHR Pump - PULL OUT	

"Step continued next page"

Section 2.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	039.K4.05		
Level of Difficulty: 4	Importance Rating	3.7		

Main and Reheat Steam: Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Automatic isolation of steam line

Question # 12

Given the following conditions:

- A plant cooldown and depressurization is in progress
- RCS pressure is 1925 psig
- The RO has taken both 1/1-PPSIRBA, PRZR PRESS SI RESET / BLOCK, and 1/1-PPSIRBB, PRZR PRESS SI RESET / BLOCK, to the 'Block' position
- The BOP has taken both 1/1-SLSIRBA, MSL ISOL SI RESET / BLOCK, and 1/1-SLSIRBB, MSL ISOL SI RESET / BLOCK, to the 'Block' position

Assuming NO further operator actions, if a large steam break were to subsequently occur outside Containment:

Safety Injection __(1)__ occur.

A Main Steamline Isolation will occur on __(2)__.

A. (1) will
(2) Low Steamline pressure

B. (1) will
(2) Negative Steamline pressure high rate

C. (1) will NOT
(2) Low Steamline pressure

D. (1) will NOT
(2) Negative Steamline pressure high rate

Answer: D

K/A Match: K/A match due to requiring knowledge of the conditions which will cause a main steam line isolation.

Explanation:

A. Incorrect. First part is incorrect. Plausible since an SI could occur if the break were inside containment. Second part is incorrect. Plausible since action is taken to block the steamline low pressure isolation, but the blocking enables a negative pressure rate signal to close the MSIVs.

B. Incorrect. First part is incorrect (See A). Second part is correct (See D).

C. Incorrect. First part is correct (See D). Second part is incorrect (See A).

D. Correct. First part is correct. SI will not occur on low steamline pressure when blocked. Second part is correct. Blocking enables a negative pressure rate signal to close the MSIVs.

Technical Reference(s)	ALM-0065A	Attached w/ Revision # See Comments / Reference
	IPO-005	
	Reactor Protection Study Guide	

Proposed references to be provided during examination: __________

Learning Objective: **EXPLAIN** the instrumentation and controls of the Reactor Protection and Engineered Safeguard Actuation Systems and predict the system response in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems and Westinghouse Drawings 7247D05. (SYS.ES1.OB04)

Question Source: Bank # 23049
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments / Reference: ALM-0065A	Revision: 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 55 OF 73

ANNUNCIATOR NOM./NO.: MSL PRESS LO TRN A SI BLK 3.8

PROBABLE CAUSE:

Manual block of low steam line pressure safety injection during cooldown

NOTE: This window is normally illuminated in Modes 3-6 when plant cooldown is required.

AUTOMATIC ACTIONS:

Blocks the main steam line low pressure safety injection
Enables the main steam line high pressure rate steam line isolation

OPERATOR ACTIONS:

None

Comments / Reference: IPO-005A	Revision: 27

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-005A
PLANT COOLDOWN FROM HOT STANDBY TO COLD SHUTDOWN	REVISION NO. 27 CONTINUOUS USE	PAGE 77 OF 131

NOTE: If RCP 1 or 4 is stopped, the associated spray valve controller should remain in MANUAL with zero demand to prevent bypassing of the spray flow.

5.2.7 B. OPEN Pressurizer spray valve(s), as necessary, to reduce RCS pressure:

- ☐ 1-PK-455B, RC LOOP 1 PRZR SPR VLV CTRL
- ☐ 1-PK-455C, RC LOOP 4 PRZR SPR VLV CTRL

/ Initials Date

C. WHEN RCS pressure is less than 2185 psig, THEN VERIFY the following annunciators are ON:

- ☐ 1-ALB-5B, 1.6, PRZR LO PRESS PORV 456 BLK
- ☐ 1-ALB-5B, 2.6, PRZR LO PRESS PORV 455A BLK

/ Initials Date

D. WHEN PRZR PRESS channels approach 1925 psig, ADJUST pressurizer heaters and spray valves to maintain RCS pressure between 1900 psig and 1950 psig UNTIL, Automatic Safety Injection Signal is blocked.

/ Initials Date

CAUTION: Maintain Pressurizer level less than 30% until SI is blocked.

[C] [22650] [b] 5.2.8 WHEN RCS pressure is below 1960 psig, THEN PERFORM the following to block SI:

A. VERIFY Measured RCS Boron Concentration and Pressurizer boron concentration are at or above the value required by prerequisite 2.14 A. for blocking SI.

/ Initials Date

B. VERIFY 1-PCIP, 2.6, PRZR PRESS SI BLK PERM P-11 is ON.

/ Initials Date

C. VERIFY the following status lights are OFF:

- ☐ 1-TSLB-9, 1.3, PRZR PRESS SI PERM PB-455B
- ☐ 1-TSLB-9, 2.3, PRZR PRESS SI PERM PB-456B
- ☐ 1-TSLB-9, 3.3, PRZR PRESS SI PERM PB-457B

/ Initials Date

Comments / Reference: IPO-005A	Revision: 27

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-005A
PLANT COOLDOWN FROM HOT STANDBY TO COLD SHUTDOWN	REVISION NO. 27 CONTINUOUS USE	PAGE 78 OF 131

5.2.8 D. TURN BOTH MSL ISOL SI RESET/BLOCK switches to BLOCK AND RELEASE:

- ☐ • 1/1-SLSIRBA, MSL ISOL SI RESET/BLOCK
- ☐ • 1/1-SLSIRBB, MSL ISOL SI RESET/BLOCK

______ / ______
Initials Date

E. VERIFY the following are ON:

- ☐ • 1-PCIP, 3.8, MSL PRESS LO TRN A SI BLK
- ☐ • 1-PCIP, 4.8, MSL PRESS LO TRN B SI BLK

______ / ______
Initials Date

F. TURN BOTH PRZR PRESS SI RESET/BLOCK switches to
BLOCK
AND
RELEASE:

- ☐ • 1/1-PPSIRBA, PRZR PRESS SI RESET/BLOCK
- ☐ • 1/1-PPSIRBB, PRZR PRESS SI RESET/BLOCK

______ / ______
Initials Date

G. VERIFY the following are ON:

- ☐ • 1-PCIP, 3.7, PRZR PRESS LO TRN A SI BLK
- ☐ • 1-PCIP, 4.7, PRZR PRESS LO TRN B SI BLK

______ / ______
Initials Date

[b] 5.2.9 PERFORM the following to raise and maintain Pressurizer Level 25% to 50%:

A. IF necessary,
THEN
PLACE 1-FK-121, CCP FLO CTRL in MANUAL
AND
RAISE charging flow.

______ / ______
Initials Date

B. IF necessary,
THEN
START an additional CCP per SOP-103A, Chemical and Volume Control System.

______ / ______
Initials Date

Comments / Reference: Reactor Protection Study Guide	Revision: 5-4-2011

Reactor Protection and ESFAS

NUCLEAR AT POWER PERMISSIVE, P-10

When 2 out of 4 Power Range Detectors are > 10% power, **P-10** is activated. **P-10** performs the following functions:

- Feeds **P-7**
- Blocks the Source Range Detectors high voltage and trip
- Allows blocking of the Intermediate Range Reactor Trip and Rod Stop
- Allows blocking of the Power Range High Flux Reactor Trip (Low Setpoint)

When the Source Range detectors are deenergized, a SR HI VOLTS FAIL alarm alerts the operator to the condition. Since the Source Range detectors are designed to be de-energized at power, this alarm is defeated whenever **P-10** is present. Also, the SR HI Flux at Shutdown annunciator and the Containment Evacuation alarm are defeated when a **P-10** signal is present.

When 3 out of 4 Power Range detectors are < 10% power, then **P-10** clears and this will automatically unblock the Intermediate Range Reactor Trip and Rod stop and the Power Range Low Setpoint Reactor Trip.

The Source Range detectors will remain blocked until the operator manually unblocks the Source Range reactor trip or the Source Range Reactor trip is automatically unblocked at < **P-6**. Also, the block of the SR HI Flux at Shutdown annunciator and the Containment Evacuation alarm will remain in until the SR reactor trip is unblocked.

When **P-10** is clear, then **P-7** will also clear (the PCIP window, RX &TURB 10% PWR **P-7** will be lit), if **P-13** is clear, blocking the following trips:

- Pressurizer Low Pressure
- Pressurizer Hi Level
- Reactor Coolant Pump Under Voltage
- Reactor Coolant Pump Under frequency
- Low Flow in 2 Reactor Coolant Loops

PRESSURIZER SI BLOCK PERMISSIVE, P-11

P-11 set at <1960 PSIG on 2 out of 3 Pressurizer pressure detectors allows for manual blocking of the low Pressurizer Pressure and low Steam Line Pressure Safety Injection signals. When the low Steam Line Pressure SI signal is blocked it arms the Steam Line Isolation for Negative Steam Line Pressure High Rate (-100 psi with a 50 second time constant). Once pressure has increased above 1960 PSIG, the alarms for the SI accumulator isolation valves (8808s) and RWST suction isolation valves (8806s) to the SI pumps not being open are armed.

Comments / Reference: Reactor Protection Study Guide	Revision: 5-4-2011

Reactor Protection and ESFAS

- Low steam line pressure - 2 of 3 PT on 1 of 4 steam lines at 605 psig decreasing, blockable at **P-11**. This is interlocked with **P-11** to allow manual blocking for an intentional cooldown. The steam line pressure signal is lead/lag compensated so that it is rate sensitive (50 second lead time constant and 5 second lag time constant) (**Figure 24**).

STEAM GENERATOR HIGH PRESSURE RATE AND LOW STEAMLINE PRESSURE ACTUATION LOGIC

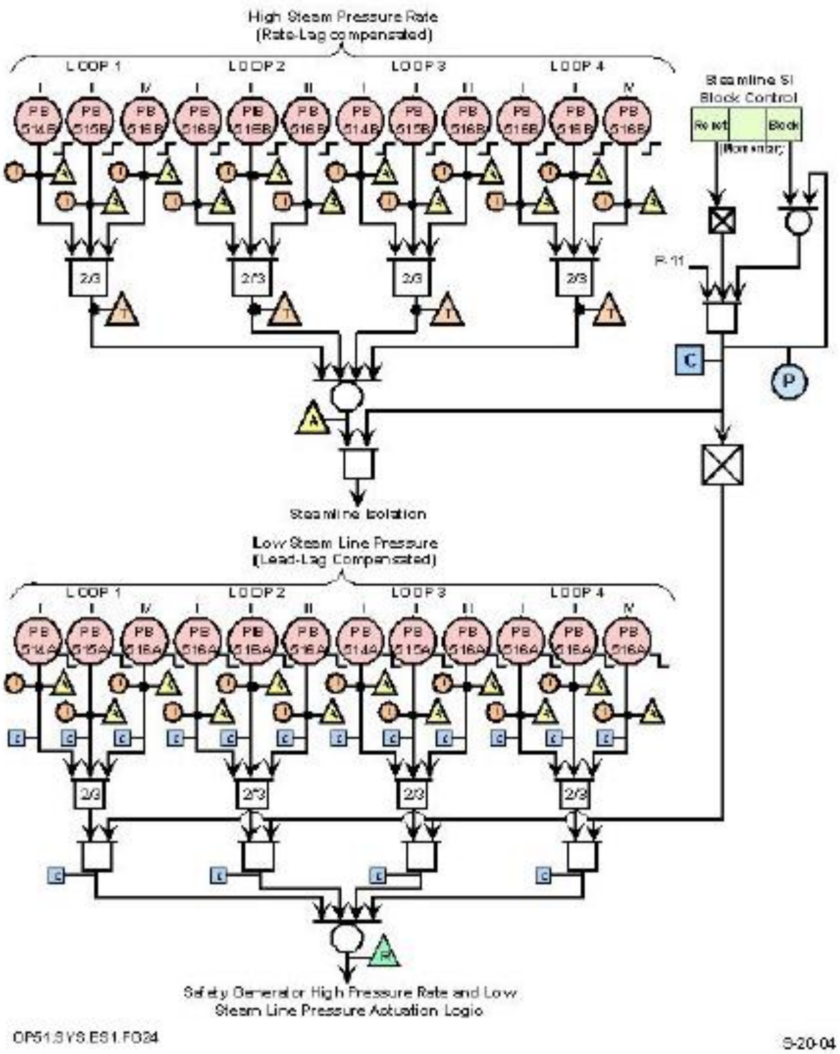


Figure 24 - Steam Generator High Pressure Rate and Low Steamline Pressure Actuation Logic

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	059.A4.11		
Level of Difficulty: 2	Importance Rating	3.1		

Main Feedwater: Ability to manually operate and monitor in the control room: Recovery from automatic feedwater isolation

Question # 13

Given the following conditions:

- Unit 1 tripped from 100% power due to a P-14 on SG 1-02
- As the crew is working through the ERG network all AFW is lost and cannot be restored
- FRH-0.1A, Response to Loss of Secondary Heat Sink has been entered
- All S/G NR levels are ~ 2%

During attempts to re-establish a MFW flow path by resetting the FWI, which of the following will be required IF the Reactor trip breakers are unable to be cycled?

A. Reset both trains of SI Sequencer, then reset the FWI signal

B. Pull universal logic card A213 from both trains of SSPS, then replace both cards

C. Cycle the FWIV hand switches to the closed and open position to allow a FWI reset

D. Open 1B3-1 and 1B4-1 breakers to remove control power from the reactor trip breakers

Answer: B

K/A Match: K/A match due to requiring knowledge of the actions required to reset feedwater isolation signal.

Explanation:

A. Incorrect. Plausible as this is part of step 7 RNO but this by itself will not reset the FWI signal.

B. Correct. Per FRH-0.1 step 7.b RNO 1.b if the breakers cannot be cycled then if card A213 is pulled on both trains this will allow power to be removed from the circuit and therefor resetting the FWI signal to allow feeding with the feedwater system.

C. Incorrect. Plausible as FRH-0.1 step 7 does have the operator place FWIV handswitches in auto after closed but this will not reset the FWI.

D. Incorrect. Plausible as this is performed by FRH-0.1 for restoration of condensate flow but without cycling the trip breakers it will not reset the FWI.

Technical Reference(s)	FRH-0.1A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination:

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRH-0.1 in accordance with FRH-0.1, Loss of Heat Sink. (ERG.FH1.OB04)

Question Source: Bank # 24393
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10
55.43

Comments / Reference: FRH-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 9	PAGE 7 OF 85

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

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

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	Establish Main FW Flow To At Least One SG:	
	a. Check Condensate system - IN SERVICE	a. Place Condensate system in service: 1) Start Condensate pump. IF Condensate system can NOT be placed in service, THEN go to Step 11. OBSERVE CAUTION AND NOTES PRIOR TO STEP 11.
	b. Reset FW Isolation:	
	1) Verify SI - NOT ACTUATED -AND- Verify SG levels - HAVE REMAINED BELOW 84% (P-14 SG HI-HI LEVEL SETPOINT)	1) Perform the following to reset FW Isolation: Perform the following to cycle Reactor Trip breakers: A) Block SI signal, if applicable: 1. Verify Containment pressure less than 3.0 psig. 2. IF PRZR pressure less than 1960 psig THEN block: • Low steamline pressure SI signal • Low PRZR pressure SI signal 3. Reset SI.

-CONT 7-

Comments / Reference: FRH-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 9	PAGE 8 OF 85

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	b. 1)	1) B) Cycle Reactor Trip breakers closed then open. OR If Reactor Trip breakers can NOT be cycled, OR Containment pressure greater than 3.0 psig, THEN perform the following to pull Universal Logic Card A213: A) Reset SI, if applicable. B) Pull Universal Logic Card A213 in both trains SSPS Logic Cabinets. C) Replace Universal Logic Card A213 in both trains SSPS Logic Cabinets.
	2) Place FW control and bypass valve controllers in manual and 0% demand.	
	3) Place FW isolation valve handswitches in auto after closed position.	
	4) Reset FW Isolation.	

-CONT 7-

Comments / Reference: FRH-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 9	PAGE 57 OF 85

ATTACHMENT 4
PAGE 5 OF 33

BASES

STEP 7: Main FW is the next source of high pressure water readily available to the operator to re-establish the secondary heat sink. Prior to restoring Main FW flow to the SGs, the operator verifies condensate system operation to ensure a source of water to the Main FW pumps. Then the Main FW isolation valve status is checked. If feedwater isolation has occurred, various actions are required to reset FW isolation signals and reopen the FW isolation valves. Reactor Trip breakers are cycled, or logic card A213 is pulled in order to reset FW isolation signal when SI has actuated, or when FW isolation has actuated from a P-14 (SG Hi-Hi Level) signal. Reactor Trip breaker cycling is the preferred method to reset FW isolation due to the step being quicker and easier to perform. However, if an active SI signal still exists after cycling the Reactor Trip breakers, the SI signal will be activated again, (e.g., Containment pressure greater than SI activation setpoint). When only a Reactor trip with low temperature signal is the FW isolation signal, resetting FW Isolation will allow restoration of feed flow. If either the condensate system cannot be placed in service or no FW isolation valves can be opened, the operator is directed to Step 11 to try to establish feed flow from any available low pressure source.

Initial attempts are made to operate the FW valves from the Control Room. This assumes the steps to reset the FW Isolation signal in step 7b were successful. If the FW Isolation signal could not be reset, the FW valves will be required to be manipulated locally.

If the condensate system is operational and FW isolation valves are open, then Main FW is established by the operator. If Main FW cannot be established, the operator is directed to attempt to establish condensate flow.

STEP 8: Following actions to establish Main FW flow to the SGs, the operator checks the SG narrow range level indications to determine if adequate flow has been established to maintain the secondary heat sink. If narrow range level has been restored to at least one SG, an adequate heat sink exists and the operator transfers to the procedure in effect. If this level does not exist but feed flow is verified to at least one SG (e.g., by core exit thermocouple indications decreasing or SG wide range level increasing), then subsequent steps to establish condensate system flow are not required and the operator transfers to the procedure in effect.

It should be noted that accurate main feed flow indication may not be available at low flow rates and the SG wide range level indication may not be accurate under adverse containment conditions.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	061.A3.03		
Level of Difficulty: 4	Importance Rating	3.9		

Auxiliary/Emergency Feedwater: Ability to monitor automatic operation of the AFW, including: AFW S/G level control on automatic start

Question # 14

Given the following conditions:

- Both MDAFWPs are operating, maintaining SG levels at 67%, with flow to each SG being controlled in MAN at 200 gpm
- The TDAFWP is secured
- An AFW autostart signal is received which starts the TDAFWP

MDAFWP flow control valves will __(1)__.

TDAFWP flow control valves will __(2)__.

A. (1) trip to AUTO and go full open
(2) trip to AUTO and go full open

B. (1) trip to AUTO and go full open
(2) remain in MAN at the current position

C. (1) trip to AUTO and throttle to maintain program SG level
(2) trip to AUTO and go full open

D. (1) trip to AUTO and throttle to maintain program SG level
(2) remain in MAN at the current position

Answer: B

K/A Match: K/A match due to requiring knowledge how an automatic start of the AFW system effects the flow control valves.

Explanation:

A. Incorrect. First part is correct. MDAFWP flow control valves trip to AUTO and go full open. Second part is incorrect, but plausible since the MDAFWP valves trip to AUTO and go full open, it might be considered that the TDAFWP valves would respond the same.

B. Correct. First part is correct (see A). Second part is correct. The TDAFWP valves remain in their current, fully open position on an autostart.

C. Incorrect. First part is incorrect, but plausible if thought that MDAFWP FCVs tripped to Auto and controlled at program level similar to the MFW FCVs. Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	AFW Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **EXPLAIN** the instrumentation and controls of the Auxiliary Feedwater system and the system response in accordance with DBD-ME-206. (SYS.AF1.OB04)

Question Source:
Bank # ____________
Modified Bank # 73907 (Note changes or attach parent)
New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments / Reference: Bank 73907	Revision:

- Unit 1 Reactor power = 100%
- Both MFW Pumps trip
- PV-2453 A&B, PV-2454 A&B, MD AFW Flow Control Valves are in MANUAL

Which of the following correctly states how the AFW system will respond to the event?

MDAFW Pumps will start ______ and their respective AFW Flow Control Valves will ______.

A. immediately
remain in MANUAL

B. immediately
shift to AUTO

C. ONLY after SG levels lower to the appropriate setpoint
remain in MANUAL

D. ONLY after SG levels lower to the appropriate setpoint
shift to AUTO

Answer: B

Answer Explanation
A. Incorrect. 1st part is correct. MD AFW pump will start automatically when both MFW pumps trip. 2nd part is incorrect because the FCVs will shift to automatic and travel full open upon a start signal. It is plausible because when most controls are in MANUAL, automatic functions cannot manipulate equipment. B. Correct. 1st part is correct (see A). 2nd part is correct. When the MD AFW pumps automatically start, their respective FCVs switch to AUTO and travel full open. C. Incorrect. 1st part is incorrect because both MD AFW pumps will automatically start when both MFW pumps trip. It is plausible because if it were the TD AFW pump, it would be correct. 2nd part is incorrect but plausible (see A). D. Incorrect. 1st part is incorrect but plausible (see C). 2nd part is correct but plausible (see B).

Comments / Reference: Bank 73907	Revision:

Question 188 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	2.00
System ID:	73907
User-Defined ID:	ILOT7478
Cross Reference Number:	SYS.AF1.OB05.055
Topic:	Unit 1 Reactor power = 100% Both MFW Pumps trip PV-2453 A&B, PV-2454 A&B, MD AFW Flow Control Valv
K/A:	061 A3.01
Question Reference:	
SRO:	
Comments:	S/R26E32 (Comp), L27E24 Ref; AFW Study Guide

Comments / Reference: AFW Study Guide	Revision: 00-0000

OP51.SYS.AF1

MDAFWP Flow Control Valves

Each MDAFW pump discharge line branches into individual lines feeding its two associated SGs. The individual AFW line to each SG is provided with a normally open, pneumatically operated flow control valve. Manual isolation valves are provided for maintenance and local flow control.

MDAFW pump flow to each SG is controlled by flow control valves, PV-2453A and B for the Train A pump, PV-2454A and B for the Train B pump. The flow control valves fail open on loss of air or electrical power.

Each flow control valve is provided with a safety class air accumulator sized for five full cycles, plus leakage and steady state consumption for 30 minutes. This allows the valve to control AFW flow following a loss of Instrument Air coincident with a plant condition which requires AFW operation, or to isolate a faulted SG when the normal motor operated isolation valves are not available. The manual isolation valves are then used to control the flow in the event the accumulators are exhausted prior to the restoration of Instrument Air.

Manual/Auto (M/A) controllers on the Main Control Board enable the operator to control flow manually from the Control Room. Upon automatic start of the MDAFW pumps, flow control valves PV-2453 A&B and PV-2454 A&B will automatically trip from manual to automatic control and position full open to ensure flow to the SGs. After a 10-second time delay the flow control valves can be manually positioned by the operator to adjust flow to the SGs. M/A controllers for these valves on the RSP enable the operator to control flow from the RSP when the RSP controllers are placed in manual. When in automatic, these controllers allow feed control to be accomplished at the Main Control Board.

A flow restricting orifice is provided downstream of each flow control valve. The orifice is designed to limit the maximum flow to a faulted SG to 700 gpm and prevent a pump runout condition.

MDAFWP Isolation Valves

A check valve is located downstream of each flow control valve. RTDs in thermowells are provided on each discharge line just upstream of the check valve. These RTDs provide input to a dual indication temperature instrument with a range of 0-300°F located on CB-09. These temperature instruments are used to monitor for potential check valve back leakage from the Main Feedwater System and SGs into the AFW System piping.

A normally open, motor operated Containment isolation valve is located downstream of each check valve. Motor operated valves HV-2491A/B, HV-2492A/B, HV-2493A/B and HV-2494A/B are used to isolate AFW flow to the SGs. These valves are operated with two-position (OPEN-CLOSE) switches on the Main Control Board. Each switch simultaneously operates two motor operated valves associated with the same SG. For example, 1-HS-2491 operates both 1-HV-2491A and 1-HV-2491B. The valve from both the MDAFW pump and the valve from the TDAFW Pump operate simultaneously to isolate AFW to one SG.

Comments / Reference: AFW Study Guide	Revision: 00-0000

OP51.SYS.AF1

Power to the local control panel is supplied from 125 VDC distribution panel uED1-1. Actuation of Train A Safety Injection deenergizes the power supply to the local TDAFWP turbine control panel, resulting in the loss of all indications powered from the panel or actuated by relays which are powered from the panel. This includes the T&TV valve position and overspeed trip indication lights, the digital turbine speed indication, locally and on the Main Control Board (CB-09), the turbine governor current/pneumatic transducer, the "TDAFWP OVRSPD TRIP" annunciator on ALB-8B, and the remote turbine trip solenoid. The net effect is that the TDAFW pump will start and accelerate to design speed without the capability to monitor turbine speed in the Control Room or to trip the turbine remotely. Until power is restored to the local control panel, the only means of stopping the turbine is by manually actuating the local trip device or by closing the steam supply valves from the main steam lines. Power to the local control panel can be restored by depressing the OPEN pushbutton (HS-2452H) on the AFWPT T&T VLV control switch on CB-09 once the SI signal (Train A) has been reset. Restoration of power to the local control panel will restore the indications, alarms, and trip capability.

TDAFWP Test Line

The TDAFW pump is provided with a test line similar to that of the MDAFW Pumps.

TDAFWP Recirculation

The TDAFW pump is protected by a continuous minimum flow recirculation line containing a flow limiting orifice (100 gpm), a check valve whose internals have been removed, and isolation valves. This recirculation line joins the test line and the common minimum flow and test line from the MDAFW Pumps.

TDAFWP Flow Control Valves

The TDAFW pump discharge line branches into individual lines feeding each of the four SGs. Each of the individual lines are provided with a normally open, pneumatically operated flow control valve. Manual isolation valves are provided for maintenance and local flow control.

TDAFW pump flow to each SG is controlled by flow control valves HV-2459, HV-2460, HV-2461, and HV-2462. The flow control valves fail open on loss of air or electrical power.

Each flow control valve is provided with a safety class air accumulator sized for five full cycles, plus additional 30 minutes. This allows the valve to control AFW flow following a loss of Instrument Air coincident with a plant condition which requires AFW operation, or to isolate a faulted SG when the normal motor operated isolation valves are not available. The manual isolation valves are then used to control the flow in the event of loss of air to the flow control valves.

Manual controllers on the Main Control Board for the TDAFW pump flow control valves enable the operator to manually control TDAFW pump flow from the Control Room. When the Control Room is inaccessible, manual flow control capability from the Remote Shutdown Panel (RSP) using M/A controllers is provided. When the RSP controllers are placed in automatic, the controllers on the RSP allow MCB control. When the RSP controllers are placed in manual, the RSP will be the overriding and controlling signal.

FOR TRAINING USE ONLY Page 21 of 35 Rev. 00.0000

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	062.A2.08		
Level of Difficulty: 3	Importance Rating	2.7		

AC Electrical Distribution: Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of exceeding voltage limitations

Question # 15

Given the following conditions:

- Unit 1 and Unit 2 Safeguards 6.9 KV Buses are aligned normally
- A low voltage condition on Unit 1 6.9 KV Buses occurs

Assuming the plant responds as designed, which of the following identifies the response of Unit 1 to the low voltage?

Unit 1 transfers from transformer …

A. XST1 to XST2 and Unit 1 EDGs are NOT running.

B. XST1 to XST2 and Unit 1 EDGs are running unloaded.

C. XST2 to XST1 and Unit 1 EDGs are NOT running.

D. XST2 to XST1 and Unit 1 EDGs are running unloaded.

Answer: C

K/A Match: K/A match due to requiring knowledge of the response of Unit 1 to a low voltage condition.

Explanation:

A. Incorrect. Plausible since the Unit 1 EDGs will not be running and also if thought that the normal alignment is Unit 1 being supplied by XST1. This would be the correct response for Unit 2.

B. Incorrect. Plausible since it may be thought that the normal alignment is Unit 1 being supplied by XST1 and that the Unit 1 EDGs will be running.

C. Correct. The normal power source to Unit 1 is XST2 so a transfer to XST1 will occur. A time delay of approximately one second will prevent the EDG from starting, allowing the buses to first be powered from XST1, the alternate supply.

D. Incorrect. Plausible since a transfer to XST1 from XST2 will occur and if there were no time delay to prevent the EDG from starting.

Technical Reference(s)	6.9Kv and 480V Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the 6.9 KV and 480 V Electrical Distribution system and **PREDICT** the system response in accordance with SOP-603, 604 and ABN-602. (SYS.AC2.OB05)

Question Source:
Bank # ______
Modified Bank # 20740 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 10
55.43 ______

Comments / Reference: Exam Bank 20740	Revision:

Which of the following are the indications that a slow transfer of 6900 V Safeguards bus 1EA1 has occurred after a loss of Startup Transformer XST2?

Breaker 1EA1-2, 1EA1 Alternate Feeder, is ...

A. open
DG1 running loaded

B. closed
DG1 NOT running

C. closed
DG1 running unloaded

D. open
DG1 running unloaded

Answer: B

Answer Explanation
A. Incorrect - Plausible if thought that the slow transfer was to the EDG, which would be running loaded after the BOS. B. Correct - ABN-602 2.2 auto actions C. Incorrect - Plausible if believes that the EDG receives a start signal prior to the slow transfer completing D. Incorrect - Plausible if thought that the slow transfer was to the EDG, but does not ascertain that the BOS would have loaded the EDG.

Comments / Reference: Exam Bank 20740	Revision:

Question 44 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	20740
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	Which of the following are the indications that a slow transfer of 6900 V Safeguards bus 1EA1 has
K/A:	062.K1.02
Question Reference:	ABN-601
SRO:	
Comments:	LC16 Audit; LC22E16RM

Comments / Reference: 6.9Kv and 480V Study Guide	Revision: 10-03-2013

6.9 kV and 480 V Electrical Distribution

SYSTEM DESCRIPTION

COMPONENTS

SAFEGUARDS 6.9KV DISTRIBUTION SYSTEM

The Safeguards 6.9KV AC Distribution Systems is equipped with preferred, alternate, and standby power sources (**Figure 2**).

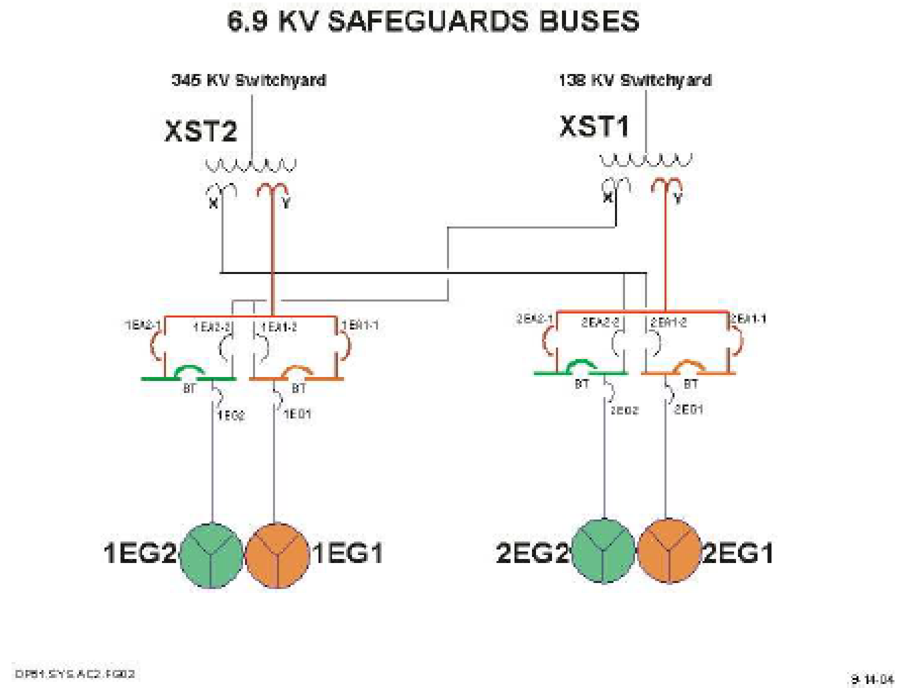


Figure 2 - 6.9KV Safeguards Buses

Preferred Power Sources

The preferred power system for Unit 1 consists of a 345Kv to 6.9KV supply via transformer XST2. Unit 2 is supplied by 138Kv to 6.9KV transformer XST1. These sources supply power to the Class 1E buses of their unit during all modes of plant operation.

XST1

Transformer XST1 is connected to the 138Kv system via the 138Kv switchyard, originating at either the De Cordova Steam Electric Station or the Stephenville Switching Station. The XST1 "Y" winding serves as the preferred power source to Unit 2 Class 1E buses 2EA1 and 2EA2 during normal plant operation.

XST2

Transformer XST2 is connected to the 345Kv switchyard. The switchyard offers a highly reliable source of power because it can be supplied by four separate transmission lines from various switching

Comments / Reference: 6.9Kv and 480V Study Guide	Revision: 10-03-2013

stations on the TXU system in conjunction with the two CPSES unit outputs. The XST2 "Y" winding serves as the preferred power source to Unit 1 Class 1E buses 1EA1 and 1EA2 during normal plant operation.

Alternate Power Sources

XST1

The XST1 "X" winding supplies alternate power to Unit 1 Class 1E buses 1EA1 and 1EA2 during normal plant operations.

XST2

The XST2 "X" winding acts as the alternate power source to Unit 2 Class 1E buses 2EA1 and 2EA2 during normal plant operations.

Therefore, the Class 1E buses of each unit can be supplied by two independent and reliable immediate access offsite power sources. Sharing of these offsite power sources between the two units has no effect on the station electrical system reliability. Each transformer is capable of supplying the required safety-related loads of both units if it becomes necessary to safely shut down both units simultaneously.

Standby Power Sources

The Standby AC Power is provided by four Emergency Diesel Generators (EDGs) which supply Class 1E loads to ensure safe plant shutdown when preferred and alternate power sources are not available. Each EDG is capable of sequentially starting and supplying the minimum power requirements for a DBA in one unit. The four EDGs are electrically and physically independent.

In the event of a loss of the normal power source to the 6.9KV AC Safeguards bus (buses), a transfer to the alternate source will be initiated in addition to bus load shedding (slow transfer). If the transfer to the alternate source is successful, the respective DG will NOT start, and loads will be sequenced on to the bus powered by the alternate power supply. If the transfer to the alternate source is not successful, the respective DG will receive a start signal (1.0 second time delay following loss of power) and loads will be sequenced on to the bus supplied by the diesel.

During outages, one or more Alternate Power Diesel Generators (APDG's) are leased and located on site. They are connected to the 6.9KV Safeguards switchgear through a transfer switch and a previously spare bus breaker. This capability does not meet technical specification operability requirements for emergency power sources. However, it does contribute to plant safety by providing another source of power, if needed, to the safeguards buses during shutdown operations. The APDG's are available to feed the selected 6.9KV Class 1E bus in modes 5 and 6. They are only used in the event of loss of offsite power coincident with failure of both onsite Class 1E Emergency Diesel Generators.

Spare Transformer XST2A

A spare transformer is connected on the high side to the 345Kv line which also supplies Transformers XST2 and 1ST. Jumpers must be obtained from the warehouse in order to connect the 6.9KV low side of the spare transformer to the bus. It is available for use in case of failure of a startup transformer.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	063.G.2.2.38		
Level of Difficulty: 3	Importance Rating	3.6		

DC Electrical Distribution: Knowledge of conditions and limitations in the facility license.

Question # 16

With Unit 1 in Mode 3 and Battery Charger BC1ED1-2 INOPERABLE, which of the following would require entry into TS 3.8.4, DC Sources – Operating, action statements?

A. Battery BT1ED2 voltage is 125 VDC.

B. Battery BT1ED3 float current is 0 amps.

C. Battery Charger BC1ED1-1 “EQUALIZE” push button is depressed.

D. Battery Charger BC1ED4-1 is placed under clearance after placing Battery Charger BC1ED4-2 in service.

Answer: A

K/A Match: K/A match due to requiring knowledge of TS entry conditions for DC electrical systems.

Explanation:

A. Correct. Each battery is required to have a minimum float voltage of 2.13 volts per cell or 128 volts total.

B. Incorrect. Plausible since 0 amps may indicate that the battery is not performing its function, but a float current of 0 amps indicates that the battery charger has fully charged the battery and the battery charger is supplying all bus loads.

C. Incorrect. Plausible since battery charger BC1ED1-2 is already out of service, but depressing the equalize push button causes battery charger output voltage to be 138 – 140 VDC which is an operable condition.

D. Incorrect. Plausible since one battery charger is already out of service, but BC1ED4-1 is a different train and BC1ED4-2 would be placed in service prior to placing BC1ED4-1 under clearance.

Technical Reference(s)	TS 3.8.4	Attached w/ Revision # See Comments / Reference
	MSE-S0-5000	

Proposed references to be provided during examination: ____________________

Learning Objective: Given system parameter indications and plant conditions, **ASSESS** from memory any required TS/TR entries, including any actions which must be completed within one hour in accordance with Technical Specifications or TRM. (SYS.DC1.OB05)

Question Source:
- Bank # 19161
- Modified Bank # ______ (Note changes or attach parent)
- New ______

Question History: Last NRC Exam 2015 NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge X
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 10
- 55.43 ______

Comments / Reference: TS 3.8.4	Revision: 170

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources -- Operating

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required battery chargers on one train inoperable.	A.1 Restore affected battery(ies) terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	AND	
	A.2 Verify affected battery(ies) float current ≤ 2 amps.	Once per 12 hours
	AND	
	A.3 Restore required battery charger(s) to OPERABLE status.	7 days

-

Comments / Reference: TSB 3.8.4	Revision: 82

BASES

BACKGROUND (continued)

not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

Each battery has adequate capacity to meet the duty cycle(s) discussed in the FSAR, Chapter 8 (Ref. 4). The battery is designed with additional capacity above that required by the design duty cycle to allow for temperature variations and other factors.

The batteries for Train A and Train B DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life and the 100% design demand. During duty cycle the batteries maintain a voltage of 105 V or greater that will provide adequate voltage for operation of all required loads.

The battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of approximately 2.065 volts per cell (Vpc). The open circuit voltage is the voltage maintained when there is no charging or discharging. Once fully charged with cell float voltage ≥ 2.07 Vpc, the battery cell will maintain its capacity for 30 days without further charging per manufacturer's instructions. The battery float charge voltage limit is established as 2.13 V per cell, which corresponds to a total minimum float voltage output of 128 V for a 60 cell battery. Optimal long term performance however, is obtained by maintaining a float voltage 2.20 to 2.25 Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge. The nominal float voltage of 2.20 Vpc corresponds to a total float voltage output of 132 V for a 60 cell battery as discussed in the FSAR, Chapter 8 (Ref. 4).

Each Train A and Train B DC electrical power subsystem battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient excess capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in the FSAR, Chapter 8 (Ref. 4).

The battery charger is normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate current to optimally charge the battery. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

(continued)

Comments / Reference: MSE-S0-5000	Revision: 7

CPNPP MAINTENANCE SECTION - ELECTRICAL MANUAL	UNIT 1 & 2	PROCEDURE NO. MSE-S0-5000
CLASS 1E STATION BATTERIES WEEKLY-MONTHLY-QUARTERLY SURVEILLANCE TESTS	REVISION NO. 7 REFERENCE USE	PAGE 25 OF 31

ATTACHMENT 10.3
PAGE 2 OF 2

DATA PACKAGE

Work Order Number:________________ Battery Number: ________________

STEP NO.	DATA RECORDING AREA	INITIALS
6.0	**PREREQUISITES**	
6.4	Charger in FLOAT ☐ Charger PLACED in FLOAT ☐	________
6.5	Battery Room ventilation VERIFIED operable.	________
6.6	Emergency eye-wash/shower station VERIFIED operable.	________
8.0	**INSTRUCTIONS**	
8.2	Tech Spec Surveillance - WEEKLY	
8.2.2	Battery float voltage:________ V	________
8.2.3	Battery float voltage VERIFIED ≥ 128 volts. SAT ☐ UNSAT ☐	________
8.5	Battery Charger Adjustment	
8.5.3	Charger position: ☐ FLOAT ☐ EQUALIZE (IF applicable)	________
8.5.3.1	Alarm condition: ☐ Actuated ☐ NOT actuated (IF applicable)	________
8.5.3.2	Alarm reset: ☐ Yes ☐ No ☐ Corrected (IF applicable)	________

COMMENTS

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Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	064.K6.07		
Level of Difficulty: 2	Importance Rating	2.7		

Emergency Diesel Generator: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers

Question # 17

Given the following conditions:

- DG 1-01 in normal alignment
- 1-ALB-10B Window 2.8, DG1 Trouble, annunciates

NEO reports the following:

- Low Press Starting Air Left Bank in alarm
- Low Press Starting Air Right Bank in alarm
- Starting air pressure in both banks 160 psig and lowering

Per TS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, the required minimum starting air pressure __(1)__ satisfied.

DG 1-01 will start __(2)__ start signal.

A. (1) is
(2) ONLY on an emergency

B. (1) is
(2) on BOTH an emergency and normal

C. (1) is NOT
(2) ONLY on an emergency

D. (1) is NOT
(2) on BOTH an emergency and normal

Answer: D

K/A Match: K/A match due to requiring knowledge of the effect of low air receiver pressure on the EDG operations.

Explanation:

A. Incorrect. First part is incorrect. The minimum air pressure required by TS 3.8.3 is ≥ 180 psig, therefore the minimum starting air pressure is not satisfied. Second part is incorrect, but plausible because this is a common misconception that the emergency start would be available at lower air pressures than a normal manual start. However, with air pressure greater than approximately 90 psig the normal start signal is available and the emergency start will be blocked below 150 psig.

B. Incorrect. First part is incorrect (see A). Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect (see A).

D. Correct. First part is correct. The minimum air pressure required by TS 3.8.3 is not met. Second part is correct. The DG will start on both an emergency and normal start because only 90 psig of air is required for a normal start and 150 psig of air is required for an emergency start.

Technical Reference(s)	TS 3.8.3	Attached w/ Revision # See Comments / Reference
	OPT-214A	
	EDG Study Guide	

Proposed references to be provided during examination: ______________________

Learning Objective: Given Emergency DG parameters and indications, **ASSESS** from memory any required TS/TR entries, including any actions which must be completed within one hour in accordance with Technical Specifications or TRM (SYS.ED1.OB24)

Question Source: Bank # 67832
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: TS 3.8.3	Revision: 150

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more DGs with stored fuel oil total particulates not within limit.	C.1 Restore fuel oil total particulates within limit.	7 days
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. Required Action and associated Completion Time not met. OR One or more DGs diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C or D.	E.1 Declare associated DG inoperable.	Immediately

Comments / Reference: TS 3.8.3	Revision: 156

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ a 7 day supply of fuel.	In accordance with the Surveillance Frequency Control Program.
SR 3.8.3.2	-------------------------------NOTE------------------------------- Not required to be performed until the engine has been shutdown for > 10 hours. -- Verify lubricating oil inventory is ≥ a 7 day supply	In accordance with the Surveillance Frequency Control Program.
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each required DG air start receiver pressure is ≥ 180 psig.	In accordance with the Surveillance Frequency Control Program.
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program.

Comments / Reference: OPT-214A	Revision: 25

CPNPP OPERATIONS TESTING MANUAL	UNIT 1	PROCEDURE NO. OPT-214A
DIESEL GENERATOR OPERABILITY TEST	REVISION NO. 25 CONTINUOUS USE	PAGE 15 OF 145

5.2 Limitations (continued)

- As a minimum, the following A.C. electrical power sources shall be OPERABLE in MODES 5 AND 6 per TS 3.8.2:
  - One circuit between the offsite transmission network AND the onsite class 1E Distribution subsystem required by LCO 3.8.10.
  - One diesel generator capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem required by LCO 3.8.10 with a fuel oil day tank containing a minimum volume of 1440 gallons of fuel.
- The Stored Diesel Fuel Oil, Lube Oil AND Starting Air Subsystem shall be within limits for each DG required to be OPERABLE per TS 3.8.3 as follows:
  - The Fuel Oil Storage Tank shall contain ≥86,000 gallons (MODES 1-6) of fuel.
  - Lubricating oil inventory shall be ≥ a level 1.75 inches below the low static level on the lube oil dipstick.
  - Either one of two, starting air systems satisfies Emergency Diesel Generator operability. Required DG starting air receiver pressure shall be ≥ 180 PSIG (actual pressure) [The surveillance requires ≥ 184 PSIG due to instrument error AND drift per CALC # IC-CA-0215-5112.]

Comments / Reference: OPT-214A	Revision: 25

CPNPP OPERATIONS TESTING MANUAL	UNIT 1	PROCEDURE NO. OPT-214A
DIESEL GENERATOR OPERABILITY TEST	REVISION NO. 25 CONTINUOUS USE	PAGE 16 OF 145

5.3 Notes

- IF a DG received a manual emergency start from the Control Room,
  THEN
  full shutdown protection may be restored by placing the MASTER SWITCH (RLMS) to LOCAL, the Remote Emergency STOP/START handswitch momentarily to STOP
  AND
  THEN RETURNING the MASTER SWITCH (RLMS) to REMOTE.
- IF a DG received a local emergency start signal,
  THEN
  full shutdown protection can be restored by returning the Local Emergency STOP/START switch to the CENTER position.
  - Full shutdown protection is restored when the SHUTDOWN SYS ACTIVATED light is ON.
- A DG will NOT accept an emergency start signal unless the pressure in at least one start air receiver is greater than 150 psig.
  - The DG will accept the start signal at approximately 150 psig, however, approximately 170 psig is required in the receiver to actually start the DG.
- IF FOST normal level indication is unavailable,
  THEN
  REFER to Attachment 10.4 for level in inches from the inside top of FOST.
- SR 3.8.2.1 allows the DG to be exempted from load testing during MODES 5 AND 6.
- Following a loss of air to the diesel generator pneumatic logic board, RESTORATION of air will require RESET of the DIESEL GENERATOR RUN/STOP MECHANICAL TRIP PRESSURE SWITCH.
  - An internal relay on the logic board will vent air until the DIESEL GENERATOR RUN/STOP MECHANICAL TRIP PRESSURE SWITCH is pushed to TRIP AND then pulled out to RUN.
- IF control air is lost (e.g., the air receiver outlet valves are closed) while the engine is in the MAINTENANCE mode,
  THEN
  the pneumatic control system will return to the Normal Mode when control air is re-established.
  - PLACE the Mode Select Switch to the MAINT position again to return the DG pneumatic control system to the MAINTENANCE mode.

Comments / Reference: EDG Study Guide	Revision: 00-0000

NOTES	LESSON OUTLINE

major air start components, except for the receivers, are the same between the two units.

3) Receiver pressure between 220 psig and 250 psig

4) Receiver Relief Valves are set to lift at 275 psig and relieve to the surrounding area.

5) Vendor test data shows that a diesel was started 5 times with an initial receiver pressure of 210 psig and a final pressure of 156 psig. A minimum of approximately 90 psig of air pressure is required to start the engine.

6) LOW PRESS STARTING AIR annunciator - ≤ 210 psig.

7) Start Air Pressure Switches ≤ 150 psig.

8) Function to disable engine emergency start signals when air pressure is low.

d. Start Air Admission

1) Four solenoid operated start air admission valves; two per start air header.

2) located on the catwalk at the generator end of the engine.

3) Powered from the safeguards DC bus associated with each train (uED1 for Train A EDG and uED2 for Train B EDG).

4) Start air admission valves close 5 seconds after a normal start signal or at 200 rpm, whichever comes first.

5) If some set of conditions ever developed to maintain the start air admission valves open and the fuel racks in the no-fuel position, the engine speed would stabilize at approximately 90 to 110 rpm.

e. Starting Air Distributors and Cylinder Air Start Valves

1) Each camshaft-driven starting air distributor regulates the action of the 8 cylinder air start valves for its side of the engine.

f. Engine Barring Device

1) It consists of a steel rod extending from the cylinder.

2) When air is admitted, the steel rod is extended approximately one foot.

3) The engine must be placed in the Maintenance Mode to use the barring device.

Comments / Reference: EDG Study Guide	Revision: 00-0000

NOTES	LESSON OUTLINE
	4) The barring device locking arrangement is intended to ensure that the barring device is never engaged in a flywheel tooth when the engine is rotated with air as the barring.
Objective 11	4. EDG Starting Air System Operation
	a. Start Air System Startup
	1) The compressor handswitch is placed in AUTO.
	2) The compressor pressurizes its associated receiver and piping. At 75 psig receiver pressure the breaker for the associated air dryer is closed. The compressor will continue to run until the receiver pressure reaches the compressor shutoff setpoint of 250 psig.
	b. The start air system on a diesel generator set is considered available to perform its intended design function if pressure in at least one receiver is > 180 psig and that receiver's associated compressor is capable of maintaining receiver pressure > 180 psig without running continuously.
TS required minimum ≥ 180 psig (OPT-214 req ≥ 184 psig due to inst error & drift)	c. A DG will not accept an emergency start signal unless the pressure in at least one start air receiver is greater than 150 psig. Approximately 170 psig is required in the receiver to actually start the DG.
	d. 1/4-inch diameter tubing connects from the intake air header to the air start header on either side of the engine block. During engine operation, a small amount of air passes from the intake air headers through these lines to the air start headers and then leaves the air start headers through a small drilled passage to atmosphere. This line continuously purges the air start headers to that any combustion gas leaking back through the closed air start valves into the air start header will be removed by the purge air flow. A continuous flow of air during engine operation from the drilled passage in each air start header ensures the hole remains unobstructed
Objective 12	**F. EDG Lube Oil System**
Figure 15	1. Flowpath
	a. Lube oil system supplies an adequate quantity of cooled, filtered oil under pressure for lubrication and cooling of engine friction surfaces which include the engine bearing, the various engine driven gears and the valve rocker arm assemblies. Oil is also supplied to both turbochargers.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	073.K1.01		
Level of Difficulty: 2	Importance Rating	3.6		

Process Radiation Monitoring: Knowledge of the physical connections and/or cause-effect relationships between the PRM system and the following systems: Those systems served by PRMs

Question # 18

A MSL Radiation Monitor trend has initially increased above the normal reading following a 1.0 gpm SG tube leak.

Assuming all procedures are implemented:

The trend on the MSL Radiation Monitor will __(1)__ as power is decreased; and,

When the MSIV is closed, the trend on the MSL Radiation Monitor will __(2)__.

A. (1) stabilize
   (2) increase rapidly to a RED alarm

B. (1) stabilize
   (2) decrease to a value above the normal reading

C. (1) decrease
   (2) increase rapidly to a RED alarm

D. (1) decrease
   (2) decrease to a value above the normal reading

Answer: D

K/A Match: K/A match due to requiring knowledge of radiation monitor trends following actions taken due to high radiation levels.

Explanation:

A. Incorrect. First part is incorrect, but plausible since lowering load causes less steam flow to leave steam line, so it could be thought that leakage is being contained in the area of the rad monitor causing rad levels to stabilize. Second part is incorrect, but plausible (see C).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible since it could be thought that containing all the leakage upstream of the MSIV after it is closed will cause rad levels to increase.

D. Correct. First part is correct. As power lowers for the same size leak the trend decreases due to less activity. Second part is correct. When the MSIV is closed no flow goes past the rad monitor so the rad monitor decreases back towards normal.

Technical Reference(s)	ABN-106	Attached w/ Revision # See Comments / Reference
	Main Steam Study Guide	
	DRMS Study Guide	

Proposed references to be provided during examination: __________

Learning Objective: **PREDICT** the response of the instrumentation and controls of the Digital Radiation Monitoring System in accordance with DBD-EE-023. (SYS.RM1.OB04)

Question Source: Bank # 17676
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 11
55.43 ________

Comments / Reference: ABN-106	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 11	PAGE 13 OF 31

3.0 STEAM GENERATOR TUBE LEAKAGE GREATER THAN OR EQUAL TO 75 GPD (0.052 GPM) for Unit 1 and GREATER THAN OR EQUAL TO 50 GPD (0.0347 GPM) for Unit 2 Cycle 19

3.1 Symptoms

a. Annunciator Alarms

None

b. Plant Indications

- Steam Generator leakage in excess of 75 gpd (0.052 gpm) for Unit 1 and in excess of 50 gpd (0.0347 gpm) for Unit 2 Cycle 19 as reported by Chemistry. The reported leak rate should be verified with a second independent radiation monitor or grab sample.
- Unidentified leakage in excess of TS Limits as determined by OPT-303 which is suspected to be Steam Generator tube leakage.
- An abnormal increase in main steamline radiation as indicated on u-RE-2325 (MSL-u78), u-RE-2326 (MSL-u79), u-RE-2327 (MSL-u80), and u-RE-2328 (MSL-u81) or leak rate indication on u-RE-2325A (N16-u74), u-RE-2326A (N16-u75), u-RE-2327A (N16-u76), and u-RE-2328A (N16-u77). Computer points R7749A(R7753A) thru R7752A(R7756A).

3.2 Automatic Actions

- Steam Generator blowdown will isolate on high radiation as indicated on u-RE-4200 (SGS-u64).
  - u-HS-2397, SG 1 BLDN ISOL VLV
  - u-HS-2398, SG 2 BLDN ISOL VLV
  - u-HS-2399, SG 3 BLDN ISOL VLV
  - u-HS-2400, SG 4 BLDN ISOL VLV

Section 3.0

Comments / Reference: ABN-106	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 11	PAGE 15 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Due to the minimum sensitivity of the MSL radiation monitors, a valid alarm indicates a leak rate of at least 3600 gpd (2.5 gpm).

☐ 1 Verify main steamline radiation alarms - CLEAR

- u-RE-2325 (MSL-u78)
- u-RE-2326 (MSL-u79)
- u-RE-2327 (MSL-u80)
- u-RE-2328 (MSL-u81)

a. Initiate power reduction to ≤50% in 1 hour

AND

Be in MODE 3 in the next 2 hours.

[C]
b. Calculate gross leak rate, refer to EPP-201.

c. GO TO Step 4.b.

(STA-732, Attachment 8.B) INFORMATION ONLY - ACTIONS DIRECTED BY THIS PROCEDURE

NOTE:
- Leakage is qualitatively confirmed when two independent radiation monitors trend in the same direction with the same order of magnitude. With two confirmed indications (i.e. N-16 and COG monitors or other combination of monitor indication and sample analyses):
- CPNPP uses the CONSTANT LEAKAGE METHOD.

LEAKAGE/LEAK RATE	ACTION
Primary to secondary leakage ≥75 gpd (0.052 gpm) for Unit 1 and ≥ 50 gpd (0.0347 gpm) for Unit 2, Cycle 19 sustained for ≥1 hour	Normal shutdown to be in MODE 3 in ≤24 hours
Primary to secondary leakage ≥100 gpd (0.07 gpm) OR Primary to secondary leakage ≥75 gpd (0.052 gpm) for Unit 1 and ≥ 50 gpd (0.0347 gpm) for Unit 2, Cycle 19 sustained for ≥1 hour AND NO condenser off-gas radiation monitor available AND main steam line leak rate radiation monitor on affected SG(s) - NOT OPERABLE	Reduce power to ≤50% in 1 hour AND Be in MODE 3 in the next 2 hours

Section 3.3

Comments / Reference: ABN-106	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 11	PAGE 16 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2 Correlate monitor readings to leak rate • (≥40% power) N16 leak rate indication. • Contact Chemistry Personnel to determine current Pri-Sec leakage based on COG readings.	

CAUTION: The RNO action of Step 3 should be initiated WHEN leak rate increases to 100 gpd (.07 gpm) (i.e. from 75 gpd (0.052 gpm) to 100 gpd for Unit 1 and 50 gpd (0.0347 gpm) to 100 for Unit 2 Cycle 19) during the 24 hour shutdown window of step 3.a.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3 Verify leak rate <100 gpd (0.07 gpm):	Perform the following:
a. Be in Mode 3 in ≤24 hours.	a. Reduce power to ≤50% in 1 hour AND Be in MODE 3 in the next 2 hours.
b. Continue monitoring leak rate and leak rate, rate of change.	[C] b. Refer to EPP-201. c. GO TO Step 4.b.

Section 3.3

Comments / Reference: ABN-106	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 11	PAGE 21 OF 31

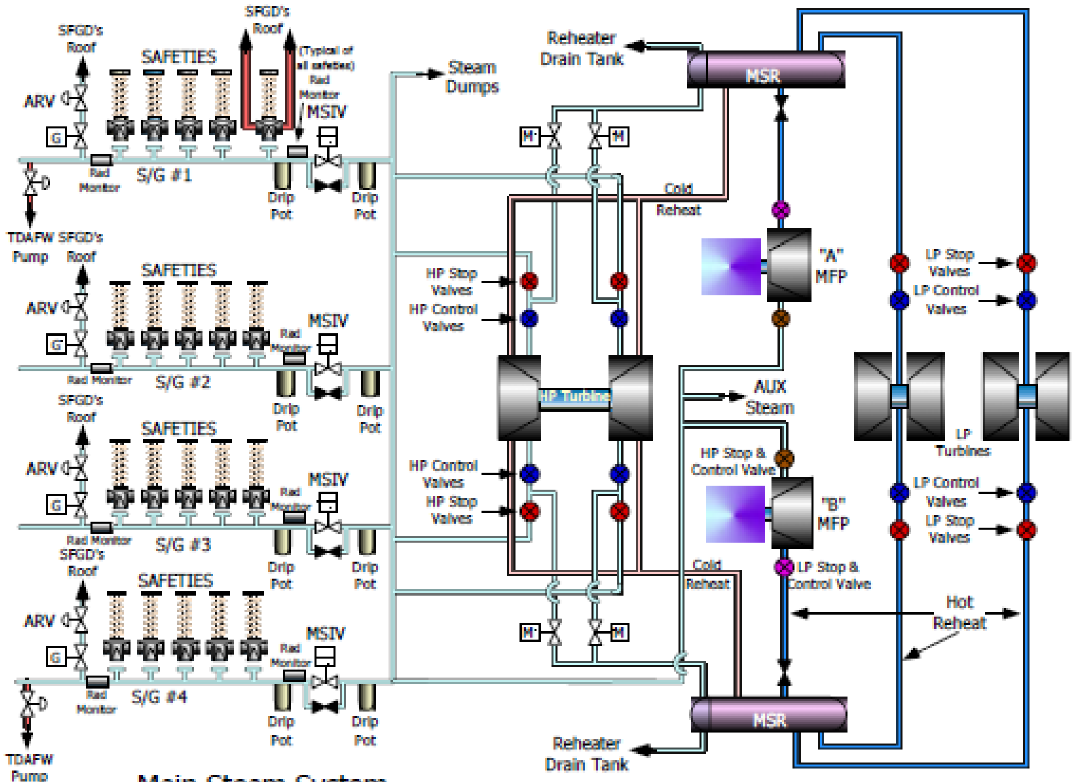
3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 10 Verify NO significant increase exists, as indicated on PC-11, from system(s) drains that discharge to the LVW: • u-RE-4200, (SGS-u64) BLOWDOWN SMPL • u-RE-2959, (COG-u82) CONDENSER OFF GAS • u-RE-2325, (MSL-u78) MAIN STEAM LINE #1 • u-RE-2326, (MSL-u79) MAIN STEAM LINE #2 • u-RE-2327, (MSL-u80) MAIN STEAM LINE #3 • u-RE-2328, (MSL-u81) MAIN STEAM LINE #4 • u-RE-2325A, (N16-u74) MAIN STEAM LINE #1 LEAK RATE • u-RE-2326A, (N16-u75) MAIN STEAM LINE #2 LEAK RATE • u-RE-2327A, (N16-u76) MAIN STEAM LINE #3 LEAK RATE • u-RE-2328A, (N16-u77) MAIN STEAM LINE #4 LEAK RATE	Notify Rad Waste Operator, -AND- Ensure any affected sump drains being discharged to LVW are STOPPED -OR- diverted to the Waste Water Holdup Tank.

Section 3.3

Comments / Reference: Main Steam Study Guide	Revision: 00-0000

STEAM EXITING THE STEAM GENERATORS



Main Steam System
Figure 5

Comments / Reference: DRMS Study Guide	Revision: 4-28-2011

Digital Radiation Monitoring System

These monitors can also be powered from Support Power during outages. SOP-613A(B) "Outage Power", provides instruction on how to shift the power supply.

AUXILIARY BUILDING TO LOW VOLUME WASTE POND

This liquid process monitor monitors the drains from the Auxiliary Building, Diesel Generator Sumps, and the CCW Drain Line. A high radiation alarm or an Operate Failure will divert the water from the Low Volume Waste Pond (X-HV-WM183 closes) to the Waste Holdup Tank (X-HV-WM182) opens.

BORON RECYCLE EVAPORATOR CONDENSATE MONITOR

This liquid monitor will annunciate on the Waste Boron Process Panel when the monitor is in an Alert condition or in Operate Failure. If the activity level increases to the High Radiation Alarm setpoint, X-RV-016 "Boron Recycle Evaporator Condensate Filter X-01 Divert Valve" will divert to the Recycle Evaporator Feed Demineralizers.

FAILED FUEL MONITOR

This liquid monitor uses a low level GM tube to detect fuel leakage (**Figure 11**). It uses system differential pressure for sample flow. No automatic actions are associated with this monitor.

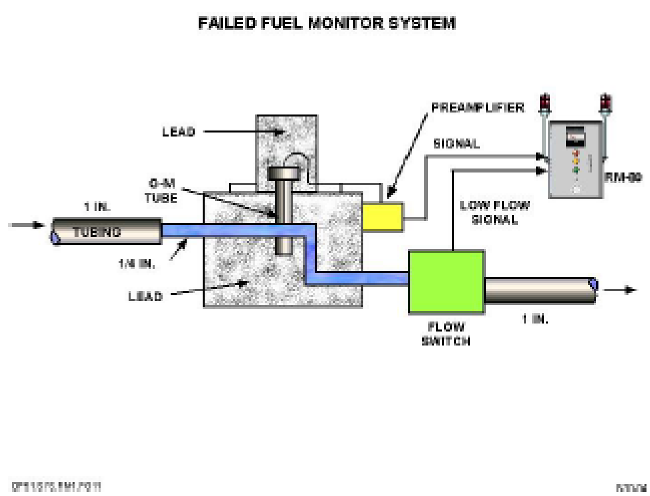


Figure 11 - Failed Fuel Monitor System

MAIN STEAM LINE MONITOR

These are a specialized area type monitors used for monitoring a process stream. The Main Steam Line monitors use GM tubes that are sensitive to a wide range of gamma energies. They monitor for gross primary to secondary leakage (~1 gpm or 3600 gpd). Or in other words greater that the tech spec limit. The MSL detectors are mounted on the Main Steam Line pipe immediately upstream of the MSIVs. It is important to understand that the readings will trend down, following the closure of the MSIVs or following a power reduction or shutdown. This is due to no flow going past the monitor

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	076.K4.06		
Level of Difficulty: 3	Importance Rating	2.8		

Service Water: Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Service water train separation

Question # 19

Given the following conditions:

- Unit 1 is in MODE 6 with the Refueling Cavity flooded
- Unit 1 PDP is supplying RCP seal injection
- Unit 2 has tripped and during post-trip actions lost both trains of SSW
- The only available Unit 1 SSW Pump, 1-01, has been aligned to supply Unit 2 SSW Train B

Which of the following describes the flowpath for this alignment per SOP-501A, Station Service Water System?

Unit 1 SSWP 1-01 is running with its discharge valve full open. Flow to Unit 2 is adjusted by throttling the __(1)__.

In this configuration, the Unit 2 Train B CCP lube oil cooler __(2)__ be supplied with SSW.

A. (1) Unit 2 SSWP (2-02) Discharge Valve
(2) can

B. (1) Unit 2 SSWP (2-02) Discharge Valve
(2) cannot

C. (1) Unit 1 to Unit 2 cross-tie valve
(2) can

D. (1) Unit 1 to Unit 2 cross-tie valve
(2) cannot

Answer: A

K/A Match: K/A match due to requiring knowledge of how SSW trains, normally maintained separated, can be cross-connected.

Explanation:

A. Correct. First part is correct. Per ABN-501, flow is from Unit 1 SSW Pump 1-01 through full open discharge, crosstie line and Unit 2 SSW Pump 2-02 discharge which is at 15% on initial pump start and then throttled full open or until pump runout flow of 18,600 gpm. Second part is correct. Unit 1 Train A CCW Heat Exchanger, Unit 2 Train B CCW Heat Exchanger, and Unit 2 Train B Centrifugal Charging Pump are aligned for flow.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible since the Unit 1 CCP cannot be supplied, but the Unit 2 CCP can be supplied.

C. Incorrect. First part is incorrect, but plausible (see D). Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible since flow is through cross-tie, but valve is opened fully. Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-501	Attached w/ Revision # See Comments / Reference
	SOP-501	

Proposed references to be provided during examination: ___

Learning Objective: **DISCUSS** the response to a Loss of All Unit u Station Service Water in accordance with ABN-501, Station Service Water System Malfunction. (ABN.501.OB104)

Question Source: Bank # 36001
Modified Bank # ___ (Note changes or attach parent)
New ___

Question History: Last NRC Exam ___

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ___

Comments / Reference: ABN-501	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 23 OF 50

5.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Cross connecting Station Service Water between units will render cross connected trains of BOTH units INOPERABLE in MODE 1, 2, 3, or 4.

-0R-

Cross connecting Station Service Water between Trains within a unit will render BOTH trains INOPERABLE in MODE 1, 2, 3, or 4.

NOTE: IF barriers designated as Fire or Security Barriers, such as manways, doors, hatchcovers, slabs, etc. are to be breached, THEN the Shift Manager and Security shall be notified and approval obtained prior to affecting the breach.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7 Restore SSW cooling flow by CROSS-TIEING Train A AND Train B as follows:	Perform the following:
☐ a. Verify at least one SSW Train - AVAILABLE.	1) Cross-tie SSW between units per SOP-501A/B, as directed. 2) Refer to TS 3.7.8
☐ b. Verify Cross Connecting SSW Trains - REQUIRED BY EQUIPMENT CONDITIONS: 1) Request permission From Emergency Coordinator to cross-connect SSW Trains. 2) Cross Connect A and B SSW Trains per SOP-501A/B. 3) Declare unaffected train INOPERABLE AND INITIATE APPROPRIATE LCOAR.	3) IF NO SSW cooling flow available, AND • Cooling is required for essential equipment, THEN perform Attachment 2, Train A and/or Attachment 3, Train B to align alternate cooling to required equipment • Cooling not available by other methods is required to the diesels, THEN perform Attachment 1, Fire Protection Water Alignment to Diesel Generators.

Section 5.3

Comments / Reference: SOP-501A	Revision: 20

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 & COMMON	PROCEDURE NO. SOP-501A
STATION SERVICE WATER SYSTEM	REVISION NO. 20 CONTINUOUS USE	PAGE 72 OF 105

5.7.3 A Single Unit 1 SSW Pump Supplying Service Water Flow to Both Units

This section describes the steps to supply one train on both units from one Station Service Water Pump. Worst Case Conditions are assumed with Unit 1 in Mode 5 or 6 and Unit 2 in Mode 3 or 4.

> CAUTION: This evolution should only be performed when directed to do so in accordance with ABN-501. PERFORMANCE of this section affects the operability of Unit 1 AND Unit 2.

A. ENSURE the following conditions for the Unit 2 SSW Train:

- ☐ Unit 2 is operating in MODE 3 OR 4.
- ☐ Service Water is only required to supply a CCW Heat Exchanger AND Charging Pump Lube Oil Coolers.
- ☐ A Safety Injection signal is NOT present.
- ☐ A Loss of Offsite power has NOT occurred.
- ☐ Steam Dumps are available.
- ☐ One Reactor Coolant Pump is available.
- ☐ RHR NOT being used for cooldown.

B. ENSURE the following conditions for the Unit 1 SSW Train:

- ☐ Unit 1 is operating in MODE 5 OR 6.
- ☐ Service Water is only required to supply the CCW Heat Exchanger.
- ☐ One pump is lined up AND operating per SOP-501A.
- ☐ A Safety Injection signal is NOT present.
- ☐ A Loss of Offsite power has NOT occurred.

Comments / Reference: SOP-501A	Revision: 20

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 & COMMON	PROCEDURE NO. SOP-501A
STATION SERVICE WATER SYSTEM	REVISION NO. 20 CONTINUOUS USE	PAGE 73 OF 105

5.7.3 C. PERFORM the following for the Unit 1 Service Water Train:

1) PLACE the selected train handswitches in PULL-OUT:

Train A SSW

- ☐ • 1/1-APSI1, SIP 1
- ☐ • 1-HS-4764, CSP 1
- ☐ • 1-HS-4765, CSP 3
- ☐ • 1/1-APCH1, CCP 1

Train B SSW

- ☐ • 1/1-APSI2, SIP 2
- ☐ • 1-HS-4766, CSP 2
- ☐ • 1-HS-4767, CSP 4
- ☐ • 1/1-APCH2, CCP 2

2) ISOLATE Service Water flow by Closing the following valves for the selected train:

Train A SSW

- ☐ • 1SW-0404, SI PMP 1-01 L\O CLR SSW STRN 1-01 IN ISOL VLV
- ☐ • 1SW-0399, CS PMP 1-01/1-03 BRG CLR SSW IN ISOL
- ☐ • 1SW-0358, CCP 1-01 L\O CLR SSW IN ISOL VLV

Train B SSW

- ☐ • 1SW-0402, SI PMP 1-02 L\O CLR SSW STRN 1-02 IN ISOL VLV
- ☐ • 1SW-0396, CS PMP 1-02/1-04 BRG CLR SSW IN VLV
- ☐ • 1SW-0356, CCP 1-02 L\O CLR SSW IN ISOL VLV

-

Comments / Reference: SOP-501A	Revision: 20

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 & COMMON	PROCEDURE NO. SOP-501A
STATION SERVICE WATER SYSTEM	REVISION NO. 20 CONTINUOUS USE	PAGE 74 OF 105

5.7.3 C. 3) ISOLATE flow to the selected Diesel Generator by CLOSING the following valve for the selected train:

Train A SSW

- ☐ • 1-HS-4393, DG 1 CLR SSW RET VLV

Train B SSW

- ☐ • 1-HS-4394, DG 2 CLR SSW RET VLV

D. PERFORM the following for the Unit 2 Service Water Train:

1) PLACE the selected train handswitches in PULL-OUT:

Train A SSW

- ☐ • 1/2-APSI1, SIP 1
- ☐ • 2-HS-4764, CSP 1
- ☐ • 2-HS-4765, CSP 3

Train B SSW

- ☐ • 1/2-APSI2, SIP 2
- ☐ • 2-HS-4766, CSP 2
- ☐ • 2-HS-4767, CSP 4

2) ISOLATE Service Water flow by Closing the following valves for the selected train:

Train A SSW

- ☐ • 2SW-0362, SI PMP 2-01 L\O CLR SSW IN ISOL VLV
- ☐ • 2SW-0420, CS PMP 2-01/2-03 BRG CLR SSW IN VLV

Train B SSW

- ☐ • 2SW-0361, SI PMP 2-02 L\O CLR SSW IN ISOL VLV
- ☐ • 2SW-0418, CS PMP 2-02/2-04 BRG CLR SSW IN VLV

Comments / Reference: SOP-501A	Revision: 20

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 & COMMON	PROCEDURE NO. SOP-501A
STATION SERVICE WATER SYSTEM	REVISION NO. 20 CONTINUOUS USE	PAGE 75 OF 105

5.7.3 D. 3) ISOLATE flow to the selected Diesel Generator by CLOSING the following valve for the selected train:

Train A SSW

☐ • 2-HS-4393, DG 1 CLR SSW RET VLV

Train B SSW

☐ • 2-HS-4394, DG 2 CLR SSW RET VLV

4) OPEN the Power Supply to the Unit 2 SSW Pump Discharge Valve on the train that is to be supplied from Unit 1.

Train A SSW

☐ • 2EB3-3/1M/BKR, SSW PUMP 2-01 DISCHARGE VALVE 4286 MOTOR BREAKER

Train B SSW

☐ • 2EB4-3/2E/BKR, SSW PUMP 2-02 DISCHARGE VALVE 4287 MOTOR BREAKER

5) PLACE the selected train handswitch in PULL-OUT:

Train A SSW

☐ • 2-HS-4250A, SSWP 1

Train B SSW

☐ • 2-HS-4251A, SSWP 2

E. Manually OPEN the Discharge Valve approximately 15% (approximately 15 turns) on the train to be supplied from Unit 1.

Train A SSW

☐ • 2-HV-4286, SSW PMP 2-01 DISCH VLV

Train B SSW

☐ • 2-HV-4287, SSW PMP 2-02 DISCH VLV

-

Comments / Reference: SOP-501A	Revision: 20

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 & COMMON	PROCEDURE NO. SOP-501A
STATION SERVICE WATER SYSTEM	REVISION NO. 20 CONTINUOUS USE	PAGE 76 OF 105

5.7.3 F. UNLOCK AND OPEN Unit 1 SSW supply to Unit 2 from the selected SSW Pump.

Train A SSW

☐ • XSW-0008, SSW PMP 1-01 DISCH HDR TO XTIE HDR ISOL VLV

Train B SSW

☐ • XSW-0007, SSW PMP 1-02 DISCH HDR TO XTIE HDR ISOL VLV

☐ G. UNLOCK AND OPEN XSW-0006, U1/U2 SSW XTIE HDR ISOL VLV.

H. UNLOCK AND OPEN the Unit 1/Unit 2 cross connect for the train to be supplied.

Train A SSW

☐ • XSW-0028, SSW PMP 2-01 DISCH HDR TO XTIE HDR ISOL VLV

Train B SSW

☐ • XSW-0029, SSW PMP 2-02 DISCH HDR TO XTIE HDR ISOL VLV

I. Slowly CYCLE XSW-0033, U1 SSW PMP TO U2 SSW PUMP XTIE HDR VNT VLV until a steady stream of water is verified.

• XSW-0033, U1 SSW PMP TO U2 SSW PUMP XTIE HDR VNT VLV

OPEN	CLOSED
☐	☐

J. Manually slowly OPEN SSW Pump Discharge Valve on the loop to be placed in service.

☐ • 2-HV-4286, SSW PMP 2-01 DISCH VLV

☐ • 2-HV-4287, SSW PMP 2-02 DISCH VLV

Comments / Reference: SOP-501A	Revision: 20

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 & COMMON	PROCEDURE NO. SOP-501A
STATION SERVICE WATER SYSTEM	REVISION NO. 20 CONTINUOUS USE	PAGE 77 OF 105

5.7.3

> CAUTION: To prevent pump runout, total pump discharge flow (flow indicated on Unit 1 added to flow indicated on Unit 2) shall NOT exceed 18,600 gpm.

K. VERIFY system pressure and flow stabilizes.

Train A SSW

- ☐ • 1-PI-4252A, SSWP 1 DISCH PRESS
- ☐ • 1-FI-4258A, SSWP 1 DISCH FLO
- ☐ • 2-FI-4258A, SSWP 1 DISCH FLO

Train B SSW

- ☐ • 1-PI-4253A, SSWP 2 DISCH PRESS
- ☐ • 1-FI-4259A, SSWP 2 DISCH FLO
- ☐ • 2-FI-3259A, SSWP 2 DISCH FLO

☐ L. ADJUST system flowrates per Attachment 3 to maintain the optimum cooling capability during single pump operation.

> NOTE: IF the alignment of the Screenwash Pump Suction is changed, THEN the SSW system status file AND the Locked Component Deviation Log should be updated.

☐ M. ENSURE Station Service Water Screenwash Pumps are supplied from an operating Station Service Water Loop.

☐ N. VERIFY adequate Screenwash System Operation per Section 5.4.1.

O. VERIFY the selected loops radiation monitor returns to a green OPERATE status at the PC-11 console:

- ☐ • 2-RE-4269, UNIT 2 STATION SERVICE WATER TRAIN A TO DISCH CANAL RAD MONITOR (SSW-265)
- ☐ • 2-RE-4270, UNIT 2 STATION SERVICE WATER TRAIN B TO DISCH CANAL RAD MONITOR (SSW-266)

COMMENTS:____

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	2		
	Group	1		
	K/A	078.A3.01		
Level of Difficulty: 3	Importance Rating	3.1		

Instrument Air: Ability to monitor automatic operation of the IAS, including: Air pressure

Question # 20

Given the following conditions:

- IAC 1-01 is running in LEAD
- IAC 1-02 is in BACKUP with the Automatic Operation light ON
- An INSTR AIR HDR PRESS LO alarm is received at 85 psig
- The NEO isolates the leak, and IA header pressure is approximately 107 psig rising

Which of the choices below describes the compressor status?

A. Both compressors are running loaded

B. Both compressors are running unloaded

C. IAC 1-01 is running loaded; IAC 1-02 is running unloaded

D. IAC 1-01 is running unloaded; IAC 1-02 is running loaded

Answer: A

K/A Match: K/A match due to requiring knowledge of starting and loading air pressure setpoints for IA compressors.

Explanation:

A. Correct. Lead started at 105, backup at 100 psig. Both will run loaded until 115 psig.

B. Incorrect. Plausible since both will be running, but compressors will not unload until 115 psig. Could think unload occurs at 105 psig.

C. Incorrect. Plausible since lead started at 105, backup at 100 psig, but both will continue to run loaded until 115 psig. Could think unload occurs at 105 psig.

D. Incorrect. Plausible since lead started at 105, backup at 100 psig, but both will continue to run loaded until 115 psig. Could think unload occurs at 105 psig.

Technical Reference(s)	ABN-301	Attached w/ Revision # See Comments / Reference
	SOP-509A	
	Instrument Air Study Guide	

Proposed references to be provided during examination: ____________________

Learning Objective: **EXPLAIN** the instrumentation and controls of the Instrument Air system and the system response in accordance with DBD-ME-218. (SYS.IA1.OB04)

Question Source: Bank # 21692
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments / Reference: ABN-301	Revision: 14

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 14	PAGE 4 OF 130

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Step 1 is a Continuous Action Step.

☐ 1 IF in MODE 1, 2, 3, OR 4 AND Instrument Air Header pressure decreases to 35 psig OR control of system(s) is lost,
THEN manually trip the reactor AND GO TO EOP-0.0A/B while other operator(s) continue this procedure section starting with step 7.

NOTE:
- Loss of Instrument Air in the Auxiliary Building will affect components in Unit 1 and Unit 2.
- Section 4.0 provides actions for "Unit 2 Response to Loss of Unit 1 Instrument Air."
- IF an air compressor is in an Auto-Start condition, THEN it will NOT start until low pressure is sensed (105 psig if in LEAD, 100 psig if in BACKUP). Once low pressure is sensed, the Compressor will restart and load.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2 VERIFY at least one Instrument Air Compressor - ALIGNED TO THE UNIT AND RUNNING: • u-HS-3451, INST AIR COMP 1 • u-HS-3463, INST AIR COMP 2 • X-ZL-3452, INST AIR COMM COMPR 1 • X-ZL-3463, INST AIR COMM COMPR 2	WHEN Instrument Air Header Pressure - LESS THAN 100 psig, THEN ENSURE the backup OR standby Instrument Air Compressor aligned to Unit - RUNNING. • u-PI-3488, INST AIR AFTFILT OUT PRESS • u-HS-3451, INST AIR COMP 1 • u-HS-3463, INST AIR COMP 2 • X-ZL-3452, INST AIR COMM COMPR 1 • X-ZL-3463, INST AIR COMM COMPR 2

Section 2.3

Comments / Reference: SOP-509A	Revision: 24

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-509A
INSTRUMENT AIR SYSTEM	REVISION NO. 24 CONTINUOUS USE	PAGE 134 OF 175

[L] ATTACHMENT 2

INSTRUMENT AIR COMPRESSORS CP1-CICACO-01 and CP1-CICACO-02 DIAGNOSTIC GUIDE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
F.	COMPRESSOR RESTART	

CAUTION:
- Motor starts are limited to ten (10) per hour with 6 minute intervals between starts.
- If system alignment and support systems are correct for operation, the following steps may be used when plant conditions require quick restart of an air compressor.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
F.1	PLACE the UNLOAD/NORMAL toggle switch in the UNLOAD position.	
F.2	DEPRESS the STOP pushbutton.	
F.3	DEPRESS the RESET/START pushbutton.	
F.4	MONITOR the compressor for proper operation. (noise, vibration, oil pressure)	STOP the compressor.
F.5	PLACE the LOAD/UNLOAD toggle switch in the LOAD position.	
F.6	ENSURE air pressure is increasing toward the pressure band based on the operating Air Compressor indicated below: • 105 to 115 psig (Lead) • 100 to 115 psig (Backup)	

ATTACHMENT 2
AID F

Comments: ______________________________

Comments / Reference: IA Study Guide	Revision: 5-7-2011

Instrument Air

Lead Setpoints

- 105 PSIG Loads
- 115 PSIG Unloads
- 20 Second Delay on Start of Compressor to load
- Auto Shutdown if Running Unloaded > 20 Minutes (1-01 or 1-02 only)

Backup Setpoints

- 100 PSIG Loads (Common IACs load 95 psig)
- 115 PSIG Unloads
- 20 Second Delay on Start of Compressor to load
- Auto Shutdown if Running Unloaded > 20 Minutes

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Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	2		
	Group	1		
	K/A	103.K3.03		
Level of Difficulty: 3	Importance Rating	3.7		

Containment: Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under refueling operations

Question # 21

Per TS 3.9.4, Containment Penetrations, which of the following constitutes a loss of Containment Integrity during Refueling Operations and requires immediate termination of Core Alterations?

A. The Emergency Airlock outer door is closed with the inner door open.

B. The Steam Generator 1-01 manway cover is open with nozzle dam installed.

C. The Equipment Hatch is open, capable of being installed and held in place with 3 bolts.

D. The Personnel Airlock inner and outer doors are open with personnel staged to close either door.

Answer: C

K/A Match: K/A match due to requiring knowledge of containment integrity requirements during refueling operations with effect of the loss requiring termination of core alterations. Explanation:	
A. Incorrect. Plausible because one door of the Emergency Airlock is open, however, in accordance with TS 3.9.4, the Emergency Airlock is required to have one door closed, therefore the other door may be open. B. Incorrect. Plausible because with the SG manway open it could be thought this created an opening from the primary to the secondary side and result in a direct air to air breach. C. Correct. In accordance with TS 3.9.4, the equipment hatch must be closed or capable of being closed with 4 bolts holding it in place. With only 3 bolts available this would constitute a loss of containment integrity during refueling operations. D. Incorrect. Plausible because both doors of the Personnel Airlock are open resulting in a direct air to air breach of containment, however, per TS 3.9.4 this is allowed as long as one door of the PAL is capable of being closed.	

Technical Reference(s)	TS 3.9.4	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **DISCUSS** specific outage items related to reduced inventory in accordance with IPO-010. (IPO.010.OB05)

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

Question History: Last NRC Exam ______

Question Cognitive Level:
- Memory or Fundamental Knowledge X
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 10
- 55.43 ______

Comments / Reference: TS 3.9.4	Revision: Amd 150

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts, or if open, capable of being closed;

b. One door in the emergency air lock closed and one door in the personnel airlock capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
2. capable of being closed by an OPERABLE containment ventilation isolation valve.

---NOTE---
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	004.K2.02		
Level of Difficulty: 3	Importance Rating	2.9		

Chemical and Volume Control: Knowledge of bus power supplies to the following: Makeup pumps

Question # 22

Which of the following accurately states the bus power supply for Boric Acid Transfer Pump 1/1-APBA1?

A. 1B1-1

B. 1EB1-1

C. XB1-1

D. XEB1-1

Answer: B

K/A Match: K/A match due to requiring knowledge of power supplies to the BAT pumps for CVCS makeup.

Explanation:

A. Incorrect. Plausible since the power supply is unit related and the pump is not needed for emergency boration on an SI, so it could be a non-safeguards unit supply.

B. Correct. This is the correct power supply per SOP-105.

C. Incorrect. Plausible since the pump takes a suction off the X-01 tank and the pump is not needed for emergency boration on an SI, so it could be a non-safeguards common supply.

D. Incorrect. Plausible since the pump takes a suction off the X-01 tank and the pump is safety related, so it could be a safeguards common supply.

Technical Reference(s)	SOP-105	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **DESCRIBE** the components of the Reactor Makeup system including interrelations with other systems to include interlocks and control loops IAW SOP-103, -104, and -105. (SYS.CS2.OB03)

Question Source:
- Bank # 52512
- Modified Bank # ______ (Note changes or attach parent)
- New ______

Question History: Last NRC Exam ______

Question Cognitive Level:
- Memory or Fundamental Knowledge X
- Comprehension or Analysis ______

10 CFR Part 55 Content:
- 55.41 7
- 55.43 ______

Comments / Reference: SOP-105	Revision: 11-29-04

11/29/04 12:21 Page 1 of

Revised Procedure: SOP-105-CS-E01, ELECTRICAL LINEUP

Type: OPS, Unit: X, Revision: 0

Step: SOP-105-CS-E01 ==> ELECTRICAL LINEUP

Equip Operator Id:	Equip Description:	Equip Location:	Required Config:	Actual Config:	Verif:	Initials	
1EB1-1/9M/BKR	BORIC ACID TRANSFER PUMP 1-01 MOTOR BREAKER	UNIT 1 TRAIN A SWITCHGEAR ROOM // N. WALL	ON		IV		
1EB4-1/8M/BKR	BORIC ACID TRANSFER PUMP 1-02 MOTOR BREAKER	BORIC ACID TRANSFER PUMP AREA // S. WALL	ON		IV		
2EB1-1/9M/BKR	BORIC ACID TRANSFER PUMP 2-01 MOTOR BREAKER	UNIT 2 TRAIN A SWITCHGEAR ROOM // U2 TRN A ELEC SWGR AREA	ON		IV		
2EB4-1/10M/BKR	BORIC ACID TRANSFER PUMP 2-02 MOTOR BREAKER	AUXILIARY BUILDING 810 CORR // U2 END OF N-S HALLWAY	ON		IV		

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	008.A1.01		
Level of Difficulty: 2	Importance Rating	2.8		

Component Cooling Water: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW flow rate

Question # 23

Given the following conditions:

- Unit 1 at 100%
- RCP 1-01 Thermal Barrier cooler leak
- CCW Surge Tank level 80% increasing slowly
- CCW flow from RCP 1-01 Thermal Barrier is 85 gpm increasing
- CCW Thermal Barrier return temperature is 170ºF increasing

1-HS-4691, RCP 1-01 THBR CLR CCW RET TEMP CTRL VLV, and 1-HV-4709, U1 THBR CLR CCW RET ISOL VLV ORC, are currently __(1)__.

1-HV-4696, THBR CLR CCW RET ISOL VLV IRC, is currently __(2)__.

A. (1) closed
(2) open

B. (1) closed
(2) closed

C. (1) open
(2) open

D. (1) open
(2) closed

Answer: D

K/A Match: K/A match due to requiring knowledge of the flow rate which will cause the CCW valves to automatically close.

Explanation:

A. Incorrect. First part is incorrect, but plausible (see B). Second part is incorrect, but plausible since flow remaining below a high setpoint will not close the valve, but setpoint not recalled.

B. Incorrect. First part is incorrect, but plausible since valves 4691 and 4709 close if temperature exceeds a high setpoint, but setpoint not recalled. Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A).

D. Correct. First part is correct. Valves 4691 and 4709 close if temperature exceeds 182.5°F, so they remain open. Second part is correct. Valve 4696 will close if flow is greater than 64 gpm.

Technical Reference(s)	ALM-0032A	Attached w/ Revision # See Comments / Reference
	ABN-502A	

Proposed references to be provided during examination: ______

Learning Objective: **DISCUSS** the response to Leakage Into the CCW System in accordance with ABN-502, Component Cooling Water System Malfunction. (ABN.501.OB106)

Question Source: Bank # 31261
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference:	Revision:

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0032A
ALARM PROCEDURE 1-ALB-3B	REVISION NO. 7	PAGE 171 OF 189

ANNUNCIATOR NOM./NO.: **ANY RCP THBR CLR CCW RET FLO HI** **4.11**

PROBABLE CAUSE:

Thermal barrier cooler failure

NOTE: When thermal barrier cooler CCW return valve closes due to a thermal barrier tube failure, CCW supply and return line to cooler is subjected to RCS pressure. Thermal barrier cooler CCW relief setpoint of 2485 psig is designed to protect associated piping.

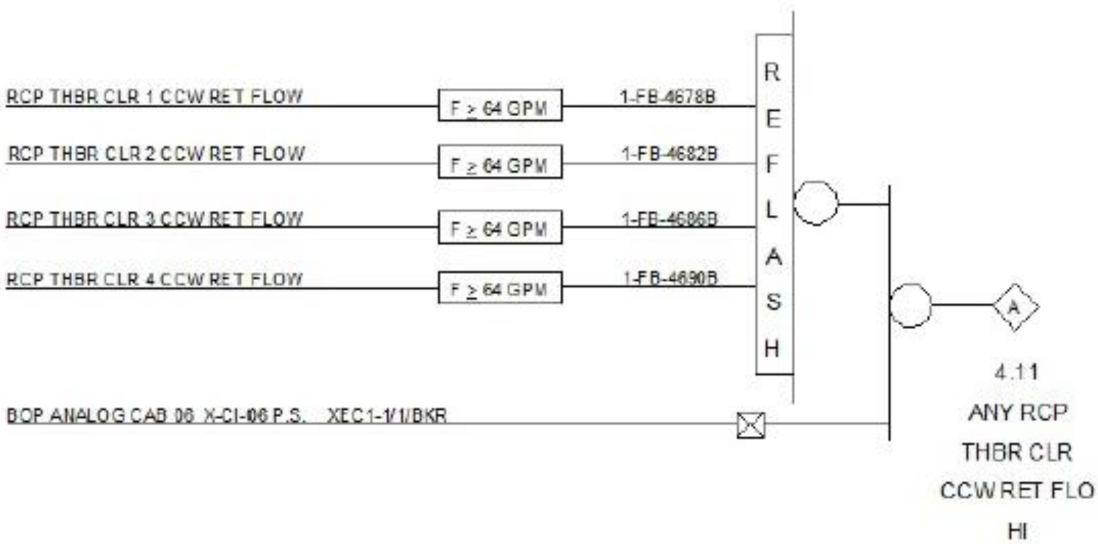
AUTOMATIC ACTIONS:

1-HS-4696, THBR CLR CCW RET ISOL VLV closes.

NOTE: If any thermal barrier cooler CCW outlet temperature reaches 182.5°F, associated cooler CCW return valve and 1-HV-4709, U1 THBR CLR CCW RET ORC ISOL VLV, will close.

OPERATOR ACTIONS:

1. REFER to ABN-101 for Loss of Component Cooling Water to any RCP.
2. REFER to ABN-502 for leakage into CCW System.



Comments / Reference: ABN-502A	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION 11	PAGE 23 OF 75

4.2 Automatic Actions

NOTE: Closure of u-HS-4709 or u-HS-4696 isolates CCW return from ALL RCPs.

a. High thermal barrier cooler CCW return temperature (182.5°F) will cause the following:

1) Auto closure of thermal Barrier Cooler CCW Return Valve for affected pump(s):

- u-HS-4691, RCP 1 THBR CLR CCW RET VLV
- u-HS-4692, RCP 2 THBR CLR CCW RET VLV
- u-HS-4693, RCP 3 THBR CLR CCW RET VLV
- u-HS-4694, RCP 4 THBR CLR CCW RET VLV

2) Auto closure of u-HS-4709, THBR CLR CCW RET ISOL VLV (ORC)

b. High thermal barrier CCW return flow will cause auto closure of u-HS-4696, THBR CLR CCW RET ISOL VLV (IRC).

Section 4.2

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	012.K5.01		
Level of Difficulty: 3	Importance Rating	3.3		

Reactor Protection: Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB

Question # 24

The OT N-16 Reactor Trip provides protection to prevent __(1)__.

Assuming the Reactor Trip setpoint is at its nominal value with the plant at NOP/NOT, if reactor power rises to 112%, the OT N-16 Reactor Trip __(2)__ actuate.

A. (1) exceeding the allowable Heat Generation Rate
(2) would

B. (1) exceeding the allowable Heat Generation Rate
(2) would NOT

C. (1) DNB
(2) would

D. (1) DNB
(2) would NOT

Answer: D

K/A Match: K/A match due to requiring knowledge of DNB reactor protection and setpoint.

Explanation:

A. Incorrect. 1st part is incorrect but plausible (see B). 2nd part is incorrect but plausible (see C).

B. Incorrect. 1st part is incorrect because the Overtemperature N 16 trip prevents DNB from occurring. It is plausible because if it were the Overpower N 16 trip, it would be correct. 2nd part is correct (see D).

C. Incorrect. 1st part is correct (see D). 2nd part incorrect because the nominal setpoint for Overtemperature N 16 trip is 115%. It is plausible because if it were the Overpower N 16 trip it would be correct.

D. Correct. 1st part correct. The Overtemperature N 16 trip prevents DNB from occurring. 2nd part is correct. The nominal setpoint for Overtemperature N 16 trip is 115%.

Technical Reference(s)	TSB 3.3.1	Attached w/ Revision # See Comments / Reference
	COLR	

Proposed references to be provided during examination: ________

Learning Objective: **EXPLAIN** the instrumentation and controls of the Reactor Protection and Engineered Safeguard Actuation Systems and predict the system response in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems and Westinghouse Drawings 7247D05. (SYS.ES1.OB04)

Question Source: Bank # 73903
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 6
55.43 ________

Comments / Reference: TS 3.3.1	Revision: 156

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
5. Source Range Neutron Flux	2[e]	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.4 E5 cps
	3[b], 4[b], 5[b]	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.4 E5 cps
6. Overtemperature N-16	1,2	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1[q][r]
7. Overpower N-16	1,2	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 112.8% RTP [q][r]
8. Pressurizer Pressure					
a. Low	1[g]	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1863.6 psig (Unit 1) ≥ 1865.2 psig (Unit 2)
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2400.8 psig (Unit 1) ≤ 2401.4 psig (Unit 2)

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 2a, 2b, 6, 7, and 14 (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
(q) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(r) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint or a value that is more conservative than the Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint, the methodology used to determine the as-found tolerance and the methodology used to determine the as-left tolerance shall be specified in the Technical Specification Bases.

Comments / Reference: TS 3.3.1	Revision: 156

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature N-16

The Overtemperature N-16 Function Allowable Values shall not exceed the following setpoint by more than 0.5% N-16 span for N-16 input, 0.5% T_{cold} span for T_{cold} input, 0.5% pressure span for pressure input, and 0.5% Δq span for Δq input.

$$Q_{setpoint} = K_1 - K_2\left[\frac{(1+\tau_1 S)}{(1+\tau_2 S)}T_c - T_c^{\circ}\right] + K_3(P - P^1) - f_1(\Delta q)$$

Where:

$Q_{setpoint}$	=	Overtemperature N-16 trip setpoint
K_1	=	*
K_2	=	*/°F
K_3	=	*/psig
T_C	=	Measured cold leg temperature, °F
T_C°	=	Indicated reference T_C at RATED THERMAL POWER, *°F
P	=	Measured pressurizer pressure, psig
P^1	≥	* psig (Nominal RCS operating pressure)
s	=	the Laplace transform operator, sec^{-1}.
τ_1, τ_2	=	Time constants utilized in lead-lag controller for T_C, $\tau_1 \geq$ * sec, and $\tau_2 \leq$ * sec
$f_1(\Delta q)$	=	*{(q_t - q_b) + *%} when (q_t - q_b) ≤ *% RTP 0% when *% RTP < (q_t - q_b) < *% RTP *{(q_t - q_b) - *%} when (q_t - q_b) ≥ *% RTP

* as specified in the COLR

Comments / Reference: U1 COLR	Revision: 0

COLR for CPNPP Unit 1 Cycle 22

2.9 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$) (LCO 3.2.2)

2.9.1 $F^N_{\Delta H} \leq F^{RTP}_{\Delta H} [1 + PF_{\Delta H} (1-P)]$

where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

2.9.2 $F^{RTP}_{\Delta H} =$ 1.60 for all Fuel Assembly Regions

2.9.3 $PF_{\Delta H} =$ 0.3

2.10 AXIAL FLUX DIFFERENCE (AFD) (LCO 3.2.3)

2.10.1 The AFD Acceptable Operation Limits are provided in Figure 9.

2.11 REACTOR TRIP SYSTEM (RTS) INSTRUMENTATION (LCO 3.3.1)

2.11.1 The numerical values pertaining to the Overtemperature N-16 reactor trip setpoint are listed below;

K_1 = 1.15

K_2 = 0.0139 /°F

K_3 = 0.00071 /psig

T_c° = indicated loop specific T_c at Rated Thermal Power, °F

P^1 ≥ 2235 psig

τ_1 ≥ 10 sec

τ_2 ≤ 3 sec

$f_1(\Delta q) = -2.78 \cdot \{(q_t-q_b) + 18\%\}$ when $(q_t-q_b) \leq -18\%$ RTP

$= 0\%$ when -18% RTP $< (q_t-q_b) < +10.0\%$ RTP

$= 2.34 \cdot \{(q_t-q_b) - 10.0\%\}$ when $(q_t-q_b) \geq +10.0\%$ RTP

6 ERX-20-002, Rev. 0

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	2		
	Group	1		
	K/A	026.K4.09		
Level of Difficulty: 2	Importance Rating	3.7		

Containment Spray: Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Prevention of path for escape of radioactivity from containment to the outside (interlock on RWST isolation after swapover)

Question # 25

Given the following conditions:

- Unit 1 100% power in a normal alignment
- CS actuation occurs due to a LOCA

Subsequently:

- The CS system is aligned for containment sump recirculation

Following the CS actuation, but prior to placing the system in containment sump recirculation, the CS Pump Recirculation Valves will be__(1)__.

Following realignment of the system for containment sump recirculation, these valves will __(2)__.

A. (1) open
(2) close

B. (1) open
(2) remain open

C. (1) closed
(2) open

D. (1) closed
(2) remain closed

Answer: D

K/A Match: K/A match due to requiring knowledge of the interlock associated with the CS recirc valves preventing flow from containment to the RWST.

Explanation:

A. Incorrect. First part is incorrect since the valves close upon the opening of the CS HX outlet valves, but plausible since the valves will not open on an SI or CS actuation signal directly. Second part is correct (see D).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since CS signal may be reset prior to going to recirculation, but valve operation is based on position of CS HX outlet or sump recirculation valve position.

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see B).

D. Correct. First part is correct. These recirc valves are normally in automatic and close automatically when the CS HX outlet valve opens on the CS actuation signal. Second part is correct. When the containment spray system is aligned for recirculation the valves will remain closed due to the heat exchanger outlet valves open, as well as the sump suction valves being opened.

Technical Reference(s)	ALM-0022A	Attached w/ Revision # See Comments / Reference
	Containment Spray Lesson Plan	

Proposed references to be provided during examination: ______

Learning Objective: **DESCRIBE** the instrumentation and controls of the Containment Spray System and predict the system response in accordance with FRZ-0.1A/B and EOS-1.3A/B. (SYS.CT1.OB04)

Question Source:
Bank # ______
Modified Bank # 66447 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments / Reference: Bank 66447	Revision:

- Unit 1 at 100% in normal alignment
- A safety injection due to a large steam break

Subsequently:

- A containment spray actuation

Which ONE of the following completes the statements below?

Following the safety injection signal and prior to the containment spray actuation the containment spray pump recirculation valves (1-HV-4772-1, 1-HV-4772-2, 1-HV-4773-1 and 1-HV-4773-2) will be______.

Following receipt of the containment spray actuation these valves will ______.

A. open
close

B. open
remain open

C. closed
open

D. closed
remain closed

Answer: A

Answer Explanation

Comments / Reference: Bank 66447	Revision:

A. Correct. These recirc valves are normally in automatic and therefore normally open. When the safety injection signal occurs the valves will initially be in the open position and will remain open. When the containment spray actuation signal is initiated the valves will go closed as the heat exchanger outlet valves open.

B. Incorrect. First part is correct Second part is incorrect but plausible if the applicant believes with the valve in automatic (normal lineup) the valves will wait for flow to rise to greater than 1090 gpm.

C. Incorrect. First part is incorrect, but plausible if the applicant believes the valve is normally closed. See A above. Second part is correct See A above.

D. Incorrect. Both parts are incorrect. See A and B above.

Question 383 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	66447
User-Defined ID:	ILOT1915
Cross Reference Number:	SYS.CT1.OB05.012
Topic:	Unit 1 at 100% in normal alignment A safety injection due to a large steam break Subsequently: A
K/A:	
Question Reference:	
SRO:	
Comments:	REF: SYS.CT1 013K1.05 Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: CSS K/A Match: This question matches the KA by testing the knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: CSS

Comments / Reference: ALM-0022A	Revision: 10

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0022A
ALARM PROCEDURE 1-ALB-2B	REVISION NO. 10	PAGE 17 OF 119

ANNUNCIATOR NOM./NO.: **CSP 1/3 RECIRC VLV FAIL TO CLOSE** **1.5**

PROBABLE CAUSE:

1-FV-4772-1, CS PMP 1-01 RECIRC VLV malfunction
1-FV-4772-2, CS PMP 1-03 RECIRC VLV malfunction
1-HV-4782, CNTMT SMP TO CS PMP 1-01/1-03 SUCT ISOL VLV inadvertently opened.

AUTOMATIC ACTIONS: None

> NOTE: 1-FV-4772-1 and 1-FV-4772-2, CS PMP 1-01/1-03 RECIRC VLV will close when 1-HV-4776, CS HX 1-01 OUT VLV or 1-HV-4782, CNTMT SMP TO CS PMP 1-01/1-03 SUCT ISOL VLV is open. When 1-HV-4782 opens, this alarm will be momentarily actuated until 1-FV-4772-1 and 1-FV-4772-2 reach their closed limit switch.

OPERATOR ACTIONS:

1. MONITOR Containment Spray Pump discharge flow.
   - 1-FI-4772-1, CSP 1 DISCH FLO
   - 1-FI-4772-2, CSP 3 DISCH FLO

   A. IF flow is <1100 GPM, THEN STOP affected pump.
      - 1-HS-4764, CSP 1
      - 1-HS-4765, CSP 3
2. DETERMINE if 1-HS-4782, CNTMT SMP TO CSP 1 & 3 SUCT ISOL VLV is open.

   A. IF valve was opened inadvertently, THEN RECLOSE valve.
3. IF 1-HS-4782, CNTMT SMP TO CSP 1 & 3 SUCT ISOL VLV is open,
   THEN
   ENSURE recirc valves are CLOSED.
   - 1-HS-4772-1, CSP 1 RECIRC VLV
   - 1-HS-4772-2, CSP 3 RECIRC VLV

   A. IF either valve can NOT be closed, THEN STOP affected CSP.
      - 1-HS-4764, CSP 1
      - 1-HS-4765, CSP 3

   B. DISPATCH an NEO to Train A ECCS Valve Room to manually close affected valve(s).
4. REFER to TS 3.6.6.
5. NOTIFY Chemistry to sample RWST for possible increased activity.
6. CORRECT the condition OR INITIATE a work request per STA-606.

Comments / Reference: CS Lesson Plan	Revision: 00-0001

NOTES	LESSON OUTLINE

o. Both room coolers receive a start signal when either pump breaker closes

8. Containment Spray Pump Recirculation Valves

   a. Each pump is provided with a normally open Recirc valve

      1) FV-4772-1, FV-4772-2, FV-4773-1 and FV-4773-2

   b. The recirculation lines are downstream of the pumps and upstream of the heat exchangers

   c. They feed into a common return line to the RWST

   d. Motor operated supplied from train related 480 vac bus

   e. Fail as-is on loss of power

   f. Controlled from a three position (CLOSE-AUTO-OPEN) spring return to center hand switch with open and closed indication located on CB-02

   g. Closed indication on MLB-4A3/4B3

   h. **AUTO CLOSE**

      1) When either the heat exchanger outlet valve or the containment recirc. sump suction valve begins to open

   i. **AUTO Open and Close**

      1) AUTO Open - when CS discharge flow drops below approximately 1090 gpm

      2) AUTO Close - when flow increases above approximately 120 gpm above 1090 gpm

      3) If the valve is not full open with a low flow signal present an alarm in annunciated on ALB-2B

      4) Interlocked with the heat exchanger outlet valve to ensure full flow is available from the pump

         a) The recirculation valve goes closed once the outlet valve begins to open

         b) Also interlocked with the recirculation sump suction valve, receiving a close signal once the sump suction valve begins to open

      5) The recirculation flowpath is utilized to add pump heat to the RWST during cold weather when necessary to maintain tank temperature above the minimum temperature

9. Containment Spray Heat Exchangers

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	039.K3.03		
Level of Difficulty: 4	Importance Rating	3.2		

Main and Reheat Steam: Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: AFW pumps

Question # 26

Given the following conditions:

- Unit 2 90% power
- 2-HS-2452-1, AFWPT STM SPLY VLV - MSL 4 fails open
- TDAFWP speed is increasing
- Crew has entered ABN-305, Auxiliary Feedwater System Malfunction, for an Inadvertent Turbine Driven AFW Pump Start

Based on plant conditions, turbine load __(1)__ required to be reduced by 50 MW.

To stop the TDAFWP, ABN-305 INITIALLY directs __(2)__ .

A. (1) is
(2) tripping the TDAFW pump

B. (1) is
(2) closing 2-HS-2452-1 by placing in PULLOUT

C. (1) is NOT
(2) tripping the TDAFW pump

D. (1) is NOT
(2) closing 2-HS-2452-1 by placing in PULLOUT

Answer: D

K/A Match: K/A match due to requiring knowledge of the requirements regarding the operation of the TDAFWP following a failure of the MS supply valves. Explanation:
A. Incorrect. First part is incorrect, but plausible. With the unit at 90% power, a power reduction is not required. If the unit was at 100%, a power reduction would be required. Second part is incorrect, but plausible because the operators are directed to trip the TDAFWP at Step 4 if the Steam Supply valve failed to close B. Incorrect. First part is incorrect, but plausible (see A). The second part is correct (see D). C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A). D. Correct. First part is correct. With the unit at 90%, a 50 MW load reduction is not required to maintain power below 100%. Second part is correct. Operators are directed to close the steam supply valves for any steam admission valve that would not close at Step 1 of the ABN.

Technical Reference(s)	ABN-305	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to Inadvertent Turbine Driven AFW Pump Start in accordance with ABN-305, Auxiliary Feedwater System Malfunction. (ABN.305.OB06)

Question Source: Bank # ______
Modified Bank # 72044 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: Bank 72044	Revision:

Unit 2 conditions:

- Unit is at 90% power
- DG 2-02 is out of service for repair
- 2-HS-2452-1, AFWPT STM SPLY VLV - MSL 4 indicates open
- 2-HS-2452-2, AFWPT STM SPLY VLV - MSL 1 indicates open
- TDAFW speed in increasing
- Operators have entered ABN-305, (AUXILIARY FEEDWATER SYSTEM MALFUNCTION), Section 6.0, (INADVERTENT TURBINE DRIVEN AFW PUMP START (STEAM SUPPLY VLV FAILS OPEN)
- Operators attempt to close the AFWPT STM SPLY Valves, only 2-HS-2452-1 indicates closed.

Which ONE of the following completes the statements below?

Based on plant conditions, operators ___**(1)**___required to reduce turbine load by 50 MW. To stop the TDAFW pump, ABN-305 directs the operators to ___**(2)**___next.

Note:
2-MS-101= MSL 2-01 TO AFWPT SPLY VLV UPSTRM ISOL VLV

	(1)	(2)
A.	are	trip the TDAFW pump
B.	are	locally close 2-MS-0101
C.	are **NOT**	trip the TDAFW pump
D.	are **NOT**	locally close 2-MS-0101

Answer: D

Answer Explanation

Comments / Reference: Bank 72044	Revision:

A. Incorrect. First part is incorrect. With the unit at 90% power, a power reduction is not required. If the unit was at 100%, a power reduction would be required. Second part is incorrect. With DG 2-02 out of service, the operators are directed to close the local manual isolation valve, and skip the step that would trip the TDAFW pump.

B. Incorrect. First part is incorrect. See A above. The second part is correct. With DG 2-02 out of service, it is not desirable to trip the TDAFW pump.

C. Incorrect. First part is correct. See A above. Second part is incorrect. See A and B above

D. Correct. With the unit at 90%, a 50 MW load reduction is not required. Operators are directed to close the steam supply valves for any steam admission valve that would not close.

Question 178 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	72044
User-Defined ID:	ILOT9516
Cross Reference Number:	SYS.AF1.OB05.009
Topic:	Unit 2 conditions: Unit is at 90% power DG 2-02 is out of service for repair 2-HS-2452-1, AFWPT
K/A:	061 A2.04
Question Reference:	
SRO:	
Comments:	

Comments / Reference: ABN-305	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-305
AUXILIARY FEEDWATER SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 76 OF 94

6.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: If the Turbine Driven AFW Pump Steam Supply Valve(s) (u-HS-2452-2 or u-HS-2452-1) are open due to a BOS actuation, the actions of ABN-601 are applicable for addressing the open steam supply valve(s).

1 CLOSE affected steam supply valve by placing handswitch in - PULL OUT	CONTINUE with Step 2.
☐ • u-HS-2452-2, AFWPT STM SPLY VLV - MSL1	
☐ • u-HS-2452-1, AFWPT STM SPLY VLV - MSL4	
▫ IF affected steam supply valve is CLOSED, THEN GO TO Step 5.	

CAUTION: A loss of efficiency due to steam supply to the TD AFWP, and flow initiation to the SGs could cause Rx Power to exceed 100% (if at or near 100% RTP).

NOTE: Step 2 is a continuous action step.

☐ 2 VERIFY Reactor Power less than or equal to 100%.	PERFORM the following: • ENSURE 1/u-RBSS, CONTROL ROD BANK SELECT in AUTO. • INITIATE a 50 MW Turbine Load reduction.

Section 6.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	1		
	K/A	061.K6.02		
Level of Difficulty: 2	Importance Rating	2.6		

Auxiliary/Emergency Feedwater: Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps

Question # 27

Given the following conditions:

- Reactor failed to trip when required
- A transition has been made to FRS-0.1A, Response to Nuclear Power Generation/ATWT
- MDAFWP 1-02 trips shortly after starting
- All SG NR levels are off-scale LOW

Placing 1-HS-2451A, MD AFWP 2 in the STOP or PULL-OUT position will reset the __(1)__ relay and may result in an automatic restart if the handswitch is returned to AUTO.

Per FRS-0.1A, the MINIMUM total AFW flow to be established is greater than __(2)__.

A. (1) 86M lockout
(2) 460 gpm

B. (1) 50/51 overcurrent
(2) 460 gpm

C. (1) 86M lockout
(2) 860 gpm

D. (1) 50/51 overcurrent
(2) 860 gpm

Answer: C

K/A Match:	K/A match due to requiring knowledge of the AFW pump operation.
Explanation:	

A. Incorrect. 1st part is correct (see C). 2nd part is incorrect, but plausible (see B).

B. Incorrect. 1st part is incorrect, but plausible (see D). 2nd part is incorrect because per FRS-0.1A, you are to establish > 860 gpm. It is plausible because if you were in EOP-0.0A, it would be correct

C. Correct. 1st part is correct, per ABN-305 if the handswitch is taken to STOP or PULL-OUT and then returned to the AUTO position the 86M relay will reset and could result in an automatic pump re-start if there are no other dropped relays on the breaker. 2nd part is correct (see D).

D. Incorrect. 1st part is incorrect but plausible because this pump does have a 50/51 overcurrent relay, however, in order to reset this relay it must be performed locally at the breaker. 2nd part is correct. FRS-0.1A directs verify that greater than 860 gpm AFW flow exists at all times until SG NR levels have reached 43% (50% adverse containment).

Technical Reference(s)	ABN-305	ing
	FRS-0.1A	

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to Motor Driven Auxiliary Feedwater Pump Malfunction in accordance with ABN-305, Auxiliary Feedwater System Malfunction. (ABN.305.OB03)

Question Source:
Bank # ______
Modified Bank # 81934 (Note changes or attach parent)
New ______

Question History: Last NRC Exam LC26 (Original Question)

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments / Reference: Bank 81934	Revision:

Unit 1 plant conditions:

- Reactor tripped
- A complete MFW Isolation has occurred
- TDAFWP is tagged out for maintenance
- MDAFWP 1-02 trips shortly after starting

Based on plant conditions, complete the following statements.

1) Per ABN-305, Auxiliary Feedwater System Malfunction, a "CAUTION" states, placing 1 HS-2451A, MD AFWP 2 in the STOP or PULL-OUT position will reset the ____(1)____ relay and may result in an automatic restart if the handswitch is returned to AUTO.

2) Per EOP-0.0A, Reactor Trip or Safety Injection, the MINIMUM total AFW flow to be established is ____(2)____.

A. (1) 50/51 overcurrent
(2) > 460 gpm

B. (1) 86M lockout
(2) > 460 gpm

C. (1) 50/51 overcurrent
(2) > 860 gpm

D. (1) 86M lockout
(2) > 860 gpm

Answer: B

Answer Explanation

Comments / Reference: Bank 81934	Revision:

Explanation:
Incorrect: 1st part is incorrect but plausible because this pump does have a 50/51 overcurrent relay, however, in order to reset this relay it must be performed locally at the breaker. 2nd part is correct. EOP-0.0A directs you to verify that greater than 460 gpm AFW flow exists.
Correct: 1st part is correct, per ABN-305 if the handswitch is taken to STOP or PULL-OUT and then returned to the AUTO position the 86M relay will reset and could result in an automatic pump re-start if there are no other dropped relays on the breaker. 2nd part is correct (see A).
Incorrect: 1st part is incorrect but plausible (see A). 2nd part is incorrect because per EOP-0.0A, you are to establish > 460 gpm. It is plausible because if you were in FRS-0.1A, it could be correct.
Incorrect: 1st part is correct, see B above. 2nd part is incorrect but plausible (see C).

Question 315 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	81934
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	Unit 1 plant conditions: Reactor tripped A complete MFW Isolation has occurred TDAFWP is tagged ou
K/A:	061.K6.02
Question Reference:	
SRO:	
Comments:	KA Match: This question matches the KA by requiring knowledge of the AFW pump operation.

Comments / Reference: ABN-305	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-305
AUXILIARY FEEDWATER SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 47 OF 94

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Placing the pump handswitch in STOP OR PULL-OUT with the pump tripped (white TRIP light) will reset the 86M relay (white TRIP light) and may result in an automatic restart if the handswitch is returned to AUTO.

☐ 1 DETERMINE which MD AFW Pump is malfunctioning AND verify affected pump - TRIPPED.	TRIP affected pump AND PLACE handswitch in PULL-OUT: • U-HS-2450A, MD AFWP 1 • U-HS-2451A, MD AFWP 2

CAUTION: Do not exceed 800 gpm total flow on one Motor Driven Auxiliary Feedwater Pump.

☐ 2 VERIFY at least one AFW pump RUNNING	START any available AFW pump per SOP-304A/B.

CAUTION: Do NOT operate both Motor-Driven Auxiliary Feedwater Pumps at the same time with the trains cross-connected.

☐ 3 VERIFY Steam Generator levels - NORMAL	IF the TD AFW Pump in available, THEN START the TD AFW Pump AND FEED the two steam generators NOT being supplied by the MD AFW Pump. If the TD AFW Pump is NOT available, THEN CROSS CONNECT AFW trains per Attachment 2 or 3 as appropriate.

Section 3.3

Comments / Reference: FRS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRS-0.1A
RESPONSE TO NUCLEAR POWER GENERATION/ATWT	REVISION NO. 9	PAGE 3 OF 33

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Verify Reactor Trip: • Reactor trip breakers - AT LEAST ONE OPEN -AND- • Neutron flux - DECREASING -AND- • All control rod position rod bottom lights - ON	Ensure manual reactor trip attempted. • IF reactor NOT tripped, THEN ensure control rods inserting at rate greater than or equal to 48 steps per minute.
2	Verify Turbine Trip: • All HP turbine stop valves - CLOSED	Manually trip turbine. IF turbine will NOT trip, THEN pull-out all EHC fluid pumps. IF turbine still NOT tripped, THEN close or verify closed main steamline isolation valves.
3	Verify Total AFW Flow - GREATER THAN 860 GPM	Manually start pump(s) and align valves as necessary.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	2		
	Group	1		
	K/A	064.A2.03		
Level of Difficulty: 2	Importance Rating	3.1		

Emergency Diesel Generator: 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: Parallel operation of ED/Gs	
Question # 28	

Given the following conditions:

- Operators are in the process of conducting OPT-214A, Diesel Generator Testing
- They have started and loaded EDG 1-01 in accordance with the loading schedule
- After reaching the full load value required for the test, an SI occurs due to a technician error on an SSPS OPT

Which of the following describes the expected response of EDG 1-01 output breaker?

A. The output breaker will open and immediately reclose due to the safeguards sequencer

B. The output breaker will open and remain open with the EDG running unloaded

C. The output breaker will remain closed until opened by the operator per the SI recovery procedures

D. The output breaker will remain closed due to the control switch alignment of the diesel for the OPT

Answer: B	

K/A Match: K/A match due to requiring knowledge of events occurring while the EDG is paralleled with offsite power. Explanation:
A. Incorrect. Plausible since the breaker will immediately open, but it will remain open unless needed to power the bus. B. Correct. If the EDG is paralleled with offsite when an SI occurs, the EDG breaker opens and remains open unless needed to carry the bus. C. Incorrect. Plausible since it may be incorrectly assumed that it is desirable to keep the safeguards bus energized by the EDG. D. Incorrect. Plausible since it may be incorrectly assumed that it is desirable to keep the safeguards bus energized by the EDG.

Technical Reference(s)	ABN-602	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **EXPLAIN** the instrumentation and controls of the Emergency Diesel Generator system including the system response in accordance with DBD-ME-011. (SYS.ED1.OB22)

Question Source: Bank # 19341
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments / Reference: OPT-214A	Revision: 25

CPNPP OPERATIONS TESTING MANUAL	UNIT 1	PROCEDURE NO. OPT-214A
DIESEL GENERATOR OPERABILITY TEST	REVISION NO. 25 CONTINUOUS USE	PAGE 38 OF 145

8.1

CAUTION:
- DO NOT EXCEED 6.0 MW until the DG load is 6.0 MW AND the DG has been running 45 minutes OR load has been maintained at 6.0 MW for at least 15 minutes.
- Grid induced load swings may cause DG load to exceed 6.0 MW prior to meeting the necessary run-time. IF this occurs, THEN DG load should be promptly adjusted back to 6.0 MW.

NOTE:
- IF an SI occurs while paralleled to OFFSITE, THEN the DG Output Breaker will OPEN AND the DG will continue to run.
- IF conditions hazardous to personnel OR equipment develop, THEN the DG can be immediately SHUTDOWN by placing the Emergency STOP/START Switch in PULLOUT. This does NOT require that the output breaker be opened first.

☐ Y. LOAD the DG to 6.0 MW over the next 20 minutes using 65-1EG1, DG 1 SPD CTRL, unless otherwise directed by the Shift Manager.

NOTE: DG load should be maintained as close to 6.4 MW as practical to ensure consistent data is taken for each DG run.

Z. WHEN DG load has been stabilized at 6.0 MW for 15 minutes OR load is at 6.0W AND DG has been running ≥45 minutes,
THEN:

☐ 1) RAISE load to 6.4 MW (6.3 to 7.0 MW).

☐ 2) RECORD time rated load is reached.

☐ 3) NOTIFY Prompt Team AND Chemistry that the DG is at full load.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	2		
	K/A	002.K6.07		
Level of Difficulty: 2	Importance Rating	2.5		

Reactor Coolant: Knowledge of the effect or a loss or malfunction on the following RCS components: Pumps

Question # 29

A loss of RCP ______ results in the greatest loss of RCS pressure control.

A. 2-01

B. 2-02

C. 2-03

D. 2-04

Answer: D

K/A Match: K/A match due to requiring knowledge of the effect of an RCP trip on RCS pressure control.

Explanation:

A. Incorrect. Plausible because RCP 2-01 is connected to an RCS Spray valve, however, it is not as effective as RCP 2-04 as it is also connected to the PRZR surge line.

B. Incorrect. Plausible because it will provide some spray flow with RCP 2-02 and RCP 2-03 running using either spray valve but not as effective as RCP 2-01 or RCP 2-04.

C. Incorrect. Plausible because it will provide some spray flow with RCP 2-02 and RCP 2-03 running using either spray valve but not as effective as RCP 2-01 or RCP 2-04.

D. Correct. RCP 2-04 is the most effective at providing spray flow and thus RCS pressure control as it is directly connected to a spray valve and the PRZR Surge Line.

Technical Reference(s)	EOS-0.2B	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **DISCUSS** the ERG background for performing Natural Circulation cooldown per EOS-0.2 Natural Circulation cooldown (ERG.E02.OB01)

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

Question History: Last NRC Exam ____________________

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ____7____
55.43 ________

Comments / Reference: EOS-0.2B	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-0.2B
NATURAL CIRCULATION COOLDOWN	REVISION NO. 9	PAGE 3 OF 60

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: If SI actuation occurs during this procedure, EOP-0.0B, REACTOR TRIP OR SAFETY INJECTION, shall be performed.

CAUTION: If RCP seal cooling had previously been lost, the affected RCP(s) should not be started prior to a status evaluation.

NOTE: RCPs should be run in order of priority to provide normal PRZR spray (RCP 4, 1 then 2 or 3).

NOTE: If conditions can be established for starting an RCP during this procedure, Step 1 should be repeated.

Comments / Reference: EOS-0.2B	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-0.2B
NATURAL CIRCULATION COOLDOWN	REVISION NO. 9	PAGE 35 OF 60

ATTACHMENT 6
PAGE 1 OF 26

BASES

CAUTION: When SI actuates, plant conditions exist which require actions not covered in this procedure. Therefore, a transition to EOP-0.0B REACTOR TRIP OR SAFETY INJECTION is made.

CAUTION: The potential for degradation in RCP seal performance and seal life increases with increasing temperature above 235°F. Hence, if RCP seal cooling is lost for a significant period of time, seal or bearing damage may occur. The potential non-uniform sealing surfaces and seal crud blockage that may exist prior to RCP start can aggravate bearing and seal damage if the RCP is started. Following restoration of seal cooling, the RCP should not be started prior to a complete RCP status evaluation in order to minimize potential RCP damage on restart.

IF RCP seal cooling is lost for only a few minutes, the inventory of cold water in the seal area should prevent excessive seal heat up. For longer periods of time, seal and bearing temperatures may increase greater than 235°F. If excessive temperatures develop, the affected RCP should not be restarted prior to a complete RCP evaluation.

RCPs should not be started prior to a status evaluation unless an extreme (red) or severe (orange) CSF challenge is diagnosed. Under such a CSF challenge the "rules of usage" apply and an RCP should be started if so instructed in the associated FRG. Under a CSF challenge, potential RCP damage is an acceptable consequence if RCP start is required to address a CSF challenge (e.g., to mitigate an inadequate core cooling condition). This is consistent with the intent of these FRGs which attempt to first establish support conditions to start an RCP, but then start the RCP whether or not the support conditions are established.

NOTE: There are PRZR connections to one RCS hot leg via the surge line and to two RCS cold legs via the spray lines. Single pump operation in the loop that provides the best spray is preferred to obtain normal PRZR spray capability. The loop that provides the most effective spray is loop 4 with connections to the PRZR via both a spray line and a surge line. Experience has demonstrated that when RCP 4 is not available, RCP 1 with a connection to the PRZR via a spray line only and at least one additional RCP should remain in operation to provide adequate driving head for spray flow.

Comments / Reference: EOS-0.2B	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-0.2B
NATURAL CIRCULATION COOLDOWN	REVISION NO. 9	PAGE 36 OF 60

ATTACHMENT 6
PAGE 2 OF 26

BASES

Analysis performed for RCP operation and spray flow results provides the following conclusions (Reference RCP TRIP/RESTART in the Generic Issues section of the Executive Volume) (See Table 1). Spray flow with any combination of RCPs operating will be more effective with a high PRZR level. Additionally, operating experience has shown that RCP vibration may be higher than normal when only one RCP is running, and that vibration is reduced when a second RCP is started.

TABLE 1

RCP(s) Running	Is Spray Flow Produced?	
	Spray Valve Loop 1 OPEN	Spray Valve Loop 4 OPEN
4	YES	YES
1	YES (1)	NO
1 AND 2 AND 3	YES	YES
1 AND 2	YES	MAYBE (1)
1 AND 3	YES	MAYBE (1)
2 AND 3	MAYBE (1)	MAYBE (1)

(1) Small amount of spray flow is produced when PRZR level is high (e.g., 90%)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	2		
	K/A	011.K1.04		
Level of Difficulty: 2	Importance Rating	3.8		

Pressurizer Level Control: Knowledge of the physical connections and/or cause-effect relationships between the PZR LCS and the following systems: RPS

Question # 30

Unit 1 is in MODE 3 with rods referenced in preparation for a Reactor Startup.

Which of the following failures will enable a Reactor Trip on PRZR High Water Level if level reaches the trip setpoint?

A. Failure low of N-43, Power Range Channel

B. Failure high of N-43, Power Range Channel

C. Failure low of PT-505, First Stage Turbine Pressure

D. Failure high of PT-505, First Stage Turbine Pressure

Answer: D

K/A Match: K/A match due to requiring knowledge of the P-7 interlock associated with pressurizer high water level reactor trip.

Explanation:

A. Incorrect. Plausible since PR NIS input to P-10 which provides P-7, but the coincidence for P-10 is 2/4 above 10% power, although it may be confused because the associated PCIP window comes on when power is below 10%.

B. Incorrect. Plausible since PR NIS input to P-10 which provides P-7, but the coincidence for P-10 is 2/4 above 10% power.

C. Incorrect. Plausible since either first stage turbine pressure will give P-13 and P-7, but the failure low will not cause the turbine to be at power, although it may be confused because the associated PCIP window comes on when power is below 10%.

D. Correct. Pressurizer high level trip is enabled above P-7 which receives an input from P-10 or P-13. Either first stage turbine pressure above 10% turbine load provides the at power condition for the turbine, providing P-13, and thus P-7, enabling the high level trip.

Technical Reference(s)	ALM-0065A	Attached w/ Revision # See Comments / Reference
	Reactor Protection/ESFAS Study Guide	

Proposed references to be provided during examination: ______

Learning Objective: **EXPLAIN** the instrumentation and controls of the Reactor Protection and Engineered Safeguard Actuation Systems and predict the system response in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems and Westinghouse Drawings 7247D05. (SYS.ES1.OB04)

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 X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ALM-0065A	Revision: 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 49 OF 73

ANNUNCIATOR NOM./NO.: RX & TURB ≤ 10% PWR P-7 3.5

PROBABLE CAUSE:

Reactor shutdown

NOTE: This window is normally illuminated when reactor and turbine power is < 10%.

AUTOMATIC ACTIONS:

The following reactor trips are blocked:

- RCP undervoltage
- RCP underfrequency
- RCS low flow
- Low pressurizer pressure
- High pressurizer level

OPERATOR ACTIONS:

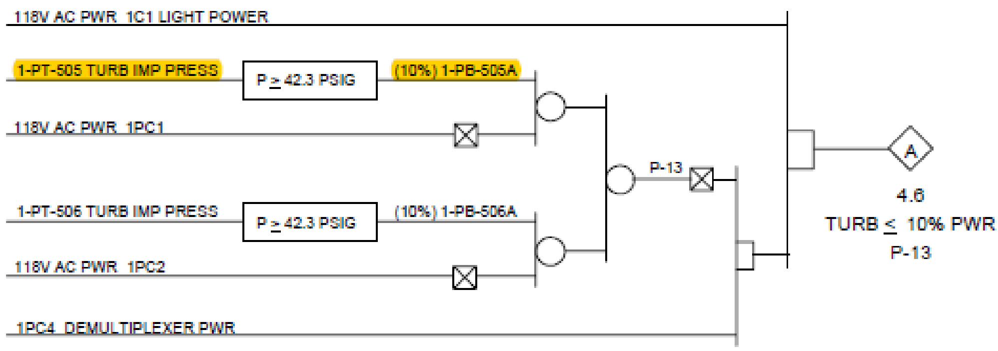
None

Comments / Reference: ALM-0065A	Revision: 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 68 OF 73

ANNUNCIATOR NO.: 4.6

LOGIC:



PLANT COMPUTER:

P0398A TURB IMP PRESS CHAN I

P0399A TURB IMP PRESS CHAN II

LOCAL INSTRUMENTS:

None

REFERENCES:

7247D05 Sh. 04,16

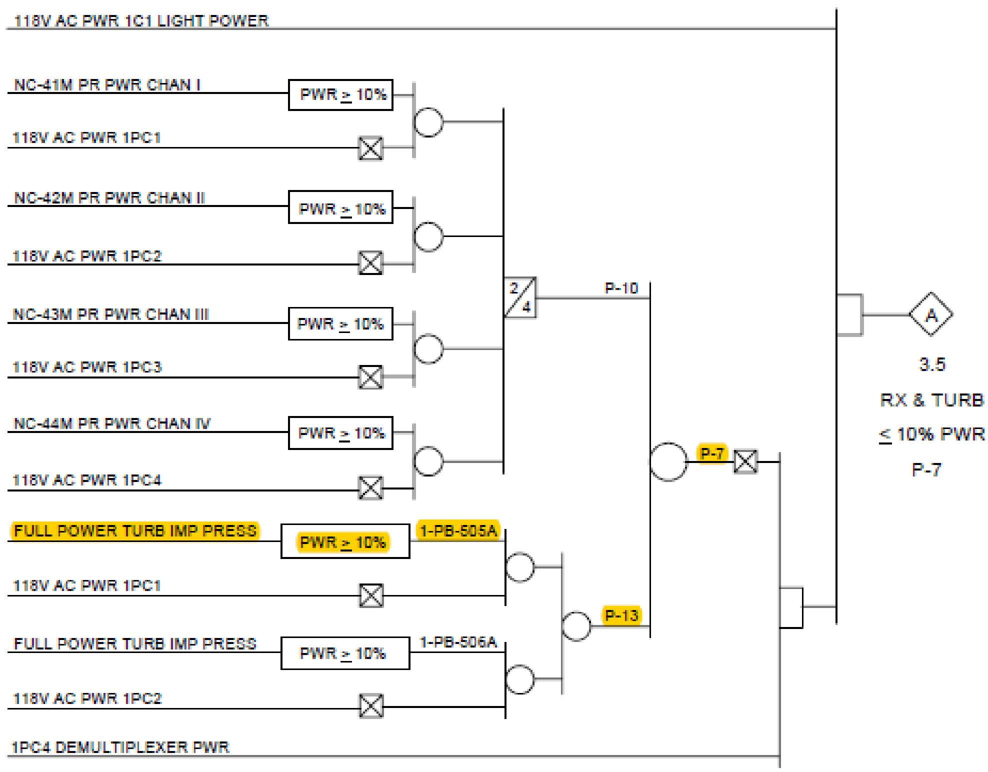
8760D60 Sh. 70

Comments / Reference: ALM-0065A	Revision: 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 48 OF 73

ANNUNCIATOR NO.: 3.5

LOGIC:



PLANT COMPUTER:

Y0001D	TURB 10% PWR (P-13) CHAN I	N0012D	RX 10% PWR (P-10) CHAN II
Y0002D	TURB 10% PWR (P-13) CHAN II	N0013D	RX 10% PWR (P-10) CHAN III
N0011D	RX 10% PWR (P-10) CHAN I	N0014D	RX 10% PWR (P-10) CHAN IV

LOCAL INSTRUMENTS:

None

REFERENCES:

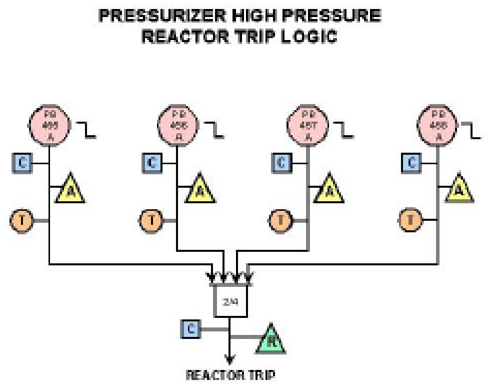
7247D05 Sh. 04,05,06,16 8760D60 Sh. 69

Comments / Reference: Reactor Protection/ESFAS Study Guide	Revision: 5-4-2011

Reactor Protection and ESFAS

Pressurizer High Pressure Trip

Set at 2385 PSIG on 2 out of 4 channels, this trip provides protection against over pressurizing the Reactor Coolant System (**Figure 15**). There are no interlocks associated with this trip.



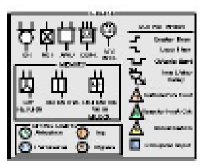


Figure 15 - Pressurizer High Pressure Reactor Trip Logic

Pressurizer High Level Trip

The Pressurizer High Level Trip is provided as a backup to the Pressurizer High Pressure Trip and to prevent passing water through the pressurizer safety valves which would damage these valves and their discharge piping. The trip setpoint is at 92% level on 2 out of 3 channels and is bypassed below **P**-7. The trip is bypassed below **P**-7 because the transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions (**Figure 16**).

Comments / Reference: Reactor Protection/ESFAS Study Guide	Revision: 5-4-2011

Reactor Protection and ESFAS

SOURCE RANGE REACTOR TRIP BLOCK PERMISSIVE, P-6

P-6 is generated from 1 out of 2 Intermediate Range channels > 10^{-10} amps. This allows us to intentionally block the Source Range Reactor trip during startup. In order to block this trip, two block switches must be placed in the "blocked" position which will also de-energize the Source Range Detectors. Train A of SSPS must be blocked for N-31 to de-energize and Train B of SSPS must be blocked to de-energize N-32.

On a shutdown when power is ~5 X 10^{-11} amps on 2 of 2 channels, **P-6** is automatically removed. The removal of **P-6** will automatically re-energize the Source Range as well as reinstate the SR trip and Flux Doubling Boron Dilution Protection.

AT POWER PERMISSIVE, P-7

The Nuclear At Power Permissive (**P-10**) or Turbine At Power Permissive (**P-13**) will generate the **P-7** permissive. Above **P-7** the following Reactor trips are automatically unblocked:

- Pressurizer Low Pressure
- Pressurizer Hi Level
- Reactor Coolant Pump Under Voltage
- Reactor Coolant Pump Underfrequency
- Low Flow in 2 Reactor Coolant Loops

Below the **P-7** setpoint, the above Reactor trips are automatically blocked. These trips are only required when operating above the **P-7** setpoint.

3-LOOP FLOW PERMISSIVE, P-8

2 out of 4 Power Range channels > 48% power generates **P-8** which allows for automatic tripping of the Reactor when a low flow condition is sensed in any Reactor Coolant Loop. Above the **P-8** setpoint, loss of flow in any one RCS loop might result in DNB conditions so the reactor must be tripped. Below the **P-8** setpoint, loss of flow in only one RCS loop will not result in DNB conditions. Therefore, this permissive allows the orderly shutdown of the Reactor if flow is lost in a single loop when power is below **P-8**.

TURBINE TRIP PERMISSIVE, P-9

2 out of 4 Power Range channels greater than 50% power automatically arms the turbine trip/reactor trip. When 3 of 4 channels are less than 50% the Rx trip on the turbine trip is blocked. Above the **P-9** setpoint, a turbine trip will cause a load rejection beyond the design capacities of the Steam Dump and Rod Control Systems. Therefore, a reactor trip is automatically initiated on a turbine trip when it is above the **P-9** setpoint, to minimize the transient on the reactor.

Comments / Reference: Reactor Protection/ESFAS Study Guide	Revision: 5-4-2011

Reactor Protection and ESFAS

Also, when pressure increases above 1960 PSIG, the low Pressurizer Pressure and low Steam Line Pressure Safety Injection signals are automatically unblocked. When the low Steam Line Pressure Safety Injection signal is unblocked, the Steam Line Isolation for Negative Steam Line Pressure High Rate is automatically blocked.

Low-Low Tavg Permissive, P-12

The **P-12** Permissive is used to automatically close all steam dump valves if the RCS temperature is below the low-low Tavg setpoint (553 F on 2/4 channels). The 3 cooldown valves of the Steam Dump System may be bypassed to allow intentional plant cooldown.

The **P-12** circuitry is not assumed to work in all accident analyses (for example, it is not assumed to work if a steam generator safety or relief valve is stuck open, in FSAR section 15.1.4.3), but it has been included in the drawing in the FSAR showing ESF features in FSAR figure 7.3-4. It is not in T.S.

Turbine At Power Permissive, P-13

P-13 is generated from turbine impulse pressure indicating a Turbine power which is equivalent to > 10% RTP, on 1 out of 2 channels (**Figure 20**). **P-13** feeds the At Power Permissive **P-7**. The figure below also shows some of the Control Interlocks (C's) which will be explained in the next section.

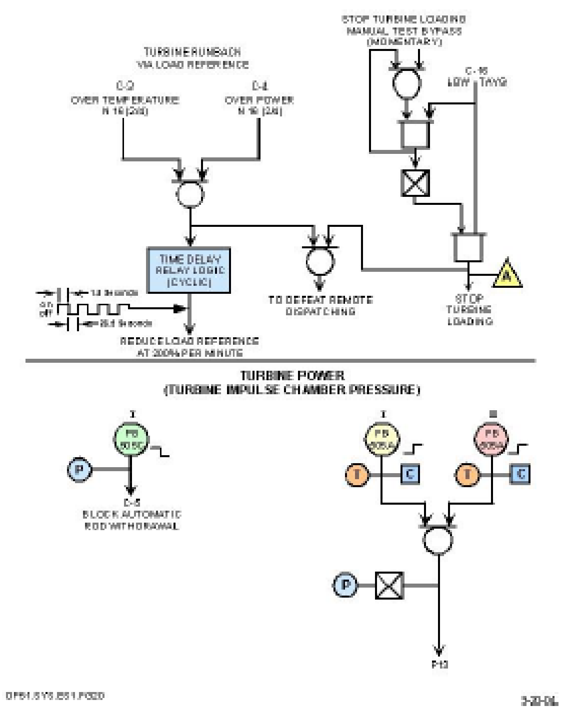


Figure 20 - Miscellaneous Turbine Signals

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Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	2		
	Group	2		
	K/A	015.K5.04		
Level of Difficulty: 2	Importance Rating	2.6		

Nuclear Instrumentation: Knowledge of the operational implications of the following concepts as they apply to the NIS: Factors affecting accuracy and reliability of calorimetric calibrations

Question # 31

During the performance of OPT-309, Unit Calorimetric, if the LEFM is NOT available to be used, the LEFM Main Display will have the "√" symbol in __(1)__.

If FW temperature points utilized for the calorimetric are reading 10ºF LOWER than actual FW temperature and PR NI adjustments are performed per the OPT, INDICATED power will be __(2)__ ACTUAL power.

A. (1) yellow
(2) greater than

B. (1) yellow
(2) less than

C. (1) red
(2) greater than

D. (1) red
(2) less than

Answer: C

K/A Match: K/A match due to requiring knowledge of how an inaccuracy in the calorimetric calibration affects the indication of power range channels.

Explanation:

A. Incorrect. First part is incorrect, but plausible since yellow is the color used for many other indications to show an alert status. Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible (see D).

C. Correct. First part is correct. A red check mark indicates that the LEFM is not available to be used and the feedwater venturis should be used. Second part is correct. Using a lower feedwater temperature indicates that more heat must be added to the feedwater to raise it to saturation due to having a lower enthalpy, so the indicated power will be greater than actual power.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible if thought that lower feedwater temperature used would cause the actual power to be greater than indicated.

Technical Reference(s)	OPT-309	Attached w/ Revision # See Comments / Reference
	Heat Transfer Generic Fundamentals	

Proposed references to be provided during examination: ____________________

Learning Objective: **DISCUSS** IPO-003, Power Operations, to include the following: Applicability; Precautions; Limitations; Notes; Instructions. (IPO.03B.OB01)

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 X

10 CFR Part 55 Content: 55.41 5
55.43 ________

Comments / Reference: OPT-309	Revision: 18

CPNPP OPERATIONS TESTING MANUAL	UNIT COMMON	PROCEDURE NO. OPT-309
UNIT CALORIMETRIC	REVISION NO. 18 MULTIPLE USE	PAGE 5 OF 34

5.2.4 Per Technical Specification SR 3.3.1.2, the calorimetric shall be performed every 24 hours when THERMAL POWER is $\geq$ 15% RTP. However, the calorimetric is not required to be performed until 24 hours after THERMAL POWER is $\geq$ 15% RTP.

5.2.5 If Calculated RTP exceeds Indicated RTP (NIS and N-16 power indications) by more than +2% RTP, the NIS and N-16 functions are not declared inoperable, but the channel gains must be adjusted consistent with the calorimetric power. If the NIS or N-16 channel outputs can NOT be properly adjusted, the channel(s) is declared inoperable.

5.2.6 Adjustment of NIS gain with the Rod Control System in AUTO can cause inadvertent rod motion. Rod control should be placed in manual when adjusting Power Range NIS gain or N-16 power channel.

5.2.7 The LEFM is checked available prior to performing a calorimetric to ensure the input is valid (TRS 13.3.34.1). The following listing provides criteria that may be used to assist in determining if the LEFM is OPERABLE (available for calorimetric calculation).

The LEFM is available for performance of SR 3.3.1.2 when the following indications are observed (normal condition of LEFM):

- LEFM Main display - the "✔" icon is displayed in green and the "wrench" icon is displayed in light gray.

WHEN the LEFM is in alert condition with the following conditions displayed, THEN the LEFM can be considered available for performance of SR 3.3.1.2:

- LEFM Main display - the "✔" icon is displayed in green and the "wrench" icon is displayed in yellow. If the "wrench" icon is displayed in yellow on the LEFM Main display, notify System Engineering of the condition.

The LEFM is NOT available for performance of SR 3.3.1.2 when the following indications are observed:

- LEFM Main display - the "✔" icon is displayed in red and the "wrench" icon is displayed in yellow.

The LEFM is not available, and is in a failed condition when the following self-diagnostics have failed.

- LEFM system fails the Uncertainty Performance test. The LEFM is unable to calculate feedwater mass flow and temperature with the accuracy sufficient to support determination of thermal power within 0.6%.
- The meter has failed self-diagnostics on calculations, measurements, and electronics which includes meter path tests and meter velocity profile tests.
- The pressure transmitter input has failed.

Comments / Reference: OPT-309	Revision: 18

CPNPP OPERATIONS TESTING MANUAL	UNIT COMMON	PROCEDURE NO. OPT-309
UNIT CALORIMETRIC	REVISION NO. 18 / MULTIPLE USE	PAGE 6 OF 34

5.2.8 The Plant Computer POWERL display may be used to perform the Unit Calorimetric calculation provided the thermal power value on the Plant Computer POWERL display is displayed in green or cyan.

- If the thermal power value on the Plant Computer POWERL screen is displayed in cyan, notify System Engineering of the condition.
- The thermal power value on the Plant Computer POWERL display should NOT be used to perform the Unit Calorimetric calculation if the thermal power value is displayed in dark blue.

5.2.9 If LEFM indication becomes unavailable during intervals between SR 3.3.1.2 performance, plant operation may continue using the power indications from the NIS and N-16 systems.

The LEFM shall be returned to service prior to the next performance of SR 3.3.1.2, or Reactor power must be reduced to $\leq$ 98.6% RTP (3562MWt) AND the calorimetric performed with the Feedwater Venturi indications of feedwater flow.

When the Feedwater Venturi indication is used for the calorimetric measurement, the core thermal power is limited to 3562 MWt in order to maintain compliance with the safety analysis.

TRM 13.3.34 provides instruction to track the compensatory actions when the LEFM is inoperable (not available for performance of the calorimetric measurement).

5.2.10 The condition and instructions for dealing with the UPFA during a unit calorimetric have been evaluated for both units individually. These instructions apply when the conditions of a UPFA exist during the performance of a Unit calorimetric AND during N16 adjustments based on calorimetric data. (A manual calorimetric calculation should not be used during a UPFA.)

On Unit 1, the UPFA is characterized by one channel of N16 power (channel 3 or 4) going high 2% to 3% and one channel (channel 1 or 2) of N16 power going low 2% to 3%, simultaneously. The duration of the UPFA ranges from a few seconds to several minutes. The average duration is about 3 minutes. During this period the channel going high and the channel going low are roughly symmetric to each other, so Tavg is not affected.

On Unit 2, the UPFA is characterized by channel 2 N16 power going high 2% to 3% and channel 3 N16 power going low 2% to 3%, simultaneously. The duration of the UPFA ranges from a few seconds to several minutes. The average duration is about 3 minutes. During this period channel 2 and channel 3 are roughly symmetric to each other, so Tavg is not affected.

The following evaluations contain additional information on the UPFA: EVAL-2002-4113-01 (Unit 1), EVAL-2003-0469-01 (Unit 1), EVAL-2005-2253-01 (Unit 2), EVAL-2008-1434-01 (Unit 2)

Comments / Reference: OPT-309	Revision: 18

CPNPP OPERATIONS TESTING MANUAL	UNIT COMMON	PROCEDURE NO. OPT-309
UNIT CALORIMETRIC	REVISION NO. 18	PAGE 11 OF 34
	MULTIPLE USE	

5.2.11 When operating above 15% Rated Thermal Power (RTP), TS SR 3.3.1.2 requires the daily adjustment of the Nuclear Instrumentation System (NIS) Power Range and N-16 Power Monitor channel outputs when the power indication exceeds the secondary side calorimetric power by more than +2% RTP. Even though the accuracy of the LEFM-based calorimetric measurement is relatively insensitive to power level, the accuracy of the secondary side calorimetric using the feedwater venturis decreases as the reactor power decreases. In addition, decreases in the downcomer temperature increase the shielding effect and reduce the NIS power indiction ; this effect is less important during normal operations than the decrease in the FW-venturi-calorimetric uncertainty. Thus, during reduced power operations following normalization of the power indications to the calorimetric measurement at a higher power level, the power indications should not be normalized to a daily calorimetric measurement unless the indicated power is more than +2% RTP greater than the calorimetric power. If a normalization is performed at reduced power against a FW-venturi-based calorimetric measurement, caution should be employed during the power ascension as the power indication may not reflect the actual power.

5.2.12 Performance of the calorimetric using venturis (PPP-VENTURIs (POWERC, POWERV, or MANUAL with venturis)) during extended power operation below 55% RTP AND a subsequent reduction in NIS Power Range or N-16 channel output has the potential to place the Unit in a condition outside the safety analysis limit (i.e., reactor trip originating from Power Range or N-16 indication may be above the value assumed in the safety analysis). Therefore, additional controls exist in ODA-308-13.3.34 for performance of a calorimetric using FW venturis with the Unit operating in extended power operation below 55% RTP.

5.2.13 During NI or N16 adjustments, there should be no Protection Set testing or maintenance to ensure adjustments in multiple cabinets do not occur.

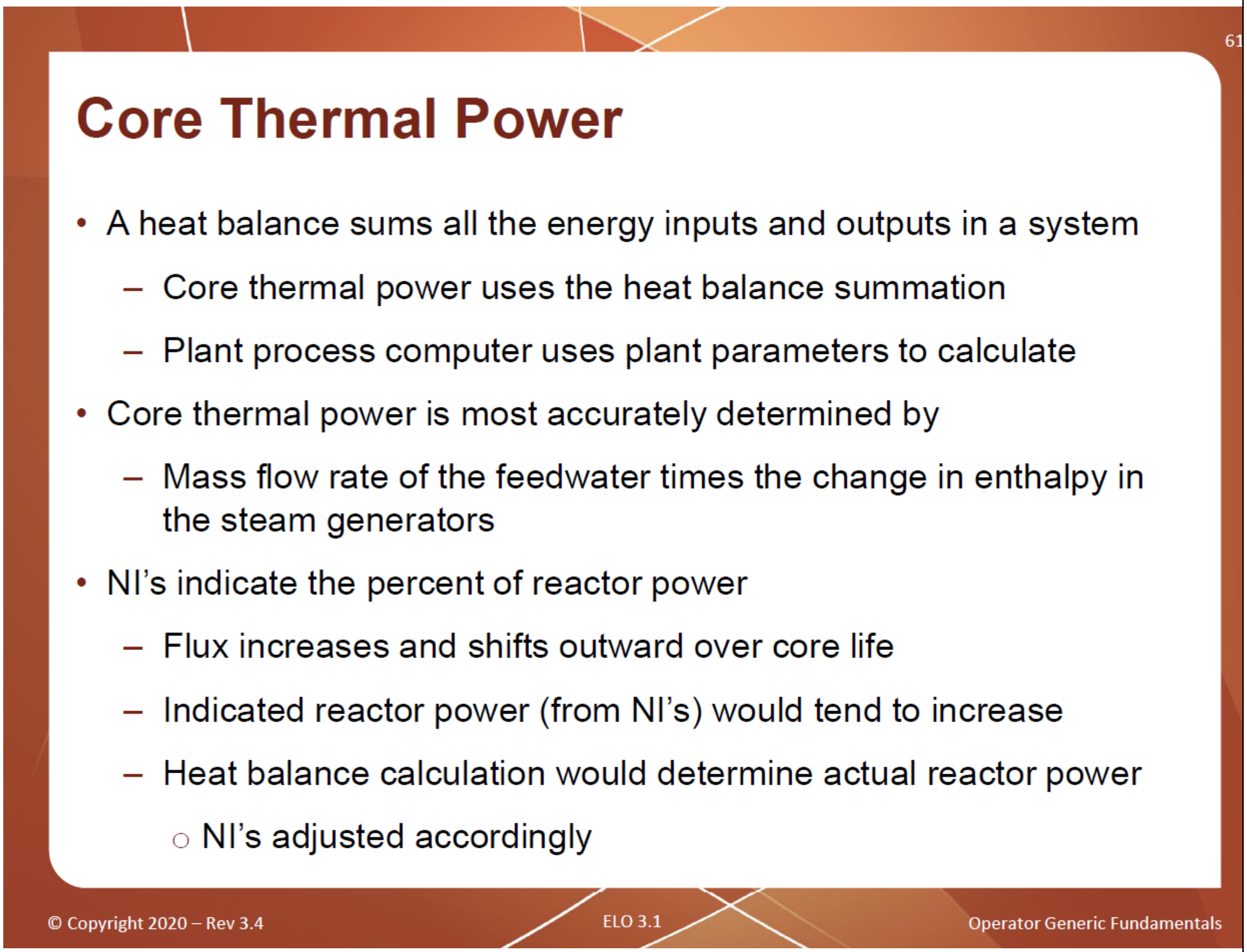
5.3 Notes

5.3.1 This procedure is common to both units. The specific unit designator (1 or 2) is represented within these instructions by the symbol "u". The appropriate unit digit may be substituted for the symbol to obtain the Unit specific equipment number. (Example u-TI-2158 represents 1-TI-2158 for Unit 1 and 2-TI-2158 for Unit 2.)

5.3.2 When Feedwater Venturi computer points (F5992A through F5999A) are used for the calorimetric, a correction factor may be used to compensate for fouling of the venturis. The correction factor is automatically calculated by the Plant Computer and updated every 24 hours. The Plant Computer Feedwater Venturi calorimetric program (POWERC) uses the correction factor in the calorimetric measurement.

5.3.3 If desired, when chemistry conditions permit, Steam Generator Blowdown may be secured for greater accuracy.

Comments / Reference: Heat Transfer Generic Fundamentals	Revision:



Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	2		
	Group	2		
	K/A	016.K3.02		
Level of Difficulty: 2	Importance Rating	3.4		

Non-nuclear Instrumentation: Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: PZR LCS

Question # 32

Given the following conditions

- Unit 2 100% power
- 1/2-LS-459D, PRZR LVL CTRL CHAN SELECT switch, is selected to 459/460
- The reference leg for level transmitter 2-LT-459A, PRZR LVL CHAN I, develops a leak

2-FCV-121, CCP Flow Control Valve will throttle __(1)__ in response to the erroneous level indication.

Pressurizer level transmitter __(2)__ will be selected for control on 1/2-LS-459D, PRZR LVL CTRL CHAN SELECT switch.

A. (1) closed
(2) 2-LT-461

B. (1) closed
(2) 2-LT-462

C. (1) open
(2) 2-LT-461

D. (1) open
(2) 2-LT-462

Answer: A

K/A Match: K/A match due to requiring knowledge of how a failure of a detector effects pressurizer level control system.

Explanation:

A. Correct. 1st part is correct. As level in the reference leg lowers, DP will lower. In the Przr level detector, this will result in indicated Przr level rising. FCV-0121 will throttle close in an attempt to reduce Przr level. 2nd part is correct. LT-461 is the only option as a substitute for LT-459. LT-462 is cold calibrated and used for indication only.

B. Incorrect. 1st part is correct (see A). 2nd part is incorrect because LT-461 is the only option as a substitute for LT-459. It is plausible because there are 4 pressure transmitters for Pressurizer pressure with each channel having its own standby transmitter. Level is different in that there is only one back which can be substituted for either LT-459 or LT-460.

C. Incorrect. 1st part is incorrect because a leak in the reference leg will cause FCV-0121 to close due to indicated Przr level rising. It is plausible because the effects of a reference leg leak are commonly mistaken. 2nd part is correct (see A).

D. Incorrect. 1st part is incorrect but plausible (see C). 2nd part is incorrect but plausible (see B).

Technical Reference(s)	PRZR Press & Lvl Control Study Guide	Attached w/ Revision # See Comments / Reference
	ABN-706	

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to a Pressurizer Level Instrument Malfunction in accordance with ABN-706, Pressurizer Level Instrument Malfunction. (ABN.705.OB02)

Question Source: Bank # 74093
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: PRZR Press & Lvl Control Study Guide	Revision: 00-0000

OP51.SYS.PP1

Each of the four level detectors is associated with a level transmitter that develops an electronic signal for remote indication. Three of the detectors, u-LT-459, 460 & 461, are calibrated for the normal operating pressurizer temperature of 653°F. These instruments provide signals used for indication, control and protection. The other detector, u-LT-462, is calibrated for 70°F and is used only for indication. This cold-calibrated instrument is the primary indication used when pressurizer temperature is below 450°F.

When pressurizer temperature is not at the calibration temperature, the level instruments will be inaccurate. This is because of the change in the density of the water in the pressurizer associated with a change in temperature. A change in pressurizer temperature results in a change in water volume, and therefore level, but no change in the pressure exerted on the variable input to the d/p detector, which is relative to the mass of the water in the pressurizer. The water in the external reference leg is at containment ambient temperature regardless of pressurizer temperature, and therefore provides a constant pressure to the d/p detector. Reducing pressurizer temperature causes the hot-calibrated instruments to read erroneously high. The volume and level of water in the pressurizer decreases as it becomes denser, while the pressure exerted on the variable input to the d/p detectors remains unchanged. Increasing pressurizer temperature causes the cold-calibrated instruments to read erroneously low. The volume and level of water in the pressurizer increases as it becomes less dense, while the pressure exerted on the variable input to the d/p detectors remains unchanged. However, the operator should realize that both hot and cold-calibrated instruments indicate accurately at approximately 20% for any temperature. Graphs comparing indicated to actual pressurizer level for various temperatures are located in the Integrated Plant Operating Procedures.

The proper level of water in the reference leg is crucial for accurate indication. Low level in a reference leg can be identified by comparing the level channel indications. If a reference leg leaks, its pressurizer level channel will read erroneously high. If the RCS suffers a rapid depressurization, hydrogen or other gasses coming out of solution could displace water in the reference leg, resulting in a level indication higher than actual. This could potentially affect all channels at the same time. There is another way to lose reference leg level during accident conditions. If steam inside containment significantly raised ambient temperature, the water in the reference leg could boil away and cause the indication to be erroneously high. This also could potentially affect all channels at the same time.

The 118 VAC instrument buses supply power to the level transmitters as follows: uPC1 to u-LT-459, uPC2 to u-LT-460, uPC3 to u-LT-461, and uPC4 to u-LT-462. Each transmitter supplies 0-100% meters on Main Control Board panel u-CB-05 and inputs to the plant computer. u-LT-459 & 460 are also indicated on the Remote Shutdown Panel. A switch on the control board selects one of the hot-calibrated channels to supply a chart recorder on u-CB-05. Signals from the three hot-calibrated level channels are sent to a channel selector switch (1/u-LS-459D) on u-CB-05. Channels 459 and 460 are normally selected, and channel 461 can be used in place of either channel, but not simultaneously.

Comments / Reference: ABN-706	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-706
PRESSURIZER LEVEL INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 4 OF 14

2.2 Automatic Actions

NOTE: For the pressurizer level and high level heater control circuits:

- CH I 0459 is the normal input.
- CH III 0461 is the alternate input.
- CH II 0460 has no input.

For the low level heater cutoff and letdown isolation circuits:

- CH I 0459 is the normal input to 1/u-LCV-459.
- CH II 0460 is the normal input to 1/u-LCV-460.
- CH III 0461 is the alternate input to 1/u-LCV-459 or 1/u-LCV-460.

a. Control response for a selected pressurizer level channel failure high.

1) Charging flow is reduced, lowering actual pressurizer level until at 17% level, low level heater block and letdown isolation occur.

2) Backup heaters come on if pressurizer level channel selected for control increases greater than or equal to 5% from programmed level (either directly due to failure or due to actual level increase).

- u-LR-459, PRZR LVL/PRZR LVL SETPT

b. Control and interlock responses for a selected pressurizer level channel failure low.

1) Charging flow is increased, raising pressurizer level.

2) Low level heater block and letdown isolation occur if channel fails to less than or equal to 17% pressurizer level.

Section 2.2

Comments / Reference: ABN-706	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-706
PRESSURIZER LEVEL INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 5 OF 14

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: To avoid thermal shock of the reactor coolant piping, the letdown flow should not be stopped without also stopping the charging flow when the reactor coolant temperature is greater than 350°F.

NOTE:
- Channels 459 and 460 are normally the controlling channels.
- Refer to Attachment 7 for program versus Tave information.

	Step	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐	1	Manually CONTROL u-LK-459, PRZR LVL CTRL OR u-FK-121, CCP CHRG FLO CTRL to maintain level at program.	
☐	2	TRANSFER 1/u-LS-459D, PRZR LVL CTRL CHAN SELECT to an operable alternate controlling channel.	
☐	3	ENSURE 1/u-LS-459E, u-LR-459 PRZR LVL SELECT selected to a valid channel.	
☐	4	VERIFY normal letdown aligned.	WHEN pressurizer level is greater than 17%, THEN RESTORE letdown per Attachment 6.
☐	5	If necessary, RECLOSE 1/u-PCPR, PRZR CTRL HTR GROUP C by placing the control switch in the "ON" position.	
☐	6	If desired, PLACE controller used in Step 1 in AUTO.	

Section 2.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	2		
	K/A	001.A4.06		
Level of Difficulty: 2	Importance Rating	2.9		

Control Rod Drive System: Ability to manually operate and/or monitor in the control room: Control rod drive disconnect/connect

Question # 33

Given the following conditions:

- Control Rod M-4 in CBD is misaligned by 18 steps
- Actions are being performed to realign the rod IAW ABN-712, Rod Control System Malfunction, using the DRPI method

Control Rod M-4 is realigned to the remaining rods in Control Bank D by placing the Rod Selector Switch to the __(1)__ position and opening the lift coil disconnect switch(es) for __(2)__.

A. (1) CBD
(2) control rod M-4

B. (1) CBD
(2) the remaining rods in Control Bank D

C. (1) MAN
(2) control rod M-4

D. (1) MAN
(2) the remaining rods in Control Bank D

Answer: B

K/A Match: K/A match due to requiring knowledge of the operation of the rod disconnect switches when realigning a misaligned rod.

Explanation:

A. Incorrect. First part is correct (see B). Second part is incorrect, but plausible since opening only disconnect switch M4 would allow the other rods in the group to be realigned to the misaligned rod, but procedure has only the misaligned rod move.

B. Correct. First part is correct. Per ABN-712, the misaligned rod is selected by using the individual bank position, CBD. Second part is correct. The disconnect switches for all other Bank D rods are opened to assure only the misaligned rod moves.

C. Incorrect. First part is incorrect, but plausible since the switch is in the MAN position when the rod is determined to be misaligned, but is placed in CBD when aligning the rod. Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	ABN-712	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **DISCUSS** the response to a Dropped or Misaligned Rod in MODE 1 or 2. (ABN.712.OB02)

Question Source: Bank # 20872
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments / Reference: ABN-712	Revision: 13

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION:
- Affected rod withdrawal should only be performed after fuel conditioning requirements have been met unless approved by Engineering.
- Do NOT withdraw an RCCA that has been misaligned for greater than 4 hours during power operation without Engineering guidance.

Note:
- The last movement of affected rod should be in the SAME direction as the last movement of affected group.
- When recovering a dropped rod using the DRPI method the group with the dropped rod should initially be moved outward to the next DRPI step up vice in so as not to drive the misaligned rod further into the core. Positive reactivity will be added during recovery.

14 RESTORE Rod to OPERABLE Status using DRPI realignment method:

☐ a. TRANSFER 1/u-RBSS, CONTROL ROD BANK SELECT, to the affected bank.

☐ b. RECORD positions for affected rod:

Affected Rod (DRPI)	______
Bank (DRPI)	______
Group 1 step counter	______
Group 2 step counter	______

☐ c. MOVE affected bank outward to the desired DRPI Light

"Step continued next page"

Comments / Reference: ABN-712	Revision: 13

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-712
ROD CONTROL SYSTEM MALFUNCTION	REVISION NO. 13	PAGE 18 OF 63

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Do NOT make any changes in plant operations during realignment of the affected rod that would require a change in bank position.

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☐ d. PLACE all lift coil disconnect switches for affected bank, groups 1 AND 2, EXCEPT for affected rod to the UP (disconnected) position.

☐ e. IF desired, RESET affected group step counter to the DRPI indicated position of affected rod.

CAUTION: Do NOT allow P/A Converter Auto-Manual selector switch to spring return to automatic until directed by this procedure.

☐ f. IF restoring a Control Bank rod, THEN Locally POSITION AND MAINTAIN P/A Converter Auto-Manual selector switch (SFGD 832 Rm u-096) - MANUAL

"Step continued next page"

Section 3.3

Comments / Reference: ABN-712	Revision: 13

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-712
ROD CONTROL SYSTEM MALFUNCTION	REVISION NO. 13	PAGE 19 OF 63

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- When moving affected rod, a CONTROL ROD CTRL URGENT FAIL alarm will be received in control room and at power cabinet containing the other group of affected bank. This is normal and will prevent the other group's step counter from operating.
- At low RCS boron concentration, excessive boration may delay return to desired power level after rod recovery.

14

☐ g. WHEN moving the affected rod for realignment,
THEN
PERFORM the following:

1) MAINTAIN Tave within 2°F of Tref by controlling the following, as necessary:
   - Turbine Power
   - Steam Dumps
   - Boration
   - Dilution

2) VERIFY that only the affected rod is moving.

3) ENSURE last movement of affected rod is in same direction as last movement of affected bank.

☐ h. WITHDRAW the affected rod in controlled increments until aligned with its group by DRPI indication.

"Step continued next page"

Section 3.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	2		
	K/A	033.G.2.1.28		
Level of Difficulty: 4	Importance Rating	4.1		

Spent Fuel Pool Cooling: Knowledge of the purpose and function of major system components and controls.

Question # 34

Given the following conditions:

- SFP Cooling Water Pump X-02 and HX X-02 are aligned to and cooling the X-01 SFP
- Level is lowering in the X-01 SFP due to a pipe rupture downstream of SFP Cooling Water Pump X-02 discharge valve

After SFP Cooling Water Pump X-02 loses suction, draining the pool via the break will be terminated by …

A. uncovering the X-01 SFP suction piping anti-siphon hole.

B. uncovering the X-01 SFP discharge piping anti-siphon hole.

C. closure of the discharge valve on SFP Cooling Water Pump X-02.

D. tripping SFP Cooling Water Pump X-02 on a low level in X-01 Spent Fuel Pool.

Answer: B

K/A Match: K/A match due to requiring knowledge of the function of the anti-siphon holes on the discharge piping located in the SFPs.

Explanation:

A. Incorrect. Plausible based on the location of the leak once the pump loses suction water will continue to drain out the SFP until the anti-siphon hole is uncovered on the discharge piping.

B. Correct. The pump must first lose suction, with no discharge pressure the SFP water will backflow through the discharge piping until the anti-siphon hole is uncovered.

C. Incorrect. Plausible since closing the discharge valve would prevent forward flow out of the downstream break, but the valve is upstream of the leak and would have no effect since the leak is downstream siphoning water out from the bottom of the pool.

D. Incorrect. Plausible because the pump will trip on low level in its associated SFP.

Technical Reference(s)	SFP Cooling and Cleanup Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **STATE** the function of the Spent Fuel Pool Cooling and Cleanup system IAW FSAR. (SYS.SF1.OB01)

Question Source: Bank # 34602
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: SFP Cooling and Cleanup Study Guide	Revision: 5-2-2011

Spent Fuel Pool Cooling and Cleanup

Cooling Return Line and Connections

Return flow from the Purification loop rejoins the main flow returning to the Spent Fuel Pools. The return header enters the pool just below the normal surface level and is routed down to a sparger located six feet above the fuel assemblies. Each return line is provided with one 1/2" anti-siphon hole located one foot below the normal pool level (858'6" normal level). The primary purpose of these holes is to prevent an inadvertent draining of the pool should a upstream line break occur on a idle line. A flow indicating switch in each return line activates an alarm on the Spent Fuel Pool panel if flow is <1600 gpm to its associated pool (windows 1.2, 1.6).

PURIFICATION LOOP (FIGURE 2)

Refueling Water Purification Pumps

There are two Refueling Water Purification Pumps located on the 790' elevation of the Auxiliary Building. These pumps have a design pressure of 150 psig, a design temperature of 200 °F, and a design flow of 250 gpm. Their primary function is to circulate Refueling Water Storage Tank water or Refueling Cavity water through the Spent Fuel Pool Filters and Demineralizers. The pumps are controlled via three position handswitches located on the Spent Fuel Pool local control panel (LV-06). A pump trip due to overload is also alarmed on this panel (windows 1.3, 1.7).

Purification suction connections

The Refueling Water Purification Pumps can be aligned to either unit's Refueling Cavity or RWST. The Wet Cask Pit or the Fuel Transfer Canals can also be sent through the Purification Loop using the Cask Pit and Transfer Canal Drain Pump.

Spent Fuel Pool Filters

The Spent Fuel Pool Filters provide mechanical filtration for the Purification Loop flow before it enters the Spent Fuel Pool Demineralizer. They are removable cartridge filters will provide 100% retention of particles from 0.45 to 6 microns. Isolation valves for the filters are operated remotely from the Auxiliary Building 852' elevation on the east side of the filter/demineralizer area. A bypass valve around each filter is provided to permit filter change out without affecting Purification Loop flow. A differential pressure indicating switch provides local indication of the filter condition. An alarm is activated on the SFP panel if the ΔP is greater than 25 psid (Windows 3.1, 3.5).

Spent Fuel Pool Demineralizers

Two demineralizers, one for each purification loop, are provided. Each demineralizer has a design flow of 250 gpm (max. 273 gpm), a design pressure of 150 psig, and a design temperature of 200 °F. They are flushable mixed bed demins with a 50 cubic foot resin volume. A pressure indicating switch provides local indication and an alarm on the Spent Fuel Pool Panel if the demin. ΔP reaches $\geq$ 25 psid (Windows 4.1, 4.5)

A resin trap is located just downstream of each demineralizer to remove resin fines that may have passed through the demineralizer retention elements.

Comments / Reference: SFP Cooling and Cleanup Study Guide	Revision: 5-2-2011

contamination levels. During this time it will be necessary to closely monitor SFP temperature and realign the SFP Cooling and Cleanup System as required to limit the temperature increase.

OPT-223, SPENT FUEL POOL COOLING SYSTEM

This procedure satisfies Spent Fuel Pool Cooling System testing for Technical Specification 4.0.5 (ASME Code Class 1, 2, and 3 components) requirements.

Section 8.1 of OPT-223 tests the SFP Cooling Pumps. Section 8.1.1 tests the X-01 SFP Cooling Pump while section 8.1.2 tests the X-02 pump. The pump to be tested is placed in service per SOP-506. Basically, the procedure has the operator establish a 3950 to 3975 gpm flow rate and record information on the data sheet. Flow greater than 3600 gpm indicates that the associated pump discharge check valve (XSF-0003 or XSF-0004) has fully stroked open. Suction pressure and discharge pressure are obtained from test gauges and a differential pressure is calculated and compared to Action Limit Low and High values. The vibration amplitude is recorded and compared to Alert and Action Limits. If Alert limits are exceeded for vibration, ODA-308-37 (LOCAR for Tech Spec 4.0.5 items) is initiated and the Shift Manager and System Engineer are notified. When alert limits are exceeded, the SFP Cooling Pump test frequency is increased. Restoration is performed in accordance with section 9.1 which has the operator align the SFP Cooling Water System per SOP-506 as directed by the Shift Manager. Maintenance is notified to remove test gauges which were installed to support the test.

Section 8.2 tests the Makeup Flowpath. Section 8.2.1 tests the flowpath from Unit 1 Reactor Makeup Water to the SFP Cooling System and section 8.2.2 tests the Unit 2 Reactor Makeup Water flowpath. Prior to each test, a portable flow meter is installed on a horizontal length of Reactor Makeup piping for the check valve to be tested. The applicable line number is listed in the procedure. Basically the makeup valve from each RMUW source is opened and the flow rate recorded. If the recorded flow rate is ≥51 gpm, the associated check valve has cycled full open. Following the test, the flow meter is removed and Chemistry notified to sample the Spent Fuel Pool for boron concentration if needed.

ABNORMAL

SPENT FUEL POOL AND REFUELING CAVITY WATER LEVELS

Tech Specs 3.7.15 requires that at least 23 feet of water be maintained over the top of irradiated fuel assemblies seated in the storage racks. This is to ensure removal of 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

To protect against loss of water from the Spent Fuel Pool, the Spent Fuel Pool Cooling Pump suction lines penetrate the pool wall and terminate approximately four feet below the normal water level and the return lines terminate six feet above the fuel assemblies. The return lines contain anti-siphon holes at the 857'6" elevation (approximately one foot below the normal level). The anti-siphon holes prevent gravity draining of the pool and ensure sufficient shielding is maintained. There are no drain lines connected to the pool.

Several things can affect Spent Fuel Pool level. Of course there is normal evaporation which is made up by using either Reactor Makeup Water (preferred source) or Demineralized Water (alternate

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	2		
	K/A	045.A3.08		
Level of Difficulty: 3	Importance Rating	3.3		

Steam Dump/Turbine Bypass Control: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure

Question # 35

Given the following conditions:

- Unit 1 is performing a cooldown
- RCS loop Tave:
  - Loop 1: 552ºF
  - Loop 2: 553ºF
  - Loop 3: 552ºF
  - Loop 4: 550ºF
- Steam Dump Mode Selector switch - STM PRESS MODE
- Steam Dump Controller - MAN set at 50% demand
- All Steam Dump valves closed
- The operator momentarily places the Train A and Train B Steam Dump Bypass Interlock switches to Bypass and then releases them

What is the status of the Steam Dump valves following the operator's actions?

A. ONLY Bank 1 valves fully open.

B. Bank 1 and 2 valves go 50% open.

C. ONLY Bank 1 valves go 50% open.

D. Bank 1 and Bank 2 valves fully open.

Answer: A

K/A Match: K/A match due to requiring knowledge of the steam dump system controls when operating in the steam pressure mode.

Explanation:

A. Correct. Bypassing the P-12 interlock allows the Bank 1 valves to open. With demand at 50%, the Bank 1 valves will be fully open (at 25%), all others are prevented from opening.

B. Incorrect. Plausible since this would be the condition above P-12 and since the controller is set to 50% demand it could be thought the valves will only open to 50%.

C. Incorrect. Plausible since only Bank 1 valves open below P-12 and since the controller is set to 50% demand it could be thought the valves will only open to 50%.

D. Incorrect. Plausible since this would be the condition above P-12, but only the Bank 1 valves can be opened below P-12.

Technical Reference(s)	IPO-005A	Attached w/ Revision # See Comments / Reference
	ALM-0065A	
	Steam Dumps Study Guide	

Proposed references to be provided during examination: ______

Learning Objective: **PREDICT** the response of the instrumentation and controls of the Steam Dump system in accordance with ABN-304, 704 and 709. (SYS.SD1.OB04)

Question Source: Bank # 19303
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: IPO-005A	Revision: 27

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-005A
PLANT COOLDOWN FROM HOT STANDBY TO COLD SHUTDOWN	REVISION NO. 27 CONTINUOUS USE	PAGE 23 OF 131

NOTE: Pressurizer level will lower as the RCS is cooled in the following step.

[C] 5.1.6 E. IF desired to cooldown with the Steam Dumps,
THEN
PERFORM the following steps. THIS IS THE PREFERRED METHOD.

☐ 1) VERIFY 43/1-SD, STM DMP MODE SELECT in STM PRESS.

☐ 2) ENSURE 1-PK-507, STM DMP PRESS CTRL is in MANUAL.

CAUTION:
- When opening the Steam Dump Valves, a rate compensated steam line low pressure Safety Injection or Main Steamline Isolation (if blocked below P-11) can result from the valves opening too quickly.
- Above P-11, Plant computer point U5533, MS SI MARGIN can be used to monitor margin to SI signal.
- Below P-11, Plant computer point U5534, MS ISOLATION RATE MARGIN can be used to monitor margin to main steamline rate isolation signal.

☐ 3) ADJUST 1-PK-507, STM DMP PRESS CTRL demand to establish an RCS cooldown rate less than 100°F in one hour.

NOTE: When RCS temperature drops below 553°F, only the Steam Dump Cooldown valves are available.

☐ 4) WHEN RCS Tavg approaches Lo-Lo Tavg P-12 (553°F),
THEN
HOLD both STM DMP INTLK SELECT switches in BYP INTLK.

5) VERIFY the following status lights are ON:

- ☐ 1-TSLB-8, 1.8, STM DMP TRN A INTLK BYP 43/1-SDA
- ☐ 1-TSLB-8, 2.8, STM DMP TRN B INTLK BYP 43/1-SDB

6) WHEN 1-PCIP, 3.6, Tavg LO-LO P-12 is ON,
THEN,
RELEASE both STM DMP INTLK SELECT switches.

_____ / _____
Initials Date

Comments / Reference: ALM-0065A	Revision: 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 51 OF 73

ANNUNCIATOR NOM./NO.: TAVE LO-LO P-12 3.6

PROBABLE CAUSE:

RCS cooldown and depressurization

NOTE: This window is normally illuminated in Mode 4-6.

AUTOMATIC ACTIONS:

Blocks operation of all steam dumps

NOTE: Auctioneered LO-LO-T_{AVE} <553°F enables C-16 Stop Turbine Loading.

OPERATOR ACTIONS:

1. To continue cooldown using steam dump cooldown valves, momentarily place 43/1-SDA, STM DMP INTLK SELECT, and 43/1-SDB, STM DMP INTLD SELECT, in INTLK BYP.

-

Comments / Reference: Steam Dumps Study Guide	Revision: 5-4-2011

Steam Dumps

SUMMARY

The Steam Dump System is designed to absorb the excess energy created from a load rejection which is greater than or equal to either a 10% step change or a rate of 5% per minute. The system works with the automatic Rod Control System to restore RCS temperature to the programmed value for the current reactor power. The Steam Dump System accomplishes its function by allowing steam which has bypassed the Main Turbine to enter the upper portion of the Main Condenser. The steam flow removes heat from the SG's which in turn remove heat from the RCS. The system also absorbs the excess energy from a plant trip. When the reactor trip breakers open, the reactor goes to a subcritical state. RCS temperature is at an elevated temperature compared to the desired temperature for a subcritical reactor. The Steam Dump System removes this heat and allows RCS temperature to reach 557°F which is the desired temperature for Hot Standby conditions.

The Steam Dump System's piping taps into the Equalization Header on the 803' level of the Turbine Building. The piping runs toward the outside wall of the Turbine Building and along this wall the length of the Main Condenser. Each pneumatic valve is attached to a piping run which taps into the piping along the outside wall of the Turbine Building. The pneumatic Steam Dump Valve has a manually operated isolation upstream and downstream of it. The piping run then enters the upper portion of the Main Condenser shells.

The Steam Dump System is comprised of 12 pneumatic valves which modulate in banks of three valves to control the amount of steam which bypasses the Main Turbine. The valve positioners use electrical signals from the three controllers within the system to position the pneumatic valves to their desired position. The Load Rejection Controller determines Steam Dump Valve position during load rejection conditions. The Plant Trip Controller is placed into service by the P-4 signal generated by the opening of the Reactor Trip Breakers. This controller controls Steam Dump Valve position during a plant trip. The Steam Pressure Controller requires manual action to place it in service. The controller is used during periods of shutdown and low power operation (<15% power). The Steam Pressure Controller maintains a desired SG pressure. The pressure of the SG increases or decreases with changes in Reactor Coolant System temperature; therefore the controller actually maintains Reactor Coolant System temperature.

The Steam Dump Valves are modulated in banks of three valves. Bank 1 valves are the first to open. These valves should be fully open when the system controller's demand is 25%. Bank 2 valves start to open at 25% demand and are fully open at 50% demand and so on. The valves close in the same manner as they open. Bank 4 valves are fully closed when Bank 3 valves start to modulate closed and so on. Bank 1 valves are designated the cooldown valves and are provided with a feature which allows bypassing the 553°F (P-12) temperature interlock. The other banks of valves are not afforded this option. The cooldown valves are used for plant cooldowns-hence the name.

The Steam Dump Valves receive air pressure to open from the Instrument Air System. Air flow must pass through four solenoid valves before the valve actuator can receive air pressure. The last two solenoid valves in the series of four solenoid valves are the protection grade solenoids. Their function is to prevent the Steam Dump System from lowering RCS temperature below 553°F. When temperature falls to 553°F, the solenoids de-energize and block the passage of air to the valve actuator. The second solenoid valve in the series is a control solenoid which is said to "arm" the Steam Dump Valves. This solenoid remains de-energized blocking air flow until one of its "arming" signals is received. One arming signal which must always exist is the Condenser available signal. The

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Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	2		
	K/A	045.A3.07		
Level of Difficulty: 2	Importance Rating	3.5		

Main Turbine Generator: Ability to monitor automatic operation of the MT/G system, including: Turbine stop/governor valve closure on turbine trip

Question # 36

Given the following conditions:

- Unit 1 100% power when a Reactor Trip occurs
- EOP-0.0A, Reactor Trip or Safety Injection, entered
- While performing Immediate Operator Actions, Main Turbine HP Stop Valve 2 indicates both RED and GREEN lights lit
- All other Main Turbine HP Stop Valves indicate GREEN Lights lit and RED lights extinguished
- All Main Turbine Control Valves indicate 0% open on the Turbine Digital Control system

Based on the above plant conditions, the turbine is __(1)__.

The reason for ensuring the turbine is tripped is to __(2)__.

A. (1) tripped
(2) prevent steaming the SGs dry

B. (1) tripped
(2) prevent an excessive RCS cooldown

C. (1) NOT tripped
(2) prevent steaming the SGs dry

D. (1) NOT tripped
(2) prevent an excessive RCS cooldown

Answer: D

K/A Match: K/A match due to requiring knowledge of the indications to verify the turbine has tripped.

Explanation:

A. Incorrect. First part is incorrect because EOP-0.0 step 2 requires that all HP turbine stop valves are closed. It is plausible because with the control valves closed virtually all steam flow to the turbine is stopped. Second part is incorrect because the reason for tripping the turbine is to stop the cooldown which would add positive reactivity and possibly initiate SI. It is plausible because analysis during an ATWT has shown that a turbine trip is necessary within 30 seconds to conserve SG inventory.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A).

D. Correct. First part is correct. Turbine is verified tripped by ALL HP stop valves indicating closed. Second part is correct. The reason for tripping the turbine on a Reactor Trip is to prevent excessive RCS cooldown.

Technical Reference(s)	EOP-0.0A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **STATE** the immediate operator actions of EOP-0.0, Reactor Trip or Safety Injection. (ERG.E0A.OB02)

Question Source: Bank # 58149
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: EOP-0.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 9	PAGE 5 OF 121

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Verify Reactor Trip:	
	a. Verify the following: • Reactor trip breakers - AT LEAST ONE OPEN -AND- • Neutron flux - DECREASING	a. Manually trip reactor from both trip switches. IF reactor will not trip, THEN momentarily de-energize 480V normal switchgear 1B3 AND 1B4. IF reactor NOT tripped, THEN go to FRS-0.1A, RESPONSE TO NUCLEAR POWER GENERATION/ATWT, Step 1.
	b. All control rod position rod bottom lights - ON	
2	Verify Turbine Trip: • All HP turbine stop valves - CLOSED	Manually trip turbine. IF the turbine will NOT trip, THEN pull-out all EHC fluid pumps. IF turbine still NOT tripped, THEN close or verify closed main steamline isolation valves.
3	Verify Power To AC Safeguards Busses:	
	a. AC safeguards busses - AT LEAST ONE ENERGIZED • AC safeguards bus voltage-6900 Volts(6500-7100 Volts)	a. Go to ECA-0.0A, LOSS OF ALL AC POWER, Step 1.
	b. AC safeguards busses - BOTH ENERGIZED	b. Restore power to de-energized AC safeguards bus per ABN-601, RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION or ABN-602, RESPONSE TO A 6900/480 VOLT SYSTEM MALFUNCTION when time permits.

Comments / Reference: EOP-0.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 9	PAGE 86 OF 121

ATTACHMENT 10
PAGE 3 OF 38

BASES

- Turbine Building area switchgear heater (CP1-VAHEDH-01) (Normally aligned to operate based on area temperature. Re-energized when power restored to 1B4.)

If MG sets are tripped to insert control rods, the rods will remain out until the flywheels on the MG sets coast down enough so that they cannot hold the rods out. Actual plant experience demonstrates that this time delay will vary from several seconds to one minute. This time delay is dependent on the amount of rod motion while the flywheels of the MG sets coast down. All rods will drop in several seconds if rod motion is occurring during flywheel coast down. If there is no rod motion during flywheel coast down, it may take approximately one minute before all rods drop into the core. (Reference DW-88-004)

When 1B3 and 1B4 are de-energized to initiate control and shutdown rod insertion (initiate reactor trip), the P-4 signal generated from the reactor trip and bypass breakers will not be present. The absence of the P-4 signal results in other automatic actions not being available. The trip of the Main Turbine from a Reactor Trip will not be present and the Feedwater Isolation signal from Low Tavg with P-4 will not occur, which may result in a prolonged RCS cooldown and a Safety Injection signal actuation. Tripping the Main Turbine in the following step helps to limit the RCS cooldown.

Steam Generator level and feed flow status are checked in subsequent recovery actions. The reactor trip and bypass breakers must still be opened in order to generate the P-4 signal for subsequent actions (e.g., Reset SI). Subsequent actions in this procedure (Attachment 2) or EOS-0.1A, REACTOR TRIP RESPONSE (Step 2) will be performed to open the reactor trip and bypass breakers.

If the reactor cannot be tripped (e.g., 1B3 or 1B4 cannot be de-energized from control room), a transition is made to FRS-0.1A, RESPONSE TO NUCLEAR POWER GENERATION/ATWT, to deal with ATWT conditions.

STEP 2: The turbine is tripped to prevent an uncontrolled cooldown of the RCS due to steam flow that the turbine would require.

The action to verify Main Steam isolation valve position is intended to include the actions to verify the Main Steam isolation bypass valves closed, in the event the bypass valves have been opened during startup operation (e.g., Main Steamline warmup).

STEP 3: AC power must be verified from either offsite sources or the diesel generators to ensure adequate power sources to operate the safeguards equipment. At least one train of safeguards equipment is required to deal with emergency conditions, if not, the operator should transfer to ECA-0.0A, LOSS OF ALL AC POWER.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	2		
	K/A	075.A2.02		
Level of Difficulty: 4	Importance Rating	2.5		

Circulating Water: Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of circulating water pumps

Question # 37

Given the following conditions:

- Unit 1 18% power
- ALB-9A, Window 8.9 – CNDSR PIT LVL TRIP/TRIP BYP, has just alarmed
- The Turbine Building NEO has just reported that leakage from a CW expansion joint has resulted in flooding in the condenser pit and level is rising
- ALL Circulating Water Pumps have tripped

Per ABN-304, Main Condenser and Circulating Water Malfunction, the __(1)__ is to be tripped and the MSIVs are to __(2)__.

A. (1) Reactor
(2) be closed

B. (1) Reactor
(2) remain open

C. (1) Turbine ONLY
(2) be closed

D. (1) Turbine ONLY
(2) remain open

Answer: A

K/A Match: K/A match due to requiring knowledge of the actions to be taken in the event of a loss of all circ water pumps.

Explanation:

A. Correct. First part is correct. With reactor power above 10%, the reactor is to be tripped. Second part is correct. With no circ water pumps available, the condenser is not available for steam dumps so the MSIVs are closed to control steam demand with the ARVs.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible (see D).

C. Incorrect. First part is incorrect, but plausible (see D). Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible since for all other conditions only a turbine trip is required up to P-9, 50% power. Second part is incorrect, but plausible since MSIVs are typically left open following a reactor or turbine trip.

Technical Reference(s)	ABN-304	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____

Learning Objective: **ANALYZE** the response to a Circulating Water Pump Trip in accordance with ABN-304, Main Condenser and Circulating Water System Malfunction. (ABN.304.OB01)

Question Source:
- Bank # 17645
- Modified Bank # ____ (Note changes or attach parent)
- New ____

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 5
- 55.43 ____

Comments / Reference: ABN-304	Revision: 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-304
MAIN CONDENSER AND CIRCULATING WATER SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 4 OF 34

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION:
- Operation with one Circulating Water Pump may result in pump runout and insufficient waterbox level, which may cause condenser tube degradation.
- With NO Circulating Water Pumps operating, a rapid shutdown of equipment cooled by TPCW, per ABN-306, may be necessary to prevent equipment damage and possible Main Generator hydrogen release.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 VERIFY Circulating Water Pumps - AT LEAST ONE OPERATING • u-HS-2800A, CWP 1 • u-HS-2801A, CWP 2 • u-HS-2802A, CWP 3 • u-HS-2803A, CWP 4	• IF Reactor Power is greater than or equal to 10%, THEN TRIP Reactor AND GO TO EOP-0.0A/B while others continue this procedure. • IF Reactor Power is less than 10%, THEN TRIP Turbine AND perform ABN-403 while continuing this procedure. • IF ALB-9A 8.9, CNDSR PIT LVL TRIP/TRIP BYP-LIT, THEN PERFORM Section 5.3, this procedure, while continuing this section. • IF NO Circulating Water Pumps operating, THEN SHUTDOWN secondary plant per ABN-306, Loss of TPCW Section, as necessary to maintain affected temperatures within limits. • IF NO Circulating Water Pumps operating, THEN SHUT MSIVs.

Section 2.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	2		
	Group	2		
	K/A	017.K4.01		
Level of Difficulty: 2	Importance Rating	3.4		

In-Core Temperature Monitor System (ITM): Knowledge of ITM design feature(s) and/or interlock(s) which provide for the following: Input to subcooling monitors

Question # 38

Concerning the Subcooled Margin Monitor:

(1) Which of the following CET temperatures are an input to the monitor; and,

(2) Assuming all other parameters remain stable, which of the following will cause RCS Subcooling Margin to be a smaller value?

A. (1) Highest CET temperature
(2) Rising RCS pressure

B. (1) Highest CET temperature
(2) Rising RCS temperature

C. (1) Average CET temperature
(2) Rising RCS pressure

D. (1) Average CET temperature
(2) Rising RCS temperature

Answer: B

K/A Match: K/A match due to requiring knowledge of the temperature inputs to the subcooling monitor.

Explanation:

A. Incorrect. First part is correct (see B). Second part is incorrect, but plausible if thought that being further from boiling would cause subcooling margin to get larger.

B. Correct. First part is correct. Along with RCS loop pressure, Pressurizer pressure, and RCS loop RTD temperatures, the Highest CET temperature is also an input. Second part is correct. Higher temperature indicates closer to boiling, so subcooling margin gets smaller.

C. Incorrect. First part is incorrect, but plausible as it could be thought that the average CET would be used as a general indication of the temperature of the RCS. Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	Core Cooling Monitor/RVLIS Study Guide	Attached w/ Revision # See Comments / Reference
	Core Cooling Monitor/RVLIS LP	

Proposed references to be provided during examination: __________

Learning Objective: **EXPLAIN** the instrumentation and controls of the Core Cooling Monitor/Reactor Vessel Level Indication System (RVLIS) IAW DBD-EE-004. (SYS.RC3.OB04)

Question Source: Bank # 19377
Modified Bank # __________ (Note changes or attach parent)
New __________

Question History: Last NRC Exam __________

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 __________

Comments / Reference: Core Cooling Monitor/RVLIS Study Guide	Revision: 4-27-2011

Core Cooling Monitor/RVLIS

SUBCOOLED MARGIN MONITOR (SMM)

In addition to the highest CET temperature, the SMM function of the CCM makes use of various other RCS measurements. Signal flow paths from the RVLIS instruments will be described later. Redundant, diverse temperature measurements are provided by RTD's located such that each SMM train employs the hot or cold leg RCS temperature from each of the four reactor coolant loops. As illustrated in Figure 5, Train A employs the hot leg temperatures from Loops 1 and 2 and the cold leg temperatures from Loops 3 and 4. Train B employs the cold leg temperatures from Loops 1 and 2 and the hot leg temperatures from Loops 3 and 4. Also employed are the Loop 4 hot leg RCS wide-range pressure measurements for Train B and redundant narrow-range pressurizer pressure measurements. Train A uses a wide range pressure transmitter which senses Reactor vessel thimble tube pressure.

These temperature and pressure measurements are used in conjunction with a stored steam table algorithm to compute the saturation margin. Conservatively, the computation is based upon the highest valid RCS temperature and lowest valid RCS pressure. Thus, under normal operating subcooled conditions, the SMM output signal represents the minimum possible margin to saturation. In the event of depressurization due to a small break LOCA, this output signal will provide the earliest indication of the existence of saturation conditions.

Subcooled Margin is a readout in EF which indicates the current temperature margin between actual highest temperature and the temperature for current RCS pressure. For example, control board display for RCS saturation margin, which indicates from -300EF to +300EF, could indicate -25EF, which is a normal value for power operations. This value means that if the hottest water in the coolant loops or core was increased greater than 25EF while maintaining RCS pressure constant, then that water would flash to steam.

The CCM microprocessor provides Core Exit Thermocouple (CET) and Subcooled Margin Monitor (SMM) output temperature signals in EF which are indicated by the side-by-side meters (Figure 6). Each meter covers the entire functional range of the corresponding instrument measurement. The CET measurement (left-side meter) provides the maximum valid core exit temperature on a scale of 0-2300EF (50 degree minimum scale intervals). The SMM measurement (right-side meter) indicates the computed RCS saturation margin on a scale ranging from 300°F subcooled, to saturation (0°F), to 300°F superheated.

During normal reactor operations, each CCM meter indicator can be expected to always remain in the bottom-half portion of the scale, giving redundant indication of adequate core cooling (i.e., CET readings significantly below the threshold for impending ICC and SMM readings well within the subcooling region). Subsequently, the CET reading would also rise (at a slower rate). The threshold of impending ICC is considered to be a CET reading of 1200°F. Since the monitored CET reading represents the maximum measured core exit temperature, it can be expected that adequate core cooling would exist as long as the CET reading does not rise well into the top-half portion of the CET scale.

Comments / Reference: Core Cooling Monitor/RVLIS LP	Revision: 00-0000

NOTES	LESSON OUTLINE

a. Two redundant trains

b. Each CCM is designed to indicate Core Exit Thermocouple temperatures (CET function) and to monitor the RCS Subcooling Margin Monitor (SMM function)

1) Core Exit Thermocouples:

a) Provide data to the CCM microprocessor.

b) Fifty CET's divided into two separate, redundant trains with each set having a distribution representative of all four quadrants of the reactor core exit area.

c) Type K (chromel-alumel) thermocouples contained within an aluminum-oxide insulated, stainless steel sheathed cable (1/8" OD).

d) Each train routed into a separate reference junction box which contains three platinum RTD's:

e) 2 of the 3 RTDs are used for reference temperature measurements

f) Other RTD is an installed spare

g) Reference measurements permit the transition from chromel-alumel leads to copper conductors for signal transmission to the CCM microprocessor.

h) CET signals used to monitor coolant temperatures over the entire range including normal operating conditions and extending to beyond accident extremes.

i) Highest valid CET signal is displayed on the Control Board and is also employed by the microprocessor to determine the RCS saturation margin

c. Subcooled Margin Monitor

1) SMM function of the CCM makes use of:

a) Highest CET temperature

b) Hot or Cold leg RCS temperature from each of the four reactor coolant loops.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	1		
	Group	1		
	K/A	000009.EA2.36		
Level of Difficulty: 2	Importance Rating	4.2		

Small Break LOCA: Ability to determine or interpret the following as they apply to a small break LOCA: Difference between overcooling and LOCA indications

Question # 39

Given the following conditions:

- EOP-1.0A, Loss of Reactor or Secondary Coolant, is being performed
- RCS WR Pressure 800 psig stable
- RCS WR Thot Temperatures 520°F stable
- RCS WR Tcold Temperatures 520°F stable
- SG Pressures 800 psig stable
- SG NR Levels 35% rising
- CCP SI Injection flow 350 gpm stable
- SI Pump flow 600 gpm each stable
- RHR Pump flow 0 gpm stable

The event causing the above indications is a _______.

A. LBLOCA

B. SBLOCA

C. Feedline Break

D. Steamline Break

Answer: B

K/A Match: K/A match due to requiring knowledge of the difference in RCS and secondary parameters during a SBLOCA events to properly diagnose the event.

Explanation:

A. Incorrect. Plausible because a LOCA is in progress, however, not a LBLOCA because RCS pressure is above the pressure at which RHR pumps begin injecting.

B. Correct. SBLOCA is in progress as RCS pressure has stabilized at a pressure above RHR injection flow. A secondary break is not evident by indications provided because SG pressures have stabilized.

C. Incorrect. Plausible because a Feedline break will cause a resultant drop in RCS pressures and temperatures and on a Feedline Break SG pressure will remain elevated until the break is uncovered, however, on a Feedline Break when the break is uncovered continued depressurization of the SGs will occur.

D. Incorrect. Plausible because RCS pressures and temperatures will drop below normal no-load values on a Steamline break, however, a Steamline Break will result in continued depressurization of the SGs.

Technical Reference(s)	EOP-1.0A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **STATE** the bases for operator actions, notes and cautions from EOP-1.0 in accordance with EOP-1.0, Loss of Reactor or Secondary Coolant. (ERG.E1A.OB04)

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 X

10 CFR Part 55 Content:
- 55.41 7
- 55.43 ______

Comments / Reference: EOP-1.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-1.0A
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 9	PAGE 10 OF 44

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

> CAUTION: RCS pressure should be monitored. If RCS pressure decreases in an uncontrolled manner to less than 325 PSIG (425 PSIG FOR ADVERSE CONTAINMENT) the RHR pumps must be manually restarted to supply water to the RCS.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 8	Check If RHR Pumps Should Be Stopped:	
	a. Check RCS pressure:	
	1) RCS pressure - GREATER THAN 325 PSIG (425 PSIG FOR ADVERSE CONTAINMENT)	1) Go to Step 10.
	2) RCS pressure - STABLE OR INCREASING	2) Go to Step 9.
	b. RHR pumps - ANY RUNNING WITH SUCTION ALIGNED TO RWST	b. Go to Step 9.
	c. Stop RHR pumps and place in standby.	
	d. Reset RHR auto switchover.	
9	Check RCS And SG Pressures:	Return to Step 1.
	• Check RCS Pressure - STABLE OR DECREASING	
	-AND-	
	• Check Pressure in All SGs - STABLE OR INCREASING	

Comments / Reference: EOP-1.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-1.0A
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 9	PAGE 14 OF 44

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[R] 11	d. Evaluate plant equipment: • Consult Plant Staff to determine equipment that should be available or started to assist in recovery.	
12	Check If RCS Cooldown And Depressurization Is Required:	
	a. RCS pressure - GREATER THAN 325 PSIG (425 PSIG FOR ADVERSE CONTAINMENT)	a. IF RHR pump flow greater than 750 gpm, THEN, go to Step 13. IF RHR pump flow is less than 750 gpm, THEN go to EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1.
	b. Go to EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1.	
13	Check If Transfer To Cold Leg Recirculation Is Required:	
	a. RWST level - LESS THAN LO-LO LEVEL	a. Return to Step 11. OBSERVE NOTE PRIOR TO STEP 11.
	b. Go To EOS-1.3A, TRANSFER TO COLD LEG RECIRCULATION, Step 1.	

Comments / Reference: EOP-1.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-1.0A
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 9	PAGE 30 OF 44

ATTACHMENT 4
PAGE 7 OF 21

BASES

STEP 9: Since procedure EOP-1.0A is used to recover from both a LOCA and secondary side break, a second check on SG pressure is necessary in case there is a faulted SG which was not fully depressurized at the time the SI termination criteria were checked. A check on RCS pressure is also necessary in case the SG pressures are stable and there is a faulted SG which is depressurizing at the time the SI termination criteria were checked. If there is a faulted SG which is still depressurizing in an uncontrolled manner or if the RCS pressure is increasing the operator is directed to return to Step 1, since the initial steps in EOP-1.0A should be rechecked.

Eventually, the faulted SG will blow down to atmospheric pressure and dry out, RCS pressure will stabilize or increase, and all SI termination criteria in EOP-1.0A should be met. If the operator proceeds past Step 9 in EOP-1.0A with a depressurizing SG, he could be directed to EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION, and encounter more restrictive SI termination criteria than necessary.

With a LOCA and no faulted SG the SG pressure could be decreasing slightly. This is considered a "stable" SG pressure. The concern addressed by this step is the presence of a secondary side break in which the faulted SG is still depressurizing in an uncontrolled manner. If this is the case, the SI flow reduction criteria may not be met at the time the check is encountered, and the operator should return to Step 1 in EOP-1.0A, and not proceed to EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION, until all SG pressures have been stabilized or are increasing and RCS pressure has stabilized or is decreasing.

"Uncontrolled" means not under the control of the operator, AND incapable of being controlled by the operator using available equipment.

STEP 10: It is recommended that the diesels not be run for more than 30 minutes unloaded. Diesels should auto-start on an SI signal, but will not load if offsite power is available. If DGs are supplying the safeguards busses, then possibly some additional equipment should be loaded to aid the recovery process.

When the diesel generators are stopped, they are placed in standby to be ready to start either manually or automatically.

Design loading requirements of the additional equipment is provided to aid the operator in ensuring the maximum load of the diesel generator is not exceeded, if it is supplying the safeguards bus.

Comments / Reference: EOP-1.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-1.0A
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 9	PAGE 33 OF 44

ATTACHMENT 4
PAGE 10 OF 21

BASES

The process sampling system is used to obtain samples required for this step. The ability to obtain RCS, or recirculation sump samples depends on RCS activity conditions, and the resulting radiation exposure that may be incurred while collecting and analyzing the samples. If radiological conditions prohibit obtaining a sample, Chemistry will notify control room personnel of the inability to obtain the sample and the Plant Staff should be notified to determine the appropriate action to support the accident recovery sequence. Alternate indications may be available to assist in applying engineering judgment for assessing Unit conditions. If it is determined that a sample is required to support recovery actions, then the Contingency Sampling Plan per CHM-111, Primary Chemistry Accident Assessment Sampling Program may be initiated with Plant Staff approval. Use of the Contingency Sampling Plan should be evaluated against the requirements of 10CFR50.54x prior to use.

An evaluation of plant equipment available following a LOCA is necessary to determine long-term recovery actions. Hence, this evaluation is initiated at this time and any additional equipment that would assist in the plant recovery is started.

STEP 12: The operator should stay in EOP-1.0A only for loss of reactor coolant accidents for which the RCS pressure is less than the RHR pump shutoff head and flow from the RHR pumps has been verified. The RHR pump flow should be verified even though the RCS pressure is less than 425 PSIG, which is the shutoff head pressure of the pumps plus allowances for normal channel accuracy and (with an adverse containment) post-accident transmitter errors. Since the post accident transmitter errors are added on to determine the pressure requirement, the actual plant pressure may be significantly less. For any break in the RCS for which the RCS pressure remains above the shutoff head pressure of the RHR pumps, the operator should transfer to procedure EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION. If the RCS pressure is less than the RHR pump shutoff head but RHR pump flow into the RCS cannot be verified, the operator should also transfer to EOS-1.2A. From this point on EOS-1.2A would be used for plant recovery.

STEP 13: Since the RHR pumps are injecting at this time, this indicates the presence of a large break LOCA and, hence, the eventual transfer to cold leg recirculation. When the switchover level in the RWST is reached, the operator should immediately go to EOS-1.3A, TRANSFER TO COLD LEG RECIRCULATION, to maintain coolant flow to the core. If, however, the switchover setpoint has not been reached when the operator encounters this step he is instructed to return to step 11. Here he would continue his evaluation of plant status while waiting for RWST level to reach the switchover setpoint.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	1		
	Group	1		
	K/A	000040.AK1.02		
Level of Difficulty: 3	Importance Rating	3.2		

Steam Line Rupture: Knowledge of the operational implications of the following concepts as they apply to the Steam Line Rupture: Leak rate versus pressure change

Question # 40

Unit 1 is in Mode 3. A Safety Valve has failed open on SG 1-02.

- RCS pressure indicates 2200 psig
- SG 1-02 pressure indicates 1000 psig
- Main Steam flow on SG 1-02 indicates 200,000 lbm/hr

Subsequently:

- A complete MSL Isolation has occurred
- RCS pressure lowers to 1700 psig
- SG 1-02 pressure lowers to 500 psig

Neglecting any change in steam density, Main Steam flow on SG 1-02 will indicate approximately __(1)__.

In accordance with EOP-2.0A, Faulted Steam Generator Isolation, AFW flow to SG 1-02 should be reduced to __(2)__.

A. (1) 100,000 lbm/hr
(2) 0 gpm

B. (1) 100,000 lbm/hr
(2) 100 gpm

C. (1) 141,000 lbm/hr
(2) 0 gpm

D. (1) 141,000 lbm/hr
(2) 100 gpm

Answer: C

K/A Match: K/A match due to requiring the ability to determine the reduction in leak rate of a SG as SG pressure lowers.

Explanation:

A. Incorrect. 1st part is incorrect but plausible because the square root of the SG DP is not taken into account (500/1000) x 200 = 100. 2nd part is correct (See C).

B. Incorrect. 1st part is incorrect but plausible (see A). 2nd part is incorrect but plausible because if all SGs were Faulted ECA-0.2A, Uncontrolled Depressurization of All SGs would direct reducing AFW flow to 100 gpm.

C. Correct. 1st part is correct, the DP between SG 1-02 and atmospheric pressure has dropped from 1000 psid to 500 psid. Taking the square root of this ratio results in a leak rate of 141 gpm $(500/1000)^{1/2}$ x 200 = 141. 2nd part is correct, EOP-2.0A directs reducing AFW flow to zero on a faulted SG.

D. Incorrect. 1st part is correct (see C). 2nd part is incorrect but plausible (see B).

Technical Reference(s)	Generic Fundamentals	Attached w/ Revision # See Comments / Reference
	EOP-2.0A	

Proposed references to be provided during examination: ____________________

Learning Objective: Given an event that results in an excessive increase in secondary steam flow transient, **DISCUSS** accident analysis assumptions, predicted plant response, and analysis conclusions as described in the Final Safety Analysis Report. (MCO.TA8.OB02)

Question Source: Bank # ____________
Modified Bank # 23157 (Note changes or attach parent)
New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ____________

Comments / Reference: Bank 23157	Revision:

Unit 1 was operating at 100% power when a Steam Generator tube leak occurred. A manual Safety Injection was initiated.

The conditions just prior to the Safety Injection were:

- RCS pressure 2200 psig and decreasing
- SG pressures at 800 psig
- Primary to secondary leakage was calculated to be 200 gpm.

The conditions following the Safety Injection are:

- RCS pressure 1700 psig and decreasing
- SG pressures at 1000 psig

Primary to secondary leakage following the SI is approximately ...

A. 100 gpm.

B. 124 gpm.

C. 141 gpm.

D. 176 gpm.

Answer: C

Answer Explanation
A. Plausible if the square root of the DP is not taken into account (700/1400) x 200 = 100.
B. Plausible if the differences between SG and RCS pressures are compared (800/1000) x (1700/2200) x 200 = 124.
C. The DP has dropped from 1400 psid to 700 psid. Taking the square root of this ratio results in a leak rate of 141 gpm $(700/1400)^{1/2}$ x 200 = 141.
D. Plausible since this is taken from a ratio, but the ratio is only RCS pressure $(1700/2200)^{1/2}$ x 200 = 176.

-

Comments / Reference: Bank 23157	Revision:

Question 61 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	23157
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	Unit 1 was operating at 100% power when a Steam Generator tube leak occurred. A manual Safety Inj
K/A:	038.EK1.02
Question Reference:	
SRO:	
Comments:	R/S20E15; R/S22E16; R/S23E25; R/S24E25; R/S25E25, S27E26 REF: EOP-3.0A; MCO.TAB

Comments / Reference: Generic Fundamentals	Revision: 3.5

Example:

A pipe with a 4 inch inner diameter contains water that flows at an average velocity of 14 feet per second. Calculate the volumetric flow rate of water in the pipe.

Solution:

Using the volumetric flow rate equation:

$$\dot{V} = (\pi r^2)\mathrm{v}$$

Volumetric Flow Rate and Pressure Relationship

Velocity is a function of the square root of the change in pressure drop; doubling the flow velocity doubles the volumetric flow rate, which quadruples the pressure drop. This relationship holds well for water, but less accurately for gases.

$$\dot{V} \propto \sqrt{DP} \quad \text{or} \quad \frac{\dot{V}_{final}}{\dot{V}_{initial}} = \frac{\sqrt{DP_{final}}}{\sqrt{DP_{initial}}}$$

Other considerations such as the friction factor, Reynolds number, and viscosity, influence this relationship but not significantly for small pressure drops and flow rates, which are discussed in more detail later in this module.

Example:

A 20 gpm leak to atmosphere has developed from a cooling water system that is operating at 200 pounds per square inch gauge (psig). If pressure decreases to 50 psig, what is the new leak rate?

Solution:

The change in pressure is now one quarter of what it was. The square root of 0.25 is 0.5. Therefore, flow and velocity decrease by half or to 10 gpm.

Mathematically:

$$\frac{\dot{V}_{final}}{\dot{V}_{initial}} = \frac{\sqrt{DP_{final}}}{\sqrt{DP_{initial}}} \text{ therefore } \frac{\dot{V}_{final}}{20} = \frac{\sqrt{50}}{\sqrt{200}}$$

$$\dot{V}_{final} = (20)\frac{\sqrt{50}}{\sqrt{200}} = (20)(0.5) = 10$$

Mass Flow Rate

The mass flow rate ($\dot{m}$) of a fluid system is a measure of the mass of fluid passing a point in the system per unit time. The mass flow rate relates to

Rev 3.5 6

Comments / Reference: EOP-2.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 9	PAGE 20 OF 121

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOP-0.0A REACTOR TRIP OR SAFETY INJECTION

1. RCP TRIP CRITERIA

NOTE: ABN-101, REACTOR COOLANT PUMP TRIP/MALFUNCTION criteria for tripping an RCP is applicable during use of the Emergency Procedures.

Trip all RCPs if BOTH conditions listed below occur:

a. RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
b. CCP or SI pump - AT LEAST ONE RUNNING

2. SHUTDOWN MARGIN CRITERIA
Emergency borate per ABN-107 if either of the following conditions below occur:

- Two or more control rods NOT fully inserted (1800 gallons of 7000 ppm boric acid for each control rod not fully inserted).
- Control rod position indication is NOT available (3600 gallons of 7000 ppm boric acid).

NOTE: During the performance of the immediate operator actions, AFW FOP verbalization is neither required nor expected.

3. CONTROL AFW FLOW TO MAINTAIN ADEQUATE HEAT SINK

- Ensure both MDAFWPs started following a Reactor Trip with NO Blackout or SI actuation. (Start TDAFWP, if necessary)
- Ensure AFW flow throttled following a Reactor Trip/SI (normally 150 gpm to 200 gpm).
  AND
  Maintain total AFW flow GREATER THAN 460 gpm UNTIL at least ONE SG NR Level greater than 43%(50% for ADVERSE CONTAINMENT).
- IF any SG identified as faulted, THEN stop AFW flow to the SG.
- IF any SG identified as ruptured, THEN:
  Stop AFW flow after ruptured SG level greater than 43%(50% for ADVERSE CONTAINMENT).
  AND
  Control AFW flow to maintain ruptured SG level greater than 43%(50% for ADVERSE CONTAINMENT).
- IF BOTH MDAFWPs are running with flow THEN, secure the TDAFWP.

4. AFW SUPPLY SWITCHOVER CRITERION
IF CST level decreases to less than 10%, THEN switch to alternate AFW water supply per ABN-305, AUXILIARY FEEDWATER SYSTEM MALFUNCTION.

5. RCP SEAL INJECTION FLOW CRITERION
Ensure 6 gpm to 13 gpm seal injection flow to all RCPs UNLESS isolated by ERG actions.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 4	Tier	1		
	Group	1		
	K/A	000015.AK1.01		
Level of Difficulty: 3	Importance Rating	4.4		

Reactor Coolant Pump Malfunctions: Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Natural circulation in a nuclear reactor power plant

Question # 41

Given the following conditions:

- While operating at 100% power all RCPs tripped
- EOS-0.1A, Reactor Trip Response, is being performed
- SG levels are 40%
- SG pressure is 1047 psig

Indication that Natural Circulation flow is NOT ADEQUATE is supported by RCS cold leg temperatures indicating __(1)__.

To enhance Natural Circulation flow, SG __(2)__ should be raised.

A. (1) 552ºF
(2) level

B. (1) 552ºF
(2) pressure

C. (1) 561ºF
(2) level

D. (1) 561ºF
(2) pressure

Answer: C

K/A Match: K/A match due to requiring knowledge of indications of natural circulation flow flowing a loss of all RCPs.

Explanation:

A. Incorrect. First part is incorrect, but plausible since the RCS must be subcooled by more than 25ºF, but that is based on RCS pressure, not SG pressure. Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since changing SG pressure could enhance natural circulation, but by lowering it due to increasing steaming rate.

C. Correct. First part is correct. To support indication of natural circulation, RCS cold leg temperatures should be at saturation temperature for SG pressure, so cold leg temperature being 9ºF above saturation is indication that natural circulation does not exist. Saturation temperature for 1047 psig is 552F. Second part is correct. Raising SG level adds cold water to the SG, increasing the difference in temperature between the heat source and heat sink, increasing the likelihood of natural circulation.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B)

Technical Reference(s)	EOS-0.1A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: **Steam Tables**

Learning Objective: **ANALYZE** the Natural Circulation generic issue in the ERG network and proper operator response per the applicable Executive Volume. (ERG.XD5.OB04)

Question Source:
- Bank #
- Modified Bank # 23172 (Note changes or attach parent)
- New

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 10
- 55.43

Comments / Reference: Bank 23172	Revision:

In accordance with EOS-0.1A, Reactor Trip Response, which of the following supports the indication that natural circulation is occurring?

A. SG pressure rising and Pressurizer pressure rising

B. SG pressure rising and Pressurizer pressure stable

C. RCS hot leg temperatures at saturation temperature for SG pressure

D. RCS cold leg temperatures at saturation temperature for SG pressure

Answer: D

Answer Explanation
A. Incorrect - Plausible as these are parameters that are referenced but not for this B. Incorrect - Plausible as these are parameters that are referenced but not for this C. Incorrect - Plausible as these are parameters that are referenced but not for this D. Correct - EOS-0.1

Question 78 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	23172
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	In accordance with EOS-0.1A, Reactor Trip Response, which of the following supports the indication
K/A:	E09.EK2.2
Question Reference:	EOS-0.1
SRO:	
Comments:	R/S21E16; R/S22E16

Comments / Reference: EOS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 9	PAGE 25 OF 40

ATTACHMENT 3
PAGE 1 OF 1

NATURAL CIRCULATION VERIFICATION

The following conditions support or indicate natural circulation flow:

- ☐ RCS subcooling - GREATER THAN 25°F.
- ☐ SG pressures - STABLE OR DECREASING.
- ☐ RCS hot leg temperatures - STABLE OR DECREASING.
- ☐ Core exit TCs - STABLE OR DECREASING.
- ☐ RCS cold leg temperatures - AT SATURATION TEMPERATURE FOR SG PRESSURE.

Comments / Reference: EOS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 9	PAGE 40 OF 40

ATTACHMENT 4
PAGE 15 OF 15

BASES

ATTACHMENT 3

Immediately following a loss of forced reactor coolant flow, RCS pressure and temperature undergo transient variations depending upon the specific initiating event and automatic control system response. Approximately 5 to 10 minutes after a loss of forced reactor coolant flow, the RCS should reach an equilibrium condition as the free convection heat transfer rates from core to reactor coolant and reactor coolant to secondary coolant begin to equilibrate. The action to verify natural circulation flow is included in the ERGs after ECCS flow is terminated. If the ECCS system is in operation, ECCS flow may affect the indications used to confirm Natural Circulation.

The following symptoms are used to verify natural circulation flow:

- RCS subcooling should be greater than instrument inaccuracies.
- The core exit TCs, RCS hot leg temperatures and SG pressures should be decreasing slowly with time, as core decay heat falls off.
- With SG pressures held relatively constant, the RCS cold leg temperatures should remain relatively constant at or slightly above the saturation temperature for the SG pressures being maintained.

In addition to the symptoms used in this attachment, the following symptoms can provide additional confirmation of natural circulation flow: the hot-to-cold leg temperature difference should be approximately equal to the full-power forced convection temperature difference, and the core exit average temperature (core exit TCs averaged reading) should be higher than the average cold leg temperature. This averaged reading should also decrease as core decay heat falls off, in step with core exit TC, hot leg temperature, and SG pressure readings in all active loops. To facilitate the verification of transient equilibrium attainment in the natural circulation process, the Natural Circulation parameters should be recorded at regular intervals beginning as soon as instructed in the ERGs.

ATTACHMENT 4

The Bases attachment provides a discussion for the steps and attachments of this procedure. The information that forms the basis steps and attachments has been taken from the WOG ERG Background Information or from specific CPNPP operating experience or information.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 4	Tier	1		
	Group	1		
	K/A	000025.G.2.4.4		
Level of Difficulty: 3	Importance Rating	4.5		

Loss of Residual Heat Removal System: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question # 42

Given the following conditions:

- RCS in Mid-Loop operations in preparation for Refueling Outage
- RCS cooldown on Train A RHR in progress
- RCS temperature 150°F stable
- Mansell indication 78” above core plate lowering
- RCS leakage has been reported

Based on the above conditions, which of the following procedures should be entered?

A. ABN-103, Excessive Reactor Coolant Leakage

B. ABN-104, Residual Heat Removal System Malfunction

C. ABN-108, Shutdown Loss of Coolant

D. ABN-909, Spent Fuel Pool/Refueling Cavity Malfunction

Answer: B

K/A Match: K/A match due to requiring knowledge of entry conditions to ABNs for an RCS leak occurring while operating on RHR.

Explanation:

A. Incorrect. Plausible since ABN-103 is entered for an RCS leak in MODES 1, 2, and 3 with pressure above 1000 psig, but the unit is in MODE 5 or 6.

B. Correct. ABN-104 is entered for an RCS leak in MODE 5 with the RCS not filled or any time at reduced inventory.

C. Incorrect. Plausible since the RCS is Mode 5 or 6 and ABN-108 would be the correct procedure for an RCS leak if the RCS was still full.

D. Incorrect. Plausible since ABN-909 would be entered if the Refueling Cavity was filled and refueling activities were actually in progress.

Technical Reference(s)	ABN-103	Attached w/ Revision # See Comments / Reference
	ABN-104	
	ABN-108	
	ABN-909	

Proposed references to be provided during examination: ______

Learning Objective: **DETERMINE** the appropriate procedural section of ABN-104, Residual Heat Removal System Malfunction or ABN-108, Shutdown Loss of Coolant for an RCS or RHR malfunction. (ABN.104.OB01)

Question Source: Bank # 32761
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2018 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: 2018 NRC Exam Q91	Revision:

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
072 Area Radiation Monitoring	Group #	2
	K/A #	2.4.8
2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	Rating	4.5
	QREV	6

Question 91

Unit 1 has just transitioned to Mode 4 in preparation for a refueling outage. RCS temp is 345°F. The RO has just validated increasing containment radiation, temperature, pressure, and humidity. Coincident with the validated containment readings, an electrical fault has tripped the running RHR pump.

As the US you are expected to initially enter which of the following procedures?

A. ABN-108, Shutdown Loss of Coolant

B. ABN-104, Residual Heat Removal System Malfunction

C. ABN-103, Excessive Reactor Coolant Leakage

D. EOP 0.0, Reactor Trip or Safety Injection

Answer: A

Explanation:
A is correct because since you've just entered Mode 4 according to ABN-108 this is the correct procedure. Also, the loops would be assumed to be filled due to not being at reduced inventory yet.
B is wrong because this would be the procedure if in Mode 5 and RCS loops not filled
C is wrong because this is the correct procedure if in Modes 1, 2, 3 and RCS >1000 psig
D is wrong because see A. This is plausible if an applicant thinks they're going to heat up into Mode 3 they may need to enter this initially to allow an entry into EOP-1, Loss of Reactor or Secondary Coolant. EOP are allowed to be used outside of Mode 1, 2, and 3 if a line by line analysis is done while performing the procedure. Not needed in this case because the ABN should have sufficient guidance.

Technical References:
ABN-108, Shutdown Loss of Coolant, Rev 4, p. 2

References to be provided to applicants during exam: None.

Learning Objective: DETERMINE the appropriate procedural section of ABN-104, Residual Heat Removal System Malfunction or ABN-108, Shutdown Loss of Coolant for an RCS or RHR malfunction. (LO21.ABN.104.OB01)

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(5)	

Comments / Reference: ABN-103	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-103
EXCESSIVE REACTOR COOLANT LEAKAGE	REVISION NO. 10	PAGE 2 OF 37

1.0 APPLICABILITY

This procedure specifies the actions to be taken in the event of excessive reactor coolant leakage and applies to MODES 1, 2 and 3 with RCS pressure greater than 1000 psig.

This procedure is common to both units. The specific unit designator (1 or 2) is represented within these instructions by the symbol "u". The appropriate unit digit may be substituted for this symbol to obtain the unit specific equipment number. (Example u-FK-121 represents 1-FK-121 for Unit 1 and 2-FK-121 for Unit 2.)

NOTE: The applicable recovery procedures for RCS leakage in various operational MODES are as follows:

• MODES 1, 2, 3 (RCS PRESS > 1000 psig)	ABN-103,	Excessive Reactor Coolant Leakage
• MODE 3 (RCS PRESS <1000 psig) MODES 4, 5 (RCS Loops filled)	ABN-108,	Shutdown Loss of Coolant
• MODE 5 (RCS Loops NOT filled including anytime at reduced inventory)	ABN-104,	Residual Heat Removal System Malfunction
• MODE 6 (Cavity NOT filled)	ABN-104,	Residual Heat Removal System Malfunction
• MODE 6 (Cavity flooded)	ABN-909,	Spent Fuel Pool/Refueling Cavity Leakage

Comments / Reference: ABN-104	Revision: 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 2 OF 134

1.0 APPLICABILITY

[C] This procedure describes the actions to be taken in the event of an RHR malfunction or loss of decay heat removal capacity in MODES 4, 5 and 6. This procedure also describes the actions to be taken for a loss of RCS inventory when the RCS is NOT filled (RCS Level at less than 872'10" (866'3") which is approximately 35% (17%) cold Calibrated Pressurizer Level OR RCS pressure < 100 psig) and when the RCS is at reduced inventory (RCS level less than 80 inches above core plate 829'8"). Once the RCS has been drained, the loops are not considered filled until vacuum fill has been performed AND RCS pressure has been raised ≥100 psig. This procedure applies to Unit 1 and Unit 2.

This procedure is common to both units. The specific unit designator (1 or 2) is represented within these instructions by the symbol "u". The appropriate unit digit may be substituted for this symbol to obtain the unit specific equipment number. (Example u-FK-121A represents 1-FK-121A for Unit 1 and 2-FK-121A for Unit 2).

Other differences between units may be identified by including Unit 2 information in parentheses immediately after Unit 1 information.

The symbol [R] has been located throughout this procedure where real or potential radiation hazards are positively identified. This identification technique should not preclude workers from following good radiation work practices throughout this procedure to ensure their occupational exposure is maintained As Low As Is Reasonably Achievable (ALARA).

NOTE:	The applicable recovery procedures for RCS leakage in various operational MODES are as follows:

- MODES 1, 2, 3 (RCS Press greater than 1000 psig) — ABN-103, Excessive Reactor Coolant Leakage
- MODE 3 (RCS press less than 1000 psig)MODES 4, 5, (RCS Loops filled) — ABN-108, Shutdown Loss of Coolant
- MODE 5 (RCS Loops NOT filled including anytime at reduced inventory) — This Procedure
- MODE 6 (Cavity NOT filled) — This Procedure
- MODE 6 (Cavity flooded) — ABN-909, Spent Fuel Pool/Refueling Cavity Leakage

Section 1.0

Comments / Reference: ABN-108	Revision: 4

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-108
SHUTDOWN LOSS OF COOLANT	REVISION NO. 4	PAGE 2 OF 63

1.0 APPLICABILITY

[C] [02494]

This procedure describes actions for excessive Reactor Coolant System leakage in MODE 3 (below 1000 psig), 4, or 5 (RCS not drained). (RCS loops are filled when RCS level is greater than 872'10" (866'3") which is approximately 35% (17%) Cold Calibrated Pressurizer Level AND RCS Pressure ≥100 psig.

This procedure is common to both units. The specific unit designator (1 or 2) is represented within these instructions by the symbol "u". The appropriate unit digit may be substituted for this symbol to obtain the unit specific equipment number. (Example u-FK-121 represents 1-FK-121 for Unit 1 and 2-FK-121 for Unit 2.)

Other differences between units may be identified by including Unit 2 information in parentheses immediately after Unit 1 information.

> NOTE: The symbol [R] has been located throughout this procedure where real or potential radiation hazards are positively identified. This identification technique should not preclude the worker from following good radiation work practices throughout this procedure to ensure his or her occupational exposure is maintained As Low As Is Reasonably Achievable (ALARA).

> NOTE: The applicable recovery procedures for RCS leakage in various operational MODEs are as follows:

- MODEs 1, 2, 3 (RCS Press greater than 1000 psig) — ABN-103, Excessive Reactor Coolant Leakage
- MODE 3 (RCS press less than 1000 psig) MODES 4, 5, (RCS Loops filled) — ABN-108, Shutdown Loss of Coolant
- MODE 5 (RCS Loops NOT filled including anytime at reduced inventory) — ABN-104, Residual Heat Removal System Malfunction
- MODE 6 (Cavity NOT filled) — ABN-104, Residual Heat Removal System Malfunction
- MODE 6 (Cavity flooded) — ABN-909, Spent Fuel Pool/Refueling Cavity Leakage
- Shutdown Loss of Coolant (Section 2.0)

Comments / Reference: ABN-909	Revision: 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-909
SPENT FUEL POOL/REFUELING CAVITY MALFUNCTION	REVISION NO. 9	PAGE 2 OF 36

1.0 APPLICABILITY

This procedure describes the actions to be taken in the event of excessive leakage within the refueling cavity and spent fuel pools, loss of spent fuel pool cooling, or a level instrument malfunction occurs. This procedure is applicable whenever spent fuel is being handled or stored within the refueling cavity or spent fuel pool(s).

This procedure is common to both units. The specific unit designator (1 or 2) is represented within these instructions by the symbol "u". The appropriate unit digit may be substituted for this symbol to obtain the unit specific equipment number (Example: u-PI-5470A represents 1-PI-5470A for Unit 1 and 2-PI-5470A for Unit 2).

Other differences between units may be identified by including Unit 2 information in parentheses immediately after Unit 1 information.

- Section 2.0, Refueling Cavity/SFP Leakage
- Section 3.0, Loss of SFP Cooling
- Section 4.0, Level Instrument Malfunction

Section 1.0

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	1		
	Group	1		
	K/A	000026.G.2.2.44		
Level of Difficulty: 3	Importance Rating	4.2		

Loss of Component Cooling Water: 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.

Question # 43

Given the following conditions:

- The Unit is in MODE 5
- RCS temperature is 165ºF
- PRZR is solid
- RCS pressure is being maintained between 260 and 285 psig
- RHR System is in service at this time
- CCW to the RHR System is lost
- RCS temperature begins to rise
- ABN-104, Residual Heat Removal System Malfunction, in progress

Based on conditions above, PCV-131, Letdown Pressure Control Valve, will throttle __(1)__ to maintain RCS pressure constant.

In accordance with ABN-104, the crew must stop all running RHR Pumps and isolate RHR if RCS temperature exceeds __(2)__.

A. (1) OPEN
(2) 350°F

B. (1) OPEN
(2) 400°F

C. (1) CLOSED
(2) 350°F

D. (1) CLOSED
(2) 400°F

Answer: A

K/A Match: K/A match due to requiring knowledge of how the plant components will respond to a loss of CCW water and understanding of what is required per the procedure when RCS temperature exceeds a certain value.

Explanation:

A. Correct. First part is correct. Loss of CCW to the RHR System will cause RCS temperature to rise which will cause pressure to rise. When solid plant pressure control in progress, PCV-131 must be opened to maintain Reactor Coolant System pressure constant. Second part is correct, per ABN-104, Section 4, Step 3 and RNO, RHR pumps must be secured and RHR isolated when RCS temperature exceeds 350°F.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible because per ABN-104 RCS pressure is required to be maintained below 400 psig. 350°F and 400 psig could be easily confused.

C. Incorrect. First part is incorrect, but plausible if thought PCV-131 throttles to control pressure downstream of the valve instead of upstream. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-104	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to a loss of RCS Temperature/Flow Control in accordance with ABN-104, RHR System Malfunctions. (ABN.104.OB03)

Question Source: Bank # ______
Modified Bank # 23349 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

Comments / Reference: Bank 23349	Revision:

Given the following conditions:

- The Unit is in MODE 5.
- Reactor Coolant System temperature is 165ºF.
- The Pressurizer is solid.
- Reactor Coolant System pressure is being maintained between 260 and 285 psig.
- Residual Heat Removal System is in service at this time.
- Component Cooling Water to the Residual Heat Removal System is lost.

Which of the following describes:

1) How PCV-131, Letdown Pressure Control Valve, will respond in attempting to maintain RCS pressure constant; and,

2) How the value of letdown flow changes?

A. 1) PCV-131 will throttle CLOSED
2) Letdown flow from RHR will decrease

B. 1) PCV-131 will throttle CLOSED
2) Letdown flow from RHR will increase

C. 1) PCV-131 will throttle OPEN
2) Letdown flow from RHR will increase

D. 1) PCV-131 will throttle OPEN
2) Letdown flow from RHR will decrease

Answer: C

Answer Explanation

Comments / Reference: Bank 23349	Revision:

A.	Plausible because PCV-131 will open, however, letdown flow will increase.
B.	Plausible if thought that a loss of CCW flow would close PCV-131, in this case Letdown flow would decrease.
C.	Loss of CCW to the RHR System will cause RCS temperature to rise. PCV-131 must be opened to maintain Reactor Coolant System pressure constant. This will cause Letdown flow to increase.
D.	Plausible because letdown flow will increase, however, PCV-131 will open because RCS temperature is rising.

Question 94 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	23349
User-Defined ID:	ILOT6161
Cross Reference Number:	IPO.005.OB04.004
Topic:	Given the following conditions: The Unit is in MODE 5. Reactor Coolant System temperature is 165º
K/A:	SF2.004.K1.30
Question Reference:	
SRO:	
Comments:	LC18 Audit; R/S21E13; R/S22E14; R/S23E23; R/S24E23, R/S27E23 REF: ABN-104; ABN-121; OP51.SYS.RH1.LN

Comments / Reference: ABN-104	Revision: 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 5 OF 134

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify RCS pressure - STABLE	IF RCS is water solid AND letdown is aligned from RHR system, THEN reduce charging flow to minimum as follows: a. Manually adjust the following, as applicable, to slowly reduce charging flow to 32 GPM WHILE maintaining RCP seal injection flows between 6 - 13 gpm: • CCP running - use u-FK-121A, CCP FLO CHRG CTRL • PDP running - use u-SK-459A, PDP SPD CTRL • u-HC-182, RCP SEAL WTR PRESS CTRL b. Manually control u-PK-131, LTDN HX OUT PRESS CTRL, to stabilize RCS pressure. c. Refer to ABN-121.

Section 2.3

Comments / Reference: ABN-104	Revision: 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 23 OF 134

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
CAUTION: The RCS temperature must be maintained less than 350°F while the RHR system is in service. The RCS pressure shall be maintained less than 400 psig.	
3 ☐ b. Verify RCS temperature does NOT - EXCEED 350°F.	b. Perform the following: 1) Stop all running RHR pumps. 2) Isolate the RHR suction from the RCS hot legs: • 1/u-8701A, RHRP 1 HL RECIRC ISOL VLV • 1/u-8702A, RHRP 1 HL RECIRC ISOL VLV • 1/u-8701B, RHRP 2 HL RECIRC ISOL VLV • 1/u-8702B, RHRP 2 HL RECIRC ISOL VLV 3) GO TO Section 2.0, this procedure.
☐ 4 Verify cold leg injection valve for running RHR pump - OPEN: • 1/u-8809A, RHR TO CL 1 & 2 INJ ISOL VLV • 1/u-8809B, RHR TO CL 3 & 4 INJ ISOL VLV	Manually open valve(s) as necessary.

Section 4.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	1		
	K/A	000027.AK2.03		
Level of Difficulty: 2	Importance Rating	2.6		

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

Question # 44

Given the following conditions:

- Unit 1 100% power
- RCS Pressure 2190 psig lowering
- RC Loop 1 PRZR Spray Valve 455B RED light extinguished and GREEN light lit
- RC Loop 4 PRZR Spray Valve 455C RED Light lit and GREEN light lit
- PRZR Master Pressure Controller demand 40% lowering
- All PRZR Backup and Control Heaters ON

Per ABN-705, Pressurizer Pressure Malfunction…

the Reactor Operator should take manual control of __(1)__.

if RCS pressure continues to lower, trip the Reactor and __(2)__.

A. (1) 1-PK-455A, PRZR Master Pressure Controller
(2) stop RCPs

B. (1) 1-PK-455A, PRZR Master Pressure Controller
(2) initiate Safety Injection

C. (1) 1-PK-455C, RC Loop 4 PRZR Spray Valve Controller
(2) stop RCPs

D. (1) 1-PK-455C, RC Loop 4 PRZR Spray Valve Controller
(2) initiate Safety Injection

Answer: C

K/A Match: K/A match due to requiring knowledge of actions required in the ABN for a Pressurizer Pressure malfunction related to a failed Spray valve controller.

Explanation:

A. Incorrect. First part is incorrect, but plausible (see B). Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible if thought that the Master Pressurizer Pressure Controller had failed, however, in this case the Master controller has responded to turn on all PRZR heaters in response to the lowering RCS pressure. Second part is incorrect, but plausible because with RCS pressure lowering it is reasonable to believe the ABN would direct SI initiation, however, the ABN directs stopping of RCPs as necessary to stop RCS Spray flow.

C. Correct. First part is correct. With only one spray valve indicating in the intermediate position and all PRZR heaters on the failure is an indication of an individual spray valve controller failure. The ABN directs taking manual control of the failed spray valve controller and attempting to close. Second part is correct. Step 1 RNO of the ABN directs stopping RCPs as necessary to stop RCS spray flow and stop the pressure reduction.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-705	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____

Learning Objective: **ANALYZE** the response to a Pressurizer Pressure Instrument Malfunction in accordance with ABN-705, Pressurizer Pressure Malfunction. (ABN.705.OB01)

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 X

10 CFR Part 55 Content:
- 55.41 7
- 55.43 ____

Comments / Reference: ABN-705	Revision: 13

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-705
PRESSURIZER PRESSURE MALFUNCTION	REVISION NO. 13	PAGE 8 OF 26

3.0 Pressurizer Spray Valve Failure

3.1 Symptoms

a. Annunciator Alarms

- PRZR 1 OF 4 PRESS LO (5B-3.4)
- PRZR 1 OF 4 SI PRESS LO (5B-4.4)
- PRZR LO PRESS PORV 456 BLK (5B-1.6)
- PRZR LO PRESS PORV 455A BLK (5B-2.6)

b. Plant Indication

- Spray valve indicated open when not called for by master controller.

3.2 Automatic Actions

a. Control response for failed open spray valve(s)

1) Control and backup heaters come on.

- 1/u-PCPR, PRZR CTRL HTR GROUP C
- 1/u-PCPR1, PRZR BACKUP HTR GROUP A
- 1/u-PCPR2, PRZR BACKUP HTR GROUP B
- 1/u-PCPR3, PRZR BACKUP HTR GROUP D

NOTE:	A reactor trip at high power and low pressure may result in an SI due to AFW flow.

2) Reactor trip at 1880 psig.

3) Safety Injection at 1820 psig.

b. Control response for failed close spray valves.

- PRZR PORV may open on a pressure transient.

Section 3.0

Comments / Reference: ABN-705	Revision: 13

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-705
PRESSURIZER PRESSURE MALFUNCTION	REVISION NO. 13	PAGE 9 OF 26

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 CLOSE Pressurizer Spray Valve(s) • u-PK-455B, RC LOOP 1 PRZR SPR VLV CTRL • u-PK-455C, RC LOOP 4 PRZR SPR VLV CTRL	a. IF Pressurizer pressure is decreasing in an uncontrolled manner, THEN PERFORM the following: 1) TRIP the Reactor 2) STOP RCP(s) as necessary to stop spray flow. 3) GO TO EOP-0.0A/B. b. INITIATE load reduction per IPO-003A/B as directed by Shift Manager. c. ENSURE ALL Pressurizer Heaters - ON d. CONTACT I&C to deenergize affected spray valve(s) (Fail Close): • REMOVE u-PCY-0455B, PRZR PRESS CONTROL DRIVER CARD (u-RK-05 card 0525) • REMOVE u-PCY-0455C, PRZR PRESS CONTROL DRIVER CARD (u-RK-05 card 0527)

Section 3.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	1		
	K/A	000029.EK1.03		
Level of Difficulty: 2	Importance Rating	3.6		

Anticipated Transient Without Scram: Knowledge of the operational implications of the following concepts as they apply to the ATWS: Effects of boron on reactivity

Question # 45

Unit 1 100% power when a Turbine Trip occurs without a Reactor Trip.

The RO adds negative reactivity by Emergency Borating per __(1)__.

The boration is achieved utilizing __(2)__ Boric Acid Transfer Pump(s).

A. (1) ABN-107, Emergency Boration
(2) BOTH

B. (1) ABN-107, Emergency Boration
(2) EITHER

C. (1) FRS-0.1A, Response to Nuclear Power Generation/ATWT
(2) BOTH

D. (1) FRS-0.1A, Response to Nuclear Power Generation/ATWT
(2) EITHER

Answer: C

K/A Match: K/A match due to requiring knowledge of the operational requirements to add negative reactivity (effects of boron on reactivity) from an emergency boration during an ATWT.

Explanation:

A. Incorrect. First part is incorrect, but plausible because ABN-107 would be utilized to Emergency Borate in all other conditions except during an ATWT. Second part is incorrect, but plausible because two pumps are available and it could be thought that ABN-107 requires both pumps the same as FRS-0.1A.

B. Incorrect. First part is incorrect, but plausible (see A) . Second part is incorrect, but plausible because ABN-107 only requires one pump, however, ABN-107 is not the correct procedure to execute the emergency boration.

C. Correct. First part is correct , during an ATWT, the US will direct Emergency Boration via FRS-0.1A, Attachment 1.F, Initiation Emergency Boration. This is the only time Emergency Boration is initiated not using ABN-107. Second part is correct, FRS-0.1A, Attachment 1.F requires the use of all available Boric Acid Transfer Pumps to initiate Emergency Boration.

D. Incorrect. First part is correct (see C). Second part is incorrect , but plausible because the 1-02 Boric Acid Transfer Pump is normally in Pull-Out as it is aligned to draw suction from the X-02 Boric Acid Tank which is dedicated to Unit 2. It could be thought that this pump would only be used during a failure of the 1-01 Boric Acid Pump.

Technical Reference(s)	FRS-0.1A	Attached w/ Revision # See Comments / Reference
	ABN-107	

Proposed references to be provided during examination: ____________________

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRS-0.1A/B, Response to Nuclear Generation/ATWT in accordance with FRS-0.1. (ERG.FS1.OB04)

Question Source: Bank # 77177
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 8
55.43 ________

Comments / Reference: FRS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRS-0.1A
RESPONSE TO NUCLEAR POWER GENERATION/ATWT	REVISION NO. 9	PAGE 4 OF 33

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1F] 4	Initiate Emergency Boration Of RCS:	
	a. CCPs - AT LEAST ONE RUNNING	a. Start one CCP. IF no CCP can be started, THEN start PD pump by performing the following: 1) Establish CCW non-safeguards loop flow. 2) IF CCW flow to RCP thermal barriers is lost, THEN isolate seal injection to affected RCP(s) before starting PD pump. 3) Open charging line isolation valves, 1/1-8105 and 1/1-8106. 4) IF the PD pump is supplied from the VCT, THEN open PD pump suction vent valves: • 1/1-8202A and 1/1-8202B 5) Place PD pump speed controller in manual for 55% demand. 6) Ensure 1/1-8109 - OPEN 7) Start PD pump. 8) Ensure 1/1-8109 - CLOSED 9) Adjust PD pump speed to establish charging flow - GREATER THAN 30 GPM 10) IF seal injection not isolated, THEN adjust seal flow to RCPs to maintain between 6 gpm and 13 gpm.

-CONT 4-

Comments / Reference: FRS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRS-0.1A
RESPONSE TO NUCLEAR POWER GENERATION/ATWT	REVISION NO. 9	PAGE 5 OF 33

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1F] 4	b. Verify Charging flow - GREATER THAN 30 GPM	b. Perform ONE of the following to establish charging flow: • IF CCP running, THEN adjust charging flow control valve to establish charging flow - GREATER THAN 30 GPM • IF PD pump running, THEN adjust PD pump speed to establish charging flow - GREATER THAN 30 GPM • IF Charging flow path can NOT be established, THEN shift charging pump suction to the RWST by performing the following: 1) Open valves 1/1-LCV-112D and 1/1-LCV-112E. 2) Close valves 1/1-LCV-112B and 1/1-LCV-112C. 3) Open the CCP High Head injection flow path. 4) Go to Step 5.
	c. Align boration flowpath by performing the following: 1) Start all available boric acid transfer pump(s). 2) Open emergency boration valve, 1/1-8104. 3) Verify emergency boration flow.	c. Shift charging pump suction to the RWST by performing the following: 1) Open valves 1/1-LCV-112D and 1/1-LCV-112E. 2) Close valves 1/1-LCV-112B and 1/1-LCV-112C. 3) Adjust charging flow control valve to establish maximum charging flow.

Comments / Reference: FRS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRS-0.1A
RESPONSE TO NUCLEAR POWER GENERATION/ATWT	REVISION NO. 9	PAGE 19 OF 33

ATTACHMENT 3
PAGE 3 OF 17

BASES

STEP 3: The MDAFW pumps start automatically on an SI signal or SG low level to provide feed to the SGs for decay heat removal. If SG levels drop below the appropriate setpoint, the TDAFW pump will also automatically start to supplement the MD pumps. The ATWT analyses have shown that actuation of AFW within 60 seconds after the failure to trip provides acceptable results. (Review of background documents indicate that this time is only used to show procedure step priority and is assumed to occur from automatic signals. There is no requirement for manual operator action to satisfy this time.) The 860 gpm gpm flow requirement is indicative of adequate Auxiliary Feedwaterflow (AFW pumps) to meet the minimum flow assumption for an ATWT.

STEP 4: After control rod trip and rod insertion functions, boration is the next most direct manner of adding negative reactivity to the core. The intended boration path here is the most direct one available, not requiring SI initiation. Ensuring that charging flow is > 30 gpm will ensure capability of delivering boron to the core. Starting all available boric acid pumps will ensure that the "Emergency Boration" flowpath is delivering the maximum available boron to the charging system. The intent is to deliver boron to the core at the maximum achievable rate using the normal charging flowpath.

If charging flow cannot be verified through the normal flowpath, direction is given in the RNO column to establish charging flow > 30 gpm.

If a normal charging flowpath cannot be established the operator manually shifts charging suction to the RWST and opens the high head injection valves. This will ensure maximum flow of 2400 ppm boron without initiating SI. The intent is to deliver boron to the core at the maximum achievable rate using this flowpath. SI initiation should be avoided in order to maintain Main Feedwater.

The symbol [1F] has been utilized to identify that Attachment 1.F exists, which allows the actions for initiating emergency boration to be delegated to a Reactor Operator by handing off the attachment. Since the action involves multiple specific actions to accomplish this evolution, having the RO perform the evolution using the attachment in a step-wise manner may benefit the overall ERG performance (e.g., minimize communications, permit SRO directing response and recovery activities to maintain higher level view of effort, provide termination criteria to RO in a written format).

Comments / Reference: ABN-107	Revision: 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-107
EMERGENCY BORATION	REVISION NO. 9	PAGE 9 OF 32

ATTACHMENT 1
PAGE 1 OF 1

EMERGENCY BORATION THROUGH EMERGENCY BORATE VALVE u-8104

[L]

☐ 1. ENSURE a charging pump is running:

- 1/u-APCH1, CCP 1
- 1/u-APCH2, CCP 2
- 1/u-APPD, PDP

2. START a Boric Acid Transfer Pump:

- ☐ 1/u-APBA1, BA XFER PMP 1 - AUTO (AFTER START)
- ☐ 1/u-APBA2, BA XFER PMP 2 - AUTO (AFTER START)

☐ 3. OPEN 1/u-8104, EMER BORATE VLV

☐ 4. VERIFY flow on u-FI-183A, EMER BORATE FLO

☐ 5. VERIFY flow on u-FI-121A, CHRG FLOW

☐ 6. IF EMER BORATE FLOW OR CHRG FLOW CANNOT be verified,
THEN
INITIATE Emergency Boration Flow per another method of ABN-107.

7. WHEN desired to terminate Emergency Boration (Reference Attachment 7 of ABN-107),
THEN
SECURE Emergency Boration by PERFORMING the following:

☐ a) CLOSE 1/u-8104 EMER BORATE VLV.

b) STOP the Boric Acid Transfer Pump started in step 2.

- ☐ 1/u-APBA1, BA XFER PMP 1 - AUTO (AFTER STOP)
- ☐ 1/u-APBA2, BA XFER PMP 2 - AUTO (AFTER STOP)

☐ 8. GO TO Step 8 of ABN-107.

Attachment 1

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	1		
	Group	1		
	K/A	000038.EA1.16		
Level of Difficulty: 2	Importance Rating	4.4		

Steam Generator Tube Rupture: Ability to operate and monitor the following as they apply to a SGTR: S/G atmospheric relief valve and secondary PORV controllers and indicators

Question # 46

Given the following conditions:

- A SGTR has occurred
- The crew has identified the affected SG
- The ruptured SG ARV controller setpoint is being adjusted

The SG ARV controller setpoint should be set to __(1)__ to prevent __(2)__.

A. (1) 1160 psig
(2) challenging the code safety valves while minimizing atmospheric releases

B. (1) 1160 psig
(2) the ruptured SG from depressurizing during the RCS maximum rate cooldown

C. (1) 1125 psig
(2) challenging the code safety valves while minimizing atmospheric releases

D. (1) 1125 psig
(2) the ruptured SG from depressurizing during the RCS maximum rate cooldown

Answer: A

K/A Match: K/A match due to requiring knowledge of how to control the SG ARV during a SGTR event.

Explanation:

A. Correct. First part is correct. 1160 psig is the setpoint selected. Second part is correct. The setpoint should be greater than no-load pressure to minimize atmospheric releases from the ruptured steam generator and less than the minimum safety valve setpoint to prevent lifting of the code safety valves.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible since it is desirable to prevent the SG from depressurizing as a result of the cooldown, but this is accomplished by isolating the SG from the intact SGs.

C. Incorrect. First part is incorrect, but plausible since this is the setpoint normally maintained on the ARVs except during SGTRs and cooldowns. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	EOP-3.0A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____

Learning Objective: Given a procedural step, or sequence of steps from EOP-3.0, **STATE** the purpose/basis for the step(s), in accordance with EOP-3.0, Steam Generator Tube Rupture. (ERG.E3A.OB04)

Question Source: Bank # 18507
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments / Reference: EOP-3.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-3.0A
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 9	PAGE 5 OF 112

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: If the TDAFW pump is the only available source of feed flow, steam supply to the TDAFW pump must be maintained from at least one SG.

CAUTION: At least two SG(s) must be maintained available for the initial RCS cooldown. At least one SG must be maintained available for the subsequent RCS cooldown to RHR system operating conditions.

NOTE: If any SG atmospheric opens the Plant Staff should be notified.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[R] 3	Isolate Flow From Ruptured SG(s):	
	a. Adjust ruptured SG(s) atmospheric controller setpoint to 1160 psig.	
	b. Check ruptured SG(s) atmospheric - CLOSED	b. WHEN ruptured SG pressure less than 1160 psig, THEN verify SG atmospheric closed. IF NOT closed, THEN place SG atmospheric controller in manual and close atmospheric. IF SG atmospheric can NOT be closed, THEN locally close SG atmospheric block valve.

-CONT 3-

Comments / Reference: EOP-3.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-3.0A
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 9	PAGE 59 OF 112

ATTACHMENT 6
PAGE 4 OF 57

BASES

As an alternative cooldown method, one could steam a ruptured steam generator. In addition to increasing radiological releases, this will result in continued primary-to-secondary leakage. If the tube failure is large, the reactor coolant makeup supply could be depleted before RHR system cooling can be established. This may also result in a steam generator overfill condition. Hence, before steaming a ruptured steam generator, one must consider potential radiological consequences, including availability of the condenser, reactor coolant activity, and meteorological conditions, and also the rate of accumulation of water in the ruptured steam generator and reactor coolant makeup supply.

NOTE: The analysis for radiological consequences assumes the block valve for a stuck open Atmospheric Relief Valve is closed within approximately 30 minutes of identifying the stuck open ARV.

STEP 3: Isolation of the ruptured steam generator(s) effectively minimizes release of radioactivity from this generator. In addition, isolation is necessary to establish a pressure differential between the ruptured and non-ruptured steam generators in order to cool the RCS and stop primary-to-secondary leakage.

[R] has been identified to alert operator that this step may be affected by changing radiation fields in plant. Local actions to be performed in plant areas where radiation levels are high could require alternative actions (i.e. limited stay times, continuous Radiation Protection coverage). If the accident involves failed fuel, local operations may be in high radiation areas.

In order to remove heat generated in the primary system the ruptured steam generator pressure and RCS pressure must be maintained greater than the non-ruptured steam generator pressures. As this pressure differential increases, so is the subcooling in the primary system. If sufficient pressure differential cannot be maintained, leakage from the RCS will continue since RCS pressure will remain greater than the ruptured steam generator pressure in order to remove decay heat. In that case, the operator is directed to ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED to minimize this leakage. In order to stop primary-to-secondary leakage, the primary pressure must be reduced to a value equal to that of the ruptured steam generator.

The ARV on the ruptured steam generator should remain available to limit steam generator pressure unless it fails open. This will minimize any challenges to the code safety valve.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	1		
	K/A	WE12.EA2.02		
Level of Difficulty: 3	Importance Rating	3.4		

Uncontrolled Depressurization of all Steam Generators: Ability to determine and interpret the following as they apply to the Uncontrolled Depressurization of all Steam Generators: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

Question # 47

Given the following conditions:

- Unit 2 was at full power when a MSL Break occurred OUTSIDE Containment
- MSIVs will NOT close from the control room
- An Operator was dispatched to locally close the MSIVs per EOP-0.0B, Reactor Trip or Safety Injection
- ECA-2.1B, Uncontrolled Depressurization of All Steam Generators, is entered

A CAUTION statement before Step 2 of ECA-2.1B directs the operator to control AFW flow to maintain a minimum flow of 100 gpm to __(1)__SG with less than a setpoint value of __(2)__.

A. (1) Each
(2) 10% NR level

B. (1) Each
(2) 18% NR level

C. (1) ONLY one
(2) 10% NR level

D. (1) ONLY one
(2) 18% NR level

Answer: A

K/A Match: K/A match due to requiring knowledge of the actions to be taken in response to all SGs depressurizing in an uncontrolled manner.

Explanation:

A. Correct. First part is correct. All SGs to be fed at a minimum of 100 gpm with all SGs faulted. Second part is correct. With no indication of adverse containment conditions, with the break outside containment, the level required is 10%.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible if adverse containment conditions existed, but with break outside containment, no adverse conditions should exist.

C. Incorrect. First part is incorrect, but plausible since any SG level above the minimum value meets heat sink requirements, but all SGs are to be fed. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ECA-2.1B	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **STATE** the bases for operator actions, notes and cautions in ECA-2.1 in accordance with ECA-2.1. (ERG.C21.OB05)

Question Source: Bank # 82372
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam LC26

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: Bank 82372	Revision:

Unit 2 was at full power when a Main Steam Line Break occurred OUTSIDE Containment.

- MSIVs will NOT close from the control room
- An Operator was dispatched to locally close the MSIVs per EOP-0.0B, Reactor Trip Or Safety Injection

Subsequently

- ECA-2.1B, Uncontrolled Depressurization Of All Steam Generators, is entered

A CAUTION statement after step one of this procedure directs the operator to control AFW flow to maintain a minimum flow of 100 gpm to ______ SG with less than ______.

A. Each
10 % Narrow Range level

B. At Least One
10 % Narrow Range level

C. Each
18 % Narrow Range level

D. At Least One
18 % Narrow Range level

Answer: A

Answer Explanation
Explanation: Note: because this is manual control of a normally automatic feature (AFW flow for level control at minimum levels) it meets the intent of EK2.1. A is correct because it states "A minimum AFW flow of 100 gpm must be maintained to each SG with a narrow range level less than 10% (18% FOR ADVERSE CONTAINMENT)." There are no adverse containment conditions because break is outside containment. B is wrong because (see A above). Plausible because could think of it in terms of heat sink but this caution is to prevent tube damage so it is all four SG's (ie EACH). C is wrong because there are no adverse containment conditions given so 18% is wrong but plausible if they don't recognize this from stem information. D is correct because both parts are incorrect (see discussion on other distracters and answer above).

Comments / Reference: Bank 82372	Revision:

Question 37 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	82372
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	Unit 2 was at full power when a Main Steam Line Break occurred OUTSIDE Containment. MSIVs will NOT
K/A:	
Question Reference:	
SRO:	
Comments:	**KA Match:** This question matches the KA by requiring knowledge of the relationship between the uncontrolled depressurization of all SGs and AFW pumps (components).

Comments / Reference: ECA-2.1B	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. ECA-2.1B
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 9	PAGE 4 OF 69

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: A minimum AFW flow of 100 gpm must be maintained to each SG with a narrow range level less than 10% (18% FOR ADVERSE CONTAINMENT).

NOTE: Shutdown margin should be monitored during RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 2	Control AFW Flow To Minimize RCS Cooldown:	
	a. Check cooldown rate in RCS cold legs - LESS THAN 100°F/HR	a. Decrease AFW flow to 100 gpm to each SG. Go to Step 2c.
	b. Check narrow range level in all SGs - LESS THAN 50%	b. Control AFW flow to maintain narrow range level less than 50% in all SGs.
	c. Check RCS hot leg temperatures - STABLE OR DECREASING	c. Control AFW flow or dump steam to stabilize RCS hot leg temperatures.
* 3	Check If RCPs Should Be Stopped:	
	a. RCS subcooling -LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)	a. Go to Step 4. OBSERVE CAUTION PRIOR TO STEP 4.
	b. ECCS pumps - AT LEAST ONE RUNNING • CCP -OR- • SI pump	b. Go to Step 4. OBSERVE CAUTION PRIOR TO STEP 4.
	c. Stop all RCPs.	

Comments / Reference: ECA-2.1B	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. ECA-2.1B
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 9	PAGE 40 OF 69

ATTACHMENT 4
PAGE 2 OF 31

BASES

CAUTION: If AFW flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable AFW flow to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased.

NOTE: This note advises the operator to monitor RCS boron concentration to verify adequate shutdown margin during the cooldown to cold shutdown. Note that since ECCS was in service, RCS boron concentration is expected to be sufficient.

Periodic samples should be taken to monitor shutdown margin, however the operator should not wait for the sample results.

STEP 2: Depending upon the size of the effective break areas for the steam generators, the cooldown rate experienced after reactor trip could exceed 100°F/hr. A reduction of AFW flow to the steam generators has three primary effects:

1) To minimize any additional cooldown resulting from the addition of AFW.

2) To prevent steam generator tube dry out by maintaining a minimum AFW flow to the steam generators, and

3) To minimize the water inventory in the steam generators that eventually is the source of additional steam flow to containment or the environment.

The 100 gpm value is representative of a minimum measurable feed flow to a steam generator.

As steam flow rate drops, the feed flow will eventually increase the steam generator inventory. Feed flow is controlled to maintain steam generator narrow range level less than 50% to prevent overfeeding the steam generators.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	1		
	K/A	000054.AA1.02		
Level of Difficulty: 3	Importance Rating	4.4		

Loss of Main Feedwater: Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): Manual startup of electric and steam-driven AFW pumps

Question # 48

Given the following conditions:

- Unit 1 20% power
- MDAFWP 1-01 out of service for maintenance
- Reactor is tripped due to trip of BOTH MFPs
- SG NR Levels are 40%

Upon entry into EOP-0.0A, Reactor Trip or Safety Injection, the TDAFWP __(1)__ started.

While performing the step to “Verify AFW Alignment” in EOP-0.0A, if AFW flow cannot be maintained > 460 gpm, go to FRH-0.1A, Response to a Loss of Secondary Heat Sink __(2)__.

A. (1) must be manually
(2) immediately

B. (1) must be manually
(2) upon completion of Attachment 2, Safety Injection Actuation Alignment

C. (1) should have automatically
(2) immediately

D. (1) should have automatically
(2) upon completion of Attachment 2, Safety Injection Actuation Alignment

Answer: A

K/A Match: K/A match due to requiring knowledge of which AFW pumps require a manual start during a loss of MFW pumps event.

Explanation:

A. Correct. First part is correct. The TDAFWP does not automatically start on a trip of both MFPs. It will require a manual start under the given SGWLs provided. Second part is correct, per EOP-0.0A, Step 6 RNO, if AFW flow cannot be maintained greater than 460 gpm and no SG NR levels are greater than 43% then a transition to FRH-0.1A is required immediately.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible because a direct entry into FRH-0.1A did not exist in the CPNPP ERGs at EOP-0.0 step 6 until Rev 9 of the ERGs (current rev). In Rev 8 and previous an entry into FRH-0.1A could not be performed until Att 2 was complete.

C. Incorrect. First part is incorrect, but plausible because an automatic start signal to the MDAFWPs will exist due to a trip of BOTH MFPs, often the MDAFWP and TDAFWP auto start signals are confused. Additionally, because since SG NR levels are below 43% it could be thought that an auto start signal would be present to the TDAFWP. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	EOP-0.0A	Attached w/ Revision # See Comments / Reference
	ODA-407	
	AFW Study Guide	

Proposed references to be provided during examination: ____________________

Learning Objective: **EXPLAIN** the instrumentation and controls of the Auxiliary Feedwater system and the system response in accordance with DBD-ME-206. (SYS.AF1.OB04)

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 X

10 CFR Part 55 Content:
- 55.41 7
- 55.43 ____________

Comments / Reference: EOP-0.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 9	PAGE 7 OF 121

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6	Verify AFW Alignment:	
	a. MDAFW Pumps - RUNNING	a. Manually start pump(s).
	b. TDAFW Pump - RUNNING IF NECESSARY	b. Manually open steam supply valve(s).
	c. AFW total flow - GREATER THAN 460 GPM	c. Check narrow range levels and perform the following: IF narrow range level greater than 43%(50% FOR ADVERSE CONTAINMENT) in any SG, THEN control feed flow to maintain narrow range level AND go to Step 6d. IF narrow range level less than 43%(50% FOR ADVERSE CONTAINMENT) in all SGs, THEN manually start pumps and align valves as necessary. -AND- IF AFW flow greater than 460 gpm can NOT be established, THEN go to FRH-0.1A, RESPONSE TO LOSS OF SECONDARY HEAT SINK, Step 1.
	d. AFW valve alignment - PROPER ALIGNMENT	d. Manually align valve(s) as necessary.

Comments / Reference: AFW Study Guide	Revision: 00-0000

OP51.SYS.AF1

The main steam line to the TDAFW pump is equipped with condensate traps to remove any moisture buildup in the lines. Turbine steam is exhausted to the atmosphere through a safety related roof vent. The turbine steam exhaust line on Unit 1 is equipped with a condensate trap to eliminate moisture buildup in the exhaust line and turbine. These traps are routed to a flash tank located in the pipe trench outside the TDAFW pump room. In both Units, selected traps are provided with level switches which provide signals to actuate annunciator window 2.6, "ANY TD AFWP D\POT LVL HI" alarm on ALB-8B on the Main Control Board. This annunciator provides indication of excessive condensation and/or moisture buildup in the steam supply line to the TDAFW Pump turbine.

The TDAFW Pump may be started or stopped from the Control Room by opening or closing the steam supply valves. The TDAFW Pump steam supply valves, HV-2452-1 and HV-2452-2, are operated using three-position (OPEN-AUTO-CLOSE) handswitches on CB-09 which spring return to Auto and Pull-To-Lock in the STOP position.

When Control Room switches are inaccessible, manual operation from the RSP is provided. Local manual control from the RSP overrides all other signals. Manual control is switched from the Main Control Board to the RSP with installed hand switches on the Switch Transfer Panel (STP) for the Train A valve (HV-2452-1 from main steam line 4) or the RSP for the Train B valve (HV-2452-2 from main steam line 1). When control is transferred, an alarm for local override is sounded in the Control Room.

The TDAFWP steam supply valves will automatically open, admitting steam to the TDAFW Pump turbine, due to:

- Low-low SG NR level at 38% (35.4% for Unit 2) on two of four detectors in any two SGs,
- Blackout Sequencer operator lockout signal, or
- AMSAC signal

Downstream of the steam supply valves is a motor operated turbine Trip & Throttle Valve (T&TV). The motor operator has been de-terminated and is not provided with electrical power due to equipment qualification concerns associated with the TDAFW pump local control panel. Since the panel is located in the TDAFW pump room, a high energy line break in the room could cause a fault condition if the motor was energized which might prevent operation of the turbine. The T&TV is used to trip the turbine by isolating the steam supply. Although Terry Turbines Co. furnishes the valve as a fail closed valve, it will be latched open at all times to ensure the ability of the turbine to start in the event of an emergency.

The turbine may be manually tripped by depressing the AFWPT TRIP pushbutton u-HS-2452F on the Main Control Board. This pushbutton energizes a solenoid which actuates the local turbine trip device, unlatching the valve from the operator and allowing the T&TV to close. The trip pushbutton on the Main Control Board requires that the trip solenoid have power available in order to function. Power to the trip solenoid is supplied from the local turbine control panel. The TDAFWP can also be locally tripped by actuating the local manual trip device attached to the T&TV linkage. A mechanical emergency overspeed trip mechanism isolates the main steam supply by tripping the T&TV closed at 116% of rated speed. The overspeed trip mechanism will reposition itself after turbine speed decreases below 3000 rpm. The T&TV must be manually reset by an operator to restart the turbine.

FOR TRAINING USE ONLY Page 17 of 35 Rev. 00.0000

Comments / Reference: ODA-407	Revision: 9

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 18 / INFORMATION USE	PAGE 29 OF 64

ATTACHMENT 8.A
PAGE 10 OF 25

ERG RULES OF USAGE

Scenarios Affecting FRZ-0.1A/B Status	Requirements for Implementing FRZ-0.1A/B
Containment Spray initiates during EOP-0.0A/B performance as Containment Pressure reaches 18 psig (FRZ-0.1A/B ORANGE priority condition exists). Step 7 of EOP-0.0A/B is performed to verify proper Containment Spray alignment. The FRZ ORANGE condition HAS CLEARED when FRG implementation is initiated.	IF FRZ ORANGE condition has CLEARED when FRG implementation is initiated (transition out of EOP-0.0A/B OR EOP-0.0A/B step initiates CSF monitoring AND automatic action verification complete), THEN performance of FRZ-0.1A/B is NOT required. Proper response for Containment Spray actuation has been verified with EOP-0.0A/B actions AND there is NOT currently a challenge to the Containment barrier.
Containment Spray initiates during EOP-0.0A/B performance as Containment Pressure reaches 18 psig (FRZ-0.1A/B ORANGE priority condition exists). Step 7 of EOP-0.0A/B is performed to verify proper Containment Spray alignment. The FRZ ORANGE condition STILL EXISTS when FRG implementation is initiated.	IF an FRZ ORANGE condition exists when FRG implementation is initiated (transition out of EOP-0.0A/B OR EOP-0.0A/B step initiates CSF monitoring AND automatic action verification complete), THEN FRZ-0.1A/B performance is required. Proper response for Containment Spray actuation has been verified with EOP-0.0A/B actions BUT a challenge to the Containment barrier may exist.
Containment Spray initiates during EOP-0.0A/B performance as Containment Pressure reaches 18 psig (FRZ-0.1A/B ORANGE priority condition exists). Step 7 of EOP-0.0A/B is performed to verify proper Containment Spray alignment. The FRZ ORANGE condition exists when FRG implementation is initiated AND clears prior to FRZ-0.1A/B entry.	IF an FRZ ORANGE condition exists when FRG implementation is initiated (transition out of EOP-0.0A/B OR EOP-0.0A/B initiates CSF monitoring AND automatic action verification complete) BUT clears prior to FRZ-0.1A/B entry, THEN FRZ-0.1A/B performance is NOT required. Proper response for Containment Spray actuation has been verified with EOP-0.0A/B actions AND a challenge to the Containment barrier does not exist as evidenced by the lowering containment pressure.
EOP-0.0A/B is performed without Containment Spray actuation (Step 7 of EOP-0.0A/B identifies Containment Spray NOT required/NOT verified). The appropriate recovery actions are initiated with transition out of EOP-0.0A/B. The FRZ ORANGE condition COMES IN AND remains in during implementation of recovery actions (after FRG implementation initiated).	All CSFSTs are monitored and FRGs are implemented per the rules of usage. FRG implementation is initiated based on the highest CSF priority. IF an FRZ ORANGE condition exists, THEN FRZ-0.1A/B performance is required. A challenge to the Containment barrier exists AND proper response for Containment Spray actuation is verified to minimize challenges to the Containment barrier.
EOP-0.0A/B is performed without Containment Spray actuation (Step 7 of EOP-0.0A/B identifies Containment Spray NOT required/NOT verified). The appropriate recovery actions are initiated with transition out of EOP-0.0A/B. The FRZ ORANGE condition COMES IN after FRG implementation has been initiated, THEN clears prior to entering FRZ-0.1A/B.	All CSFSTs are monitored and FRGs are implemented per the rules of usage. FRG implementation is initiated based on the highest CSF priority. IF an FRZ ORANGE condition has previously existed AND FRZ-0.1A/B has NOT been performed, THEN FRZ-0.1A/B performance is required. Proper response for Containment Spray actuation is verified to ensure challenges to the Containment barrier have been addressed.

11. Monitoring of the CSFSTs and implementation of the FRGs are initiated per the following instructions to ensure appropriate priority is used for ERG response and recovery actions.

A. CSFST Monitoring is initiated by any of the following conditions:
- The step in EOP-0.0A/B directs the operator to initiate monitoring of the CSFSTs, or
- The operator transitions out of EOP-0.0A/B to some other procedure.

B. FRG Implementation is initiated when:
1) directed by ERG procedure step, or
2) when CSFST monitoring criteria is satisfied AND EOP-0.0A/B automatic action verification is complete (example: EOP-0.0A/B Attachment 2 following SI actuation, EOP-0.0A/B Steps 1 through 4 following Reactor Trip without SI actuation).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	1		
	Group	1		
	K/A	000055.G.2.4.35		
Level of Difficulty: 2	Importance Rating	3.8		

Station Blackout: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Question # 49

During a Station Blackout, an NEO is directed to locally perform additional load shedding per ECA-0.0A, Loss of All AC Power, Attachment 2C, DC Load Shed when DC Voltage Less than 110V.

This attachment is performed to allow for __(1)__ and __(2)__.

A. (1) battery charger restoration with portable generator
(2) Containment Phase A isolation capability

B. (1) battery charger restoration with portable generator
(2) Safeguards Bus supply breaker closure

C. (1) Diesel Generator field flashing
(2) Containment Phase A isolation capability

D. (1) Diesel Generator field flashing
(2) Safeguards Bus supply breaker closure

Answer: D

K/A Match: K/A match due to requiring knowledge of the effect of local actions taken during a station blackout.

Explanation:

A. Incorrect. Plausible because Attachments 2.A and 2.B when performed ensure sufficient time to restore battery chargers using a portable generator, however this is not what is accomplished by Attachment 2.C. The second part is incorrect but plausible (See C).

B. Incorrect. First part is incorrect but plausible (See A). Second part is correct (See D).

C. Incorrect. First part is correct (See D). The second part is incorrect but plausible because load shedding per Attachment 2.B is not performed until Phase A is verified, but Phase A does not require DC power to perform, only to verify.

D. Correct. If battery voltage lowers to less than 110 volts the associated bus is further load shed to ensure adequate voltage remains for flashing the diesel generator field or closing safeguards bus supply breakers for power restoration.

Technical Reference(s)	ECA-0.0A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given a procedural step, or sequence of steps from ECA-0.0, **DISCUSS** the purpose/basis for the step(s) in accordance with ECA-0.0. (ERG.C00.OB05)

Question Source: Bank # 75812
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam LC24

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: Bank 75812	Revision:

During a Station Blackout additional load shedding is performed when safeguards battery voltage is less than 110 volts to allow for ________________ and ________________.

A. battery charger restoration with portable generator
plant monitoring and control until AC power restored

B. battery charger restoration with portable generator
Safeguards Bus supply breaker closure

C. Diesel Generator field flashing
plant monitoring and control until AC power restored

D. Diesel Generator field flashing
Safeguards Bus supply breaker closure

Answer: D

Answer Explanation	
A.	Incorrect. Plausible because Attachments 2.A and 2.B when performed ensure sufficient time to restore battery chargers using a portable generator, however this is not what is accomplished by Attachment 2.C. The second part is incorrect but plausible (See C below).
B.	Incorrect. First part is incorrect but plausible (See A above). Second part is correct (See D below).
C.	Incorrect. First part is correct (See D below). The second part is incorrect but plausible because load shedding does provide for plant monitoring and control until AC power is restored during initial load shedding not the load shedding performed per Attachment 2.C.
D.	Correct. If battery voltage lowers to less than 110 volts the associated bus is further load shed to ensure adequate voltage remains for flashing the diesel generator field or closing safeguards bus supply breakers for power restoration.

Comments / Reference: Bank 75812	Revision:

Question 177 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	75812
User-Defined ID:	ILOT9441
Cross Reference Number:	ERG.CO0.0B05.014
Topic:	During a Station Blackout additional load shedding is performed when safeguards battery voltage is
K/A:	EPE 054 AK3.01
Question Reference:	
SRO:	
Comments:	LC24 NRC K/A Match: Question matches K/A as applicant must know that the reason for tripping the reactor during a low power loss of feedwater.

-

Comments / Reference: ECA-0.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 9	PAGE 18 OF 136

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18	Check DC Bus Loads:	
	a. Initiate shedding of DC loads per Attachment 2.A.	
	b. Check ELAP - HAS BEEN DECLARED	b. Perform the following: 1) IF Voltage is less than 110 Volts, THEN determine necessity of shedding additional DC loads per Attachment 2.C. 2) Go to Step 19.
	c. Check vital instrumentation - INSTRUMENTATION FOR REQUIRED FUNCTION AVAILABLE • CNTMT PRESS (IR) • RCS HL & CL TEMP (WR) • RCS PRESS (WR) • PRZR LVL • CORE EXIT TEMP • MSL PRESS • SG LVL (NR) • SG AFW FLO • CST LVL • RVLIS • SR NI	c. Perform FSI-7.0A, LOSS OF VITAL INSTRUMENTATION OR CONTROL POWER.

Comments / Reference: ECA-0.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 9	PAGE 52 OF 136

ATTACHMENT 2.C
PAGE 1 OF 4

DC LOAD SHED WHEN DC VOLTAGE LESS THAN 110V

IF DC Bus Voltage is LESS THAN 110 VOLTS AND Plant Staff determines it necessary to conserve Battery BT1ED1 or BT1ED2 for subsequent Diesel Generator starts OR Offsite Power breaker closure, THEN perform the following:

1. IF Train A Safeguards bus is most probable to be restored, THEN perform the following load shed of 1ED1:

   a. Reference ABN-603, LOSS OF PROTECTION OR INSTRUMENT BUS to:

   ☐ • Evaluate equipment and indication that will be lost when 1PC1 and 1EC1 are de-energized.

   ☐ • Verify equipment and indication supplied from BT1ED2 via 1PC2 and 1EC2 is available.

   ☐ • IF sufficient equipment and indication that will be supplied from BT1ED2 (via 1PC2, 1EC2) is NOT available, THEN Plant Staff should evaluate plant conditions to determine which DC loads to shed.

   b. Due to loss of input signals, place Pressurizer PORV handswitches to CLOSE:

   ☐ • 1/1-PCV-455A, PRZR PORV

   ☐ • 1/1-PCV-456, PRZR PORV

ECB 792 U1 Train A UPS Room

   c. Place following breakers in OFF:

   ☐ • 1ED1/2-10/BKR, IV1PC1 SPLY

   ☐ • 1ED1/2-11/BKR, IV1EC1/3 DC BUS 1 SPLY

   ☐ • 1ED1/2-13/BKR, IV1EC1 SPLY

☐ d. Place 1ED1/1-7/DSW, 125 VDC DISTRIBUTION PANEL 1ED1-1 PREFERRED FUSED DISCONNECT SWITCH in OFF.

☐ e. Place 1ED1/1-5/DSW, 125 VDC DISTRIBUTION PANEL 1ED1-2 FUSED DISCONNECT SWITCH in OFF.

Comments / Reference: ECA-0.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 9	PAGE 123 OF 136

ATTACHMENT 9
PAGE 45 OF 58

BASES

ATTACHMENT 2

These attachments provides instructions for shedding of DC safeguards bus loads in order to conserve capacity to assist in future actions to restore AC power, while maintaining that minimum instrumentation necessary to monitor plant conditions. The intent of DC load shedding is to remove large loads not necessary for the Loss of All AC Power event as soon as practical. However, the specific battery sizing calculation for CPNPP Station Blackout shows that, even without load shedding, the heaviest loaded battery with an assumed electrolyte temperature of 65°F, has sufficient capacity to not only carry its loads for a four (4) hour period, but also provide sufficient DC power for Diesel Generator field flashing.

The load shed strategy is separated into 4 sub-attachments:

Attachment 2.A Shedding of loads not essential for cooldown. This strategy involves attempting to transfer common loads to Unit 2 when possible prior to de-energizing equipment. The attachment is separated into 4 sections, one for each of the possible situations:
1) both Unit 2 safeguard busses de-energized.
2) 2EA1 energized and 2EA2 de-energized.
3) 2EA1 de-energized and 2EA2 energized.
4) both Unit 2 safeguard busses energized.

Attachment 2.B Shedding of loads no longer requiring power after the equipment has repositioned following Containment Isolation Phase A and Containment Ventilation Isolation.

Attachment 2.C Shedding of loads in the event that DC bus voltage reaches 110 Volts. Each safeguards battery is capable of carrying all loads for a period of 4 hours in the event of a Loss of All AC Power condition. Load shedding performed in Attachments 2.A & 2.B will extend the time battery voltage is maintained. The minimum system voltage required for equipment operation is approximately 105 volts. This ensures a sufficient DC voltage is available to flash the diesel generator field and to operate the associated output breaker for the diesel generator. This attachment will de-energize DC Bus 1ED1 OR 1ED2 to conserve DC power for subsequent power restoration activities bases on the bus most likely to be restored.

Attachment 2.D Restoration of DC power. This attachment provides any additional considerations for recovering the DC loads that were shed in Attachments 2.A, 2.B and 2.C.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	1		
	K/A	000056.AK3.02		
Level of Difficulty: 2	Importance Rating	4.4		

Loss of Offsite Power: Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Actions contained in EOP for loss of offsite power

Question # 50

Given the following conditions:

- LOOP has occurred
- Bus 1EA1 de-energized
- EOS-0.1A, Reactor Trip Response, in progress
- RCS temperature 561ºF stable
- Letdown has been lowered to 45 gpm with CCP 1-02 charging and FCV-121, Charging Flow Control Valve, full open
- RCS pressure 1920 psig trending down slowly
- Pressurizer level 5% slowly lowering

Which of the following actions is required per EOS-0.1A, Reactor Trip Response, Fold Out Page?

A. Due to the loss of RCPs, verify Natural Circulation per EOS-0.1A, Reactor Trip Response.

B. Due to the loss of 1EA1, attempt to restore Bus 1EA1 per ABN-602, Response to a 6900V/480V System Malfunction.

C. Due to the low pressurizer level, manually actuate SI and return to EOP-0.0A, Reactor Trip or Safety Injection.

D. Due to the high RCS temperature, increase AFW flow to the Steam Generators per EOS-0.1A, Reactor Trip Response.

Answer: C

K/A Match: K/A match due to requiring knowledge of the reason for actions required following a loss of offsite power.

Explanation:

A. Incorrect. Plausible because Natural Circulation would be verified if SI was not required per the Foldout Page.

B. Incorrect. Plausible because it would have been performed in EOP-0.0A, however, priority is Safety Injection (SI).

C. Correct. EOS-0.1A Foldout Page requires manual initiation of SI when PRZR level cannot be maintained greater than 6% and a transition back to EOP-0.0A.

D. Incorrect. Plausible because RCS temperature is above the no-load value of 557F, and increasing AFW flow would serve to cooldown the RCS, however, 561F RCS temperature is expected for the given conditions and this is not an action required per the FOP of EOS-0.1A.

Technical Reference(s)	EOS-0.1A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **ANALYZE** the recovery technique used and the procedure steps of EOS-0.1, Reactor Trip Response. (ERG.E01.OB02)

Question Source: Bank # 34861
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam LC21

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ________

Comments / Reference: Bank 34861	Revision:

- A Loss of Offsite Power has occurred
- Bus 1EA1 is de-energized
- EOS-0.1A, Reactor Trip Response, is in progress
- Reactor Coolant System (RCS) temperature is 561°F and stable
- RCS pressure is 1920 psig and trending down slowly
- Pressurizer level is 5% and slowly lowering

Which of the following actions is required per EOS-0.1A, Reactor Trip Response?

A. Raise Condenser Steam Dumps to maintain T_{AVE} at 557°F per EOS-0.1A, Reactor Trip Response.

B. Attempt to restore Bus 1EA1 per ABN-602, Response to a 6900V/480V System Malfunction.

C. Manually actuate Safety Injection and return to EOP-0.0A, Reactor Trip or Safety Injection.

D. Isolate Letdown and verify Natural Circulation per EOS-0.1A, Reactor Trip Response.

Answer: C

Answer Explanation
A. Plausible because this is a Step 1 RNO action of EOS-0.1A, however, Steam Dump will not be available without Circulating Water Pumps. T_{AVE} is where it should be for the conditions.
B. Plausible because it would have been performed in EOP-0.0A, however, priority is Safety Injection (SI).
C. EOS-0.1A Foldout Page requires manual initiation of SI when PRZR level cannot be maintained greater than 6% and a transition back to EOP-0.0A.
D. Plausible because Natural Circulation would be verified if SI was not required per the Foldout Page. Additionally, letdown is isolated if Pressurizer Level is less than 17%.

-

Comments / Reference: Bank 34861	Revision:

Question 94 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	34861
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	A Loss of Offsite Power has occurred Bus 1EA1 is de-energized EOS-0.1A, Reactor Trip Response, is
K/A:	056.AA1.05
Question Reference:	EOS 0.1
SRO:	
Comments:	LC16 NRC; LC21 NRC; R/S22E14; R/S23E23; R/S24E23, LC25 Comp(Remedial), R/S27E25

Comments / Reference: EOS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 9	PAGE 18 OF 40

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOS-0.1A, REACTOR TRIP RESPONSE

1. SI ACTUATION CRITERIA

Actuate SI and go to EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION, Step 1, if EITHER condition listed below occurs:

- RCS subcooling - LESS THAN 25°F
- PRZR level - CANNOT BE MAINTAINED GREATER THAN 6%

2. SHUTDOWN MARGIN CRITERIA

Emergency borate per ABN-107 if either of the following conditions below occur:

- Two or more control rods NOT fully inserted (1800 gallons of 7000 ppm boric acid for each control rod not fully inserted).
- Control rod position indication is NOT available (3600 gallons of 7000 ppm boric acid).

3. SG LEVEL/AFW FLOW CONTROL CRITERIA

Control AFW total flow as necessary to maintain an adequate Heat Sink (Narrow Range level greater than 43% in any SG OR AFW total flow GREATER THAN 460 gpm).

IF not required, secure TDAFWP.

4. AFW SUPPLY SWITCHOVER CRITERION

IF CST level decreases to less than 10%, THEN switch to alternate AFW water supply per ABN-305, AUXILIARY FEEDWATER SYSTEM MALFUNCTION.

5. RCP SEAL INJECTION FLOW CRITERION

Ensure 6 gpm to 13 gpm seal injection flow to all RCPs UNLESS isolated by ERG actions.

Comments / Reference: EOS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 9	PAGE 36 OF 40

ATTACHMENT 4
PAGE 11 OF 15

BASES

ATTACHMENT 1.A

SI ACTUATION CRITERIA - If RCS subcooling is lost or if pressurizer level cannot be maintained, control of the Unit is lost and SI actuation is necessary. Although SI Actuation Criteria are identical to the ones found in the SI Reinitiation Criteria, the actions are different. The operator is instructed to actuate safety injection rather than start ECCS pumps as necessary.

SHUTDOWN MARGIN CRITERIA - If two or more control rods are not inserted or the operator cannot verify that the rods are inserted, the operator is instructed to emergency borate. The amount of boration provides a conservative value based upon a bounding value for the most reactive rod worth:

- Two or more control rods NOT fully inserted requires boration of 1800 gallons (130 ppm) of 7000 ppm boric acid for each control rod not fully inserted.
- Control rod position indication is NOT available requires boration of 3600 gallons (260 ppm) of 7000 ppm boric acid. If the operator has verified rod bottom lights, then the rod bottom lights subsequently become unavailable, proper insertion of the rods has been verified and emergency boration is not required.

These values have been made conservative to bound the most reactive rod worth which changes from cycle to cycle.

SG LEVEL/AFW FLOW CONTROL CRITERIA - The operator is reminded of minimum requirements for heat sink. The operator should take action to control intact SG levels as necessary to mitigate the severity of possible accident conditions.

AFW SUPPLY SWITCHOVER CRITERION - This criterion is on the FOLDOUT PAGE to remind the operator that the supply of water from the CST to the suction of the AFW pumps is limited, and if it is depleted, an alternate suction supply of water to the AFW pumps is necessary.

RCP SEAL INJECTION FLOW CRITERIA - The effectiveness of the RCP number 1 seal is not affected by pump rotation. To ensure continued performance of the seal, cool filtered water should be continuously supplied within the required flow band.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	1		
	K/A	000058.AA2.03		
Level of Difficulty: 3	Importance Rating	3.5		

Loss of DC Power: Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems

Question # 51

Given the following conditions:

- Unit 2 100% power
- Steam Break occurs outside containment
- Reactor Trip and Safety Injection initiated
- Battery Charger 2D2 load sheds

Crew is performing EOP-2.0B, Faulted Steam Generator Isolation, Step 6 "Verify Faulted SG(s) Break Inside Containment."

Per Step 6, the crew will direct an NEO to align Battery Charger __(1)__ to DC Bus 2D2 to ensure the __(2)__ remain closed.

A. (1) 2D4
(2) ARVs

B. (1) 2D4
(2) MSIVs

C. (1) 2D24
(2) ARVs

D. (1) 2D24
(2) MSIVs

Answer: D

K/A Match: K/A match due to requiring knowledge of the effect of a loss of a component supporting DC power (load shed of BC 2D2) on equipment monitored in the control room (MSIVs).

Explanation:

A. Incorrect. First part is incorrect, it is plausible to think BC 2D4 could supply DC Bus 2D2 since BC 2D24 may supply DC Bus 2D2, however, BC 2D4 cannot supply DC Bus 2D2. Second part is incorrect, but plausible since the ARVs are located within the same room, however, their DC solenoids are controlled by Safeguards DC power which are environmentally qualified.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A).

D. Correct. First part is correct. EOP-2.0B Attachment 2 directs alignment of BC 2D24 to DC Bus 2D2 after a Fault outside of containment to assure power is maintained to the batteries that supply power to the MSIV hydraulic oil pump solenoid operated valves. The SOVs open to align air to the hydraulic oil pump, which is required to open the MSIVs. Ensuring Battery 2D2 is powered from Battery Charger BC2D24 will ensure that the solenoids for the air to the oil pump remain closed. Second part is correct. The MSIVs are the affected component.

Technical Reference(s)	EOP-2.0B	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **GIVEN** a procedure step, or sequence of steps from EOP-2.0, state the purpose/bases for the step in accordance with EOP-2.0. (ERG.E2A.OB04)

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

Question History: Last NRC Exam ____________________

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ____7____
55.43 ________

Comments / Reference: EOP-2.0B	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-2.0B
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 9	PAGE 4 OF 14

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	CAUTION: If the TDAFW pump is the only available source of feed flow, steam supply to the TDAFW pump must be maintained from at least one SG.	
4	Isolate Faulted SG(s): • Isolate main feedline. • Isolate AFW flow. • Place TDAFW Pump steam supply valve(s) in PULL-OUT. (SG 1 or 4) • Isolate blowdown and sample lines. • Ensure SG atmospheric(s) - CLOSED • Ensure main steamline drippot isolation valve(s) - CLOSED	IF SG atmospheric(s) can NOT be closed, THEN dispatch operator to locally close block valve(s). IF other valves can NOT be closed, THEN dispatch operator to locally close valves or block valves.
5	Check CST Level - GREATER THAN 10%	Perform ABN-305, AUXILIARY FEEDWATER SYSTEM MALFUNCTION while continuing with this procedure.
6	Verify Faulted SG(s) Break Inside Containment	Perform Attachment 2.

Comments / Reference: EOP-2.0B	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-2.0B
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 9	PAGE 7 OF 14

ATTACHMENT 2
PAGE 1 OF 2

MSIV ELECTRICAL REQUIREMENT VERIFICATION

To prevent inadvertent opening of the MSIVs, the following steps should be performed to maintain power aligned to Battery BT2D2.

☐ 1. Locally verify 2D2/2-6/BKR, 125 VDC BATTERY CHARGER BC2D24 TO 125/250 VDC SWBD 2D2 POS FEEDER BKR is CLOSED to confirm the Positive-Neutral side of Switchboard 2D2 is powered from Battery Charger BC2D24 (ECB 792, X-126, Unit 2 Train C UPS Room).

2. IF necessary to align Battery Charger BC2D24 to supply Switchboard 2D2 (Battery BT2D2), THEN perform the following:

a. Ensure the following breakers on BC2D24 in OFF (ECB 792, X-126, Unit 2 Train C UPS Room).

- ☐ BC2D24/CB1/BKR, 480 VAC MCC 2B1-1 TO BATTERY CHARGER BC2D24 INPUT BREAKER
- ☐ BC2D24/CB2/BKR, BATTERY CHARGER BC2D24 TO 125/250 VDC SWITCHBOARD 2D2 OUTPUT BREAKER

b. Ensure BC2D24 AC feeder breaker in ON (TB 803, Near Main Feedwater Pumps).

- ☐ 2B1-1/6BR/BKR, 125 VDC BATTERY CHARGER BC2D24 SUPPLY BREAKER

☐ c. Place 2D2/2-6/BKR, 125 VDC BATTERY CHARGER BC2D24 TO 125/250 VDC SWBD 2D2 POS FEEDER BKR in ON (This will align BC2D24 to BT2D2).

d. Place BC2D24 AC INPUT breaker in ON.

- ☐ BC2D24/CB1/BKR, 480 VAC MCC 2B1-1 TO BATTERY CHARGER BC2D24 INPUT BREAKER

e. Ensure the following indications on BC2D24:

- ☐ FLOAT light (green) is LIT.
- ☐ DC VOLTS indicates FLOAT voltage, 128-135 VDC.
- ☐ EQUALIZE TIMER set to Zero (0).

f. Place BC2D24 DC OUTPUT breaker in ON.

- ☐ BC2D24/CB2/BKR, BATTERY CHARGER BC2D24 TO 125/250 VDC SWITCHBOARD 2D2 OUTPUT BREAKER

Comments / Reference: EOP-2.0B	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-2.0B
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 9	PAGE 11 OF 14

ATTACHMENT 3
PAGE 3 OF 6

BASES

STEP 6: The intent of this step is to determine the necessity for performing the MSIV electrical requirement verification of Attachment 2. If the steam break is located inside the MSIV rooms, the environmental conditions are adverse to the equipment located in the area, and the electrical alignment of Attachment 2 is necessary to ensure the closed position of the MSIVs is maintained. Determination that the secondary break is located inside containment is a relatively simple check (e.g., containment pressure, containment temperature, etc.) and can be used to eliminate the need for performance of the electrical alignment of Attachment 2. If the secondary break is not located inside containment, then performance of Attachment 2 is initiated to assure power is maintained to MSIV solenoids without additional actions to check the secondary fault in the MSIV rooms. Interpretation of the actions to confirm the secondary break location NOT in the MSIV rooms could result in personnel entering into a hazardous environment, which is not the intent of the step. The action to perform the MSIV electrical alignment is only required when the secondary break is located inside the MSIV rooms; however, it is performed anytime the leak is outside containment as a general recourse to minimize the potential for personnel exposure to harsh environments (e.g., steam atmosphere, radioactivity, etc). The MSIVs are opened by an air operated oil pump. The air is controlled from a solenoid operated valve, energized to close.

The MSIVs also have a solenoid operated hydraulic dump valve which opens (energizes) to dump oil and allow the MSIVs to close. The hydraulic dump solenoid valve is not qualified for extended periods in a harsh environment, such as a main steamline break in the MSIV rooms. The hydraulic dump valve solenoid may de-energize and allow the MSIVs to open unless the air is isolated to the oil pump. Ensuring Battery 2D2 is powered from Battery Charger BC2D2 or BC2D24 will ensure that the solenoids for the air to the oil pump remain closed.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	1		
	Group	1		
	K/A	000062.AK3.03		
Level of Difficulty: 2	Importance Rating	4.0		

Loss of Nuclear Service Water: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Guidance actions contained in EOP for Loss of nuclear service water

Question # 52

Given the following conditions:

- Unit 1 70% power
- 86-2 LOR fault occurs on Safeguards Bus 1EA2
- Both 1EA2-1, INCOMING BKR, and 1EA2-2, INCOMING BKR trip open
- DG 1-02 automatically starts and DG 2 BKR 1EG2 automatically closes
- 1-ALB-1, Window 1.8 – SSWP 1/2 OVERLOAD/TRIP, alarms
- SSW Pump 1-02 handswitch indications are as follows:
  - Amber MISMATCH light LIT
  - White TRIP light LIT

Per ABN-501, Station Service Water System Malfunction, which of the following actions should be taken?

A. (1) Wait for the Blackout Sequencer to complete its timing and verify proper SSW flow to EDG 1-02
(2) Continue plant operations at approximately 70% power

B. (1) Wait for the Blackout Sequencer to complete its timing and verify proper SSW flow to EDG 1-02
(2) Enter and perform the actions of EOP-0.0A, Reactor Trip or Safety Injection

C. (1) Place DG 1-02 EMER STOP/START handswitch in PULL-OUT
(2) Enter and perform the actions of EOP-0.0A, Reactor Trip or Safety Injection

D. (1) Place DG 1-02 EMER STOP/START handswitch in PULL-OUT
(2) Continue plant operations at approximately 70% power

Answer: D

K/A Match: K/A match due to requiring knowledge of the actions to be taken in the event of a loss of SSW.

Explanation:

A. Incorrect. First part is incorrect, but plausible because the BOS will fire to load the EDG and the SSW Pump will have an amber MISMATCH light until the sequencer sends a signal to start the pump, however, the white TRIP light is not expected and this is an indication that the pump has tripped and will not start on the sequencer timing. The immediate action per ABN-501 is to place the EDG in PULLOUT. Second part is correct (see D).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since the unit would be tripped if both buses were lost.

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see B)..

D. Correct. First part is correct. With the SSW Pump tripped, the EDG must be immediately tripped whether or not it is carrying the bus. Second part is correct. A loss of a single Safeguards Bus does not require the Unit be tripped.

Technical Reference(s)	ABN-501A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: __________

Learning Objective: **DISCUSS** the response to Station Service Water Pump Trip in accordance with ABN-501, Station Service Water System Malfunction. (ABN.501.OB101)

Question Source: Bank # 23103
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2018 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ABN-501A	Revision:

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 4 OF 50

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- The diesel generator can be operated, with load, for approximately one minute without SSW flow and not affect diesel performance.
- When a fault exists on the 6.9KV safeguard bus, the SSW pump will not be running to supply cooling water to the DG. The time this condition exists should be minimized (approximately 15 minutes) to prevent damage to the DG.
- Diamond step 1 denotes Initial Operator Actions.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ ◇1 Place affected train diesel generator handswitch, CS-uDGuE (emergency stop/start) in PULLOUT.	
☐ 2 Verify unaffected train SSW Pump - RUNNING	Manually start the SSW pump in the unaffected train. IF the pump fails to start, THEN GO TO Section 5.0 of this procedure.

NOTE: Opposite train's SSW Pump and CCW Pump DO NOT provide cooling to CCW loads from the Ultimate Heat Sink.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3 Verify unaffected train CCW Pump - RUNNING	Perform the following: a. Manually start the CCW pump in the unaffected train. b. IF the CCW pump in the unaffected train will not start, THEN perform the following: 1) TRIP the Reactor 2) GO TO EOP-0.0A/B while other qualified operators continue this procedure. 3) TRIP ALL RCPs. 4) GO TO Section 5.0 of this procedure

Section 2.3

Comments / Reference: ABN-501A	Revision:

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 20 OF 50

5.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 4 Verify Standby Service Water Pump - RUNNING	Perform the following: a. Manually start Standby SSW pump. b. Check discharge pressure on operating pump. • u-PI-4252A, SSWP 1 DISCH PRESS • u-PI-4253A, SSWP 2 DISCH PRESS IF discharge pressure on operating pump does NOT increase to greater than 32 psig, THEN STOP the affected pump AND place handswitch in PULL OUT. c. IF the standby pump does NOT start, THEN perform the following: 1) Manually TRIP the Reactor AND GO to EOP-0.0A/B while other operators continue this procedure. 2) Trip ALL RCPs.

"Step continued next page"

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	1		
	Group	1		
	K/A	000065.AK3.08		
Level of Difficulty: 2	Importance Rating	3.7		

Loss of Instrument Air: Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Actions contained in EOP for loss of instrument air

Question # 53

Given the following conditions:

- Unit 1 100% power
- After maintenance, the TDAFWP is recirculating to the CST at 2000 rpm per SOP-304A, AFW System, when:
  - Unit 1 experiences a Loss of Instrument Air
  - Instrument Air pressure is 34 psig and lowering

Which of the following:

(1) Identifies the impact on TDAFWP speed?

(2) What action should be taken to mitigate the situation?

A. (1) TDAFWP speed rises to maximum
(2) Trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection

B. (1) TDAFWP speed rises to maximum
(2) Maintain plant conditions stable and manually control SG level by controlling the MFW Flow Control Valves

C. (1) TDAFWP speed lowers to minimum
(2) Trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection

D. (1) TDAFWP speed lowers to minimum
(2) Maintain plant conditions stable and manually control SG level by controlling the MFW Flow Control Valves

Answer: A

K/A Match: K/A match due to requiring knowledge of how the TDAFWP responds to a loss of instrument air and the actions needed to be taken as IA pressure lowers.

Explanation:

A. Correct. First part is correct. TDAFW Pump speed rises to maximum. Second part is correct. Given the listed instrument air pressure, the Reactor should be tripped and EOP-0.0A entered

B. Incorrect. First part is correct (see A). Second part is incorrect, plausible because it could be thought that the MFW Flow Control valves have accumulators to allow continued operation after a loss of instrument air similar to the accumulators for the AFW flow control valves.

C. Incorrect. First part is incorrect, but plausible since if thought that the TDAFW Pump speed will lower vice rise on a loss of air. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-301	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to Loss of Instrument Air in accordance with ABN-301 Instrument Air System Malfunction. (ABN.301.OB04)

Question Source: Bank # 32803
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: ABN-301	Revision: 14

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 14	PAGE 4 OF 130

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Step 1 is a Continuous Action Step.

☐ 1 IF in MODE 1, 2, 3, OR 4 AND Instrument Air Header pressure decreases to 35 psig OR control of system(s) is lost,
THEN manually trip the reactor AND GO TO EOP-0.0A/B while other operator(s) continue this procedure section starting with step 7.

NOTE:
- Loss of Instrument Air in the Auxiliary Building will affect components in Unit 1 and Unit 2.
- Section 4.0 provides actions for "Unit 2 Response to Loss of Unit 1 Instrument Air."
- IF an air compressor is in an Auto-Start condition, THEN it will NOT start until low pressure is sensed (105 psig if in LEAD, 100 psig if in BACKUP). Once low pressure is sensed, the Compressor will restart and load.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2 VERIFY at least one Instrument Air Compressor - ALIGNED TO THE UNIT AND RUNNING: • u-HS-3451, INST AIR COMP 1 • u-HS-3463, INST AIR COMP 2 • X-ZL-3452, INST AIR COMM COMPR 1 • X-ZL-3463, INST AIR COMM COMPR 2	WHEN Instrument Air Header Pressure - LESS THAN 100 psig, THEN ENSURE the backup OR standby Instrument Air Compressor aligned to Unit - RUNNING. • u-PI-3488, INST AIR AFTFILT OUT PRESS • u-HS-3451, INST AIR COMP 1 • u-HS-3463, INST AIR COMP 2 • X-ZL-3452, INST AIR COMM COMPR 1 • X-ZL-3463, INST AIR COMM COMPR 2

Section 2.3

Comments / Reference: ABN-301	Revision: 14

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 14	PAGE 86 OF 130

ATTACHMENT 2
PAGE 27 OF 47

AIR OPERATED EQUIPMENT FAILURE POSITIONS

BLDG		EQUIPMENT	FAILURE POSITION
Rm u-067, TRN A ECCS VLV RM			
SFGD 790	u-HCV-0606	RHR HX u-01 FLO CTRL VLV	F.O.
SFGD 790	u-FCV-0618	RHR HX u-01 BYP FLO CTRL VLV	F.C.
Rm u-070, HALLWAY			
SFGD 790	u-HC-0606	RHR HX 01 FLO CTRL	INOP
SFGD 790	u-FC-0618	RHR HX 01 BYP FLO CTRL	INOP
SFGD 790	2-HV-5660B	SI PMP 2-01 RM VENT EXH DMPR 2-13 OPER	F.C.
SFGD 790	2-HV-5663B	SI PMP 2-02 RM VENT EXH DMPR 2-16 AO	F.C.
Rm u-072, AFW PMP u-01 RM			
SFGD 790	u-PV-2453A	MD AFW PMP u-01 DISCH TO SG u-01 FLO CTRL VLV	F.O.
SFGD 790	u-PV-2453B	MD AFW PMP u-01 DISCH TO SG u-02 FLO CTRL VLV	F.O.
SFGD 790	u-FV-2456	MD AFW PMP u-01 TO CST RECIRC FLO VLV	F.O.
Rm u-073, AFW PMP u-02 RM			
SFGD 790	u-PV-2454A	MD AFW PMP u-02 DISCH TO SG u-03 FLO CTRL VLV	F.O.
SFGD 790	u-PV-2454B	MD AFW PMP u-02 DISCH TO SG u-04 FLO CTRL VLV	F.O.
SFGD 790	u-FV-2457	MD AFW PMP u-02 TO CST RECIRC FLO VLV	F.O.
Rm u-074, TD AFW PMP RM			
SFGD 790	u-SY-2452	TDAFWP SPEED CTRL	MAX
SFGD 790	u-HV-2459	TD AFW PMP u-01 DISCH TO SG u-01 FLO CTRL VLV	F.O.
SFGD 790	u-HV-2460	TD AFW PMP u-01 DISCH TO SG u-02 FLO CTRL VLV	F.O.
SFGD 790	u-HV-2461	TD AFW PMP u-01 DISCH TO SG u-03 FLO CTRL VLV	F.O.
SFGD 790	u-HV-2462	TD AFW PMP u-01 DISCH TO SG u-04 FLO CTRL VLV	F.O.

Attachment 2

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. ~~3~~ 4	Tier	1		
	Group	1		
	K/A	000077.AK2.06		
Level of Difficulty: 3	Importance Rating	3.9		

Generator Voltage and Electric Grid Disturbances: Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Reactor power

Question # 54

Given the following conditions:

- Unit 2 90% power
- Main Generator load 1140 MWe
- Grid frequency has lowered from 60 Hz to 59.7 Hz

When the Turbine Load Control system has restored Main Generator load to 1140 MWe, reactor power will be __(1)__ than 90%.

Per ABN-601, Response to a 138/345 KV System Malfunction, if Grid Frequency continues to lower, an IMMEDIATE Reactor Trip is FIRST required at __(2)__.

A. (1) higher
(2) ~~57.5 Hz~~ 57.2 Hz

B. (1) higher
(2) ~~59.4 Hz~~ 58.5 Hz

C. (1) lower
(2) ~~57.5 Hz~~ 57.2 Hz

D. (1) lower
(2) ~~59.4 Hz~~ 58.5 Hz

Answer: A

K/A Match: K/A match due to requiring knowledge of the effect of changing grid frequency on turbine load and reactor power.

Explanation:

A. Correct. First part is correct. Generator load will decrease with decreasing frequency (due to reduction in generator efficiency) and automatic load control will restore the load to the set load. This will result in increasing steam flow and increased reactor power. Second part is correct. An immediate Reactor Trip will be required as Grid Frequency lowers to ~~57.5~~ 57.2 Hz per ABN-601, Section 9, Step 3 table.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible because ~~this is a recent change to ABN-601 in which continued operation at 59.4 Hz is now allowed~~ automatic grid frequency relays provide for load shedding at frequency between 59.3 Hz and 58.5 Hz. ~~In previous revisions a Reactor Trip was required if Grid Frequency had lowered to 59.4 Hz for 9 minutes. There are also significant actions taken in the ABN at this Hz level to divorce the Safeguards Busses from the grid.~~

C. Incorrect. First part is incorrect, but plausible because it could be thought Grid Frequency lowering would cause less load on the generator and therefore cause Reactor power to lower as Main Generator Control valves throttle down. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-601	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____

Learning Objective: **ANALYZE** the response to events described in accordance with ABN-601, Response to 138/345 kV System Malfunction and ABN-602, Response to 6900/480 V System Malfunction. (ABN.601.OB00)

Question Source: Bank # ____
Modified Bank # 44282 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43 ______________

Comments / Reference: Bank 44282	Revision:

Given the following conditions:

- Unit 2 is in MODE 1, Reactor power is 90%.
- Main Generator load is 1140 MWe.
- Main Generator reactive load is 270 MVAR.
- ABN-601, Response to a 138/345 KV System Malfunction, Section 9.0, Grid Frequency Fluctuations/Loss of QSE Generation Controller Communications, is in progress.
- Grid frequency has lowered from 60 Hz to 59.5 Hz.

Which of the following identifies main generator response to the lowering frequency?

Main Generator reactive load will ______________ due to generator ______________.

A. increase over-excitation

B. decrease over-excitation

C. increase under-excitation

D. decrease under-excitation

Answer: A

Answer Explanation
A. Correct. If Grid Frequency lowers the main generator will become over excited and reactive load will increase. Grid frequency lowering also causes terminal voltage to decrease which leads to an increase in reactive load.
B. Incorrect. Plausible if it is thought that over-excitation cause reactive load to lower.
C. Incorrect. Plausible if it is thought that as grid frequency lowers the main generator will become under excited and reactive load will increase.
D. Incorrect. Plausible because under excitation will cause reactive load to decrease, however, grid frequency lowering causes over excitation.

Comments / Reference: Bank 44282	Revision:

Question 429 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	44282
User-Defined ID:	ILOT8484
Cross Reference Number:	ABN.601.OB00.001
Topic:	Given the following conditions: Unit 2 is in MODE 1, Reactor power is 90%. Main Generator load i
K/A:	
Question Reference:	
SRO:	
Comments:	LC20 NRC Ref: ABN-601 Sect. 9; GFC.MTR

Comments / Reference: ABN-601	Revision: ~~16~~ 17

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-601
RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION	REVISION NO. 17	PAGE 132 OF 256

9.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- Automatic grid frequency relays provide for load shedding at frequency between 59.3 Hz and 58.5 Hz. If the grid frequency has not recovered after 9 minutes, other plants under-frequency protection may lead to additional lowering of grid frequency.
- IF frequency recovers to >59.4 Hz, STOP the timer AND RESET. START a new 9 minute timer if frequency drops below 59.4 Hz.
- Reactor trip will occur on Reactor Coolant Pump under-frequency trip at 57.2 Hz (2 of 4 RCPs) when power is above P-7(10%).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 4 CHECK Frequency-GREATER THAN 59.4 Hz (1782 rpm)	IF Frequency is LESS THAN 57.2 Hz THEN ENSURE Reactor Tripped AND GO TO EOP-0.0A/B PERFORM the following steps for both units. A. START 9 minute timer AND CALL QSE. B. STABILIZE plant power level. C. IF Grid Frequency is still <59.4 Hz after 2 minutes, THEN PERFORM the following to divorce the safeguards busses from the grid: 1) PERFORM an Emergency Start of the Train A diesel generator. 2) TURN ON the synchroscope for the Train A diesel output breaker AND PARALLEL the diesel with off-site power. 3) CLOSE the diesel output breaker AND TURN OFF the synchroscope.

"Step continued next page"

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-601
RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION	REVISION NO. 16	PAGE 131 OF 256

9.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- Generator load will decrease with decreasing frequency (due to reduction in generator efficiency) and automatic load control will restore the load to the set load. This will result in increasing steam flow and increased reactor power. MAINTAIN reactor power less than 100%.
- Steps 1 through 3 should be considered continuous action steps during periods of grid instability.

☐ 1 MAINTAIN Reactor Power - less than or equal to 100%.

☐ 2 VERIFY QSE Generation Controller communications - AVAILABLE — CONTROL frequency as necessary per Attachment 2.

CAUTION: WHEN CPNPP trips both Reactors it is highly probable that the grid will be lost AND a loss of all offsite power will occur.

☐ 3 PERFORM the following as applicable:

FREQUENCY	ACTION
>60.6 Hz (1818 rpm)	• MAINTAIN contact with QSE • STABILIZE plant power for load reduction • IF immediate recovery is NOT evident after 9 minutes, THEN COORDINATE with QSE to determine actions (turbine overspeed trip occurs between 1830-1845 rpm).
≤57.5 Hz (1725 rpm)	**IMMEDIATELY** TRIP reactor AND GO to EOP-0.0A/B
≤58.0 Hz (1740 rpm)	**AFTER 2 sec** TRIP reactor AND GO to EOP-0.0A/B
≤58.4 Hz (1752 rpm)	**AFTER 30 sec** TRIP reactor AND GO to EOP-0.0A/B
≤59.4 Hz (1782 rpm)	**GO to STEP 4**
>59.4 Hz (1782 rpm)	Continuous operation allowed.

Section 9.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	1		
	Group	1		
	K/A	WE11.EA1.03		
Level of Difficulty: 3	Importance Rating	3.7		

Loss of Emergency Coolant Recirculation: Ability to operate and / or monitor the following as they apply to the Loss of Emergency Coolant Recirculation: Desired operating results during abnormal and emergency situations

Question # 55

Given the following Unit 1 conditions:

- ECA-1.1A, Loss of Emergency Coolant Recirculation, entered following a LOCA
- RCS subcooling margin 88ºF
- One Train of ECCS equipment has been secured
- RWST Make-up is being initiated

Per ECA-1.1A, makeup to the RWST may be accomplished utilizing __(1)__.

Subsequently, ECA-1.1A directs the RCS be depressurized to establish a subcooling margin between 25ºF and 35ºF in order to __(2)__.

A. (1) RHUT X-01
(2) maximize CCP injection flow prior to RWST depletion

B. (1) U2 RWST
(2) maximize CCP injection flow prior to RWST depletion

C. (1) RHUT X-01
(2) minimize RCS break flow

D. (1) U2 RWST
(2) minimize RCS break flow

Answer: D

K/A Match: K/A match due to requiring knowledge of loss of coolant recirc procedures to obtain desired results.

Explanation:

A. Incorrect. First part is incorrect, but plausible because this is a borated makeup source utilized in other procedures to provide a makeup volume (i.e. SFPs). Second part is incorrect, but plausible since this would provide maximum makeup to RCS, but conflicts with objective to conserve RWST inventory.

B. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A).

C. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D).

D. Correct. First part is correct. ECA-1.1A, Attachment 3 allows use of U2 RWST as a source of makeup to the U1 RWST. Second part is correct. Depressurizing to minimize subcooling margin while cooling down ensures that the RCS is at the minimum pressure allowable while still maintaining the RCS subcooled. This minimizes RCS break flow and thereby minimizes the amount of makeup flow required to the RCS.

Technical Reference(s)	ECA-1.1A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given a procedural step, or sequence of steps from ECA-1.1, **STATE** the purpose/basis for the step(s). (ERG.C11.OB04)

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 X

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: ECA-1.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.1A
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 9	PAGE 17 OF 83

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: The upper head region may void during RCS depressurization if RCPs are not running. This will result in a rapidly increasing PRZR level.	
27	Depressurize RCS To Decrease RCS Break Flow:	
	a. RCS subcooling - GREATER THAN 35°F (65°F FOR ADVERSE CONTAINMENT)	a. Go to Step 28.
	b. Use normal PRZR spray.	b. Use one PRZR PORV. IF PORV NOT available OR effective, THEN use auxiliary spray.
	c. Depressurize RCS until either of the following conditions satisfied: • RCS subcooling - BETWEEN 25°F AND 35°F (55°F AND 65°F FOR ADVERSE CONTAINMENT). -OR- • PRZR level - GREATER THAN 75% (65% FOR ADVERSE CONTAINMENT)	c. IF RCS subcooling is less than 25°F (55°F FOR ADVERSE CONTAINMENT), THEN increase RCS makeup flow as necessary to restore subcooling.
	d. Stop RCS depressurization.	

Comments / Reference: ECA-1.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.1A
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 9	PAGE 69 OF 83

ATTACHMENT 7
PAGE 15 OF 29

BASES

CAUTION: One of the goals of ECA-1.1A is to minimize ECCS flow to reduce RWST depletion and delay the time until injection flow is lost. The criterion to increase ECCS or makeup flow is based on the prevention of inadequate core cooling. The criteria to increase makeup flow based on RVLIS indication is the same criteria that is used to diagnose degraded core cooling in the Critical Safety Function Status Trees. If level in the vessel cannot be maintained above the top of the core, then it is only a matter of time before the core will start to heat up and additional makeup flow is needed. Using the RVLIS criterion creates an ERG priority conflict between the Orange path based on RVLIS indication less than 11 IN Core Cooling Critical Safety Function Status Tree and the ECCS reinitiation criteria in ECA-1.1A. Restoration of ECCS or makeup flow should first be attempted in ECA-1.1A and, only if the operator is not successful in restoring RVLIS level to greater than 11 IN, then the transfer to the FRGs on low RVLIS level should be made for additional actions.

STEP 26: This step instructs the operator to verify that a makeup flow increase is not required by checking that the RVLIS indication is above the top of the core and that core exit TCs are stable or decreasing. This ensures that the makeup flow reduction performed in the previous steps was done properly; i.e., if the makeup flow is now inadequate due to the actions of Steps 21 through 25, it will be detected in this step and makeup flow will be increased.

This is a Continuous Action Step.

NOTE: Without RCPs running, there is very little flow into the upper head region. Liquid in that region remains relatively hot even though the liquid temperature in the active regions of the RCS has been significantly reduced during the RCS cooldown. As the RCS is subsequently depressurized, the hotter liquid in the upper head may flash to steam, forming an upper head void. Steam formation in the upper head will displace water into the PRZR, causing rapidly increasing PRZR level with the potential for water relief through the PRZR PORVs. The PRZR may fill with water within a few minutes. This note informs the operator of the potential for this condition, so that RCS depressurization can be stopped quickly to avoid a water solid PRZR.

STEP 27: The RCS pressure reduction that is performed in this step is done to decrease RCS break flow for small break LOCAs when the RCS is in a subcooled condition. The purpose of the depressurization is to decrease RCS pressure to the lowest pressure possible without losing subcooling. For large break LOCAs, the RCS would already be depressurized and RCS subcooling would not exist and this step should not be performed. When this step is entered, RCS subcooling is checked first. If RCS subcooling is not adequate, the operator will be directed to the next step and this step will not be performed. If RCS subcooling is adequate, the operator will proceed on and perform this step.

Comments / Reference: ECA-1.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.1A
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 9	PAGE 35 OF 83

ATTACHMENT 3
PAGE 3 OF 17

RWST MAKEUP METHODS

[R]
2. Makeup to Unit 1 RWST From Unit 2 RWST (A PB Key is required to enter the rooms).

CAUTION: Reducing water volume in Unit 2 RWST may exceed Unit 2 operating limitations (TS 3.5.4, and TRM 13.1.31 and 13.1.32). Unit 2 operating restrictions should be evaluated prior to transferring water to Unit 1 RWST. Notify the Plant Staff if time permits prior to transferring Unit 2 RWST to Unit 1 RWST

a. Ensure the following valves open:

- ☐ • 2SI-0047, RWST 2-01 TO SI ISOL VLV (Unit 2 RWST Tank room).
- ☐ • 2SI-8977, RWST 2-01 TO SFPCS WTR PURIF PMP X-01/X-02 ISOL VLV (AB 790 in overhead)

b. Open Unit 2 RWST Isolation valves (Unit 2 MCB, CB-02)

- ☐ • 1/2-8800A, RWST TO SFPCS DRN VLV
- ☐ • 1/2-8800B, RWST TO SFPCS DRN VLV

-Cont 2-

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	1		
	K/A	WE05.EK2.01		
Level of Difficulty: 3	Importance Rating	3.7		

Loss of Secondary Heat Sink: 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

Question # 56

Given the following conditions:

Time = 0800:

- Reactor power = 100%
- Loss of all MFW occurs

Time = 0820:

- FRH-0.1A, Response to Loss of Secondary Heat Sink, in progress
- Condensate flow is to be established to the SGs
- CVCS Letdown is isolated

In accordance with FRH-0.1A, before SG pressure is reduced, RCS pressure will be reduced to approximately __(1)__ to allow blocking of automatic SI actuation signals.

The PREFERRED method of RCS depressurization is to use __(2)__.

A. (1) 1910 psig
(2) a Pressurizer PORV

B. (1) 1910 psig
(2) Auxiliary Spray

C. (1) 1880 psig
(2) a Pressurizer PORV

D. (1) 1880 psig
(2) Auxiliary Spray

Answer: A

K/A Match: K/A match due to requiring knowledge of the method of RCS depressurization during a loss of heat sink event.

Explanation:

A. Correct. 1st part is correct. RCS pressure is reduced to ~ 1910 psig to allow blocking of the low steam line pressure and low Przr pressure SI signals. 2nd part is correct. If letdown is not in service, the PORV is used to reduce pressure.

B. Incorrect. 1st part is correct (see A). 2nd part is incorrect because with letdown not in service, the PORV is used to reduce pressure. It is plausible because if letdown were in service, it would be correct.

C. Incorrect. 1st part is incorrect because FRH-0.1A directs you to reduce RCS pressure to ~ 1910 psig. It is plausible because 1880 psig is the trip value and above the value where the SI signal occurs. 2nd part is correct (see A).

D. Incorrect. 1st part is incorrect but plausible (see C). 2nd part is incorrect but plausible (see B).

Technical Reference(s)	FRH-0.1A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRH-0.1 in accordance with FRH-0.1, Loss of Heat Sink. (ERG.FH1.OB04)

Question Source: Bank # 77325
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: FRH-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 9	PAGE 12 OF 85

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	CAUTION: Following block of automatic SI actuation, manual SI actuation may be required if conditions degrade.	
	NOTE: After the low steamline pressure SI signal is blocked, main steam isolation will occur if the high steam pressure rate setpoint is exceeded.	
9	Establish Feed Flow From Condensate System:	
	a. Depressurize RCS to less than 1910 psig:	
	1) Turn off all PRZR heaters.	
	2) Check letdown - IN SERVICE	2) Use one PRZR PORV. IF NOT, THEN use auxiliary spray. Go to Step 9b.
	3) Use auxiliary spray.	3) Use one PRZR PORV.
	b. Block SI signals: • Low steamline pressure SI • Low PRZR pressure SI	
	c. Depressurize at least one SG to less than 500 psig:	
	1) Dump steam to condenser at maximum rate and avoid main steam isolation.	1) Manually or locally dump steam from SGs using intact SG atmospheric(s). IF NOT, THEN go to Step 11. OBSERVE CAUTION AND NOTES PRIOR TO STEP 11.

-CONT 9-

Comments / Reference: FRH-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 9	PAGE 58 OF 85

ATTACHMENT 4
PAGE 6 OF 33

BASES

CAUTION: The next step blocks the low steamline pressure SI and the low PRZR pressure SI signals in order to depressurize a SG, and establish condensate flow to the SG. Blocking the SI signal prevents feedline and steamline isolation on low steam pressure which, if allowed to occur, could hamper or delay recovery.

Operator attention to plant conditions is necessary to ensure that SI is not required. Manual SI actuation may be required if conditions deteriorate.

The level of plant condition degradation required before manual SI actuation is based on operator judgment after assessing the plant parameters of RCS subcooling and PRZR level. Using these parameters as indicators for the level of degradation is consistent with the parameter requirements of SI Reinitiation used throughout the ERG network.

NOTE: Following low steamline pressure SI signal block, the high steam pressure rate main steamline isolation signal will be enabled. If automatic MSIV closure occurs, steam flow to condenser is terminated requiring SG depressurization to be continued by dumping steam to the atmosphere. In addition to delaying recovery, closure of the MSIVs increases the radiological releases and reduces feedwater supply. With RCPs off, establishing the steaming rate will need to be performed more slowly. If MSIV closure occurs, the rapid cooldown should be continued using the SG atmospherics.

STEP 9: The condensate system is the next source of water readily available to the operator for use in re-establishing the secondary heat sink.

In order to depressurize at least one SG to less than the shutoff head pressure of the condensate system pumps, the RCS must be depressurized to 1910 psig (Conservatively below the P-11 setpoint of 1960 psig) to allow blocking of the low steamline pressure SI and low PRZR pressure SI signals. If these signals were allowed to actuate, feedline and steamline isolation actuation signals may have to be reset. Feedline isolation may still occur on a reactor trip signal coincident with the low Tavg signal.

Auxiliary spray is used to depressurize the RCS, if letdown is in service, since it provides a maximum cooling to the primary system while allowing no loss of primary water inventory. Normal spray is not available since RCPs are stopped. If letdown is not in service, PRZR PORVs are used to avoid thermal stresses to the auxiliary spray nozzles. However, if the PRZR PORVs cannot be used, auxiliary spray must be used. If SI is actuated for RCS bleed and feed, auxiliary spray flow will not be effective since the charging flow path is aligned through the CCP SI flowpath, and it is not desired to isolate CCP SI flow.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	1		
	Group	2		
	K/A	000001.AK2.06		
Level of Difficulty: 3	Importance Rating	3.0		

Continuous Rod Withdrawal: Knowledge of the interrelations between the Continuous Rod Withdrawal and the following: T-ave./ref. deviation meter

Question # 57

Given the following conditions:

- Unit 1 60% power
- Rod Control in AUTO
- PT-505, First Stage Pressure Instrument fails, causing the T_{ave} - T_{ref} Deviation Meter to indicate -10°F
- Control rods begin stepping
- Crew enters ABN-709, Steam Line, Steam Header & Turbine 1st Stage Pressure & Feed Header Pressure Instrument Malfunction

Control rods are stepping __(1)__.

Per ABN-709, Control Rods should be placed in MANUAL and __(2)__.

A. (1) inward
(2) are required to remain in MANUAL until the failed Turbine First Stage Pressure channel is repaired

B. (1) inward
(2) may be placed in AUTO after the alternate Turbine First Stage Pressure channel is selected

C. (1) outward
(2) are required to remain in MANUAL until the failed Turbine First Stage Pressure channel is repaired

D. (1) outward
(2) may be placed in AUTO after the alternate Turbine First Stage Pressure channel is selected

Answer: D

K/A Match: K/A match due to requiring knowledge of the relationship between the Tave-Tref deviation meter and rod movement.

Explanation:

A. Incorrect. First part is incorrect, but plausible since a Tave-Tref deviation will cause rod motion, but indication of -10ºF will result in outward rod motion. Second part is incorrect, but plausible because other systems that utilize first stage pressure (i.e. Steam Dumps) are required to be maintained in off-normal condition (steam dumps must be maintained in Steam Pressure Mode vs Tave Mode). Also, on a Tcold failure the rods must remain in Manual until the failed Tcold channel is repaired.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D)

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A).

D. Correct. First part is correct. Tave-Tref deviation meter indicating -10ºF will result in outward rod motion. Second part is correct, per ABN-709 after selecting the alternate channel on 1-PS-505Z rods may be placed back in AUTO.

Technical Reference(s)	ABN-709	Attached w/ Revision # See Comments / Reference
	Rod Control Study Guide	

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to a Turbine Impulse Pressure Instrument Malfunction in accordance with ABN-709 Steam Line Pressure, Steam Header Pressure, Turbine 1st-Stage Pressure and Feed Header Pressure Instrument Malfunction. (ABN.710.OB07)

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 X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ABN-709	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-709
STM LINE, STM HDR & TURB 1st STAGE PRESS. & FEED HDR PRESS. INSTR MALFUNCTION	REVISION NO. 10	PAGE 13 OF 35

4.0 TURBINE IMPULSE PRESSURE INSTRUMENT MALFUNCTION

4.1 Symptoms

a. Annunciator Alarms

- AVE Tave-Tref DEV (6D-1.10)
- Tref - AUCT LO Tave MISMATCH (6D-3.13)

b. Plant Indications

- Turbine impulse pressure channels not indicating the same.

1) u-PI-505, TURB IMP PRESS CHAN I

2) u-PI-506, TURB IMP PRESS CHAN II

4.2 Automatic Actions

NOTE:	TURB IMP PRESS CHAN SELECT u-PS-505Z should normally be in the "BOTH" position and input for control rod response will be from PT-505. With this switch in the PS-506 position, input for control rod response will be from PT-506.

a. Pressure Transmitter u-PT-505 (normally selected for Tref)

- u-PT-505 failing high will cause control rods to withdraw if in automatic.
- u-PT-505 failing low will cause control rods to insert if in automatic, open steam dumps if an arming signal is present and disable AMSAC actuation (PCIP 1.3, AMSAC BLK TURB <40% PWR C-20, LIT).

b. Pressure Transmitter u-PT-506

- u-PT-506 failing high will prevent steam dump actuation on an actual loss of load .
- u-PT-506 failing low will arm the steam dumps (the signal seals in) and disable AMSAC actuation (PCIP 1.3, AMSAC BLK TURB <40% PWR C-20, LIT).

Section 4.0

Comments / Reference: ABN-709	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-709
STM LINE, STM HDR & TURB 1st STAGE PRESS. & FEED HDR PRESS. INSTR MALFUNCTION	REVISION NO. 10	PAGE 14 OF 35

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 PLACE 1/u-RBSS, CONTROL ROD BANK SELECT Switch in - MANUAL	

NOTE: The following step will prevent automatic steam dump actuation on an actual load rejection, if RNO step is applied.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2 VERIFY Steam Dumps - CLOSED WITH NO OPEN DEMAND ☐ a. u-UI-500, STM DMP DEMAND, indicating 0% DEMAND. ☐ b. STM DMP VLV ZL lights indicating - CLOSED.	IF steam dump operation NOT required, THEN PLACE at least one steam dump interlock select switch - OFF: • 43/u-SDA, STM DMP INTLK SELECT • 43/u-SDB, STM DMP INTLK SELECT

CAUTION: A briefing should be conducted to evaluate steam dump response and contingency actions should a subsequent runback or trip occur. Reference Section 4.2.

NOTE:
- If transferring dumps to steam pressure mode, steam demand will be erroneously high if PT-505 is failed low.
- The following step ensures steam dumps available for subsequent runbacks or trips.
- Attachment 8 is available to aid in brief (L)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3 RESTORE steam dump availability by placing Steam Dumps in STM PRESS Mode per Attachment 7.	
☐ 4 TRANSFER u-PS-505Z, TURB IMP PRESS CHAN SELECT to operable channel	
☐ 5 ENSURE Tave within 1°F of Tref.	

Section 4.3

Comments / Reference: ABN-709	Revision: 10

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-709
STM LINE, STM HDR & TURB 1st STAGE PRESS. & FEED HDR PRESS. INSTR MALFUNCTION	REVISION NO. 10	PAGE 15 OF 35

4.3 Operator Actions

		ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐	6	RETURN 1/u-RBSS, CONTROL ROD BANK SELECT Switch to AUTO, if desired.	
☐	7	CHECK Reactor Plant in - MODE 1	IF Reactor Plant is in MODE 2, THEN DO NOT attempt to enter MODE 1 until failed channel is repaired.
☐	8	CHECK Turbine Power - GREATER THAN 10% POWER.	ENSURE affected channel bistable listed on Attachment 5 - RESET (normal).

NOTE: The following step will prevent the automatic block of several reactor trips when Reactor power is below 10% power.

		ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐	9	Within 1 hour, VERIFY PCIP window 4.6, TURB ≤10% PWR P-13 - IN PROPER STATE for existing plant conditions. (TS Table 3.3.1-1, item 18.f)	PERFORM the following: a. Within 1 hour, HAVE an I&C Technician place bistable test switches for failed channel in CLOSE utilizing Attachments 1 and 3. b. VERIFY appropriate alarms AND trip status lights ON per Attachment 5 AND NOTE this verification in Unit Log. IF I&C Technician NOT available, THEN a Reactor Operator may perform this step with Shift Manager/Unit Supervisor concurrence.
☐	10	VERIFY PCIP window 1.3, AMSAC BLK TURB <40% PWR C-20 - IN PROPER STATE for actual turbine power.	IF AMSAC actuation blocked AND turbine power >40%, THEN ENSURE Automatic Actions of ALB-9B 3.7, AMSAC ACT TURB TRIP as necessary.
☐	11	INITIATE a Condition Report per STA-421, as applicable.	

END OF SECTION

Section 4.3

Comments / Reference: Rod Control Study Guide	Revision: 00-0000

OP51.SYS.CR1

Failure of PT-505, Turbine First Stage Pressure

Indications of this condition are receipt of the AVG Tave-Tref DEV or Tref-AUCT LO Tave MISMATCH alarms and/or inadvertent rod motion. If PT-505 fails high, the control rods would withdraw if in the automatic mode. If PT-505 fails low, the control rods will insert and, with the presence of an arming signal, operate the steam dumps. Response to this condition is in accordance with ABN-709 and includes placing the control rod in manual to prevent inadvertent operation.

Abnormal Control Rod Response

The indications of this condition include abnormal rod speed, failure to achieve rod motion when called for, inadvertent rod motion, abnormal changes in plant parameters for a rod motion, improper rod sequencing, or improper bank overlap. This condition could lead to other plant actions such as a reactor trip, turbine runback, or steam dump operation. Because of this, immediate operator response is necessary. Response to this condition is in accordance with ABN-712 Section 2.0.

Dropped or Misaligned Rod in Mode 1 or 2

A control rod is considered to be misaligned when two DRPI's in the same group disagree by greater than or equal to 12 steps or one DRPI disagrees with its group step counter by greater than or equal to 12 steps. Possible annunciator alarms that would indicate a misaligned rod(s) condition are PR Chan Dev, DRPI Rod Dev and/or a Quadrant PWR Tilt. The procedure for a misaligned rod recovery is ABN-712.

The recovery procedure initially places the reactor in a safe condition by verifying various plant parameters and taking actions accordingly to bring these back into operating bands. These include T_{avg}-T_{ref}, Reactor Power and the Axial Flux Difference. The Axial Flux Difference is defined as the difference in the normalized flux signals between the top and the bottom halves of a four section excore neutron detector. Basically it is the relationship of how the neutron flux is distributed across the core from top to bottom. The Quadrant Power Tilt Ratio (QPTR) is then verified to be within specifications by verifying the alarm clear. The Quadrant Power Tilt is defined as the ratio of the maximum upper half excore detector calibrated output, to the average of the upper half excore detector calibrated outputs, or the ratio of the maximum lower half excore detector calibrated output to the average of the lower half excore calibrated outputs, **whichever is greater**. If the alarm is in, the QPTR is calculated in accordance with OPT-302 or by the plant computer.

The cause of the misaligned or dropped control rod is investigated and must be determined within an hour or else implement the requirements of Tech. Spec. 3.1.4.1.

To retrieve a dropped rod, the operator places the bank selector switch to the affected group and records the position of that group (and P/A converter value). At the back of the control board, the lift coil disconnect switches are placed in the open position for all rods in the affected bank with exception of the affected rod. The operator then recovers the dropped rod using either the DRPI recovery method or the referencing method.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	1		
	Group	2		
	K/A	000024.AA1.10		
Level of Difficulty: 2	Importance Rating	3.3		

Emergency Boration: Ability to operate and / or monitor the following as they apply to Emergency Boration: Boric acid storage tank

Question # 58

Given the following conditions:

- Unit 1 Reactor has tripped
- During the Reactor Trip, the RO notes the following indications on DRPI:
    - Control Rod F-14 at 212 steps
    - Control Rod H-2 at 228 steps
    - Control Rod M-12 at 6 steps
    - All other rods at bottom
- 1-ALB-6B, Window 3.1 – BAT 1 LVL LO-LO, in alarm
- Boric Acid Tank X-01 level 54%

The crew is required to Emergency Borate __(1)__ in response to these conditions.

BAT X-01 __(2)__ have the required Technical Specification MINIMUM level for the given plant conditions.

A. (1) 3600 gallons
   (2) does

B. (1) 5400 gallons
   (2) does

C. (1) 3600 gallons
   (2) does NOT

D. (1) 5400 gallons
   (2) does NOT

Answer: B

K/A Match: K/A match due to requiring knowledge of the BAT operability requirements.

Explanation:

A. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

B. Correct. First part is correct. Three rods not fully inserted requires 5400 gallons of boric acid to ensure adequate SDM. Second part is correct. Minimum required volume is 50%.

C. Incorrect. First part is incorrect, but plausible since a rod stuck at 6 steps could be considered to be fully inserted, but the requirement is that 1800 gallons of boric acid is required for any rod not fully inserted. Second part is incorrect, but plausible since it could be thought that the LO-LO level alarm is indicative of the BAT being inoperable.

D. Incorrect. First part is correct (see B). Second part is incorrect, but plausible (see C).

Technical Reference(s)	EOP-0.0A	Attached w/ Revision # See Comments / Reference
	OPT-104A-1	

Proposed references to be provided during examination: ______

Learning Objective: Given the Reactor Makeup system parameter indications and plant conditions, **ASSESS** from memory any required Technical Specification/TRM entries, including any actions which must be completed within one hour in accordance with Technical Specifications and TRM. (SYS.CS2.OB06)

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

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: EOP-0.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 9	PAGE 20 OF 121

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOP-0.0A REACTOR TRIP OR SAFETY INJECTION

1. RCP TRIP CRITERIA

NOTE: ABN-101, REACTOR COOLANT PUMP TRIP/MALFUNCTION criteria for tripping an RCP is applicable during use of the Emergency Procedures.

Trip all RCPs if BOTH conditions listed below occur:

a. RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
b. CCP or SI pump - AT LEAST ONE RUNNING

2. SHUTDOWN MARGIN CRITERIA
Emergency borate per ABN-107 if either of the following conditions below occur:

- Two or more control rods NOT fully inserted (1800 gallons of 7000 ppm boric acid for each control rod not fully inserted).
- Control rod position indication is NOT available (3600 gallons of 7000 ppm boric acid).

NOTE: During the performance of the immediate operator actions, AFW FOP verbalization is neither required nor expected.

3. CONTROL AFW FLOW TO MAINTAIN ADEQUATE HEAT SINK

- Ensure both MDAFWPs started following a Reactor Trip with NO Blackout or SI actuation. (Start TDAFWP, if necessary)
- Ensure AFW flow throttled following a Reactor Trip/SI (normally 150 gpm to 200 gpm).
  AND
  Maintain total AFW flow GREATER THAN 460 gpm UNTIL at least ONE SG NR Level greater than 43%(50% for ADVERSE CONTAINMENT).
- IF any SG identified as faulted, THEN stop AFW flow to the SG.
- IF any SG identified as ruptured, THEN:
  Stop AFW flow after ruptured SG level greater than 43%(50% for ADVERSE CONTAINMENT).
  AND
  Control AFW flow to maintain ruptured SG level greater than 43%(50% for ADVERSE CONTAINMENT).
- IF BOTH MDAFWPs are running with flow THEN, secure the TDAFWP.

4. AFW SUPPLY SWITCHOVER CRITERION
IF CST level decreases to less than 10%, THEN switch to alternate AFW water supply per ABN-305, AUXILIARY FEEDWATER SYSTEM MALFUNCTION.

5. RCP SEAL INJECTION FLOW CRITERION
Ensure 6 gpm to 13 gpm seal injection flow to all RCPs UNLESS isolated by ERG actions.

-

Comments / Reference: OPT-104A-1	Revision: 21

OPERATIONS WEEKLY SURVEILLANCES

MODE	TECH SPEC	PARAMETERS	ACCEPTANCE CRITERIA	CHANNEL NUMBERS	READING	NOTES
ALL	3.5.4.2 13.1.32.7 (7 DA*)	REFUELING WATER STORAGE TANK LEVEL (%)	LEVEL ≥ 95% IN MODES 1 THROUGH 4. LEVEL ≥ 24% IN MODES 5 AND 6 (CCP). LEVEL ≥ 15% IN MODE 6 (SIP with RPV head removed).	1-LI-930 (CB-02)		IN MODE 5 OR 6, EITHER THE BORIC ACID STORAG TANK OR THE RWST MUST BE OPERABLE.
				1-LI-931 (CB-02)		
				1-LI-932 (CB-04)		
				1-LI-933 (CB-04)		
ALL	13.1.31.3 13.1.32.4 13.1.32.5 (7 DA*)	BORIC ACID STORAGE TANK LEVEL (%)	LEVEL ≥ 50% IN MODES 1 THROUGH 4. LEVEL ≥ 10% IN MODES 5 AND 6. INDICATE THE BAT USED FOR UNIT 1 BY CIRCLING THE OPERABLE LIs. N/A THE BAT READINGS FOR THE TANK USED FOR UNIT 2.	X-LI-102 (CB-06) BA TK 1 LVL		IN MODE 5 OR 6, EITHER THE BORIC ACID STORAG TANK OR THE RWST MUST BE OPERABLE.
				X-LI-104 (CB-06) BA TK 1 LVL		
				X-LI-105 (CB-06) BA TK 2 LVL		
				X-LI-106 (CB-06) BA TK 2 LVL		
ALL	13.1.31.1 13.1.32.2 (7 DA*)	BORIC ACID STORAGE TANK TEMPERATURE (°F)	TEMPERATURE ≥ 65°F. INDICATE THE BAT USED FOR UNIT 1 BY CIRCLING THE OPERABLE TI. N/A THE BAT READING FOR THE TANK USED FOR UNIT 2.	X-TI-103 (1-CB-06) BA TK 1 TEMP		
				X-TI-107 (1-CB-06) BA TK 2 TEMP		
ALL	13.1.31.1 13.1.32.2 (7 DA*)	BORIC ACID FLOW PATH TEMPERATURE(°F) (RECORD SAT OR UNSAT)	ALL TEMPERATURES IN THE FLOW PATH ARE ≥ 65°F. XHT-9B CIRCUITS ARE BACKUP CIRCUITS FOR XHT-8B. EITHER CIRCUIT FROM XHT-8B OR XHT-9B MAY BE USED TO SATISFY THE ACCEPTANCE CRITERIA.	XHT-8B (AB 832) XHT-9B (AB 832) List non functioning circuits:		CHECK EACH CHANNEL BY PLACING THE CAL/RD IN THE RD POSITION. ONLY REQUIRED IF THE BO ACID STORAGE TANKS ARE BEING USED AS ONE REQUIRED SOURCES OF BORATED WATER. CIRCUITS REQUIRED ON XHT-8B OR 9B ARE: 1, 2, 5, 8, 9, 11, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 2 31, 32, 33.
ALL	3.8.1.1 3.8.2.1 3.8.7.1 3.8.8.1 3.8.9.1 3.8.10.1 (7 DA*)	AC AND DC POWER SOURCES (SAT OR UNSAT)	AC AND DC POWER ALIGNED PER THE APPLICABLE SECTIONS OF OPT-215.	PERFORM OPT-215-1 AND OPT-215-11		

REFERENCE USE

OPT-10
PAGE
REV. 2

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	1		
	Group	2		
	K/A	000037.AK3.05		
Level of Difficulty: 3	Importance Rating	3.7		

Steam Generator Tube Leak: Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: Actions contained in procedures for radiation monitoring, RCS water inventory balance, S/G tube failure, and plant shutdown

Question # 59

Given the following conditions:

- Unit 1 100% power
- Current plant conditions as follows:
  - PRZR level 60% stable
  - Letdown flow 130 gpm
  - CCP 1-01 in service
  - COG-182, Condenser Off-Gas Radiation Monitor, in RED alarm
  - N16-175, # 2 Main Steam Line N16 Radiation Monitor, in RED alarm
  - MSL-179, # 2 Main Steam Line Radiation Monitor, in RED alarm
- ABN-106, High Secondary Activity, in progress

Subsequently:

- Unit 1 performed a rapid shutdown in accordance with IPO-003A, Power Operations

Based on indications provided, the primary-to-secondary leak rate is __(1)__.

Per ABN-106, the subsequent RCS cooldown is to be__(2)__.

A. (1) ≥ 75 gpd, but < 3600 gpd (2.5 gpm)
   (2) limited to a rate of 100ºF per hour

B. (1) ≥ 75 gpd, but < 3600 gpd (2.5 gpm)
   (2) established at maximum achievable rate without causing a MSLI

C. (1) ≥ 3600 gpd (2.5 gpm)
   (2) limited to a rate of 100ºF per hour

D. (1) ≥ 3600 gpd (2.5 gpm)
   (2) established at maximum achievable rate without causing a MSLI

Answer: C

K/A Match: K/A match due to requiring knowledge of the indications of a SG tube leak and the reason for actions taken to recover.

Explanation:

A. Incorrect. First part is incorrect, but plausible since N16-175 minimum sensitivity is 1 gpd and has a range up to 150 gpd. Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since the initial cooldown during a SGTR, during the performance of EOP-3.0, is at the maximum achievable rate.

C. Incorrect. First part is correct (see D). Second part is correct. The cooldown rate used in ABN-106 is limited to 100°F per hour.

D. Correct. First part is correct. MSL-179 minimum sensitivity and being in alarm indicates the leak is greater than 2.5 gpm (3600 gpd). Second part is incorrect (see B).

Technical Reference(s)	ABN-106	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to a Steam Generator Tube Leakage greater than or equal to 75 gpd in accordance with ABN-106, High Secondary Activity. (ABN.106.OB02)

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 X

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: ABN-106	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 11	PAGE 15 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE: Due to the minimum sensitivity of the MSL radiation monitors, a valid alarm indicates a leak rate of at least 3600 gpd (2.5 gpm).	
☐ 1 Verify main steamline radiation alarms - CLEAR • u-RE-2325 (MSL-u78) • u-RE-2326 (MSL-u79) • u-RE-2327 (MSL-u80) • u-RE-2328 (MSL-u81)	a. Initiate power reduction to ≤50% in 1 hour AND Be in MODE 3 in the next 2 hours. [C] b. Calculate gross leak rate, refer to EPP-201. c. GO TO Step 4.b.

(STA-732, Attachment 8.B) INFORMATION ONLY - ACTIONS DIRECTED BY THIS PROCEDURE

NOTE:
- Leakage is qualitatively confirmed when two independent radiation monitors trend in the same direction with the same order of magnitude. With two confirmed indications (i.e. N-16 and COG monitors or other combination of monitor indication and sample analyses):
- CPNPP uses the CONSTANT LEAKAGE METHOD.

LEAKAGE/LEAK RATE	ACTION
Primary to secondary leakage ≥75 gpd (0.052 gpm) for Unit 1 and ≥ 50 gpd (0.0347 gpm) for Unit 2, Cycle 19 sustained for ≥1 hour	Normal shutdown to be in MODE 3 in ≤24 hours
Primary to secondary leakage ≥100 gpd (0.07 gpm) OR Primary to secondary leakage ≥75 gpd (0.052 gpm) for Unit 1 and ≥ 50 gpd (0.0347 gpm) for Unit 2, Cycle 19 sustained for ≥1 hour AND NO condenser off-gas radiation monitor available AND main steam line leak rate radiation monitor on affected SG(s) - NOT OPERABLE	Reduce power to ≤50% in 1 hour AND Be in MODE 3 in the next 2 hours

Section 3.3

Comments / Reference: ABN-106	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 11	PAGE 25 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Control cooldown to maintain pressurizer level greater than 17%.
PZR level shall not be raised to >30% until SI is blocked.
The RCS shall not be cooled down below 510 deg prior to SI block.
SI shall not be blocked until adequate SDM for 350 deg, xenon free, has been verified.

(Provide Attachment 2 to RO to track cooldown requirements)

NOTE:
- An initial cooldown rate of 30-60 deg/hr is recommended to enhance PZR level control and allow time to adjust AFW flow.
- Just prior to commencing cooldown, raise charging flow approximately 1gpm for each deg/hr of cooldown rate, to offset RCS contraction.

17 Cooldown the RCS:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ a. Initiate monitoring of the RCS pressure AND temperature per OPT-407.	IF Steam Dumps to condenser can NOT be used, THEN cool down the RCS using intact Steam Generator Atmospheric Relief(s)
☐ b. Ensure Steam Dumps in STM PRESS mode in manual.	
☐ c. Adjust u-PK-507, STM DMP PRESS CTRL to maintain Cooldown Rate - LESS THAN OR EQUAL TO 100°F/hr	
☐ d. WHEN P-12 (553° F TAVG) is reached, THEN select bypass interlock on Steam Dumps and continue cooldown.	
☐ e. Adjust charging flow as needed to control PZR level 17%-30%.	

Section 3.3

Comments / Reference: ABN-106	Revision: 11

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 11	PAGE 31 OF 31

ATTACHMENT 2
PAGE 1 OF 1

COOLDOWN LIMITATIONS

Prior to SI Block

- Maintain PZR level >17% but < 30%
- Maintain PZR pressure 1900 - 1950 psig
- Maintain RCS temperature >510°F
- Control non-affected SG levels 60-75%
- Control leaking SG level > 43% (10% Unit 2)
- Max cooldown rate is 100°F/Hr (but should be less for control)

After SI Block

- Maintain sub-cooling 60-70°F
- Maintain RCS pressure >900 psig until the accumulators are isolated
- Control non-affected SG levels 60-75%
- Control leaking SG level > 43% (10% Unit 2)
- Max cooldown rate is 100°F/Hr

Attachment 2

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	2		
	K/A	000059.AA1.01		
Level of Difficulty: 2	Importance Rating	3.5		

Accidental Liquid Radwaste Release: Ability to operate and / or monitor the following as they apply to the Accidental Liquid Radwaste Release: Radioactive-liquid monitor	
Question # 60	

PC-11, Digital Radiation Monitoring System is alarming and the display for 1-RE-5100, Turbine Building Sump 1-02 Radiation Detector is RED.

1-RE-5100, Turbine Building Sump 1-02 Radiation Detector has a(an) __(1)__ alarm.

In accordance with ALM-3200, Alarm Procedure DRMS, the Radwaste Operator should verify Turbine Building drains have shifted to the __(2)__.

A. (1) OPERATE FAILURE
(2) Co Current Waste System

B. (1) OPERATE FAILURE
(2) Low Volume Waste Pond

C. (1) HIGH
(2) Co Current Waste System

D. (1) HIGH
(2) Low Volume Waste Pond

Answer: C

K/A Match: K/A match due to requiring the ability to monitor radiation monitor alarms during liquid waste releases.

Explanation:

A. Incorrect. First part is incorrect, but plausible because PC-11 can have an OPERATE FAILURE alarm with automatic actions but has a BLUE display. Second part is correct. Procedure ALM-3200 directs the crew to verify the proper automatic actions have occurred per Attachment 3. RE-5100 in alarm or with an OPERATE FAILURE diverts from LVW to Cocurrent Waste.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since the path is shifted between LVW and Cocurrent Waste, but from LVW to Cocurrent Waste, not the reverse.

C. Correct. First part is correct. Red alarm is indication of an actual high radiation condition. Second part is correct (see A).

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ALM-3200	Attached w/ Revision # See Comments / Reference
	ABN-903	

Proposed references to be provided during examination: ______________________

Learning Objective: **DESCRIBE** the response to an Accidental Release of Radioactive Liquid in accordance with ABN-903, Accidental Release of Radioactive Liquid. (ABN.201.OB13)

Question Source:
Bank # 36101
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2009 NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments / Reference: ABN-903	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-903
ACCIDENTAL RELEASE OF RADIOACTIVE LIQUID	REVISION NO. 8 CONTINUOUS USE	PAGE 3 OF 16

2.0 ACCIDENTAL RELEASE OF RADIOACTIVE LIQUID

2.1 Symptoms

a. Annunciator Alarms

Main Control Board

- LWPS PNL TRBL (6B-4.7)

LWPS PANEL

- LWPS EFFLUENT MONITOR ALERT (2.6)

b. Plant Indications

1) An unexpected increase in any of the following liquid process effluent monitors:

- X-RE-5251A (ABP074) LVW/EVAP POND VNT & DRN HDR RADIATION DETECTOR 5251A
- X-RE-5253 (LWE076) LIQUID WASTE PROCESSING DISCHARGE RADIATION DETECTOR
- u-RE-4269 (SSWu65) UNIT u STATION SERVICE WATER TRAIN A TO DISCH CANAL RAD DETECTOR
- u-RE-4270 (SSWu66) UNIT u STATION SERVICE WATER TRAIN B TO DISCH CANAL RAD DETECTOR
- 1-RE-5100 (TBD172) TURBINE BUILDING SUMP 1-02 RADIATION DETECTOR
- 2-RE-5100 (TBD272) TURBINE BUILDING SUMP 2-04 RADIATION DETECTOR

2) Waste Water Hold-up Tank or piping leak or spill reported by Plant Personnel.

2.2 Automatic Actions

a. A High Alarm on X-RE-5251A will realign sump discharge from the LVW system to the COW system.

b. A High Alarm on X-RE-5253 the Liquid Waste discharge process radiation monitor closes X-RV-5253 Liquid Waste Discharge Isolation Valve.

c. A High Alarm on the Turbine Building Sump 1-02/2-04 discharge monitor u-RE-5100 will realign sump discharge from the LVW system to the COW system.

Section 2.0

Comments / Reference: ABN-903	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-903
ACCIDENTAL RELEASE OF RADIOACTIVE LIQUID	REVISION NO. 8 / CONTINUOUS USE	PAGE 7 OF 16

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4 CHECK Radioactivity in turbine building sump.	
☐ a. VERIFY turbine building sump monitor on PC11 - NOT IN ALERT OR HI ALARM (GREEN/OPERATE) • Unit 1 only, TBD172 (1-RE-5100), TURBINE BUILDING SUMP 1-02 RADIATION DETECTOR • Unit 2 only, TBD272 (2-RE-5100), TURBINE BUILDING SUMP 2-04 RADIATION DETECTOR	a. NOTIFY Rad Waste to Perform following: 1) ENSURE Co-current WWHUT aligned correctly. 2) ENSURE affected turbine building sump discharge aligned per RWS-108, u-RE-5100 Radiation Monitor Alarm (Channel # TBD-u72) 3) REFER to STA-653.
☐ b. CONTACT Chemistry to sample affected sump for confirmation of indicated increase in activity.	
☐ c. NOTIFY Radiation Protection of possible contamination in turbine building.	
☐ 5 VERIFY Low Volume Waste Oil Colexer - NOT IN OPERATION. (U2 TB 778 NE Wall)	STOP oil colexer per RWS-107.

Section 2.3

Comments / Reference: ALM-3200	Revision: 5

CPNPP ALARM PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ALM-3200
ALARM PROCEDURE DRMS	REVISION NO. 5	PAGE 89 OF 100

ATTACHMENT 3
Page 1 of 1

AUTOMATIC ACTIONS

NOTE: A loss of power to the RM-80 will result in the Automatic Actions for the associated monitor.

TITLE	CHANNEL	FUNCTION	PRINT
Plant Vent Stack Wide Range Gas Monitor	X-RE-5570A S. X-RE-5570B N.	Closes HCV-014 on High Radiation or any OPERATE FAILURE	E1-0046 Sh 62/63
Auxiliary Building Exhaust	X-RE-5701	Closes HCV-014 on High Exhaust Radiation or any OPERATE FAILURE	E1-0065 Sh 22
Liquid Waste to Circulating Water	X-RE-5253	Closes discharge to Circulating Water Circulating Water (X-RV-5253) on High Radiation or any OPERATE FAILURE	E1-0065 Sh 29
Turbine Building Drains	u-RE-5100	Closes the discharge to Low Volume Waste (u-RV-5100A) and opens discharge to Co-Current Waste on High Radiation or any OPERATE FAILURE	E1-0055 Sh 61/62 E2-0055 Sh 61/62
Containment Air Gaseous and Particulate	u-RE-5503 u-RE-5502	Causes Containment Ventilation Isolation on High Radiation	E1-0046 Sh 62/64 E2-0046 Sh 62/64
Control Room Air Supply (Gas)	X-RE-5895A/B X-RE-5896A/B	Initiates Control Room Emergency Recirculation on High Radiation	E1-0046 Sh 62/63 E1-0035 Sh 76/77
Secondary Sample	u-RE-4200	Isolates Steam Generator Blowdown and Sampling System on High Radiation	E1-0040 Sh 97 E2-0040 Sh 97
Common discharge AB, DG Sumps and CCW Drain Tanks	X-RE-5251A	Diverts to Cocurrent Waste System Wastewater Holdup Tanks on High Radiation or any OPERATE FAILURE	E1-0065 Sh 58

Comments / Reference: ALM-3200	Revision: 5

CPNPP ALARM PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ALM-3200
ALARM PROCEDURE DRMS	REVISION NO. 5	PAGE 99 OF 100

ATTACHMENT 8
PAGE 2 OF 2

GENERAL INFORMATION

The PC-11 console alarm conditions may also be printed on the alarm printer. Specific alarm codes are used to flag when alarms are received and cleared. These alarm codes are summarized as follows:

PC-11 INDICATION			ALARM PRINTER INDICATION ALARM	CLEAR
PC-11 POLL STATUS	MONITOR OFFLINE	WHITE	ALM	RN
PC-11 COMMUNICATIONS	MONITOR COMMUNICATIONS FAILURE CHANNEL NOT RESPONDING TO POLL	MAGENTA MAGENTA	blank blank	blank blank
OPERATE FAILURE	MONITOR DATA BASE UNKNOWN MONITOR LOSS OF SAMPLE FLOW CHANNEL OUT OF SERVICE CHANNEL FILTER NOT MOVING CHANNEL NO PULSES RECEIVED CHANNEL CHECK SOURCE TEST FAILED CHANNEL LOSS OF SAMPLE FLOW CHANNEL HIGH TEMPERATURE CONDITION CHANNEL OPERATE FAILURE	BLUE BLUE BLUE BLUE BLUE BLUE BLUE BLUE BLUE	ALM ALM ALM blank ALM ALM ALM ------ ------	RN blank RN blank RN RN RN --------- ---------
CHANNEL HIGH ALARM	CHANNEL IN HIGH ALARM	RED	ALM	RN
CHANNEL ALERT ALARM	CHANNEL IN ALERT ALARM	YELLOW	ALM	RN
EQUIPMENT FAILURE	MONITOR LOSS OF PROCESS FLOW MONITOR IN SCAN OVERLOAD MONITOR LOSS OF FLOW CONTROL MONITOR LOSS OF RM-23 COMMUNICATION MONITOR EQUIPMENT FAILURE MONITOR LOW PRESSURE ALARM MONITOR HIGH PRESSURE ALARM	LIGHT BLUE LIGHT BLUE LIGHT BLUE LIGHT BLUE LIGHT BLUE LIGHT BLUE LIGHT BLUE	EF blank EF EF ------ EF EF	blank not logged blank blank -------- blank blank

ALM	-Monitor or channel status entering alarm	ERR	-Change request incomplete
blank	-Return to normal or non alarm condition	LV	-Leaving condition
BRD	-Broadcast error	RJT	-Operator request rejected
DSA	-Buffer unavailable to complete request	RN	-Monitor or channel status returning to normal
DWN	-System disc error during data base change		
EF	-Equipment failure alarm	RTE	-Communications routing error between PC-11s
ENT	-Entering condition		

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	2		
	K/A	000068.G.2.4.34		
Level of Difficulty: 3	Importance Rating	4.2		

Control Room Evacuation: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Question # 61

Given the following conditions:

- Both units 100% power
- A fire in the Control Room requires it to be evacuated
- ABN-803A, Response to a Fire in The Control Room or Cable Spreading Room is being performed

In accordance with ABN-803A:

ONLY the ____(1)____ Safeguards Bus will be energized by its associated EDG when complete.

The EDG is started from the ____(2)____.

A. (1) Train A
(2) RSP

B. (1) Train A
(2) EDG room

C. (1) Train B
(2) RSP

D. (1) Train B
(2) EDG room

Answer: B

K/A Match: K/A match due to requiring knowledge of specific RO tasks performed outside the control room for control room evacuation as well as the resultant operational effect.

Explanation:

A. Incorrect. 1st part is correct. Only 1EA1 will be energized with the EDG. 2nd part is incorrect because the EDG is started locally. It is plausible because the EDG breaker and bus feeder breaker controls are located at the RSP.

B. Correct. 1st part is correct (see A). 2nd part is correct. The EDG is started locally before the operator at the RSP gains control of it.

C. Incorrect. 1st part is incorrect because ONLY the Train A EDG will be powering its respective bus. It is plausible because both 1EA1 and 1EA2 are de energized in the procedure prior to energizing Train A from its associated EDG. 2nd part is incorrect but plausible (see A).

A. Incorrect. 1st part is incorrect but plausible (see C). 2nd part is correct (see B).

Technical Reference(s)	ABN-803A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: __________

Learning Objective: **ANALYZE** the response to a Fire in the Electrical or Control Building in accordance with ABN-803, Response To A Fire In The Control Room Or Cable Spreading Room. (ABN.803.OB01)

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

Question History: Last NRC Exam __________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: ABN-803A	Revision: 13

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-803A
RESPONSE TO A FIRE IN THE CONTROL ROOM OR CABLE SPREADING ROOM	REVISION NO. 13	PAGE 27 OF 67

ATTACHMENT 1
PAGE 3 OF 6

REACTOR OPERATOR ACTIONS TO ACHIEVE HOT SHUTDOWN

NOTE: Attachment 13, ABN-803A Job Aid may be used to track actions of RRO, NEO 1 and NEO 2.

l. DEENERGIZE 1EA2 as follows:

☐ 1) TRANSFER following controls from **CR** to **HSP**:

- 43-1EG2, DG 2 BKR 1EG2 CTRL XFER
- 43-1EA2-2, BKR 1EA2-2 CTRL XFER
- 43-1EA2-1, BKR 1EA2-1 CTRL XFER

☐ 2) PLACE following handswitches in **PULL-OUT**:

A. CS-1EG2-L, DG 2 BKR 1EG2
B. CS-1EA2-2L, INCOMING BKR 1EA2-2
C. CS-1EA2-1L, INCOMING BKR 1EA2-1

☐ m. WHEN control has been transferred to RSP,
THEN
ENSURE the following:

- 1/1-APRH1F, RHRP 1 - **OFF**
- 1/1-APCH1L, CCP 1 - **OFF**

Attachment 1

Comments / Reference: ABN-803A	Revision: 13

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-803A
RESPONSE TO A FIRE IN THE CONTROL ROOM OR CABLE SPREADING ROOM	REVISION NO. 13	PAGE 31 OF 67

ATTACHMENT 2
PAGE 1 OF 4

RELIEF REACTOR OPERATOR ACTIONS TO ACHIEVE HOT SHUTDOWN

SFGD 832 (Rm 1-096)

☐ a. PROCEED to the Remote Shutdown Panel, AND OBTAIN a copy of this procedure (will need an ABA1 key to get into panel).

☐ b. ENSURE 43/1-456FT, PRZR PORV CTRL XFER - **HSP**

SFGD 810 CP1-ECPRLV-15, Trn A Electrical Switchgear Room, (Rm 1-083)

☐ c. PROCEED to Shutdown Transfer Panel (will need a CAT60 key to get into panel).

NOTE: All transfer switches are located in the two right hand panels.

☐ d. Rapidly TRANSFER **ALL** transfer switches at Shutdown Transfer Panel.

☐ e. NOTIFY RO at RSP that all transfer switches have been transferred to HSP position.

SFGD 810 TRN A Diesel Room

☐ f. PLACE 1-HS-3413-3B, RLMS (MASTER SWITCH) (1-DG-01B) - **LOCAL**.

g. PERFORM the following to ensure proper Train A Diesel Generator operation:

☐ 1) PLACE 1-HS-3413-4B LOCAL EMERG STOP OFF START switch (1-DG-01B) - **START**.

IF Train A Diesel Generator does not start,
THEN
PERFORM the following:

☐ • PLACE 1-HS-3413-2B LOCAL NORMAL STOP-START switch - **START**.

☐ • WHEN Diesel Generator running,
THEN
PLACE 1-HS-3413-4B LOCAL EMERG STOP OFF START switch (1-DG-01B) - **START**.

☐ 2) VERIFY proper voltage and frequency (1-DG-01B).

- AC VOLTS, 6.6 - 7.2 kv
- FREQUENCY, 59.5 - 60.5 Hz

☐ h. NOTIFY RO at RSP that Train A Diesel Generator is running and may be loaded as necessary.

Attachment 2

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	2		
	K/A	000076.AK2.01		
Level of Difficulty: 2	Importance Rating	2.6		

High Reactor Coolant Activity: Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors

Question # 62

Given the following conditions:

- Unit 1 in Mode 3 during a cool down for a mid-cycle outage to replace a damaged seal package on an RCP
- RCS temperature 390ºF and pressure 780 psig
- CCP 1-01 in operation with one 75 gpm orifice isolation valve open
- 1-RE-0406 (FFL160), Gross Failed Fuel Monitor, has alarmed
- Chemistry reports RCS specific activity has increased steadily over the past several days

Per ABN-102, High Reactor Coolant Activity, letdown flow should be _____ to minimize personnel radiation exposure during the outage.

A. lowered to 0 gpm

B. lowered to 45 gpm

C. raised to 120 gpm

D. raised to 195 gpm

Answer: C

K/A Match: K/A match due to requiring knowledge of the response of the radiation monitoring system to high RCS activity.

Explanation:

A. Incorrect. Plausible since isolating letdown will prevent the activity from circulating in the Auxiliary and Safeguards Buildings, but it will not reduce RCS activity impacting future dose.

B. Incorrect. Plausible since reducing letdown will minimize the activity circulating in the Auxiliary and Safeguards Buildings while still allowing some cleanup of the RCS, but it will not maximize the reduction in RCS activity impacting future dose.

C. Correct. Letdown flow should be increased to a maximum value, but less than 140 gpm, to allow mechanical filtration of the letdown flow via the mixed bed demineralizers, minimizing future dose.

D. Incorrect. Plausible as all letdown valves open would give this value but flow is limited to 140 gpm when RCS temp is ≥ 500 degrees.

Technical Reference(s)	ABN-102	Attached w/ Revision # See Comments / Reference
	SOP-103A	

Proposed references to be provided during examination: __________

Learning Objective: **DISCUSS** the response to High Reactor Coolant Activity in accordance with ABN-102, High Reactor Coolant Activity. (ABN.103.OB01)

Question Source: Bank # 75843
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam LC-24

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: Exam Bank 75843	Revision:

- Unit 1 at 100%
- Letdown flow is 75 gpm
- 1-RE-0406 (FFL160), GROSS FAILED FUEL MONITOR, has alarmed
- Chemistry reports that Reactor Coolant System specific activity has increased steadily over the past several days

Per ABN-102, High Reactor Coolant Activity letdown flow should be __________ to minimize personnel radiation exposure.

A. lowered to 0 gpm

B. lowered to 45 gpm

C. raised to 120 gpm

D. raised to 195 gpm

Answer: C

Answer Explanation	
A.	Incorrect. Plausible since isolating letdown will prevent the activity from circulating in the Auxiliary and Safeguards Buildings, but it will not reduce RCS activity impacting future dose.
B.	Incorrect. Plausible since reducing letdown will minimize the activity circulating in the Auxiliary and Safeguards Buildings while still allowing some cleanup of the RCS, but it will not maximize the reduction in RCS activity impacting future dose.
C.	Correct. Letdown flow should be increased to a maximum value, but less than 140 gpm, to allow mechanical filtration of the letdown flow via the mixed bed demineralizers, minimizing future dose.
D.	Incorrect. Plausible as all letdown valves open would give this value but flow is limited to 140 gpm when RCS temp is ? 500 degrees.

Comments / Reference: Exam Bank 75843	Revision:

Question 59 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	75843
User-Defined ID:	ILOT9451
Cross Reference Number:	ABN.103.OB01.012
Topic:	Unit 1 at 100% Letdown flow is 75 gpm 1-RE-0406 (FFL160), GROSS FAILED FUEL MONITOR, has alarmed C
K/A:	APE 076 AA2.02
Question Reference:	
SRO:	
Comments:	LC24 NRC K/A Match: The question is a K/A match as it requires the applicant to understand the ABN procedure and know what corrective action is taken based on evaluation of current plant conditions (RCS high activity).

Comments / Reference: ABN-102	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-102
HIGH REACTOR COOLANT ACTIVITY	REVISION NO. 8	PAGE 4 OF 6

2.3 Operator Actions

NOTE:
- Reactor Coolant System transients such as power changes, temperature changes, pressure changes, and starting and stopping RCPs can cause temporary increases in RCS activity.
- Monitor spiking and return to normal is not a real indication of failed fuel and as such does not require sampling. A steady or sustained increase over time would be a real indication of failed fuel/RCS activity problems.

☐ 1. REQUEST additional reactor coolant specific activity samples be taken in accordance with CHM-111 for isotopic content analysis per Technical Specification 3.4.16, SURVEILLANCE REQUIREMENTS.

☐ 2. NOTIFY Chemistry to review chemistry data and Core Performance Engineering to review chemistry data and core follow trends. Chemistry will determine if a "CRUD" burst has occurred. Core Performance Engineering will determine if the source of RCS activity is failed fuel and the extent of failed fuel, if any.

☐ 3. INCREASE letdown flow to 120-140 gpm as follows:

a) IF PDP is in operation,
THEN
START a centrifugal charging pump AND SHUTDOWN PDP per SOP-103A/B.

b) INCREASE letdown flow to 120-140 gpm per SOP-103A/B.

☐ 4. NOTIFY Radiation Protection that radiation levels may increase in Auxiliary and Safeguards Buildings AND on any ARMs.

☐ 5. MAKE a plant announcement via Gai-Tronics of indication of an increase in RCS Activity AND a possibility of increased radiation in Auxiliary and Safeguards Buildings.

NOTE: A rapid increase of RCS fission product isotopes during steady state operation may indicate fuel cladding damage. (e.g., Xe-133, Kr-85M, Cs-137, Cs-136, Sr-84, Sr-90, Iodine).

☐ 6. IF Core Performance Engineering Review of the chemistry data indicates failed fuel,
THEN
PROCEED as follows:

a) REFER to EPP-201.

b) REFER to Technical Specifications 3.4.16.

c) REVIEW logs for any known RCS to Secondary Leakage.

Section 2.3

Comments / Reference: SOP-103A	Revision: 18

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-103A
CHEMICAL AND VOLUME CONTROL SYSTEM	REVISION NO. 18 CONTINUOUS USE	PAGE 11 OF 164

4.1 Limitations (continued)

[C] • The PDP suction stabilizer gas supply and vent valves should be closed and the PDP should be stopped if the charging pump suction is switched from the VCT to the RWST due to VCT low-low level or operator action. This is applicable when VCT pressure is greater than RWST pressure. Higher VCT pressure will disable the PDP stabilizer vent path and may cause gas binding of the CCP's if 1CS-8200, PD CHRG PMP 1-01 SUCT STAB VNT CHK VLV leaks.

[C] • When the PDP is running and 1/1-8204, H2/N2 SPLY VLV indicates open (red light on), 1/1-8210A, H2/N2 SPLY VLV and 1/1-8210B, H2/N2 SPLY VLV may be opened no more than 10 seconds to clear the high level (1/1-8204 green light on). When 1-ALB-6A, 1.8 "PDP SUCT STAB LVL HI-HI" alarms, operator actions will provide steps to start a CCP and stop the PDP.

- Charging flow through the Regenerative Heat Exchanger is limited to 300 gpm. Due to indication (1-FI-121A), flow is limited to 270 gpm.
- The minimum charging flow from the CCP's with 1-FK-121 in AUTO is 55 gpm. Any charging flow less than 55 gpm will require placing 1-FK-121 in MANUAL.
- Seal injection to the RCP No. 1 seals should not exceed 130°F.
- Seal injection to any RCP No. 1 seal should not exceed 13 gpm.

[C] • Seal injection to any RCP No. 1 seal shall not be less than 6 gpm.

- When RCS temp is $\geq$ 500 degrees, letdown flow is limited to 140 gpm with the 45 gpm orifice and ONE 75 gpm orifice in service.
- Letdown flow is limited to 170 gpm (when RCS temp is < 500 degrees) with 1 Mixed Bed Demineralizer in service. (Reference EVAL-2005-001409-01-00)
- Letdown flow is limited to 195 gpm (when RCS temp is < 500 degrees) when 2 demineralizers are in service. (Reference FDA-2007-001435-01-00)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	2		
	K/A	WE02.EK1.02		
Level of Difficulty: 2	Importance Rating	3.4		

SI Termination: Knowledge of the operational implications of the following concepts as they apply to the SI Termination: Normal, abnormal and emergency operating procedures associated with SI Termination.

Question # 63

Given the following conditions:

- An SI has occurred due to a fault on SG 1-02
- The operators have completed the actions of EOP-2.0A, Faulted Steam Generator Isolation, and transitioned to EOP-1.0A, Loss of Reactor or Secondary Coolant
- Containment pressure 7 psig slowly decreasing
- Total AFW flow to the intact SGs is 480 gpm
- All intact SG levels between 7% and 22%
- PRZR pressure 1725 psig rapidly increasing
- All PRZR level channels indicate between 23% and 28%
- CETs indicating 540ºF stable

Which of the following would PROHIBIT terminating Safety Injection under these conditions?

A. Subcooling

B. RCS pressure

C. RCS inventory

D. Secondary heat sink

Answer: C

K/A Match: K/A match due to requiring knowledge of the parameters used to determine if SI Termination can be implemented.

Explanation:

A. Incorrect. Plausible since this is one of the parameters required to be met to terminate SI (> 55°F), but it is met as subcooling is approximately 76°F.

B. Incorrect. Plausible since this is one of the parameters required to be met to terminate SI (stable or increasing), but it is met as pressure is increasing.

C. Correct. RCS inventory is not adequate since 34% level is required with adverse containment conditions. The crew would be directed to stabilize pressure and transition to EOS-1.1A when SI flow restores adequate pressurizer level.

D. Incorrect. Plausible since this is one of the items required to be met to terminate SI (> 460 gpm or at least one intact SG level > 26%), but it is met as AFW is 480 gpm.

Technical Reference(s)	EOP-1.0A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: **IDENTIFY** the proper transitions out of EOP-1.0 in accordance with EOP-1.0, Loss of Reactor or Secondary Coolant. (ERG.E1A.OB05)

Question Source: Bank # 62538
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments / Reference: EOP-1.0A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-1.0A
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 9	PAGE 8 OF 44

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6	Check If ECCS Flow Should Be Reduced:	
	a. Secondary heat sink: • Total AFW flow to intact SGs - GREATER THAN 460 GPM -OR- • Narrow range level in at least one intact SG - GREATER THAN 43% (50% FOR ADVERSE CONTAINMENT)	a. IF neither condition satisfied, THEN go to Step 7. OBSERVE CAUTIONS PRIOR TO STEP 7.
	b. RCS subcooling - GREATER THAN 25°F(55°F FOR ADVERSE CONTAINMENT)	b. Go to Step 7. OBSERVE CAUTIONS PRIOR TO STEP 7.
	c. RCS pressure - STABLE OR INCREASING	c. GO to Step 7. OBSERVE CAUTIONS PRIOR TO STEP 7.
	d. PRZR level - GREATER THAN 13% (34% FOR ADVERSE CONTAINMENT)	d. Stabilize RCS pressure with normal PRZR spray. Go to Step 7. OBSERVE CAUTIONS PRIOR TO STEP 7.
	e. Go To EOS-1.1A, SAFETY INJECTION TERMINATION, Step 1.	

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	2		
	K/A	WE13.EK3.02		
Level of Difficulty: 4	Importance Rating	2.9		

Steam Generator Overpressure: Knowledge of the reasons for the following responses as they apply to the Steam Generator Overpressure: Normal, abnormal and emergency operating procedures associated with Steam Generator Overpressure.

Question # 64

Given the following conditions:

- Unit 1 entered FRH-0.2A, Response to Steam Generator Overpressure, due to high pressure in SG 1-01
- Attempts to lower SG 1-01 pressure are unsuccessful
- The crew is directed to cool down the RCS using the remaining three SGs

Which of the following describes the reason for cooling the RCS hot leg temperature to less than 535ºF?

Ensure …

A. SG pressure is below the highest SG safety valve setpoint.

B. adequate subcooling in the RCS exists to continue RCP operation.

C. steam pressure supplied to the TDAFW pump is within design limits.

D. excessive heat transfer from the RCS is not the cause of the overpressure.

Answer: D

K/A Match: K/A match due to requiring knowledge of the reasons for performing actions in response to a SG overpressure condition.

Explanation:

A. Incorrect. Plausible since it is preferred to not lift a safety valve, but maintaining the SG at normal temperature would keep the pressure below the safety setting.

B. Incorrect. Plausible since it is desirable to operate RCPs, but higher pressure would assure adequate subcooling.

C. Incorrect. Plausible since the TDAFWP may be the supply of AFW to the SGs, but higher pressure will not affect the TDAFWP capability.

D. Correct. Cooling down the RCS to less than 535°F ensures excessive heat transfer from the RCS is eliminated thereby reducing temperature and pressure of the affected SG.

Technical Reference(s)	FRH-0.2A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: __________

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRH-0.2 in accordance with FRH-0.2, Response to Steam Generator Overpressure. (ERG.FH2.OB04)

Question Source: Bank # 32622
Modified Bank # __________ (Note changes or attach parent)
New __________

Question History: Last NRC Exam __________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 __________

Comments / Reference: FRH-0.2A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.2A
RESPONSE TO STEAM GENERATOR OVERPRESSURE	REVISION NO. 9	PAGE 4 OF 13

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 5	Check Affected SG(s) Pressure:	
	a. Pressure - DECREASING	a. Go to Step 6. OBSERVE CAUTION PRIOR TO STEP 6.
	b. Pressure - LESS THAN 1235 PSIG	b. Return to Step 3.
	c. Control steam release to maintain SG pressure less than 1235 psig.	
	d. Return to procedure and step in effect.	

CAUTION: AFW flow should remain isolated to affected SG(s) until a steam release path is established.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	Isolate AFW Flow To Affected SG(s).	
7	Check RCS Hot Leg Temperatures - LESS THAN 535°F	Cooldown RCS to less than 535°F by dumping steam from the unaffected SG(s).
* 8	Continue Attempts To Manually Or Locally Dump Steam From Affected SG(s): • SG atmospheric -OR- • Locally with the main steamline isolation bypass valve -OR- • Steam supply valve to TDAFW pump (SG 1 or 4) -OR- • Before MSIV drip pot isolation valve	

Comments / Reference: FRH-0.2A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.2A
RESPONSE TO STEAM GENERATOR OVERPRESSURE	REVISION NO. 9	PAGE 9 OF 13

ATTACHMENT 2
PAGE 2 OF 6

Bases

STEP 5: Steam release should result in affected SG(s) pressure decreasing. If this does not occur, the operator is directed to Step 6. If pressure is decreasing but is not below 1235 psig, the operator is directed to return to Step 3 and continue monitoring level and releasing steam. If steam release drops the affected SG(s) pressure to less than 1235 psig, then the steam release is controlled to maintain pressure and the operator is instructed to return to the procedure in effect.

This action contains the action verb "control" which implies continuous performance; therefore, this step has been identified as a continuous action step.

CAUTION: If AFW flow is re-established to an affected SG prior to establishing a steam release path, the AFW flow could further increase the affected SG pressure.

STEP 6: If the operator has been unsuccessful in releasing steam to lower the affected SG pressure below design pressure, the operator should proceed to isolate AFW flow to the affected SG since the AFW system is a high pressure water source. This eliminates an additional source of over pressurization of the affected SG(s) and may prevent a potential failure of secondary integrity.

STEP 7: Excessive heat transfer from the primary system may be the cause of the affected SG over pressurization. Therefore, a check on RCS hot leg temperatures is made to determine this. If RCS hot leg temperatures are greater than 535°F, a cooldown is initiated by dumping steam from the unaffected SG(s) to aid in reducing the temperature and pressure in the affected SG(s).

STEP 8: The operator should continue attempts to manually or locally release steam from the affected SG(s), utilizing the four alternative release paths plus any plant specific means identified until the challenge to the SG pressure boundary is mitigated.

This action contains the action verb "control" which implies continuous performance; therefore, this step has been identified as a continuous action step.

STEP 9: The operator has done everything possible to mitigate the overpressure condition and has isolated AFW flow to the affected steam generator to mitigate a potential failure of secondary integrity. Therefore, the operator should continue plant recovery operations by returning to the procedure and step that was in effect at the time FRH-0.2A was entered.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	1		
	Group	2		
	K/A	WE09.EA2.02		
Level of Difficulty: 2	Importance Rating	3.4		

Natural Circulation Operations: Ability to determine and interpret the following as they apply to the Natural Circulation Operations: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Question # 65

During EOS-0.2A, Natural Circulation Cooldown, following a reactor trip, which of the following criteria determines if the amount of RCS subcooling required is 75ºF or 125ºF?

A. Pressurizer level

B. RCS cooldown rate

C. Number of CCPs running

D. Number of CRDM fans running

Answer: D

K/A Match: K/A match due to requiring knowledge of the subcooling limit based on plant conditions during a natural circulation cooldown.

Explanation:

A. Incorrect. Plausible since ERG decisions are based upon pressurizer level, but the subcooling required in EOS-0.2A is based upon the amount of vessel head cooling available from the CRDM fans.

B. Incorrect. Plausible since ERG decisions are based upon RCS pressure, but the subcooling required in EOS-0.2A is based upon the amount of vessel head cooling available from the CRDM fans.

C. Incorrect. Plausible since ERG decisions are based upon the number of CCPs operating, but the subcooling required in EOS-0.2A is based upon the amount of vessel head cooling available from the CRDM fans.

D. Correct. The amount of subcooling required in EOS-0.2A is based upon the amount of vessel head cooling available, which is dependent on the number of CRDM fans running.

Technical Reference(s)	EOS-0.2A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________

Learning Objective: **ANALYZE** the recovery technique used and the procedure steps of EOS-0.2, Natural Circulation Cooldown. (ERG.E02.OB02)

Question Source: Bank # 62529
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments / Reference: EOS-0.2A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.2A
NATURAL CIRCULATION COOLDOWN	REVISION NO. 9	PAGE 16 OF 60

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

<u>NOTE</u>: If at any time it is determined that a natural circulation cooldown and depressurization must be performed at a rate that may form a steam void in the vessel, EOS-0.3A, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS), or EOS-0.4A, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS) should be used.

15 Establish Required RCS Subcooling:

a. Maintain RCS pressure between 1600 psig and 1910 psig until subcooling in Step 15b is established.

b. Establish RCS subcooling as follows:

- Both CRDM fans running - GREATER THAN 75°F
- One or less CRDM fans running - GREATER THAN 125°F

Comments / Reference: EOS-0.2A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.2A
NATURAL CIRCULATION COOLDOWN	REVISION NO. 9	PAGE 48 OF 60

ATTACHMENT 6
PAGE 14 OF 26

BASES

STEP 14: The core exit thermocouples (TCs) and the reactor coolant hot leg temperatures are monitored to verify that the reactor coolant is being cooled by the discharge of steam from the steam generators at the cooldown rate previously described. In addition, it is verified that the subcooling of the reactor coolant is increasing. This ensures that adequate core cooling is being provided. The subcooling is determined by use of RCS wide range pressure, RCS wide range temperature and core exit thermocouples.

After the natural circulation cooldown has been established, the reactor coolant hot leg temperature should trend down with the decreasing steam pressure.

Trended readings for core exit thermocouples, loop THOT readings, and TCOLD readings should be used to monitor cooldown and subcooling with readings at 10-15 minute intervals recommended. The observed loop temperatures and temperature differences (THOT, TCOLD.) can be expected to vary from loop-to-loop and may deviate at any single observation. These variations of individual readings from the nominal responses are normal, and therefore only trended values are useful for diagnosis of possible problematic conditions in natural circulation flow.

These conditions are expected to be monitored during the subsequent cooldown, therefore, this step has been identified as a Continuous Action Step.

NOTE: From this point onward the operator has the option of changing procedures if a need to cooldown and depressurize more quickly than at the present rate exists. Procedure EOS-0.3A NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS), or EOS-0.4A NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS) should be used in this case depending upon the availability of RVLIS to monitor upper head void growth.

The major factor which could require a more rapid cooldown/depressurization than this procedure allows is limited condensate storage.

STEP 15: To prevent possible void formation in the upper head, the required RCS subcooling should be maintained. With CRDM fans available, a subcooling margin of at least 75°F should be maintained during depressurization. Without the availability of CRDM fans, maintain a minimum subcooling of 125°F during depressurization.

Comments / Reference: EOS-0.2A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.2A
NATURAL CIRCULATION COOLDOWN	REVISION NO. 9	PAGE 49 OF 60

ATTACHMENT 6
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BASES

RCS pressure is maintained between 1910 psig and 1600 psig as the required subcooling is established, to prevent upper head void formation. Once subcooling is established, pressurizer pressure should periodically be decreased to maintain the reactor coolant and pressurizer pressure-temperature relationship in accordance with the Technical Specifications and the minimum subcooling requirement.

The analysis supporting the strategy in this procedure assumes either "all" or no CRDM fans are operating when RCS depressurization is initiated in this step. If some, but less than "all", CRDM fans are operating, the conservative subcooling requirements specified in this step for no CRDM fans operating will apply throughout the remainder of this procedure.

STEP 16: Once subcooling is established, RCS cooldown and depressurization should continue while maintaining the minimum subcooling requirement and remaining within the Technical Specification pressure-temperature limits. The depressurization should be accomplished using pressurizer auxiliary spray or pressurizer PORVs, depending upon whether letdown is in service. The RCS cooldown rate must be maintained less than 50°F/hr if all RCS loops are active or in accordance with the requirements for RCS cooldown with an inactive loop(s). Attachment 5 provides a curve which is used to determine the maximum cooldown rate versus the active loop(s) ΔT when at least one RCS loop is inactive. If at any time the required subcooling cannot be maintained, the RCS depressurization should be stopped until the required subcooling is reestablished.

When Auxiliary spray flow is used for RCS depressurization, the amount of auxiliary spray flow may be reduced if the charging flow control valve (FCV-121) is not fully open. In addition, the normal spray valves (455B and 455C) and the charging loop isolation valves (8146 and 8147) should be closed in order to achieve maximum auxiliary spray flow. Precise RCS pressure control can be obtained by throttling the normal spray valve(s) (455B and 455C). Throttling closed the normal spray valves will force auxiliary spray into the pressurizer and reduce RCS pressure. Throttling open the normal spray valves will reduce auxiliary spray flow into the pressurizer and raise flow into the RCS loops to increase RCS pressure. Pressure control may be more precise if one of the normal spray valves (455B or 455C) is fully closed and the other is modulated closed. Pressurizer inventory can be regulated with charging flow control valve (FCV-0121). When auxiliary spray flow is initiated, the Plant Staff may consider gathering pertinent data to assist in the post event analysis of charging nozzle thermal cycles.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	3		
	Group			
	K/A	2.1.23		
Level of Difficulty: 2	Importance Rating	4.3		

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question # 66

Procedures or work instructions with complex or infrequent work activities for which consequences of an improper action could have immediate, possibly irreversible impact on safety, production or reliability are classified as __(1)__ procedures.

__(2)__ are generally classified as these type of procedures.

A. (1) Information Use
(2) Operations Department Work Instructions

B. (1) Information Use
(2) System Operating Procedures

C. (1) Continuous Use
(2) Operations Department Work Instructions

D. (1) Continuous Use
(2) System Operating Procedures

Answer: D

K/A Match: K/A match due to requiring knowledge of how procedures are be implemented in accordance with administrative requirements.

Explanation:

A. Incorrect. First part is incorrect, but plausible since these are procedures or work instructions with work activities, but are performed frequently and have no immediate consequences if performed improperly. Second part is incorrect, but plausible since OWIs are considered Information Use procedures.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see B).

D. Correct. First part is correct. Continuous Use procedures are procedures or work instructions with complex or infrequent work activities for which consequences of an improper action could have immediate, possibly irreversible impact on safety, production or reliability. Second part is correct. SOPs are considered to be Continuous Use.

Technical Reference(s)	STA-201	Attached w/ Revision # See Comments / Reference
	ODA-407	

Proposed references to be provided during examination: ______

Learning Objective: **STATE** requirements for Procedure Use and Adherence in accordance with ODA-102, ODA-407, STA-201 and Operations Guideline 3. (ADM.XA3.OB01)

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

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: STA-201	Revision: 18

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-201
PROCEDURE AND WORK INSTRUCTION USE AND ADHERENCE	REVISION NO. 18 INFORMATION USE	PAGE 6 OF 48

4.8 Critical Step - A procedure step, series of steps, or action that, if performed improperly, will cause irreversible harm to plant equipment or people, or will significantly impact plant operation.

EXAMPLE:

> CAUTION: Measuring resistance in the following step may result in starting DG 1–01 if the test equipment is not selected to measure resistance prior to installation of the test leads.

8.10 IF DG 1-01 starting air receiver 1-01 is NOT in service with pressure ≥ 150 psig, THEN verify the K609A contact in the CHANNEL II starting circuit is open by measuring resistance across terminals TB621-3 and TB621-4 in Train A SSPS Output Cabinet No. 1. (OPT-467A)

4.9 eSPOC - electronic Smart Procedure Organization and Control. An application utilized to manage procedure life cycle including: authoring, reviewing, approving, using, requesting changes, and retirement.

4.10 Infrequently Performed Test or Evolution (IPTE) - An evolution requiring a Senior Line Manager to participate in the pre-evolution briefing and to exercise Continuous Responsibility for Management Oversight. (Reference STA-122)

4.11 Intent - The overall objective or purpose of a procedure or procedure section.

4.12 Level of Use - A procedure or work instruction classification that designates the minimum requirements for procedure use during an activity. The instructions of this procedure require classification of procedures as:

- CONTINUOUS USE - procedure or work instruction with complex or infrequent work activities for which consequences of an improper action could have immediate, possibly irreversible impact on safety, production or reliability.
- REFERENCE USE - procedure or work instruction with work activities for which consequences of an improper action are not immediate and are not irreversible.
- INFORMATION USE - procedure or work instruction with work activities, usually administrative in nature, that do not involve direct contact with plant equipment, are performed frequently, have no immediate consequences if performed improperly, and are within the knowledge and skills of experienced individuals.
- MULTIPLE USE - procedures or work instruction in which sections or subsections are designated with different level of use.

Comments / Reference: ODA-407	Revision: 17

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 17 INFORMATION USE	PAGE 62 OF 63

ATTACHMENT 8.D
PAGE 2 OF 3

OPERATIONS PROCEDURE LEVEL OF USE DESIGNATIONS

Procedure Group	**Level of Use (1)**	**Exceptions / Individual Procedure Designations (1)**
Operation Department Work Instructions (OWIs)	INFORMATION USE	**OWI-206**: MULTIPLE USE with: Section 6.1.3 - CONTINUOUS USE Section 6.3.2 - CONTINUOUS USE Section 6.3.3 - INFORMATION USE Section 6.3.3.1 - CONTINUOUS USE Section 6.3.3.2 - CONTINUOUS USE Attachment 8.E - REFERENCE USE Attachment 8.G - CONTINUOUS USE Attachment 8.H - REFERENCE USE Attachment 8.K - CONTINUOUS USE Attachment 8.P - CONTINUOUS USE Attachment 8.Q - CONTINUOUS USE All other Sections and Attachments - INFORMATION USE **OWI-208**: MULTIPLE USE with: Attachments 1 through 10 - REFERENCE USE All other Sections and Attachments - INFORMATION USE **OWI-404**: MULTIPLE USE with: Attachment 3 - REFERENCE USE All other Sections and Attachments - INFORMATION USE **OWI-801**: MULTIPLE USE with: Attachments 1 through 4 - REFERENCE USE All other Sections and Attachments - INFORMATION USE **OWI-802**: MULTIPLE USE with: Section 6.1 - REFERENCE USE All other Sections and Attachments - INFORMATION USE **OWI-912**: MULTIPLE USE Sections 6.2 through 6.7 - REFERENCE USE Attachments 1 through 4 - REFERENCE USE Attachments 5 through 8 - CONTINUOUS USE All other Sections and Attachments - INFORMATION USE
Station Refueling Procedures (RFOs) *	See ➜	**RFO-102**: CONTINUOUS USE **RFO-401**: REFERENCE USE **RFO-402**: REFERENCE USE **RFO-403**: REFERENCE USE **RFO-404**: CONTINUOUS USE **RFO-501**: REFERENCE USE **RFO-502**: REFERENCE USE
Radwaste System Procedures (RWSs)	CONTINUOUS USE	None
System Operating Procedures (SOPs)	CONTINUOUS USE	**SOP-907A/B**: MULTIPLE USE with ALL CONTINUOUS USE except Attachment 5 which is INFORMATION USE

-

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	3		
	Group			
	K/A	2.1.29		
Level of Difficulty: 2	Importance Rating	4.1		

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Question # 67

As part of a clearance, several normally locked closed valves were tagged.

During restoration, a _______ locking tab should be reapplied to these valves.

A. Red

B. Green

C. Blue

D. Yellow

Answer: B

K/A Match: K/A match due to requiring knowledge of how to identify locked closed valves during clearance restorations.

Explanation:

A. Incorrect. Plausible as this is the color used to identify normally locked open or off positions.

B. Correct. This is the color used to identify normally locked closed positions.

C. Incorrect. Plausible as this is the color used to identify normally locked throttled positions.

D. Incorrect. Plausible as this is the color used for personal safety.

Technical Reference(s)	ODA-403	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ________

Learning Objective: **EXPLAIN** the administrative controls of valves, breakers and other equipment required to be secured-in-position in accordance with ODA-403 and OWI-103. (ADM.XA1.OB10)

Question Source:
- Bank # ________
- Modified Bank # 75848 (Note changes or attach parent)
- New ________

Question History: Last NRC Exam ________

Question Cognitive Level:
- Memory or Fundamental Knowledge X
- Comprehension or Analysis ________

10 CFR Part 55 Content:
- 55.41 10
- 55.43 ________

Comments / Reference: Bank 75848	Revision:

As part of a clearance several normally sealed throttled valves were closed. During restoration, a ________ seal should be reapplied to these valves.

A. Red

B. Green

C. Blue

D. Yellow

Answer: C

Answer Explanation	
A.	Incorrect. Plausible as this is the color used to identify normally sealed open or off positions.
B.	Incorrect. Plausible as this is the color used to identify normally sealed closed positions.
C.	Correct. This is the color used to identify normally sealed throttled positions.
D.	Incorrect. Plausible as this is the color used for personal safety

Comments / Reference: Bank 75848	Revision:

Question 27 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	75848
User-Defined ID:	ILOT9457
Cross Reference Number:	ADM.XA1.OB09.007
Topic:	As part of a clearance several normally sealed throttled valves were closed. During restoration, a
K/A:	G.2.2.13
Question Reference:	
SRO:	
Comments:	LC24 NRC K/A Match: The question is a K/A match as it requires the applicant to demonstrate knowledge the color of seal that would be applied when restoring a clearance.

Comments / Reference: ODA-403	Revision: 8

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL	UNIT COMMON	PROCEDURE NO. ODA-403
OPERATIONS DEPARTMENT LOCKED COMPONENT CONTROL	REVISION NO. 8 INFORMATION USE	PAGE 3 OF 16

3.0 Continued

- STA-602, "Temporary Modifications"
- STA-605, "Clearance and Safety Tagging"
- STA-694, "Station Verification Activities"
- STA-696, "Hazard Barrier Controls"
- STA-738, "Fire Protection Systems/Equipment Impairments"
- TDM-901A/B, "Systems Data Throttled Valves/Flow Rates"

4.0 DEFINITIONS/ACRONYMS

4.1 Component - Any valve or circuit breaker determined to be administratively controlled by this procedure. This definition only applies to ODA-403 and OWI-103.

4.2 Design or Licensing Document - Document which contains specific details or references to plant design or licensing criteria. Design or Licensing Documents include, but are not limited to M1 and M2 drawings, Design Basis Documents (DBD), FSAR, Offsite Dose Calculation Manual (ODCM), Technical Requirements Manual (TRM), Fire Protection Report and Technical Specifications.

4.3 Locked Component Deviation Log - A log containing information about components that are required to be secured-in-position, but have been unlocked or repositioned (OWI-103-3).

4.4 Operations Locked Component List - Current listings of all components which are required to be secured-in-position, found in OWI-103.

4.5 Lock - A device used to secure-in-position a valve, breaker or other component. This device can be a padlock, plastic seal, lead seal or other device used with plastic ty-wraps, stainless steel cables, chain or permanently installed clasps.

- Red - The red color is used to identify normally sealed open or off positions
- Green - The green color is used to identify normally sealed closed positions
- Blue - The blue color is used to identify normally sealed throttled positions
- Yellow - The yellow color is used for personal safety and is not covered by this procedure

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	3		
	Group			
	K/A	2.1.43		
Level of Difficulty: 2	Importance Rating	4.1		

Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Question # 68

In accordance with ODA-102, Conduct of Operations, Attachment 8, Operations Reactivity Management, complete the following statements.

When performing a reactor startup with a designated Reactivity SRO providing oversight, the RO is required to inform the designated Reactivity SRO __(1)__ the Unit Supervisor PRIOR TO each 50 step rod pull.

Informing the designated Reactivity SRO and/or the Unit Supervisor when each control rod movement is COMPLETE __(2)__ required.

A. (1) AND
(2) is

B. (1) AND
(2) is NOT

C. (1) OR
(2) is

D. (1) OR
(2) is NOT

Answer: C

K/A Match: K/A match due to requiring knowledge of the administrative procedures governing reactivity manipulations of the plant.

Explanation:

A. Incorrect. First part is incorrect, but plausible because the Unit Supervisor is in charge of the startup so it would make sense that they are informed prior to the first rod pull. Second part is correct. Notification after each rod pull is required.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible because if performing multiple rod pulls (example given in ODA-102 is for restoring CBD after an instrument failure), notification of completion of the last movement is required.

C. Correct. ODA-102 states the Reactor Operator informs the Unit Supervisor or the designated SRO providing reactivity oversight prior to moving control rods and at the completion of rod movement.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ODA-102	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ________________

Learning Objective: **EXPLAIN** the requirements for manipulating reactor controls in accordance with ODA-102, STA-102 and STA-601. (ADM.XA1.OB06)

Question Source: Bank # 72316
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 10
55.43 ________

Comments / Reference: ODA-102	Revision: 34

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 34 / INFORMATION USE	PAGE 46 OF 151

ATTACHMENT 8.B
PAGE 5 OF 19

OPERATIONS REACTIVITY MANAGEMENT

5.3 REACTIVITY MANAGEMENT DURING NORMAL AT-POWER OPERATION

5.3.1 The RO/BOP should:

- Conduct Control Board / Equipment Monitoring per Attachment 8.C.
- Reactivity manipulations are perfomed by the Reactor Operator utilizing procedures in-hand. Exceptions to the use of procedures in-hand for reactivity manipulations may be specified by Operations Management.
- Peer-checks are provided for all planned activities that are known to have an impact on reactivity. Peer-checks are conducted for reactivity control manipulations commensurate with available resources and the need for timely operator response actions during transient conditions.
- Inform the Unit Supervisor prior to any dilution, boration, rod movement, or turbine load changes. Include the magnitude and direction (for rods and turbine load) of change.

5.3.2 The Unit Supervisor is expected to monitor reactivity manipulations.

- The degree of monitoring depends on the experience level of the operator and the complexity of the reactivity manipulation.
- As a minimum, visually spot check switch positions/manipulations and control adjustments.
- Position in proximity to the Reactor Operator for all planned activities that are known to have an impact on reactivity and maintain this oversight until the evolution is complete.
- During transient conditions, overview reactivity manipulations during the transient as necessary to ensure methods for monitoring reactor power are being utilized and requested manipulations are appropriate.
- Both the Unit Supervisor and the Reactor Operator are to assess the impact of a proposed reactivity manipulation (change in power/temperature) before the manipulation occurs, AND determine whether the desired impact was achieved after the manipulation is performed.

Comments / Reference: ODA-102	Revision: 34

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 34 INFORMATION USE	PAGE 47 OF 151

ATTACHMENT 8.B
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OPERATIONS REACTIVITY MANAGEMENT

5.3.3 There are times when additional reactivity oversight is desired (such as reactor startup, shutdown, testing, and ramping the unit). During these times a "Reactivity SRO" may be assigned to assist the Unit Supervisor in providing this oversight. WHEN a Reactivity SRO is assigned , THEN the following apply:

- The Reactivity SRO may provide peer checks for reactivity manipulations.
- These peer checks should not be routinely used, but are available during times when there isn't an additional RO available for peer checks.
- The Unit Supervisor provides additional oversight for the "at the controls area" and is not required to be positioned in the proximity of the Reactor Operator.

5.3.4 For reactivity manipulations which are required in response to equipment malfunctions, the RO verbalizes the action as it is being taken, allowing a brief pause for the Unit Supervisor to intervene if necessary. Such manipulations are to be performed in a controlled and deliberate manner.

5.3.5 Expectations for adherence with BEACON Reactivity Projections for runbacks, rapid downpowers, rapid shutdown and scheduled load reductions (Self Assessment Recommendation, CR-2010-001177).

- Trust but verify Core Performance Engineering (CPE) projections, AND stop and collaborate with Operation's management and CPE if the projection is suspect or not fully understood.
- CPE ensures the projections are conservative for boron amounts such that the projected boration amount will maintain control rods above the RIL, but will not over borate the core for the current burnup. Therefore, when borating per the BOS briefing and using the RXM Projection, the expectation is to add the total amount of boron per the projection.
- CPE updates the BEACON projections for runbacks and rapid downpowers or shutdowns monthly until RCS boron is < 100 ppm, then they are updated weekly.
- CPE tracks BEACON projection performance and results are reviewed with licensed operators in LORT during Core Operating Cycle characteristic updates.

5.3.6 During steady-state operation with fully conditioned fuel, control rods should normally be operated in automatic. Rod control may be taken to manual as necessary to make minor adjustments to AFD or at the discretion of the Unit Supervisor in anticipation of reactivity changes from other sources.

Comments / Reference: ODA-102	Revision: 34

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 34 INFORMATION USE	PAGE 51 OF 151

ATTACHMENT 8.B
PAGE 10 OF 19

OPERATIONS REACTIVITY MANAGEMENT

5.5 REACTIVITY MANAGEMENT DURING A REACTOR STARTUP

5.5.1 Prior to conducting a reactor startup, all control room operators assigned to the startup ensure they are familiar with the precautions and limitations for reactivity management contained in IPO-002A/B.

5.5.2 A pre startup brief should be conducted by the Unit Supervisor utilizing STI-429.04 and IPO-002A/B.

5.5.3 Multiple methods for anticipating criticality are to be used, such as:

- ECC
- ICRR plot
- 5-7 doublings of source range counts

5.5.4 No other responsibilities are assigned to the Reactor Operator during a startup.

5.5.5 The Reactor Operator informs the Unit Supervisor or the designated SRO providing reactivity oversight prior to moving control rods and at the completion of rod movement.

- If direction provided requires multiple incremental moves (e.g. restoring CBD to 215 after instrument failure), the Unit Supervisor or designated SRO providing reactivity oversight is to be informed prior to first movement and at completion of directed movements.

5.5.6 For non-refueling startups the Shift Manager should perform a "back of the envelope calculation" to ensure a sanity check of the ECC.

5.5.7 The Unit Supervisor and Reactor Operator monitor control rod movement, NIS, and Gamma metrics indications during the startup.

- Monitoring by the Unit Supervisor includes ensuring that other methods of controlling RCS temperature (steam dumps, turbine load, SG blowdown flow, feed flow, etc.) are being utilized as necessary to limit changes in RCS temperature.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	3		
	Group			
	K/A	2.2.4		
Level of Difficulty: 2	Importance Rating	3.6		

Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.	
Question # 69	

Unit 1 plant equipment labels are __(1)__.

Common equipment/alarms are normally controlled by __(1)__.

A. (1) blue
(2) Unit 1

B. (1) yellow
(2) Unit 2

C. (1) blue
(2) Unit 2

D. (1) yellow
(2) Unit 1

Answer: A

K/A Match: K/A match due to requiring knowledge of differences between units.

Explanation:

A. Correct. First part is correct. Unit 1 uses blue labeling and Unit 2 uses yellow labeling. Second part is correct. Common systems are normally controlled by Unit 1.

B. Incorrect. First part is incorrect, but plausible since yellow is used for labeling for one of the units, but it is Unit 2. Second part is incorrect, but plausible since only one of the units normally controls common equipment, but it is Unit 1, not Unit 2.

C. Incorrect. First part is correct (see A). Second part is incorrect, but plausible (see B).

D. Incorrect. First part is incorrect, but plausible (see B). Second part is correct (see A).

Technical Reference(s)	OWI-402	Attached w/ Revision # See Comments / Reference
	ODA-102	

Proposed references to be provided during examination: ______

Learning Objective: **DESCRIBE** the administrative requirements for operating plant equipment; performing routine watchstanding evolutions and maintaining system status and plant configuration control in accordance with ODA-106, STI-604.03, OWI-207, ODA-102, ODA-410, ODA-407, OWI-107, STA-694, STA-601 and OWI-409. (ADM.XA1.OB09)

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

Question History: Last NRC Exam 2017 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: 2017 NRC Exam Q69	Revision:

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	3		
	Group			
	K/A	G2.2.3		
Level of Difficulty: 3	Importance Rating	3.8		

(multi-unit license) Knowledge of the design, procedural, and operational differences between units.

Question # 69

Initial Unit 1/2 plant conditions:

- Unit 1 100% power 14 days from Refueling Outage Shutdown
- Unit 2 100% power and stable for 100 days

Subsequently:

- Both Units are responding to events in the ERGs

Complete the statements below regarding procedure response in accordance with ODA-407, Operations Department Procedure Use and Adherence.

1. ___(1)___ has the lead for monitoring common equipment status and operating common equipment in the ERGs.

2. Based on the severity of the event to the unit responsible for monitoring and operating common equipment, this responsibility may be transferred to the other unit by the ___(2)___.

A. (1) Unit 1
(2) Shift Manager

B. (1) Unit 1
(2) Shift Ops Manager

C. (1) Unit 2
(2) Shift Manager

D. (1) Unit 2
(2) Shift Ops Manager

Answer: A

Comments / Reference: 2017 NRC Exam Q69	Revision:

K/A Match:

The question matches the K/A as it requires knowledge of operational differences between units while both units are operating in the ERGs.

Explanation:

A. Correct. 1st part is correct, with both units responding to events in the ERGs, Unit 1 will monitor and operate common equipment. 2nd part is correct, with both units responding to events in the ERGs the Shift Manager may request responsibility of monitoring and operating common equipment transferred to Unit 2 based on the severity of the event on Unit 1.

B. Incorrect. 1st part is correct (see A above). 2nd part is incorrect but plausible as the Shift Ops Manager is responsible for directing station operations for both Unit 1 and Unit 2 per ODA-102, Conduct of Operations.

C. Incorrect. 1st part is incorrect but plausible as with Unit 1 14 days from shutting down for a refueling outage IPO-003A will have directed transferring common equipment to Unit 2. However, this is specifically referring to which unit will provide power to the common equipment and not the unit that will be responsible for monitoring and operation of the equipment if both units are operating in the ERGs. 2nd part is correct (see A above).

D. Incorrect. 1st part is incorrect but plausible (see C above). 2nd part is incorrect but plausible (see B above).

Technical Reference(s)	ODA-407	Attached w/ Revision # See Comments / Reference
	ODA-102	
	IPO-003A/B	

Proposed references to be provided during examination: __________

Learning Objective: **DISCUSS** the operator role in plant operation including the interface with procedures. (LO21.ERG.XG1.OB101)

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

Question History: Last NRC Exam __________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: OWI-402	Revision: 5

CPNPP OPERATIONS DEPARTMENT WORK INSTRUCTION MANUAL		PROCEDURE NO. OWI-402
LABEL STANDARD GUIDE	REVISION NO. 5 INFORMATION USE	PAGE 16 OF 21

ATTACHMENT 8.A
PAGE 2 OF 2

Label Border Color Coding

UNIT	OUTER BORDER	OUTER BORDER TEXT
Unit 1	Blue	White
Unit 2	Yellow	White
Common	White with Black border	Black
No Unit	White	N/A
TRAIN/CHANNEL	**INNER BORDER**	**INNER BORDER TEXT**
Train A	Orange	White
Train B	Green	White
Train A/B	Orange/Green	White
Associated Train A	Orange	White
Associated Train B	Green	White
Train C	Black	White
Channel I	Red	White
Channel II	White with Black border	Black
Channel III	Blue	White
Channel IV	Yellow	Black
Associated Channel I	Red	White
Associated Channel II	White with Black border	Black
Associated Channel III	Blue	White
Associated Channel IV	Yellow	Black

Comments / Reference: ODA-102	Revision: 35

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 35 / INFORMATION USE	PAGE 17 OF 152

[C]6.5 O. Conduct a meeting of representatives of work groups on site. This meeting is held early in the shift to ensure groups are working together to complete the station objectives [4401800].

P. Coordinate all power changes with the Generation Controller and operate the unit at the output requested by the Generation Controller, to the extent allowed by operating procedures, equipment availability and Technical Specifications.

[C] Q. Review routine operating data to assure safe plant operation. [08292]

R. As time permits, notify the Security Shift Supervisor of emergent conditions or activities that may impact safety related plant equipment or site physical protective strategy per STA-919.

[C]6.6 Duties/Responsibilities of the Unit Supervisor (US) [01076, 01078]

- The US is responsible to the SM for the operation of the assigned unit and the supervision of the operating personnel on that shift. The Unit 1 US is responsible for Unit 1 and common systems operation. The Unit 2 US is responsible for Unit 2.
- [C] Implement the ERGs and direct the required procedure steps to stabilize the plant following an emergency or abnormal condition. [09447]
- [C] Monitor or assign another Licensed Operator to monitor the Plant Computer for the affected unit following a reactor trip. [27148]
- Maintain cognizance of the current status of the Critical Safety Functions and provide updates as required.
- It is necessary for the Unit Supervisor to identify the critical parameters necessary for control of the unit. Once these are identified, the US should establish a band within which the critical parameter should be controlled, with a specific owner for each critical parameter.
- Assist in the review and modification of operating procedures as required.
- Assist in on-shift training of operating personnel.
- Approve changes in equipment and system operational status.
- Approve changes to locked/sealed valves or breaker positions on his assigned unit.
- Be cognizant of all clearances which affect the unit and consider the impact of the clearance on any Technical Specification LCO and its effect on system component operations.
- Ensure the SM is aware of any unanticipated changes in plant conditions as they occur.
- [C] The US's normal station is in the Control Room. The US should inform the Unit Control Room Staff when leaving the "At The Controls" area. [08296]

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	3		
	Group			
	K/A	2.2.14		
Level of Difficulty: 2	Importance Rating	3.9		

Knowledge of the process for controlling equipment configuration or status.

Question # 70

The Shift Manager has directed a Verification Lineup performed on a system.

While performing the Verification Lineup, independent verification of the lineup __(1)__ required to be completed.

If a valve is found out of position, the operator performing the lineup should __(2)__.

A. (1) is
(2) obtain specific permission prior to repositioning the valve

B. (1) is
(2) position the valve to the required position and document the repositioning on the lineup attachment

C. (1) is NOT
(2) obtain specific permission prior to repositioning the valve

D. (1) is NOT
(2) position the valve to the required position and document the repositioning on the lineup attachment

Answer: C

K/A Match: K/A match due to requiring knowledge of procedures associated with maintaining system configuration control.

Explanation:

A. Incorrect. First part is incorrect, but plausible since independent verification is required for Initial Lineups. Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since this is how an Initial Lineup would be performed.

C. Correct. First part is correct. Independent verification is not required for Verification Lineups unless a component must be repositioned. Second part is correct. Component repositioning requires specific permission prior to being operated.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ODA-410	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: __________

Learning Objective: **DESCRIBE** the administrative requirements for operating plant equipment; performing routine watchstanding evolutions and maintaining system status and plant configuration control in accordance with ODA-106, STI-604.03, OWI-207, ODA-102, ODA-410, ODA-407, OWI-107, STA-694, STA-601 and OWI-409. (ADM.XA1.OB09)

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

Question History: Last NRC Exam __________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 __________

Comments / Reference: ODA-410	Revision: 16

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6.3 Verification Lineups

A. Verification Lineups are performed at the Shift Operations Manager's discretion, to ensure that systems are properly aligned and capable of performing their intended function. Component positioning should not be performed during the performance of Verification Lineups without specific authorization of the Shift Manager, Unit Supervisor, or Work Window Manager, as applicable. Once directed to do so, component positioning activities are to be accomplished in accordance with STA-694, "Station Verification Activities", OWI-206, "Guidelines for Operation of Manual and Power Operated Valves", and this procedure.

B. The attachments performed for Initial System Lineups may be the same as those performed for Verification Lineups on certain systems. Verification Lineups need not be independently verified for completion.

C. Verification Lineup attachments need only be completed for flowpath valves. The following valves may be marked N/A (not applicable) during the performance of Verification Lineup attachments at the Shift Manager's discretion.

- Vent valves
- Drain valves
- Test connection valves
- Sample valves
- Throttle valves during "at power, system in-service" verification lineup

D. RT valves do not require lineup, but are to be put on the discrepancy sheet with "alternate verification used" to confirm position.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	3		
	Group			
	K/A	2.2.35		
Level of Difficulty: 2	Importance Rating	3.6		

Ability to determine Technical Specification Mode of Operation.

Question # 71

If a Unit is in the process of cooling down for refueling, complete the following statements regarding the Mode of operation:

If average coolant temperature = 195^{o}F, the plant is considered to be in __(1)__.

Mode 6 is INITIALLY declared when the __(2)__ Reactor Vessel Head Closure Bolt is detensioned.

A. (1) MODE 4, Hot Shutdown
(2) first

B. (1) MODE 4, Hot Shutdown
(2) last

C. (1) MODE 5, Cold Shutdown
(2) first

D. (1) MODE 5, Cold Shutdown
(2) last

Answer: C

K/A Match: K/A match due to requiring knowledge of the definitions of different modes of operation.

Explanation:

A. Incorrect. First part is incorrect, but plausible since if it were 5 degrees hotter, it would be correct. Second part is correct. With one RPV head bolt not fully tensioned, the unit is considered to be in Mode 6.

B. Incorrect. First part is incorrect but plausible (see A). Second part is incorrect, but plausible because it could be thought that all RPV head closure bolts must be detensioned to enter Mode 6 as this is similar to exit when all bolts must be tensioned.

C. Correct. First part is correct. If RCS temperature is < 200°F, the unit is considered to be in Cold Shutdown (Mode 5). 2nd part is correct (see A).

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	TS 1.1-1	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **DISCUSS** the terms defined in Technical Specifications. (RLS.SL1.OB01)

Question Source: Bank # ______
Modified Bank # 71459 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: Bank Question 71459	Revision:

6 71459

If a unit is in the process of cooling down for refueling, complete the following statements regarding the Mode of operation:

If average coolant temperature = 195ºF, the plant is considered to be in Mode ________ , if a MINIMUM of ________ Reactor Vessel Head Closure Bolts are less than fully tensioned, the plant is considered to be in Mode 6.

A. 4, Hot Shutdown
one

B. 4, Hot Shutdown
four

C. 5, Cold Shutdown
one

D. 5, Cold Shutdown
four

Answer: C

Answer Explanation
A. Incorrect: 1st part is incorrect because if RCS temperature is < 200ºF, you are considered to be in Cold Shutdown (Mode 5). It is plausible because if you were 5 degrees hotter, you could be correct. 2nd part is correct. With one RPV head bolt not fully tensioned, you are considered to be in Mode 6. B. Incorrect: 1st part is incorrect but plausible (see A). 2nd part is incorrect because only one head bolt detensioned makes you in Mode 6 by definition. It is plausible because 4 bolts is used at Equipment hatch closure criteria. C. CORRECT: 1st part is correct. If RCS temperature is < 200ºF, you are considered to be in Cold Shutdown (Mode 5). 2nd part is correct (see A). D. Incorrect: 1st part is correct (see C). 2nd part is incorrect but plausible (see B).

Comments / Reference: Bank Question 71459	Revision:

Question 6 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	71459
User-Defined ID:	ILOT8387
Cross Reference Number:	RLS.SL1.OB01.005
Topic:	If a unit is in the process of cooling down for refueling, complete the following statements regar
K/A:	
Question Reference:	
SRO:	
Comments:	R/S27E12 This question matches the KA by requiring the ability to determine the TS mode of operability. TS 1.1-1

Question 6 Table-Item Links

10CFR55-41

41.7

General Question

Fundamental / Lower

Initial

Comments / Reference: TS 1.1-1	Revision: 150

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[(b)]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.
(b) All reactor vessel head closure bolts fully tensioned.
(c) One or more reactor vessel head closure bolts less than fully tensioned.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	3		
	Group			
	K/A	2.3.4		
Level of Difficulty: 2	Importance Rating	3.2		

Knowledge of radiation exposure limits under normal or emergency conditions.

Question # 72

Per STA-655, Exposure Monitoring Program, the normal annual administrative exposure limit that can be received, without an administrative extension or Planned Special Exposure, by an operator is __(1)__ mRem Total Effective Dose Equivalent.

Per EPP-305, Emergency Exposure Guidelines and Personnel Dosimetry, the Emergency Exposure limit to save a life is __(2)__ Rem.

A. (1) 2000
(2) 10

B. (1) 2000
(2) 25

C. (1) 4000
(2) 10

D. (1) 4000
(2) 25

Answer: B

K/A Match: K/A match due to requiring knowledge of annual and emergency radiation exposure limits.

Explanation:

A. Incorrect. First part is correct (see B). Second part is incorrect, but plausible because 10 REM is the emergency limit for protecting valuable property.

B. Correct. First part is correct. Per STA-655 ATT 8.A the administrative limit without an extension has been set at CPNPP for operators at 2000 mrem. Second part is correct. Per EPP-305, the emergency dose limit to save a life is 25 REM.

C. Incorrect. First part is incorrect, but plausible since 4000 mrem is the limit with an approved extension but no extension is allowed in the stem of the question. Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	STA-655	Attached w/ Revision # See Comments / Reference
	EPP-305	

Proposed references to be provided during examination: ____________________

Learning Objective: Given a known total exposure, **CALCULATE** the remaining exposure permitted before exceeding an administrative exposure limit in accordance with STA-655, Exposure Monitoring Program, and an NRC exposure limit in accordance with 10CFR20, Standards for Protection Against Radiation. (ADM.RAD.OB05)

Question Source: Bank # ____________
Modified Bank # 75850 (Note changes or attach parent)
New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 12
55.43 ____________

Comments / Reference: Bank 75850	Revision:

Per STA-655, Exposure Monitoring Program, the normal MAXIMUM annual administrative exposure levels that can be received by an...

Escorted Radiation Worker (with DLR) is ______ mRem Deep Dose Equivalent.

Operator is ______ mRem Total Effective Dose Equivalent.

A. 2000
2000

B. 2000
4000

C. 4000
2000

D. 4000
4000

Answer: A

Answer Explanation
A Correct. Per STA-655 ATT 8.A these are the administrative limits set for CPNPP.
B Incorrect. Part 1 is correct per current revision of STA-655. Part 2 is incorrect but plausible as 4000 mrem was the previous administrative limit at CPNPP until 2008. It is plausible to believe that the escorted radiation worker would be allowed a dose of one-half of that of a full time radiation worker.
C Incorrect. Part 1 is plausible since 4000 mrem was the administrative limit until 2008 at CPNPP for Escorted Radiation Worker. Part 2 is correct for the operator. It is plausible that the applicant knows their administrative limit and believes that the limits for CPNPP full time employees are lower than a temporary assignee of a vendor based on the site's strict adherence to ALARA principles. Prior limits were 4000 mrem for both the escort and the escorted radiation worker.
D Incorrect. Part 1 is plausible as described in 'C'. Part 2 is plausible as described in 'B'.

Comments / Reference: Bank 75850	Revision:

Question 62 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	75850
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	Per STA-655, Exposure Monitoring Program, the normal MAXIMUM annual administrative exposure levels
K/A:	2.3.4
Question Reference:	STA-655
SRO:	
Comments:	LC24 NRC

Comments / Reference: STA-655	Revision: 23

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ATTACHMENT 8.A
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ADMINISTRATIVE EXPOSURE LEVELS

RADIATION WORKERS

PERIOD	CALCULATION	LEVEL
Annual	TEDE (Total Effective Dose Equivalent)	2000 mrem
Annual	Skin Dose	40 REM/year
Annual	Extremities	40 REM/year
Annual	Lens of the Eye	12 REM/year
Annual	Total Organ Dose	40 REM/year

PERIOD	EVENT	LEVEL
Annual	Planned Special Exposure (PSE)	4000 mrem
	NOT TO EXCEED:	
Lifetime	Planned Special Exposure (PSE)	Five times the annual dose limit.

Comments / Reference: EPP-305	Revision: 14

CPNPP EMERGENCY PLAN MANUAL		PROCEDURE NO. EPP-305
EMERGENCY EXPOSURE GUIDELINES AND PERSONNEL DOSIMETRY	REVISION NO. 14 INFORMATION USE	PAGE 10 OF 11

ATTACHMENT 1
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EMERGENCY EXPOSURE GUIDELINES

Dose Limit [1] (rem)	Activity	Condition
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Life saving or protection of large populations	Lower dose not practicable
>25	life saving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved (See Attachment 2)

[1] This is the Total Effective Dose Equivalent (TEDE) to non-pregnant adults from exposure and intake during an emergency condition at CPNPP. Workers performing services during emergencies should limit dose to the lens of the eye to three times the listed value and doses to any other organ (including skin and body extremities) to ten times the listed value. These limits apply to all doses received from an incident, except those received in unrestricted areas as members of the public during the intermediate (ingestion) phase of the incident.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	3		
	Group			
	K/A	2.3.15		
Level of Difficulty: 3	Importance Rating	2.9		

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question # 73

You are exiting an RCA and approach the portable frisker which is reading 275 cpm background radiation.

(1) Are you allowed to perform your whole body frisk in an area with background radiation at this level?

(2) You should be considered to have received contamination if the frisker rises to a MINIMUM value of __(2)__.

A. (1) NO
(2) 100 cpm above background

B. (1) NO
(2) double the background

C. (1) YES
(2) 100 cpm above background

D. (1) YES
(2) double the background

Answer: A

K/A Match:	K/A match due to requiring knowledge of personnel monitoring equipment use.
Explanation:	

A. Correct. First part is correct. Frisking may only be performed in areas with the background radiation < 200 cpm. Second part is correct. Contamination is identified by being 100 cpm above background.

B. Incorrect. 1st part is correct (see A). 2nd part is incorrect. It is plausible because this would be indication of contamination in this case, but minimum level to be considered contaminated is 100 cpm above background.

C. Incorrect. 1st part is incorrect because background count rates are > 200 cpm, therefore NO (not allowed). It is plausible because it is well over the preferred background level (100 cpm). 2nd part is correct (see A).

D. Incorrect. 1st part is incorrect, but plausible (see C). 2nd part is incorrect, but plausible (see B).

Technical Reference(s)	STA-653	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given a set of contamination conditions, **DETERMINE** the posting requirements for the area in accordance with RPI-304, Radiological Posting and Labeling. (ADM.RAD.OB02)

Question Source: Bank # ______
Modified Bank # 82575 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 12
55.43 ______

Comments / Reference: Bank 82575	Revision:

You are exiting an RCA and approach the portable frisker which is reading 350 cpm background radiation. Based on this information, answer the following questions:

1. Are you allowed to perform your whole body frisk in an area with background radiation at this level?

2. When you do perform a whole body frisk (in this area or another) you should:

A. (1) NO
(2) Start on the lowest scale and move up one scale at a time until the meter is on scale.

B. (1) NO
(2) Start on the highest scale and go down one scale at a time until the meter is on scale.

C. (1) YES
(2) Start on the lowest scale and go up one scale at a time until the meter is on scale.

D. (1) YES
(2) Start on the highest and go down one scale at a time until the meter is up on scale.

Answer: A

Answer Explanation
A. Explanation: B. A is correct. Frisking may only be performed in areas with the background radiation < 300 cpm. Frisking should be started on the X1 (lowest) scale. C. B is wrong. 1st part is correct. 2nd part is incorrect because frisking should be started on the X1 scale. It is plausible because scales are set up such that they go up by a factor of 10 for every scale. Using that knowledge of how the scales work, it would figure that the applicant could think that X1 (0-10), X10 (0-100), X100 (0-1000) cpm. With this philosophy, it would reason that they would start on the highest scale so as not to "peg" the meter. D. C is wrong 1st part is incorrect because background count rates are > 300 cpm, therefore NO (not allowed). It is plausible because it is well over the preferred background level (100 cpm). 2nd part is correct. E. D is wrong. 1st part is incorrect but plausible (see C). 2nd part is incorrect but plausible (see B).

Comments / Reference: Bank 82575	Revision:

Question 61 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	2.00
System ID:	82575
User-Defined ID:	ILOT
Cross Reference Number:	ADM.XA1.OB03.032
Topic:	You are exiting an RCA and approach the portable frisker which is reading 350 cpm background radia
K/A:	G.2.3.5
Question Reference:	
SRO:	
Comments:	LC22 RO Retake NRC; R/S25E31 (Admin) REF: STA-653

Comments / Reference: STA-653	Revision: 21

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[C] 6.7 Personnel Monitoring
[00816]

6.7.1 Contamination monitoring requirements should be posted at the exit of Satellite/Alternate RCA's.
[CR-2011-005658]

6.7.2 Unless otherwise posted or authorized, all personnel shall monitor themselves after handling contaminated materials or exiting a contaminated area, at the nearest available frisker or PCM, and when exiting at the access control point.

NOTE:	Where available, Personnel Portal Monitors can be used in lieu of frisking.

6.7.3 Personnel exiting from the following RCA's into an outside RCA should perform a frisk of the hands and feet prior to proceeding.
[AI-CR-2016-002617-9]

- Fuel Building door 100 when the Protected Area RCA Yard is posted as an RCA
- Warehouse C Building RCA when entering into the Posted Yard Area.
- Entry into the Protected Area RCA Yard via the Radioactive Protection Building (3J43).

6.7.4 The frisker is most commonly used for monitoring after exiting a contaminated area or after handling contaminated material. Frisking should be done with a background count rate of less than 200 counts per minute (cpm).

6.7.5 The Personnel Contamination Monitor (PCM) is most commonly used at the access control point, although it may be used to replace the frisker at locations such as the Reactor Building Personnel Air Lock.

6.7.6 Hand-held friskers should be available at or near normally established step-off pads when PCM's are not readily available.
[CR-2011-005658]

6.7.7 Hand held friskers should be available at or near each normally established RCA egress point for RP Technician use to respond to quantify contamination alarms.
[CR-2011-005658]

6.7.8 Portal Monitors will most commonly be used when exiting the protected area (e.g., PAP, AAP).

6.7.9 See Attachment 3 for guidelines on the use of friskers, PCMs and portal monitors.

Comments / Reference: STA-653	Revision: 21

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ATTACHMENT 3
PAGE 1 OF 2

GUIDELINES FOR PERSONAL MONITORING

Monitoring With a Frisker

> **NOTE:** Due to background radiation levels some friskers may indicate a background count rate greater than 200 cpm. These friskers may be used to perform a gross contamination check. In-plant low background frisker stations are provided as necessary.

1. Ensure meter is turned on and the scale switch is set at X1. Observe background level momentarily.

2. Without picking up the probe, frisk both sides of one hand. The probe should be about ½ inch away from the surface area being frisked.

3. Pick up probe and frisk remainder of body, scanning at a slow rate. Special attention shall be given to the face, soles of feet, hands, knees, posterior, and any surface left exposed while wearing protective clothing and dosimetry.

4. If an increase in the count rate is noted (visual or audible), return the probe to the spot and verify count rate. A significant and abrupt rise/drop in the count rate may indicate the presence of a DRP. Notify Radiation Protection.

5. If the frisker alarms or a continuous count rate of 100 cpm above background or greater is noted, remain at that point and notify, or have a co-worker notify, Radiation Protection for assistance. If contamination is not detected, proceed as usual.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier	3		
	Group			
	K/A	2.4.6		
Level of Difficulty: 2	Importance Rating	3.7		

Knowledge of EOP mitigation strategies.

Question # 74

Which of the following actions should NOT be performed prior to direction from the US in an Emergency Response Guideline?

A. Isolating AFW flow to a single faulted SG.

B. Throttling AFW flow to control a ruptured SG level within the required band.

C. Adjusting RCP seal injection flow to maintain within the required range of flow.

D. Securing a CCP to prevent overfilling the Pressurizer following an inadvertent SI.

Answer: D

K/A Match: K/A match due to

Explanation:

A. Incorrect. Plausible because this is a numbered step in EOP-2.0, but the ERG Rules of Usage addresses this as being acceptable.

B. Incorrect. Plausible because this is a numbered step in EOP-3.0, but the ERG Rules of Usage addresses this as being acceptable.

C. Incorrect. Plausible because this is a numbered step in EOS-0.1, but the ERG Rules of Usage addresses this as being acceptable.

D. Correct. Performing steps out of sequence is allowed, but must be done with caution to prevent masking symptoms or defeating the intent of the EOP being used. Although terminating SI early might be beneficial to prevent filling the Pressurizer if the only event is a spurious SI, this may result in further degradation of the plant if another undiagnosed event is in progress.

Technical Reference(s)	ODA-407	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________

Learning Objective: **DESCRIBE** the requirements associated with deviating from an ERG in accordance with ODA-407, Operations Department Procedure Use and Adherence. (ERG.XD2.OB25)

Question Source: Bank # 23501
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam LC20

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments / Reference: Bank 23501	Revision:

Which of the following actions would be INAPPROPRIATE to perform prior to direction in an Emergency Response Guideline?

A. Isolating Auxiliary Feedwater flow to a single faulted Steam Generator.

B. Throttling Auxiliary Feedwater flow to control a ruptured Steam Generator level within the required band.

C. Securing a Centrifugal Charging Pump to prevent overfilling the Pressurizer following an inadvertent Safety Injection.

D. Closing the Main Steam Isolation Valves to isolate a steam line break which has not resulted in a Safety Injection.

Answer: C

Answer Explanation
A. Incorrect. Plausible because this is a numbered step in EOP-2.0, but the ERG Rules of Usage addresses this as being acceptable.
B. Incorrect. Plausible because this is a numbered step in EOP-3.0, but the ERG Rules of Usage addresses this as being acceptable.
C. Correct. Performing steps out of sequence is allowed, but must be done with caution to prevent masking symptoms or defeating the intent of the EOP being used. Although terminating SI early might be beneficial to prevent filling the Pressurizer if the only event is a spurious SI, this may result in further degradation of the plant if another undiagnosed event is in progress.
D. Incorrect. Plausible because this is a numbered step in EOS-0.1, but the ERG Rules of Usage addresses this as being acceptable.

Comments / Reference: Bank 23501	Revision:

Question 22 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	23501
User-Defined ID:	ILOT6313
Cross Reference Number:	ERG.XD2.OB11.003
Topic:	Which of the following actions would be INAPPROPRIATE to perform prior to direction in an Emergenc
K/A:	G.2.4.12
Question Reference:	
SRO:	
Comments:	LC20 NRC; R/S22E14; R/S23E24 REF: ODA-407 Att. 8.A

General Question

Comprehension / Higher

Initial

LC20 NRC

Comments / Reference: ODA-407	Revision: 17

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ATTACHMENT 8.A
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ERG RULES OF USAGE

15. • When performing YELLOW status FRGs, continuous actions and foldout page items of ORGs are monitored and implemented as required, therefore, YELLOW path procedures are considered to be performed in parallel. If ORG conditions degrade, performance of a YELLOW path FRG may be suspended.

EXAMPLE 3 - While in EOS-1.1A/B, the SRO makes a decision to implement FRH-0.5A/B due to SG low level. Subsequently, RCS subcooling is lost while in FRH-0.5A/B. The SRO should terminate activities of FRH-0.5A/B unless adequate personnel coverage exists, and implement actions of EOS-1.1A/B foldout page to reinitiate ECCS flow.

• Operators may take action to control the AFW System to enhance plant control, consistent with the intent of the ERGs, prior to steps directing such action in the ERGs. These actions include but are not limited to throttling flow and stopping pumps.

EXAMPLE 4 - If excessive cooldown is observed before reaching EOP-0.0A/B Step 9, AFW flow should be throttled or the TDAFW pump should be stopped as the step directs. If excessive cooldown is observed before reaching EOS-0.1A/B Step 1, the TDAFW pump should be stopped or AFW flow should be throttled as the step directs.

Stopping AFW flow to a faulted SG prior to entry into EOP-2.0A/B, is allowed consistent with meeting requirements of >(Unit 1, 43%)(Unit 2, 10%) NR (Unit 1, 50% adverse)(Unit 2, 18% adverse) or 460 gpm minimum as specified in EOP-0.0A/B. Similarly, it is permissible to stop AFW flow to a ruptured SG but maintain >(Unit 1, 43%)(Unit 2, 10%) NR (Unit 1, 50% adverse)(Unit 2, 18% adverse) in that SG, prior to entry into EOP-3.0A/B, consistent with meeting requirements of >(Unit 1, 43%)(Unit 2, 10%) NR (Unit 1, 50% adverse)(Unit 2, 18% adverse) or 460 gpm minimum as specified in EOP-0.0A/B.

16. The following rules apply to the use of instrumentation during ERG performance:

• ERG response and recovery actions should not be based solely on a single plant indication when more than one of the same parameter is available. Indications from non-qualified instruments exposed to post LOCA environments should be checked against qualified instruments if possible.

• For Control Board analog meters, the most precision an operator can obtain is one-half the distance between the face plate increment markings.

Example: a tank level indicator face plate has 2 percent (%) increments. When the indicator pointer is between 66% and 68%, the accurate response for current level is 67%.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier	3		
	Group			
	K/A	2.4.17		
Level of Difficulty: 2	Importance Rating	3.9		

Knowledge of EOP terms and definitions.

Question # 75

While performing an Emergency Operating Procedure, the Unit Supervisor observes an asterisk next to a step number.

The asterisk indicates the step...

A. is only applicable under Adverse Containment conditions.

B. is to be performed from memory and verbalized upon completion.

C. must be completed in its entirety before the subsequent step is started.

D. is a continuous action that applies until superseded or no longer applies.

Answer: D

K/A Match: K/A match due to requiring knowledge of symbols used in the ERG network.

Explanation:

A. Incorrect. Plausible as Adverse containment parameters are specified in ERG's by parentheses.

B. Incorrect. Plausible as Initial Action are designated by diamond symbol.

C. Incorrect. Plausible as steps in ABNs and ERGs do not have to be completed prior to proceeding to next step. They just have to be started and have assurance that it is progressing. Step where this does not apply have to be designated.

D. Correct. An asterisk designates a continuous action step, which applies from that point in the procedure until superseded or no longer applies.

Technical Reference(s)	ODA-407	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **DESCRIBE** the expectations associated with performing immediate action, continuous action and radiation hazard steps and how each is identified in the Emergency Response Guidelines. (ERG.XD2.OB05)

Question Source: Bank # 41924
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: ODA-407	Revision: 17

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 17 INFORMATION USE	PAGE 20 OF 63

ATTACHMENT 8.A
PAGE 2 OF 25

ERG RULES OF USAGE

4. Some ERGs contain a foldout page which provides information or actions that are normally applicable at any step in the related procedure. Some foldout page actions may be stated to apply after performance of a certain procedure step. The foldout page should be continuously monitored during ERG performance.

   Copies of the foldout page are maintained in the procedure being implemented and in separate locations for the RO and BOP positions. The SRO directing emergency response activities identifies when foldout page information is applicable upon entry into the ERG. Each member of the operating crew reviews the foldout page information to ensure the criteria is implemented as required during recovery actions.

   Only the foldout page for the current controlling procedure is applicable, and it is no longer applicable after a transition is made to a different controlling procedure. For example, the foldout page of EOP-0.0 does not apply when either FRS-0.1 or FRH-0.1 is the controlling procedure and EOP-0.0 steps are to be performed for verification of auto actions.

5. Additional information may be displayed by the ERG Step numbers to assist the operator in performance of the associated step.

   A. ERG steps that are performed in an area where radiation hazards are positively identified or which, when performed, create a radiation hazard, should be specified in the left margin by using a [R]. A substep is included when a plant announcement or notification of Plant Staff is required by the ERG action, and information may also be included in the Step's Bases to identify the radiological concern.

      When ERG instructions require local actions to be performed, the Control Room Operators should inform personnel being dispatched of the associated radiation concerns, as required for the current plant conditions.

   B. Some steps require "continuous" performance throughout remainder of the procedure. Continuous Action Steps are identified by an asterisk (*) and shading on the procedure flow chart, by an asterisk next to the step number in the procedure, and as part of the Continuous Action Step listing on Attachment 1.B of the associated ERG. The following are examples of Continuous Action Steps:

      - Step instruction uses action verbs such as monitor, maintain, or control.
      - Step instruction to perform action when contingent condition satisfied (e.g., WHEN, THEN).
      - A note precedes a step to identify that the step is applicable during subsequent steps.

Comments / Reference: ODA-407	Revision: 17

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 17 INFORMATION USE	PAGE 21 OF 63

ATTACHMENT 8.A
PAGE 3 OF 25

ERG RULES OF USAGE

5. B. A Continuous Action Step is applicable from the point at which it is first encountered until superseded by alternate instruction OR stated to be no longer applicable. The course of direction stated by the Continuous Action Step should be followed until subsequent procedure instruction OR a change in the response and recovery strategy is encountered that supercedes the previous instruction.

EXAMPLE 1 - EOP-1.0A/B, "Loss of Reactor or Secondary Coolant" Step 3 provides a Continuous Action Step to "Check Intact SG Levels". The continuous instructions for the step include:

- Maintain total AFW flow greater than 460 gpm until narrow range level is greater than 43% (50% FOR ADVERSE CONTAINMENT) for Unit 1, and 10% (18% FOR ADVERSE CONTAINMENT) for Unit 2 in at least one intact SG.
- Control AFW flow to maintain narrow range level between 43% (50% FOR ADVERSE CONTAINMENT) and 60% for Unit 1, and 10% (18% FOR ADVERSE CONTAINMENT) and 50% for Unit 2.

The continuous instruction for AFW flow and SG level maintenance applies with the subsequent transition to EOS-1.1A/B, "Safety Injection Termination" until specific instruction in EOS-1.1A/B provides guidance for AFW flow and SG level maintenance.

EXAMPLE 2 - EOS-1.3A/B, "Transfer to Cold Leg Recirculation" Step 5 provides a Continuous Action Step to "Verify Pumps Aligned From Containment Recirculation Sump NOT Affected By Sump Blockage". The continuous instructions for the step requires reporting of abnormal operating parameters of any ECCS pump or Containment Spray pump so that proper actions can be taken to protect the affected ECCS pump(s) and Containment Spray pumps.

The Continuous Action Steps attachments, when applicable are maintained in the procedure being implemented and in a separate location for the RO and BOP positions. The SRO directing emergency response activities identifies when a continuous action is applicable upon performance of the step. The operating crew has responsibility to ensure the action is performed when it is required by applicable Unit conditions. The Continuous Action Step attachment provides a summary of the continuous action attribute for each step and can be referenced during response and recovery actions.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			2
	Group			1
	K/A	006.A2.13		
Level of Difficulty: 2	Importance Rating			4.2

Emergency Core Cooling: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent SIS actuation

Question # 76

Given the following conditions:

- Unit 1 in MODE 3 waiting to enter MODE 2
- An Inadvertent Train A Safety Injection occurs

The Unit Supervisor __(1)__ expected to exercise procedure expediency as described in Operations Guideline 3, Attachment 6, Strategies for Successful Transient Mitigation.

An Inadvertent Safety Injection places a challenge on the __(2)__.

A. (1) is NOT
(2) PRZR safeties

B. (1) is
(2) PRZR safeties

C. (1) is NOT
(2) RCS cooldown rate

D. (1) is
(2) RCS cooldown rate

Answer: B

K/A Match: K/A match due to requiring knowledge of the procedural strategy in response to an inadvertent safety injection.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Explanation:

A. Incorrect. First part is incorrect, but plausible as no accident has occurred and it may be thought that terminating SI is secondary to stabilizing the plant. Second part is correct (see B).

B. Correct. First part is correct. Procedure expediency is required to prevent overfilling the pressurizer. Second part is correct. Overfilling the pressurizer could result in water damage to the pressurizer safety valves.

C. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since adequate heat removal existed without SI and it could be thought that the additional heat removal would result in an excessive RCS cooldown condition.

D. Incorrect. First part is correct (see B). Second part is incorrect, but plausible (see C).

Technical Reference(s)	Ops Guideline 6, Att 3	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **LIST** three beneficial functions and one possible problem caused by SI flow during accident conditions. (ERG.XD4.OB01)

Question Source: Bank # ____________

Modified Bank # 75863 (Note changes or attach parent)

New ____________

Question History: Last NRC Exam ____________

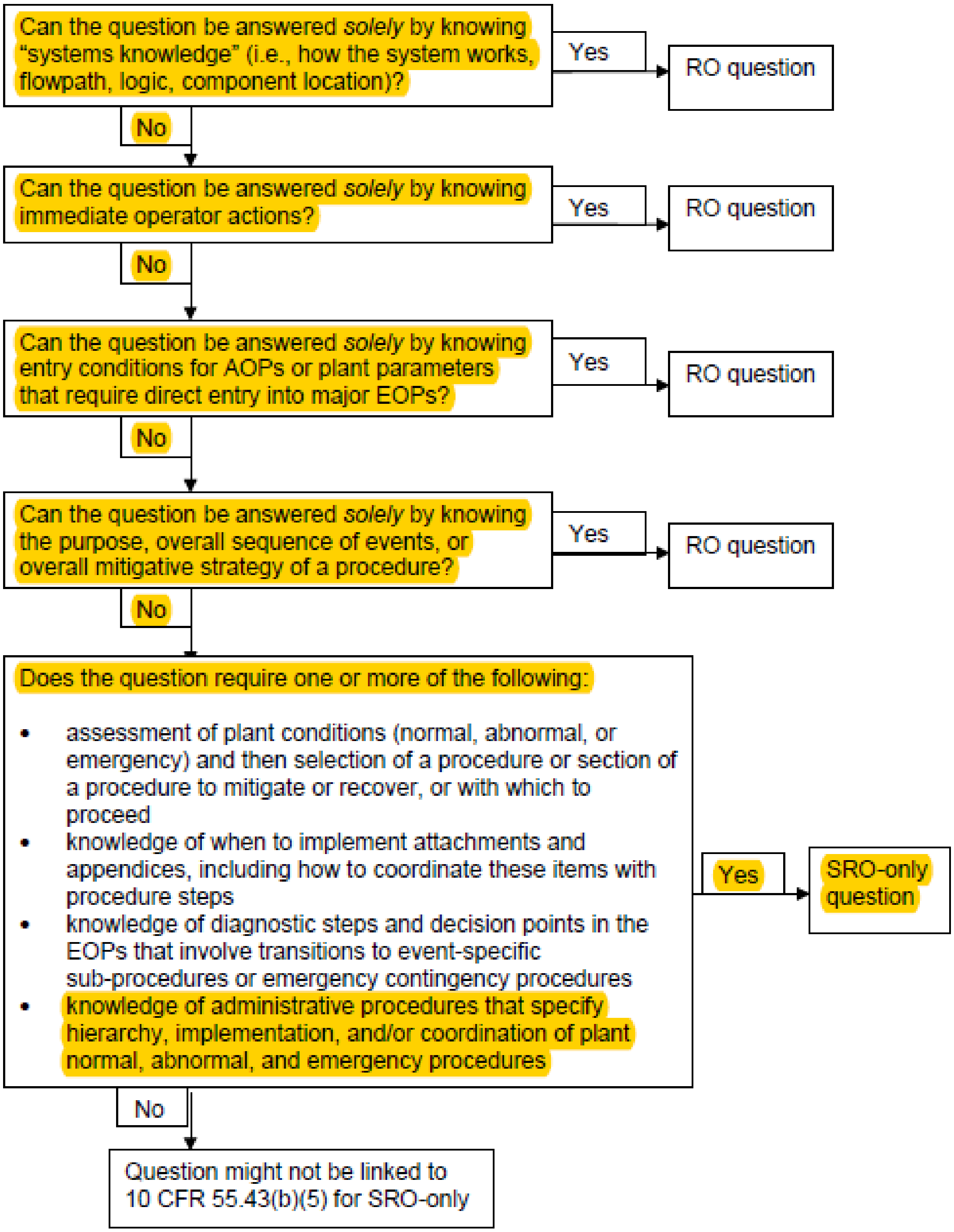
Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____________

55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: Bank 75863	Revision:

- Unit 1 in MODE 3 waiting to enter MODE 2
- An Inadvertent Safety Injection occurs

Minimum flow protection for any running Centrifugal Charging Pump is provided by recirculation flow to the ________________________.

Unit Supervisor ________ expected to exercise Procedure Expediency as described in Operations Guideline 3, Attachment 6, Strategies for Successful Transient Mitigation.

A. Refueling Water Storage Tank
is NOT

B. Refueling Water Storage Tank
is

C. Charging Pump suction header
is NOT

D. Charging Pump suction header
is

Answer: B

Answer Explanation

Comments / Reference: Bank 75863	Revision:

A Incorrect. First part is correct (See B below). Second part is incorrect but plausible if believed that in Mode 3 the procedure expediency is not required in the ERG network to satisfy the inadvertent SI event.

B Correct. First part is correct in that with a Safety Injection actuation the CCP discharge aligns such that normal miniflow and charging lines are isolated and safety injection line and an alternate miniflow back to the RWST are opened. As the RCS pressure remains high (2235 psig) and will increase as fluid is injected from the inadvertent SI, the CCPs are protected by the alternate miniflow lines back to the RWST. Second part is correct per OPGD 3, Att. 6, procedure expediency is expected to be used by the Unit Supervisor to prevent the pressurizer from going solid.

C Incorrect. First part is incorrect but plausible because normal miniflow from the CCPs is directed back to the Charging Pump suction header. Second part is incorrect but plausible (See A above).

D Incorrect. First part is incorrect but plausible (See C above). Second part is correct (See B above).

Question 43 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	75863
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	Unit 1 in MODE 3 waiting to enter MODE 2 An Inadvertent Safety Injection occurs Minimum flow prot
K/A:	006 A2.13
Question Reference:	EOS 1.1
SRO:	YES
Comments:	R/S27E28 (Comp)

Comments / Reference: Ops Guideline 6, Att 3	Revision: 10-21-2019

Operations Guideline 3
Attachment 6

7.3. Procedure Expediency

The Unit Supervisor's pace through ERGs / ABNs is always critical until the Reactor has been placed in a stable condition.

The Unit Supervisor should consider temporarily suspending expectations that may otherwise be in effect during ERG/ABN performance (e.g. Crew Briefings). any time delay in working through the procedures may contribute to the event severity. The following are examples of conditions associated with ERG/ABN performance.

- **Inadvertent SI**: To preclude overfilling the pressurizer with the increased risk of damaging the pressurizer safeties, the ERG network should be worked expeditiously to the point of re-establishing letdown (terminating the pressurizer fill.)

- **SGTR**: In the event of a SGTR in which an uncontrolled level rise is clearly observable in the ruptured SG, any unnecessary delay prior to RCS depressurization and termination of ECCS flow (closing u-HV-8801 A & B) increases the probability of SG overfill and a radioactive release.
- **Main Steam Break**: To preclude overfilling the pressurizer with the increased risk of damaging the pressurizer safeties, the ERG network should be worked expeditiously to the point of re-establishing letdown (terminating the pressurizer fill.)
- **Transition and Completion of EOS 1.3 Cold Leg Recirculation**: To preclude uncovering the fuel during a large break LOCA.

- **Transition to FRH-0.1 upon loss of Heat Sink:** The need to establish RCS Feed & Bleed upon loss of a Heat Sink is critical to ensuring core cooling is properly maintained.

- **Response to Fire in the Control Room or Cable Spreading Room**: To mitigate the potential for inadvertent component operation (PORVs), provide cooling flow to SSCs, and provide controls for RCS Inventory and Decay Heat Removal. Actions taken should be performed expeditiously to the point that CCWP A is started from the RSP.

Additional methods:

- The direction to complete Attachment 2 of EOP-0.0 should take precedence over the completion of throttling AFW. This can be accomplished by interrupting the BOP while throttling AFW and providing direction for the performance of Attachment 2. The BOP can then quickly complete actions with AFW and proceed with Attachment 2 while the US and RO proceed in EOP-0.0.

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Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier			2
	Group			1
	K/A	013.G.2.2.25		
Level of Difficulty: 2	Importance Rating			4.2

Engineered Safety Features Actuation: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question # 77

In accordance with TSB 3.3.2, ESFAS Instrumentation, the reason the following signals use a 2 of 4 logic to actuate, vice a 2 of 3 logic to actuate, is:

Low Pressurizer Pressure Safety Injection logic __(1)__.

High-3 Containment Pressure logic __(2)__.

A. (1) is used for both protection and control functions
(2) requires additional redundancy because it energizes to trip

B. (1) is used for both protection and control functions
(2) is used for both protection and control functions

C. (1) requires additional redundancy because it energizes to trip
(2) requires additional redundancy because it energizes to trip

D. (1) requires additional redundancy because it energizes to trip
(2) is used for both protection and control functions

Answer: A

K/A Match: K/A match due to requiring knowledge of TS bases for ESFAS instrumentation.

SRO Only: SRO Only due to 10 CFR 55.43(b)(2) Analysis requiring knowledge of TS bases that is required to analyze TS-required actions and terminology.

Explanation:

A. Correct. First part is correct. Pressurizer pressure provides both control and protection functions and actuation logic must be able to withstand both an input failure to control system, and a single failure in the other channels providing the protection function actuation. Thus, four channels are required to satisfy the requirements with a two-out-of-four logic. Second part is correct. This is one of the only functions that requires the bistable output to energize to perform its required action. Four channels are used in a two-out-of-four logic configuration because this function is energize to trip.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible if thought that reason for pressurizer pressure logic also applied to high-3 logic.

C. Incorrect. First part is incorrect, but plausible if thought that pressurizer pressure was energized to actuate. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	TSB 3.3.2	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given ESFAS operability status or parameter indications, various plant conditions, and a copy of regulatory requirements (TS, TRM, etc.), **ASSESS** any LCO entries, applicable conditions, and required actions (including completion time) in accordance with the associated regulatory requirement(s) and their bases. (SYS.ES1.OB08)

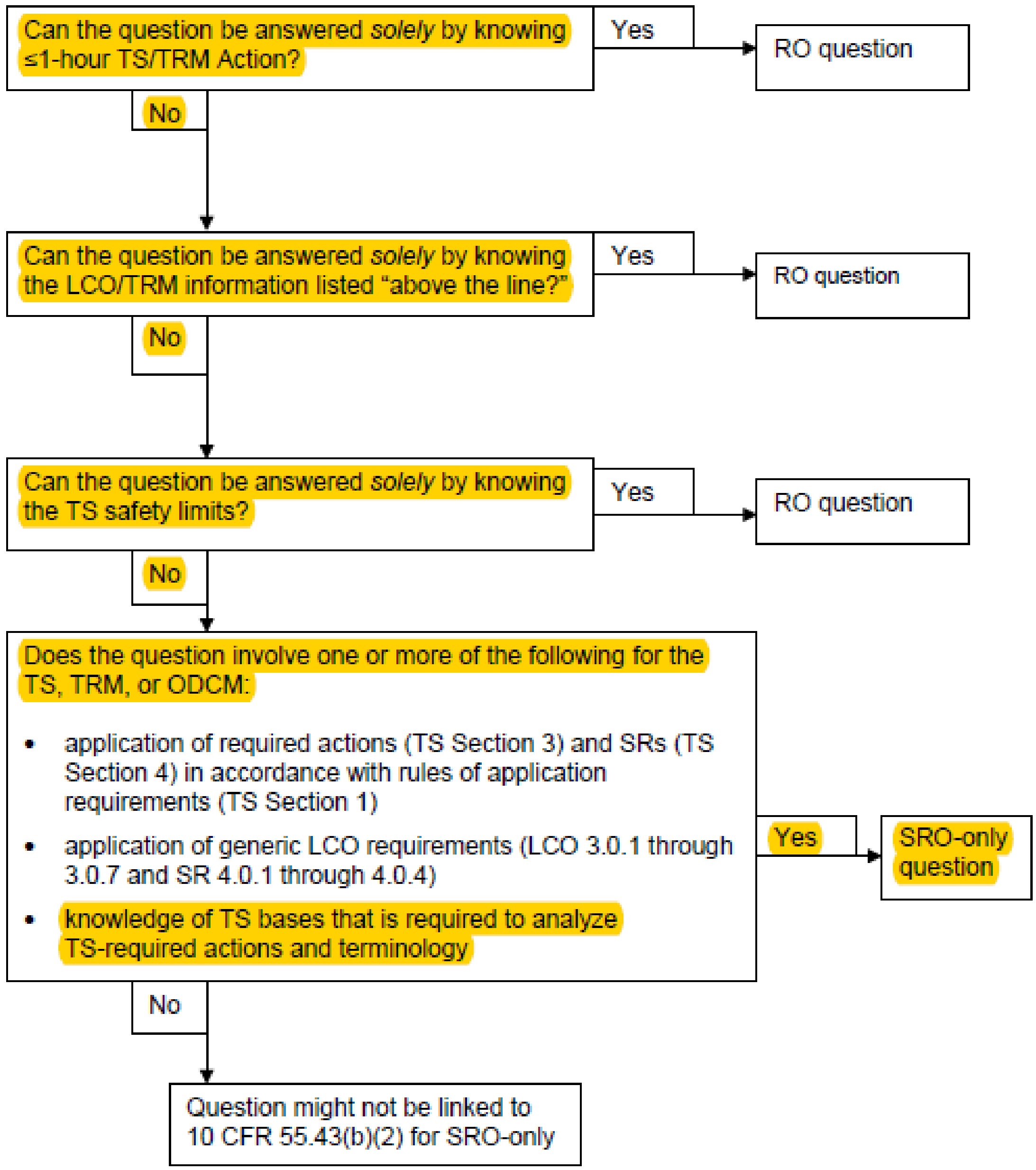
Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:



Comments / Reference: TSB 3.3.2	Revision: 77

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Containment Pressure-High 1 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters (d/p cells) and electronics are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment.

Thus, the high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure-High 1 must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to significantly pressurize the containment.

d. Safety Injection - Pressurizer Pressure-Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) relief or safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

The pressurizer pressure provides both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, and SI. Therefore, the actuation logic must be able to withstand both an input failure to control system, which may then require the protection function actuation, and

(continued)

Comments / Reference: TSB 3.3.2	Revision: 77

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic.

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11 and below P-11, unless the Safety Injection - Pressurizer Pressure-Low Function is blocked) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure-High 1 signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

e. Safety Injection - Steam Line Pressure-Low

Steam Line Pressure-Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure-Low provides no input to any control functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line.

(continued)

Comments / Reference: TSB 3.3.2	Revision: 77

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation hand switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray - Containment Pressure

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

This is one of the only Functions that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

Four channels are used in a two-out-of-four logic configuration. This configuration is called the Containment Pressure-High 3 Setpoint.

(continued)

Comments / Reference: TSB 3.3.2	Revision: 77

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Additional redundancy is warranted because this Function is energize to trip. Containment Pressure-High 3 must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure-High 3 setpoint.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except component cooling water (CCW) to the reactor coolant pumps, at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW on the low pressure Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating CCW on the low pressure signal may force the use of feed and bleed cooling, which could prove more difficult to control.

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of CCW, are isolated.

CCW is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers, motor air coolers, and upper and lower bearing coolers. All process lines not equipped with remote operated isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains.

(continued)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 4	Tier			2
	Group			1
	K/A	022.G.2.4.18		
Level of Difficulty: 3	Importance Rating			4.0

Containment Cooling: Knowledge of the specific bases for EOPs.

Question # 78

Given the following conditions:

- A LOCA has occurred inside Unit 1 Containment
- Containment pressure rose to 21.8 psig, resulting in a Containment Spray actuation
- Containment pressure has subsequently lowered to 9.4 psig and is continuing to lower slowly
- Due to multiple RHR malfunctions, entry has been made to ECA-1.1A, Loss of Emergency Coolant Recirculation
- All Containment Fan Coolers were stopped by the SI signal

Per ECA-1.1A, Containment Fan Coolers may _______, WITHOUT consulting Plant Staff.

A. be restarted, under the above conditions

B. be restarted, once Containment pressure drops below 5 psig

C. NOT be restarted, due to a lack of CCW flow to HVAC Chill Water

D. NOT be restarted, due to potential water hammer damage to HVAC Chill Water

Answer: D

K/A Match: K/A match due to requiring knowledge of the bases when non-safety containment cooling equipment can be started in the event of an accident.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

Explanation:

A. Incorrect. Plausible since CS is to be secured and it is desirable to restore some form of containment cooling and a step in the procedure directs restarting the coolers, but only if pressure has remained below 5 psig (without consulting plant staff).

B. Incorrect. Plausible since CS is to be secured and pressure must be below 5 psig to operate the fan coolers and a step in the procedure directs restarting the coolers, but it must have remained below 5 psig for the duration of the event otherwise a Plant Staff evaluation is required prior to starting the coolers.

C. Incorrect. Plausible since CCW flow must be verified to the Chillers or they cannot be started, but there is no reason to think that CCW flow cannot be restored.

D. Correct. With containment pressure above 5 psig, the cooling water for the containment coolers may be saturated and water hammer may occur in the system if chill water is restored. Plant Staff must be consulted to evaluate the effects of water hammer prior to restarting.

Technical Reference(s)	ECA-1.1	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given a procedural step, or sequence of steps from ECA-1.1, **STATE** the purpose/basis for the step(s). (ERG.C11.OB04)

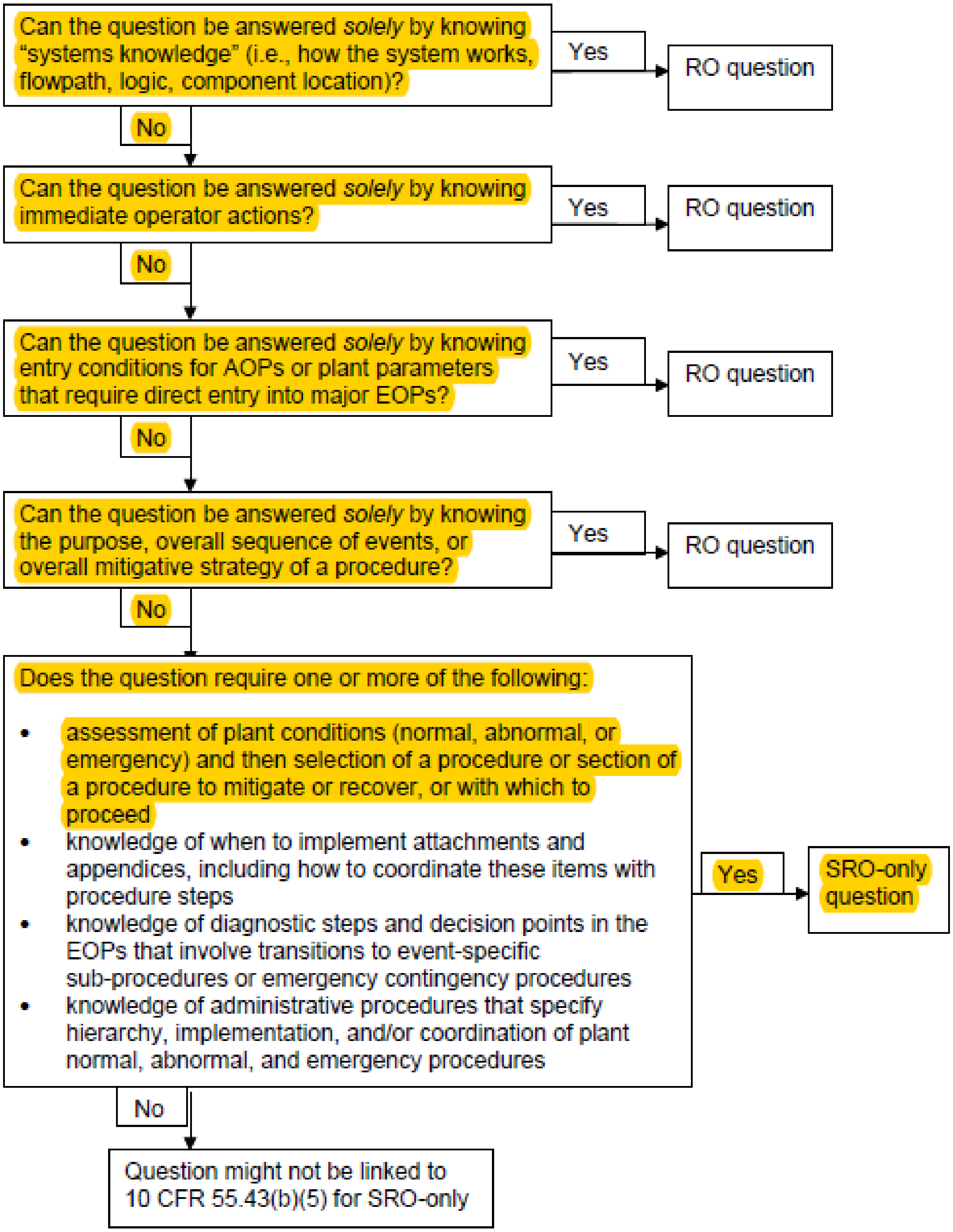
Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: ECA-1.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.1A
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 9	PAGE 5 OF 83

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Reset SI Sequencers If Necessary.	
6	Reset Containment Isolation Phase A and Phase B	
7	Reset Containment Spray Signal	
8	Reset RHR Auto Switchover.	
9	Check If Containment Fan Coolers Should Be Started	
	a. Verify containment pressure - HAS REMAINED LESS THAN 5 PSIG.	a. Notify Plant Staff to determine if Containment Fan Coolers should be started to provide containment cooling. Go to Step 10.
	b. Ensure 1-ALB-3B Window 1.16 SEAL WTR HX CCW RET FLO LO - DARK.	
	c. Open VENT CHLR CCW SPLY & RET VLV: • 1-HS-4650	
	d. Start HVAC Ventilation Chillers as necessary.	
	e. Open CH WTR SPLY & RET ISOL VLVS: • 1-HS-6084 • 1-HS-6083 • 1-HS-6082	
	f. Start Containment Fan Coolers	
	g. Start CRDM Fans	
*10	Check RWST Level - GREATER THAN RWST EMPTY	Go to step 33.

-

Comments / Reference: ECA-1.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.1A
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 9	PAGE 59 OF 83

ATTACHMENT 7
PAGE 5 OF 29

BASES

The intent of this step is to provide containment heat removal capability by using containment fan coolers when conditions inside containment are NOT adverse (i.e., LOCA outside containment). The containment fan coolers and its associated ventilation chilled water cooling are not qualified for post accident operation. High temperature conditions inside containment can cause flashing/waterhammer to occur and chilled water to containment should not be aligned under these conditions, and consequently, the containment fan coolers should not be started under these conditions. A check that containment pressure has remained less than 5 psig ensures that the chilled water inside the containment fan coolers cooling supply will not be under saturated conditions when cooling water is realigned. If containment pressure increased above 5 psig prior to reaching this step, the containment fan coolers cooling supply may be under saturated conditions and the Plant Staff is notified to evaluate the potential for the cooling water flashing to steam, and the subsequent waterhammer that may be experienced if cooling flow is realigned.

A check of the alarm for the Seal Water Heat Exchanger is to verify flow in the Non-Safeguards loop of the CCW system to ensure cooling flow is available to the HVAC Centrifugal Water Chillers. If this component is not available, any indication of flow in the Non-Safeguards loop is sufficient to satisfy the substep.

For the reset of the Containment isolation signals, this part of the automatic logic requires a deliberate operator action to remove the "close" signal. No valve will reposition upon actuation of the resets, but subsequent control actions will open the valves. These valves should remain closed, unless necessary process streams are being established, until the cause of the SI is determined or corrected.

The maximum CCW pump flow is 17,500 GPM to prevent pump runout. If the CCW pump runs out during performance of this alignment, non-essential CCW loads will need to be isolated to prevent CCW pump damage.

STEP 10: If the RWST is not empty, the operator proceeds with Steps 11 through 32, which are concerned with minimizing the RWST outflow and, therefore, extending time that fluid for core cooling is providing by the RWST. This is accomplished by stopping the containment spray pumps and decreasing the ECCS pumps flow rates. However, if the RWST is empty, the operator is instructed to skip to Step 33 to stop pumps taking suction from the empty RWST.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier			2
	Group			1
	K/A	063.A2.02		
Level of Difficulty: 2	Importance Rating			3.1

DC Electrical Distribution: Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of ventilation during battery charging

Question # 79

Given the following conditions:

- Unit 1 100% power
- Battery 1ED1 in EQUALIZE on 125 VDC BATTERY CHARGER BC1ED1-1
- Battery Room Exhaust Fan 7 running with Battery Room Exhaust Fan 8 in standby
- RO reports Battery Room Exhaust Fan 7 tripped and Battery Room Exhaust Fan 8 did not AUTO start and will NOT manually start
- Battery Room temperature 81ºF stable

Which of the following completes the statement below?

To ensure ___(1)___ is maintained within limits, the battery should be placed in FLOAT per ___(2)___.

A. (1) room temperature
(2) ALM-0112A, 2.4, BATT RM EXH FN 8 AUTO START FAIL

B. (1) room temperature
(2) ALM-0112A, 2.3, BATT RM TRN A TEMP HI/LO-LO

C. (1) hydrogen concentration
(2) ALM-0112A, 2.4, BATT RM EXH FN 8 AUTO START FAIL

D. (1) hydrogen concentration
(2) ALM-0112A, 2.3, BATT RM TRN A TEMP HI/LO-LO

Answer: C

K/A Match: K/A match due to requiring knowledge of the concern of a loss of ventilation while charging a battery and the procedure and actions required to address this.

SRO Only: SRO Only due to requiring an assessment of plant conditions and then the selection of a section of a procedure to mitigate or recover, or with which to proceed.

Explanation:

A. Incorrect. 1st part is incorrect, but plausible because there is a TRM limit for temperature and the exhaust fan draws air through the battery room, however, the reason for placing the battery on float is for hydrogen concentration. 2nd part is correct. The procedure that directs placing the battery on float is correct, ALM-0112A, 2.4.

B. Incorrect. 1st part is incorrect, but plausible (see A). 2nd part is plausible since room temperature would direct starting a second fan, but the correct procedure is ALM-0112A, 2.4.

C. Correct. The battery should be placed in FLOAT to minimize the buildup of hydrogen gas. This is directed by ALM-0112A, Window 2.4, BATT RM EXH FN 8 AUTO START FAIL.

A. Incorrect. 1st part is correct (see C). 2nd part is incorrect, but plausible (see B).

Technical Reference(s)	ALM-0112A, 2.4	Attached w/ Revision # See Comments / Reference
	SOP-805A	
	ALM-0112A, 2.3	

Proposed references to be provided during examination: ______________________

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the DC Electrical Distribution system in accordance with procedures:
1) SOP 605A 125 VDC SWITCHGEAR AND DISTRIBUTION SYSTEMS, BATTERIES AND BATTERY CHARGERS
2) SOP 606A 24/48V & 125/250 VDC SWITCHGEAR AND DISTRIBUTION SYSTEMS, BATTERIES & BATTERY CHARGERS
3) Review loss of DC loads in accordance with DBD-EE-044 DC POWER SYSTEMS. (SYS.DC1.OB04)

Question Source: Bank # 50415
Modified Bank # ______ (Note changes or attach parent)
New ______

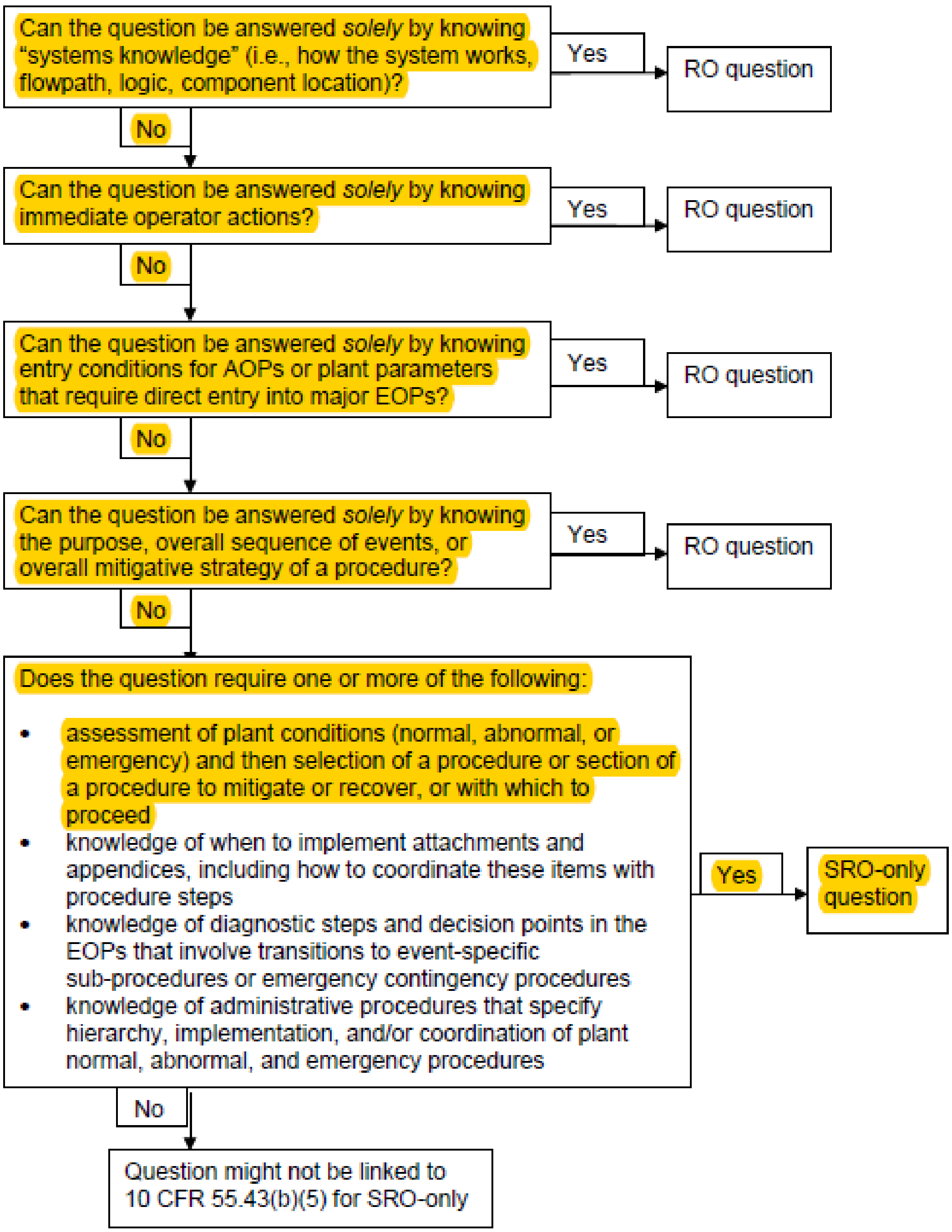
Question History: Last NRC Exam 2020 Retake (Previous)

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Comments / Reference: ALM-0112A	Revision: 6

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0112A
ALARM PROCEDURE ALARM PROCEDURE 1-ALB-11B	REVISION NO. 6	PAGE 2 OF 155

ALARM DIAGRAM

	1	2	3	4	5	6
1	UPS & DISTR RM EXH FN 5 DMPR CLOSE	UPS & DISTR RM FN 5/6 START	ERF CMPTR BATT RM EXH FN DISCH PRESS LO	BATT RM EXH FN 7 DMPR CLOSE	BATT RM EXH FN 9 DMPR CLOSE	BATT RM EXH FN 11 DMPR CLOSE
2	UPS & DISTR RM EXH FN 5 ΔP LO	UPS & DISTR RM FN 5/6 TRIP	BATT RM TRN A TEMP HI/LO-LO	BATT RM EXH FN 8 AUTO START FAIL	BATT RM EXH FN 10 AUTO START FAIL	BATT RM EXH FN 12 AUTO START FAIL
3	UPS & DISTR RM EXH FN 6 DMPR CLOSE	ANY BATT RM HTR TRIP	BATT RM TRN B TEMP HI/LO-LO	BATT RM EXH FN 8 DMPR CLOSE	BATT RM EXH FN 10 DMPR CLOSE	BATT RM EXH FN 12 DMPR CLOSE
4	UPS & DISTR RM EXH FN 6 ΔP LO	ANY BATT RM EXH FN TRIP	BATT RM TRN C TEMP LO-LO	BATT RM EXH FN 7 AUTO START FAIL	BATT RM EXH FN 9 AUTO START FAIL	BATT RM EXH FN 11 AUTO START FAIL

	7	8	9	10	11	12
1	SFGD BLDG SWGR RM ANY FN CLR FN LOC CTRL	SFGD BLDG MS & FW PIPE AREA EXH DMPR CLOSE	SFGD BLDG ELEC AREA INTK FILT ΔP HI		SFGD BLDG MS & FW PIPE AREA INTK FILT ΔP HI	ROD CTRL CAB TEMP HI
2	SFGD BLDG SWGR RM ANY FN CLR FN TRIP	SFGD BLDG MS & FW PIPE AREA HDR PRESS HI	SFGD BLDG ELEC AREA SPLY FN 15 ΔP LO	SFGD BLDG MS & FW PIPE AREA SPLY TEMP HI/LO	SFGD BLDG MS & FW PIPE AREA SPLY FN 17 ΔP LO	SFGD BLDG ANY EXH PLENUM TEMP HI
3	SFGD BLDG SWGR RM TRN A TEMP HI-HI	SFGD BLDG MS & FW PIPE AREA EXH FN 3/4 TRIP	SFGD BLDG ELEC AREA SPLY FN 16 ΔP LO	SFGD BLDG ELEC AREA INTK DMPR CLOSE	SFGD BLDG MS & FW PIPE AREA SPLY FN 18 ΔP LO	ANY RHRP/SIP/AFWP RM FN CLR FN TRIP
4	SFGD BLDG SWGR RM TRN B TEMP HI-HI	SFGD BLDG ELEC AREA EXH FN 1/2 TRIP	SFGD BLDG ELEC AREA SPLY FN 15/16 TRIP	SFGD BLDG ELEC AREA SPLY TEMP HI/LO	SFGD BLDG MS & FW PIPE AREA SPLY FN 17/18 TRIP	ANY RHRP/SIP/AFWP RM FN CLR FN LOC CTRL

	13	14	15	16	17	18
1	SIP 1 RM TEMP HI/LO	CSP 1 & 3 RM TEMP HI/LO	ANY CSP RM FN CLR FN LOC CTRL	SFGD BLDG PRESS APPROACHING ATMOS PRESS	ANY DG RM TEMP LO-LO	DG 1 RM TEMP HI
2	SIP 2 RM TEMP HI/LO	CSP 2 & 4 RM TEMP HI/LO	ANY CSP RM FN CLR FN TRIP		O/A TEMP LO	DG 2 RM TEMP HI
3	RHRP 1 RM TEMP HI/LO	AFWP 1 RM TEMP HI/LO	HP CHEM FD RM 100 TEMP HI-HI	ANY DG RM FN ΔP LO	ANY DG RM FN TRIP	ANY DG RM FN LOC CTRL
4	RHRP 2 RM TEMP HI/LO	AFWP 2 RM TEMP HI/LO	HP CHEM FD RM 100 SPLY/EXH FN TRIP		DG BLDG VENT CTRL SW MISALIGN	

Comments / Reference: ALM-0112A, 2.4	Revision: 6

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0112A
ALARM PROCEDURE ALARM PROCEDURE 1-ALB-11B	REVISION NO. 6	PAGE 51 OF 155

ANNUNCIATOR NOM./NO.: **BATT RM EXH FN 8 AUTO START FAIL** **2.4**

PROBABLE CAUSE:

Fan 1-08 NOT operating with a low ΔP on Fan 1-07
Fan 1-08 AND Fan 1-07 NOT operating

AUTOMATIC ACTIONS: None

> NOTE: Fan 1-08 will auto start with a low ΔP on Fan 1-07.
> Fan 1-07 will auto start with a low ΔP on Fan 1-08.
> Fan 1-08 AND Fan 1-07 will trip if associated intake damper fails to open.

OPERATOR ACTIONS:

1. DETERMINE cause of Fan 1-08 NOT operating.
   - 1-HS-5949, BATT RM A EXH FN 8 & EXH DMPR, NOT in AUTO
   - Exhaust Fan 1-08 tripped (Window 4.2)
   - Exhaust Fan 1-08 inlet damper failed to open (Window 3.4)

2. IF conditions permit,
   THEN
   START Fan 1-08.

3. DETERMINE cause of low ΔP on Fan 1-07.
   - Exhaust Fan 1-07 tripped (Window 4.2)
   - Exhaust Fan 1-07 inlet damper closed (Window 1.4)

4. IF Fan 1-08 will NOT start AND Fan 1-07 can be started,
   THEN
   START Fan 1-07.

5. IF ventilation can NOT be restored,
   THEN
   SECURE any equalizing charge in progress on Train A Batteries.

6. CORRECT the condition OR INITIATE a condition report per STA-421.

Comments / Reference: ALM-0112A, 2.3	Revision: 6

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0112A
ALARM PROCEDURE ALARM PROCEDURE 1-ALB-11B	REVISION NO. 6	PAGE 49 OF 155

ANNUNCIATOR NOM./NO.: **BATT RM TRN A TEMP HI/LO-LO** **2.3**

PROBABLE CAUSE:

High Temperature

Battery room cooling malfunction OR NOT in operation

Low Temperature

Battery room heater malfunction OR NOT in operation

AUTOMATIC ACTIONS: None

OPERATOR ACTIONS:

1. DISPATCH an operator to determine if temperature is high OR low.

NOTE: The TR limit for UPS/Battery Rooms is 104 °F.

2. A. IF temperature is high,
   THEN
   PERFORM the following:
   - ENSURE UPS AND Distribution Room Cooling operating per SOP-803 AND Battery Room Ventilation operating per SOP-805A.

   B. MONITOR room temperature to ensure it is returning to normal.

NOTE: IF power is lost to the heater, THEN a heater trip alarm will not actuate. (Window 3.2)

3. IF temperature is low,
   THEN
   PERFORM the following:
   A. ENSURE heater control switch is in automatic.
   B. MONITOR room temperature to ensure it is returning to normal.

4. REFER to TS 3.8.6.

5. CORRECT the condition OR INITIATE a condition report per STA-421.

Comments / Reference: SOP-805A	Revision: 11

CPNPP SYSTEM OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. SOP-805A
BATTERY ROOM VENTILATION SYSTEM	REVISION NO. 11 CONTINUOUS USE	PAGE 3 OF 13

1.0 APPLICABILITY

This procedure describes the steps for operation of the Battery Room Ventilation System.

2.0 PREREQUISITES

2.1 Startup

2.1.1 Startup of the Battery Room Exhaust Fans

- ☐ The control switch lineup per Section 1.0 of Attachment 1 is complete.
- ☐ The electrical lineup per Section 1.0 of Attachment 2 is complete.
- ☐ I&C has completed the instrument lineup per INC-2100A. |
- ☐ The instrument air valve lineup per Attachment 3 is complete.

2.1.2 Startup of the Battery Charger Distribution Panel Exhaust Fans

- ☐ The control switch lineup per Section 2.0 of Attachment 1 is complete.
- ☐ The electrical lineup per Section 2.0 of Attachment 2 is complete.
- ☐ I&C has completed the instrument lineup per INC-2100A. |
- ☐ The instrument air valve lineup per Attachment 3 is complete.

3.0 PRECAUTIONS

- One exhaust fan shall be in service and operating in each battery room at all times to prevent excessive hydrogen concentrations in the battery rooms.

4.0 LIMITATIONS AND NOTES

4.1 Limitations

- Battery Room temperatures shall not exceed 104°F for more than 8 hours per TR 13.7.36.
- Battery Room temperatures shall not exceed 113°F at any time per TR 13.7.36.

4.2 Notes

- Battery Room Exhaust Fans 1-08 and 1-10 may be aligned to Outage Power. The steps to align these fans to and from Outage Power are contained in SOP-613A.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier			2
	Group			1
	K/A	103.A2.05		
Level of Difficulty: 3	Importance Rating			3.9

Containment: Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency containment entry

Question # 80

Unit 1 is operating in Mode 1.

- A confirmed report of sabotage has occurred rendering the PAL INNER door INOPERABLE due to damaged hinges inside containment
- The PAL OUTER door remains OPERABLE
- No further sabotage has occurred
- The Security threat has ended
- An Emergency Containment Entry has been approved to repair the PAL INNER door

1) Per STA-620, CONTAINMENT ENTRY, which of the following is required to be completed PRIOR to EMERGENCY Containment Entry?

2) Per Technical Specification Bases 3.6.2, Containment Air Locks, the PREFERRED method of Containment Entry to repair the INOPERABLE PAL INNER door is to enter ________________.

A. (1) RWP issuance
(2) the PAL from the OUTER door

B. (1) RWP issuance
(2) through the EAL and access the PAL from inside containment

C. (1) Containment briefing by Shift Manager
(2) the PAL from the OUTER door

D. (1) Containment briefing by Shift Manager
(2) through the EAL and access the PAL from inside containment

Answer: D

K/A Match: K/A match due to requiring knowledge of procedures used for emergency containment entry requirements.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis, assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed, as well as requiring knowledge of Tech Spec bases.

Explanation:

A. Incorrect. First part is incorrect, but plausible since STA-620 allows for the RWP to be completed after entry has occurred in a timely manner consistent with related emergency response activities. Second part is incorrect, but plausible as TS allows entry and exit of affected Airlock doors to perform repairs, however, an SRO must understand that TSB 3.6.2 Bases states this is not the preferred method to access the affected component.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A)

D. Correct. First part is correct. STA-620 states that a Containment briefing should be held by the Shift Manager. Each RWO entering into Containment should ensure that all questions on the checklist have been reviewed and that all entry members understand their responsibilities. Second part is correct. TSB 3.6.2 explicitly states that it is preferred the air lock be accessed from inside primary containment by entering through another operable airlock door.

Technical Reference(s)	TS 3.6.2	Attached w/ Revision # See Comments / Reference
	TSB 3.6.2	
	STA-620	

Proposed references to be provided during examination: ____________________

Learning Objective: **APPLY** the administrative requirements of the Containment system including Technical Specifications, TRM and ODCM: 3) Containment Air Locks 3.6.2 (SYS.CY1.OB05)

Question Source: Bank # ________
Modified Bank # 82582 (Note changes or attach parent)
New ________

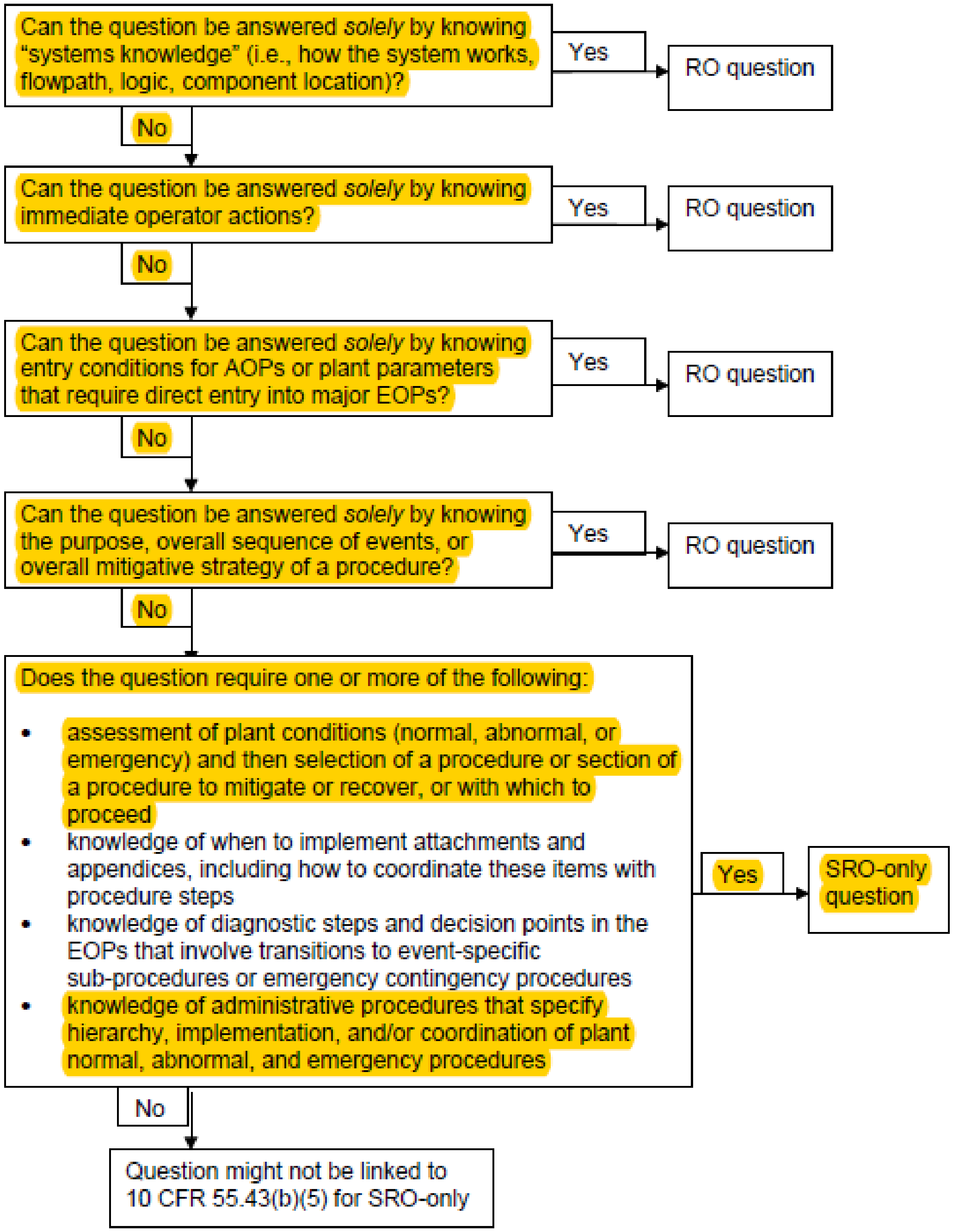
Question History: Last NRC Exam 2018 NRC Exam Q89

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

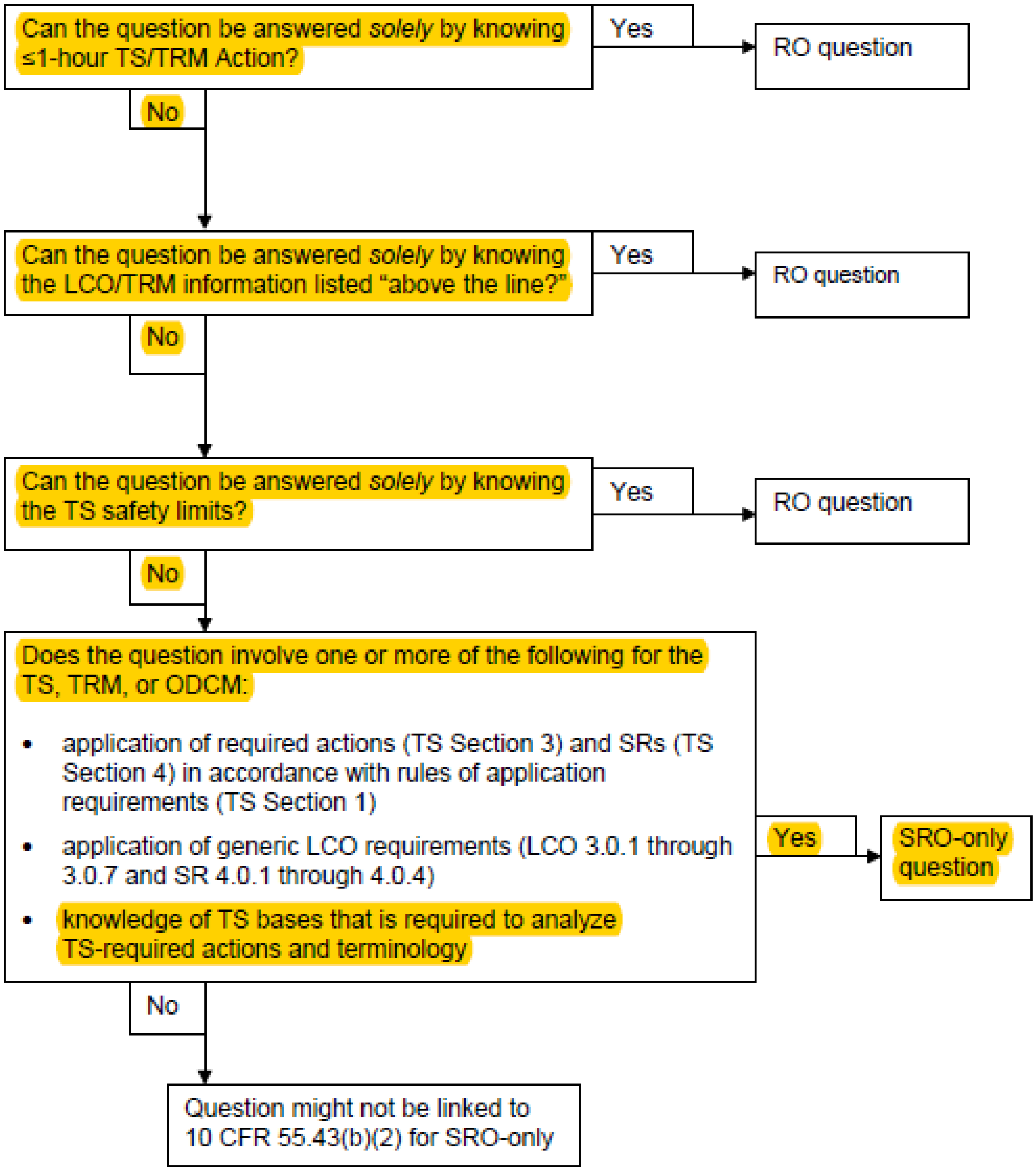
10 CFR Part 55 Content: 55.41 ________
55.43 5/2

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:



Comments / Reference: 2017 NRC Q89	Revision:

Examination Outline Cross-Reference

103 (SF5 CNT) Containment

Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.05 Emergency containment entry

Level	SRO
Tier #	2
Group #	1
K/A #	A2.05
Rating	3.9
QREV	6

Question 89

Unit 1 is operating in Mode 1.

- A confirmed report of sabotage has occurred rendering the Personnel Air Lock INNER door INOPERABLE.
- The Personnel Air Lock OUTER door remains OPERABLE.
- No further sabotage has occurred. The Security threat has ended.
- An Emergency Containment Entry has been approved

(1) Per STA-620, CONTAINMENT ENTRY, which of the following is required to be completed PRIOR to emergency containment entry?

(2) Per Technical Specification Bases 3.6.2, Containment Air Locks, Containment Entry via the Personnel Air Lock outer door _______________ to perform repairs on the INNER door.

A. (1) Containment briefing by Shift Manager
(2) IS NOT permitted

B. (1) RWP issuance
(2) IS permitted

C. (1) Containment briefing by Shift Manager
(2) IS permitted

D. (1) RWP issuance
(2) IS NOT permitted

Answer: C

Explanation:

Comments / Reference: 2017 NRC Q89	Revision:

NOTE:	During emergencies, an RWP need NOT be completed or issued prior to entry. The required paperwork is expected to be completed in a timely manner consistent with related emergency response activities.

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact

(continued)

ACTIONS (continued)

(during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A is wrong because TSB 3.6.2 explicitly makes an allowance to enter an affected air lock via the operable outer door.

B is wrong because STA-620 allows for the RWP to be completed after entry has occurred in a timely manner consistent with related emergency response activities. TSB 3.6.2 does allow for the outer door of the affected air lock to be briefly opened then closed if necessary to access the affected inner door.

C is correct. STA-620 states that “A Containment briefing should be held by the Shift Manager. Each RWO entering into Containment should ensure that all questions on the checklist have been reviewed and that all entry members understand their responsibilities.” There is no NOTE stating that this need not occur. Also, TSB 3.6.2 explicitly makes an allowance to enter an affected air lock via the operable outer door.

D is wrong because STA-620 allows for the RWP to be completed after entry has occurred in a timely manner consistent with related emergency response activities.

Technical References:
Technical Specification Bases 3.6.2, Containment Air Locks
STA-620, CONTAINMENT ENTRY

References to be provided to applicants during exam: None.

Learning Objective: APPLY the administrative requirements of the Containment

Comments / Reference: 2017 NRC Q89	Revision:

system including Technical Specifications, TRM and ODCM: 3) Containment Air Locks 3.6.2 (LO21.SYS.CY1.OB05)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41 55.43(b)(5) / (b)(2)	

Definitions:

Emergency Entry - Entry into Containment to prevent actual or potential personnel injury or damage to equipment and facilities or as requested by the Shift Manager to address plant operational concerns.

General Entry - Includes but are not limited to Operator tours, inspection and system manipulations, RP or Safety surveys, or initial entries to evaluate equipment performance.

4.12 Scheduled Entry - Entry into Containment for the purpose of conducting routine tasks necessary to support unit operations.

4.13 Unplanned Entry - Entry into Containment that has NOT been included in the POD/Outage schedule.

Comments / Reference: TS 3.6.2	Revision: 150

3.6 CONTAINMENT SYSTEM

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

--NOTES--

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.

Comments / Reference: TS Bases 3.6.2	Revision: 74

BASES

APPLICABLE SAFETY ANALYSES (continued)

containment leakage rate at the calculated peak containment internal pressure P_a = 48.3 psig, following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact

(continued)

Comments / Reference: STA-620	Revision: 15

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-620
CONTAINMENT ENTRY	REVISION NO. 15	PAGE 26 OF 38
	INFORMATION USE	

6.3 Emergency Containment Entry Requirements

NOTE 1: Personnel are not allowed to enter Containment when pressurized for an Integrated Leakage Rate Test (ILRT).

NOTE 2: Some NPRCS are too large to adequately sample at the access point (e.g., Containment and Main Condenser Shells). In these cases, after the atmosphere is sampled near the entrance and entry is made, personnel should carry an instrument with them to sample the atmosphere in the area which they are working, prior to beginning the work activity.

6.3.1 A minimum of two people are required to enter Containment, one of which should be a Radiation Protection Technician.

6.3.2 All personnel entering Containment should wear the necessary protective clothing recommended by Radiation Protection. Personnel should also wear self contained breathing apparatus if airborne concentration levels are unknown or subject to change due to worsening plant conditions OR IF Nitrogen or other gas leaks are suspected. IF Containment has been sampled and analyzed for Class C Atmosphere as defined by STI-211.01, THEN Respiratory Protection may NOT be required if entry is made for a non-operational type emergency (e.g., medical injury).

6.3.3 A Containment briefing should be held by the Shift Manager. Each RWO entering into Containment should ensure that all questions on the checklist have been reviewed and that all entry members understand their responsibilities.

6.3.4 The Shift Manager shall approve all entries.

6.3.5 IF the Containment PIG (1/2RE5502, 1/2RE5503 and 1/2RE5566) is NOT operable, the Shift Manager should notify Radiation Protection to obtain Containment air samples in accordance with Section 6.2.5 of this procedure.

6.3.6 The Shift Manager should also notify Security prior to an emergency entry into Containment through the Emergency Airlock.

Comments / Reference: STA-620	Revision: 15

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-620
CONTAINMENT ENTRY	REVISION NO. 15	PAGE 27 OF 38
	INFORMATION USE	

<u>NOTE</u>:	During emergencies, an RWP need <u>NOT</u> be completed or issued prior to entry. The required paperwork is expected to be completed in a timely manner consistent with related emergency response activities.

6.3.7 Containment entry should be documented by filling out an RWP. The RWP should be filled out by Radiation Protection.

6.3.8 A back-up team should be ready for Containment entry to provide assistance, if required.

-

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier			2
	Group			2
	K/A	034.K4.03		
Level of Difficulty: 4	Importance Rating			3.3

Fuel-Handling Equipment: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Overload protection	
Question # 81	

In accordance with TR LCO 13.9.33, Refueling Machine bases, the Refueling Machine Auxiliary Monorail Hoist is used for __(1)__ and has a load indicator used to prevent lifting of loads in excess of __(2)__.

A. (1) latching, unlatching and movement of control rod drive shafts
(2) 2800 pounds

B. (1) lifting a fuel assembly with a control rod inserted
(2) 2800 pounds

C. (1) latching, unlatching and movement of control rod drive shafts
(2) 600 pounds

D. (1) lifting a fuel assembly with a control rod inserted
(2) 600 pounds

Answer: C	

K/A Match: K/A match due to requiring knowledge of the overload protection associated with refueling cranes.

SRO Only: SRO Only due to being addressed by 10 CFR 55.43(b)(7), assessment of surveillance requirements for the refueling mode.

Explanation:

A. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

B. Incorrect. First part is incorrect, but plausible since the main hoist is used for lifting a fuel assembly with a control rod inserted. Second part is incorrect, but plausible since 2800 pounds is the overload limit for the main hoist.

C. Correct. First part is correct. The auxiliary hoist is typically used for latching, unlatching and movement of control rod drive shafts. Second part is correct. 600 pounds is the cutoff limit for the auxiliary hoist.

D. Incorrect. First part is incorrect, but plausible (see B). Second part is correct (see C).

Technical Reference(s)	TR/TRB 13.9.33	Attached w/ Revision # See Comments / Reference
	RFO-102	

Proposed references to be provided during examination: ______

Learning Objective: Given access to Technical Specifications and a refueling activity, **APPLY** the appropriate specification. (OPD1.G16.OB04)

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

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 7

Comments / Reference: TR 13.9.33	Revision: 77

13.9 REFUELING OPERATIONS

TR 13.9.33 Refueling Machine

TR LCO 13.9.33 The refueling machine main hoist and auxiliary monorail hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine main hoist used for movement of fuel assemblies having:

1. A minimum capacity of 2850 pounds, and

2. An overload cutoff limit less than or equal to 2800 pounds.

b. The auxiliary monorail hoist used for latching, unlatching and movement of control rod drive shafts having:

1. A minimum capacity of 610 pounds, and

2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

- NOTE -

Special evolutions may require the use of hoists and load indicators in addition to the auxiliary monorail hoist.

APPLICABILITY: During movement of fuel assemblies and/or latching, unlatching or movement of control rod drive shafts within the reactor vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements for hoist OPERABILITY not satisfied.	A.1 Suspend use of any inoperable hoist from operations involving the movement of fuel assemblies and/or latching, unlatching or movement of control rod drive shafts within the reactor vessel.	Immediately

Comments / Reference: TRB 13.9.33	Revision: 77

B 13.9 REFUELING OPERATIONS

TRB 13.9.33 Refueling Machine

BASES

The OPERABILITY requirements for the refueling machine main hoist and auxiliary monorail hoist ensure that: (1) the main hoist will be used for movement of fuel assemblies, (2) the auxiliary monorail hoist will typically be used for latching, unlatching and movement of control rod drive shafts, (3) the main hoist has sufficient load capacity to lift a fuel assembly (with control rods), (4) the auxiliary monorail hoist has sufficient load capacity to latch, unlatch and move the control rod drive shafts, and (5) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. The LCO is modified by a note which provides for the use for additional hoists and load indicators for special planned evolutions such as a bent control rod drive shaft.

Comments / Reference: RFO-102	Revision: 13

CPNPP STATION REFUELING MANUAL	UNIT COMMON	PROCEDURE NO. RFO-102
REFUELING OPERATION	REVISION NO. 13 CONTINUOUS USE	PAGE 11 OF 95

4.1.5 Direct communications shall be maintained between the Control Room and personnel at the Refueling Station during CORE ALTERATIONS (TR 13.9.32). The primary means of communications will be the Wireless Intercom System and the Intraplant Radio System will be the backup.

4.1.6 During movement of fuel assemblies and/or latching, unlatching or movement of control rod drive shafts within the reactor vessel, the refueling machine main hoist and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with (TR 13.9.33):

A. The refueling machine main hoist used for latching, unlatching and movement of control rod drive shafts having:
   1) A minimum capacity of 2850 pounds
   2) An overload cutoff limit less than or equal to 2800 pounds

B. The auxiliary monorail hoist used for latching and unlatching drive rods having:
   1) A minimum capacity of 610 pounds
   2) A load indicator which shall be used to prevent the lifting of loads in excess of 600 pounds

[C] 4.1.7 Loads in excess of 2150 pounds shall be prohibited from travel over fuel assemblies in the storage pool (TR 13.9.34). [10005]

4.1.8 At least one Residual Heat Removal (RHR) loop shall be OPERABLE and in operation when the water level above the top of the Reactor Vessel Flange $\geq$23 feet (TS 3.9.5).

The basis for requiring one RHR loop $\geq$23 feet is to prevent the water above the core from boiling in the event of a loss of RHR cooling. However, it has been determined that when upper internals are installed in the reactor vessel, there is insufficient communication with the water above the core for adequate decay heat removal by natural circulation if decay heat is >7.6 MWth (as determined by Core Performance/Safety Analysis). Therefore, the availability of additional cooling equipment (e.g., the other RHR train) is necessary above this decay heat level to ensure redundant cooling capability is maintained with the upper internals installed.

4.1.9 Two Residual Heat Removal (RHR) loops shall be OPERABLE and at least one RHR loop shall be in operation when the water level above the top of the Reactor Vessel Flange is less than 23 feet with irradiated fuel in the Reactor Vessel (TS 3.9.6).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier			2
	Group			2
	K/A	056.A2.04		
Level of Difficulty: 3	Importance Rating			2.8

Condensate: Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of condensate pumps

Question # 82

Given the following conditions:

- Unit 1 80% power
- B Condensate Pump trips
- ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, entered
- During the runback:
  - MFP suction pressure lowers to 210 psig for 35 seconds and recovers to 350 psig and stable
  - 1-ALB-6D, window 2.7 – ANY CONTROL ROD BANK AT LO-LO LIMIT, alarms

Per ABN-302, the US should ensure __(1)__.

In accordance with TS 3.1.6, Control Bank Insertion Limits, rods must be restored above the rod insertion limit within __(2)__.

A. (1) Main Feedwater Pump A has tripped
(2) 2 hours

B. (1) Main Feedwater Pump A has tripped
(2) 4 hours

C. (1) 1-PV-2286, LP FW HTR BYP VLV has opened
(2) 2 hours

D. (1) 1-PV-2286, LP FW HTR BYP VLV has opened
(2) 4 hours

Answer: C

K/A Match: K/A match due to requiring knowledge of the effect of a loss of a Condensate Pump will have on the condensate system.

SRO Only: SRO Only due to 10 CFR 55.43(b)(2) Analysis requiring application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1) and 10 CFR 55.43(b)(5) .

Explanation:

A. Incorrect. First part is incorrect, but plausible because MFP A would have tripped if Feedwater pressure had lowered to 190 psig for 30 seconds. Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible (see A) . Second part is incorrect, but plausible since many more TS actions are 4 hour requirements than 2 hour requirements.

C. Correct. First part is correct. When Feedwater pressure falls below 250 psig, ABN-302, Section 3.0, Step 2.b RNO requires the US to ensure PV-2286 is open. Second part is correct. Rods are to be restored above RIL within 2 hours.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-302	Attached w/ Revision # See Comments / Reference
	TS 3.1.6	

Proposed references to be provided during examination: ____________________

Learning Objective: **ANALYZE** the response to a Condensate Pump Trip in accordance with ABN-302, Feedwater, Condensate, Heater Drains System Malfunction**.** (ABN.302.OB02)

Question Source: Bank # ____________

Modified Bank # 41844 (Note changes or attach parent)

New ____________

Question History: Last NRC Exam ____________

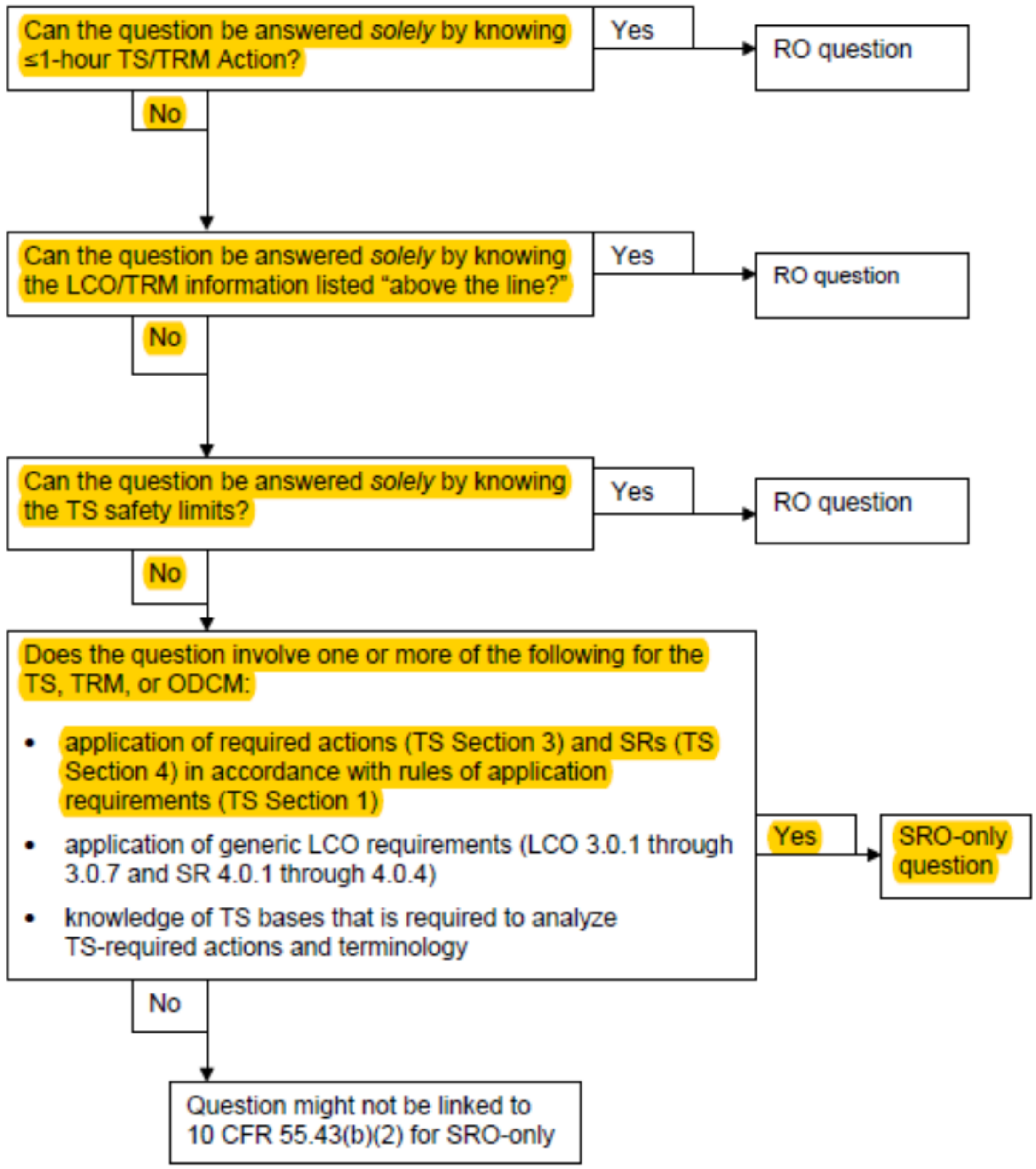
Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____________

55.43 2/5

Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:



Comments / Reference: Bank 41844	Revision:

248 **41844**

- Unit 1 is operating at 90% power with Rod Control in MANUAL due to troubleshooting the automatic control circuitry
- B MFP trips
- The Reactor Operator begins inserting control rods in MANUAL
- ALB-06D, 2.7, ANY CONTROL ROD BANK AT LO-LO LIMIT, alarms
- T-ave is approximately 8ºF greater than T-ref

Which of the following actions is to be taken?

A. Stop inserting rods and initiate emergency boration

B. Continue inserting rods until Tave is within 1ºF of Tref

C. Continue inserting rods until Tave is within 5ºF of Tref

D. Stop inserting rods and start withdrawing rods to restore SDM

Answer: C

Answer Explanation
A Plausible since rods are below the rod insertion limit with temperature high, requiring further rod insertion, but this is acceptable provided rods are restored above the rod insertion limit within TS allotted times and emergency boration would allow rods to be withdrawn now to restore SDM B Plausible since it is desirable to have T-ave matched to T-ref, but driving control rods in this far will result in an overshoot low on temperature. C Correct - When driving rods manually following a runback or other large load reduction, good technique would be a continuous insertion until temperature is approximately 5 degrees high. This ensures a timely reduction in steam flow (via steam dumps) while preventing an overshoot low on temperature.(Ops Guideline #3 ATT6) D. Incorrect - Plausible as rods below the rod insertion limit does challenge SDM but rods are not withdrawn to restore this until temp is under control first

Comments / Reference: Bank 41844	Revision:

Question 248 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	41844
User-Defined ID:	LORT
Cross Reference Number:	
Topic:	Unit 1 is operating at 90% power with Rod Control in MANUAL due to troubleshooting the automatic c
K/A:	054.AA2.10
Question Reference:	OPGD-03
SRO:	
Comments:	

Comments / Reference: ABN-302	Revision: 14

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-302
FEEDWATER, CONDENSATE, HEATER DRAIN SYSTEM MALFUNCTION	REVISION NO. 14	PAGE 13 OF 80

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2 ☐ b. Main Feedwater pump suction pressure - GREATER THAN 250 PSIG • u-PI-2295, FWP A SUCT PRESS • u-PI-2297, FWP B SUCT PRESS	b. Perform the following: • Ensure u-HS-2286, LP FW HTR BYP VLV - OPEN • Verify both HDP - RUNNING

NOTE: Differential pressure between feedwater and steamline may decrease following a Turbine Runback. The following computer points may aid the operator:

- U5002A FW-MS HDR DP
- U5003A DELTA PROGRAM-ACTUAL DP
- P5446A FW STM FLOW SETPOINT

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ c. Feedwater header pressure - MAINTAINED GREATER THAN MAIN STEAM HEADER PRESSURE.	c. Adjust the Main Feed Pump master or individual speed controller to maintain adequate differential pressure.
☐ d. Main Feedwater - ALIGNED	d. Ensure Main Feedwater aligned
☐ 3 Verify SG water level - STABLE OR TRENDING TO NORMAL OPERATING RANGE.	Perform the following: a. Manually control feedwater flow to Steam Generator to maintain level between 60% and 75%. b. IF Reactor Power is above the capability of available feed flow, THEN reduce power using rod control, turbine control or boration until steam generator levels can be maintained while continuing this procedure.

Comments / Reference: ABN-302	Revision: 14

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-302
FEEDWATER, CONDENSATE, HEATER DRAIN SYSTEM MALFUNCTION	REVISION NO. 14	PAGE 14 OF 80

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Control Rod insertion should be allowed to continue even if ΔI is outside the band. Continued rod insertion is required to return Tave to Tref as soon as possible so that steam demand is reduced.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 4 Verify Tave - TRENDING TO TREF. • u-TI-412A, AVE TAVE - TREF DEV	Perform the following: a. Ensure rods stepping in. b. Operate steam dumps to maintain stable plant conditions.

CAUTION: Reactor power must be established at a value within the capability of available feedwater. Auxiliary feedwater pumps can supply approximately 6% reactor power.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5 Stabilize Reactor using one or more of the following: • Control rods • Steam dumps • Boration • Turbine Load	
☐ 6 Verify SG FW FLO CTRL Valves - AUTO • u-FK-510, SG 1 FW FLO CTRL • u-FK-520, SG 2 FW FLO CTRL • u-FK-530, SG 3 FW FLO CTRL • u-FK-540, SG 4 FW FLO CTRL	Place SG FW FLO CTRL Valves in AUTO AND monitor steam generator levels to verify proper response of control valves.

Section 3.3

Comments / Reference: ABN-302	Revision: 14

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-302
FEEDWATER, CONDENSATE, HEATER DRAIN SYSTEM MALFUNCTION	REVISION NO. 14	PAGE 15 OF 80

3.3 Operator Actions

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	7 Verify the following:	
	☐ a. Rods - ABOVE ROD INSERTION LIMIT	a. Verify SDM or initiate boration to restore SDM within 1 hour and restore Rods above insertion limits within 2 hours per TS 3.1.6.

NOTE: • The PPC plot for AFD Limits utilizes NIS power. Other power indications may differ significantly following large transient conditions. For the step below consider using the best indication of actual reactor power to evaluate operating limits.

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	☐ b. ΔFlux - (AFD) WITHIN LIMITS	b. Borate as necessary to restore ΔFlux to within limits or reduce power within 30 minutes while continuing with this procedure. Refer to TS 3.2.3.
☐	8 WHEN steam dumps have closed, THEN reset steam dump arming signal (C-7 interlock) • 43/u-SD, STM DMP MODE SELECT	
☐	9 Verify u-HS-2286, LP FW HTR BYP VLV - CLOSED.	Perform Section 7.0 of this procedure while continuing this section.
☐	10 Notify QSE Generation Controller and update GAPS to "Create Current Condition" for the down power.	
☐	11 Initiate equipment repairs per STA-606.	
	12 Check Chemistry Sampling Requirement:	
	☐ a. SG ARVS - REMAINED CLOSED AND TDAFW Pump - REMAINED STOPPED	a. Notify Chemistry that a release has occurred and for Chemistry to determine if a release permit is required per STA-603.
[C]	☐ b. Verify Reactor Power change - LESS THAN 15% RTP WITHIN ONE HOUR.	b. Notify Chemistry to perform RCS Isotopic analysis for iodine between 2 and 6 hours after power change.

Section 3.3

Comments / Reference: TS 3.1.6	Revision: 156

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

-----NOTE-----
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank insertion limits not met.	A.1.1 Verify SDM to be within the limits provided in the COLR. OR A.1.2 Initiate boration to restore SDM to within limit. AND A.2 Restore control bank(s) to within limits.	1 hour 1 hour 2 hours

-

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier			2
	Group			2
	K/A	086.G.2.1.20		
Level of Difficulty: 2	Importance Rating			4.6

Fire Protection: Ability to interpret and execute procedure steps.

Question # 83

Given the following conditions:

- Due to a fire, the Shift Manager has directed the Control Room evacuated
- A Unit 1 Natural Circulation cooldown is to be performed

The cooldown should be performed using __(1)__ per the guidance contained in __(2)__.

A. (1) Steam Dumps
(2) ABN-803A, Response to a Fire in the Control Room or Cable Spreading Room

B. (1) Steam Dumps
(2) EOS-0.2A, Natural Circulation Cooldown

C. (1) SG ARVs
(2) ABN-803A, Response to a Fire in the Control Room or Cable Spreading Room

D. (1) SG ARVs
(2) EOS-0.2A, Natural Circulation Cooldown

Answer: C

K/A Match: K/A match due to requiring knowledge of the content of procedures related to fires in the Control Room or cable spreading room.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) analysis involves both (1) assessing plant conditions (normal, abnormal, or emergency) and then (2) selecting a procedure or section of a procedure to mitigate or recover, or with which to proceed. One area of SRO-level knowledge (with respect to selecting a procedure) is knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

Explanation:

A. Incorrect. First part is incorrect, but plausible since a cooldown via the steam dumps is normally preferred over using the ARVs. Second part is correct. ABN-803A contains a note that once the decision to evacuate the control room is made, ERGs are no longer applicable.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible because if control room evacuation was not performed, the natural circulation cooldown would be performed per the guidance of EOS-0.2A.

C. Correct. First part is correct. Any plant cooldown performed from outside the control room is performed using the SG ARVs. Second part is correct (see A).

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-803A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **ANALYZE** the response to a Fire in the Electrical or Control Building in accordance with ABN-803A, Response to a Fire in the Control Room or Cable Spreading Room . (ABN.803.OB01)

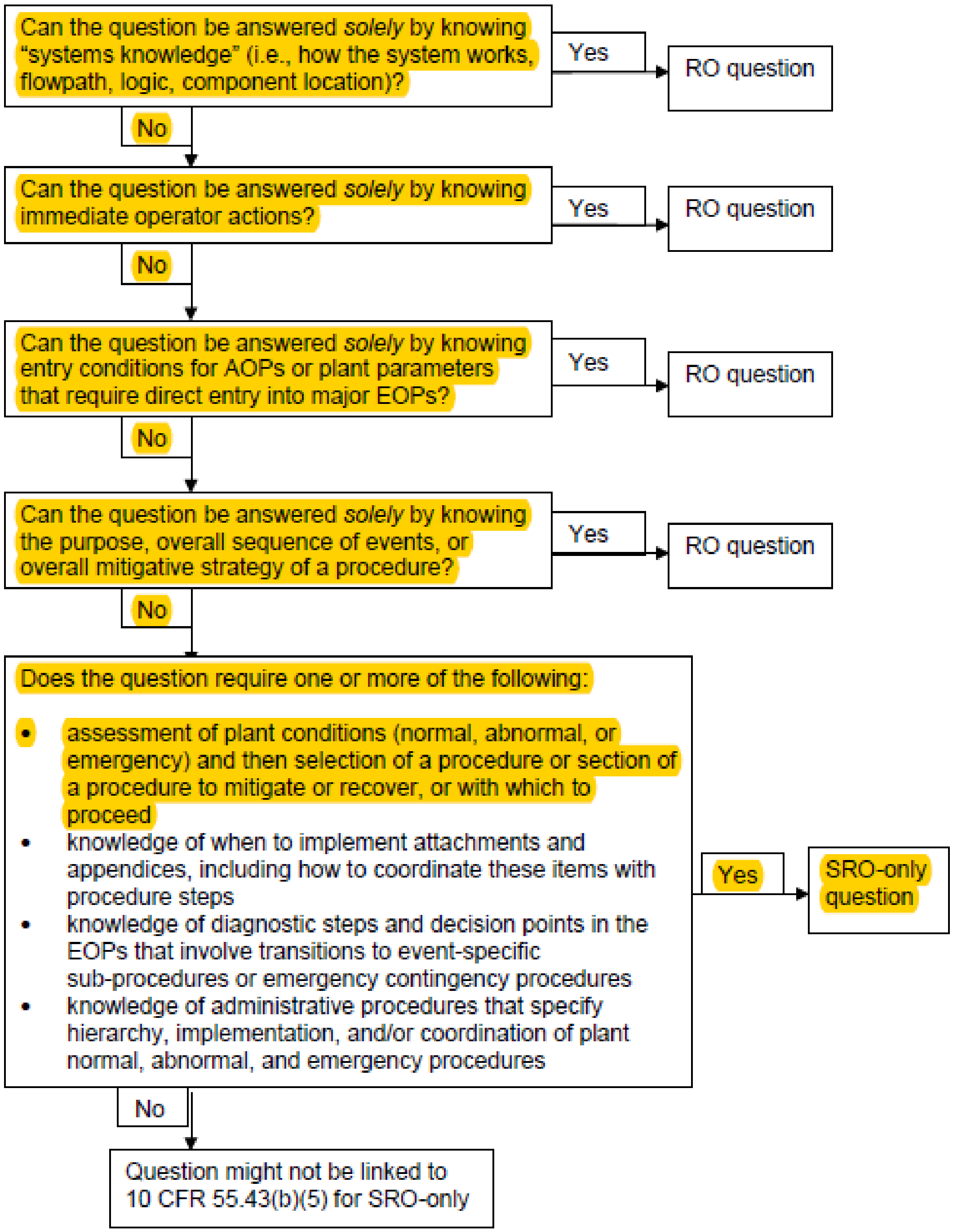
Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: ABN-803A	Revision: 16

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-803A
RESPONSE TO A FIRE IN THE CONTROL ROOM OR CABLE SPREADING ROOM	REVISION NO. 16	PAGE 6 OF 68

2.3 Operator Actions

CAUTION:
- Use of this procedure may result in abnormal configuration. Management review of steps performed is necessary to ensure configuration tracking and restoration.
- The Control Room should be evacuated for any of the following conditions:
  - SCBAs should not to be used to stay in control room. If needed, then an evacuation should be performed.
  - Fire induced failure of any of the following Train A systems. AFW, CVCS, CCW, SSW or electrical (AC or DC).
  - Fire induced component/system failure that could compromise SG inventory control, RCS inventory or pressure control. PORV, RWST, BAT
    (See Attachment 18 for additional information, as time allows)

NOTE:
- The decision to evacuate the Control Room shall be made by the Shift Manager, based upon the ability to safely control the plant from the Control Room.
- Once the decision to leave the Control Room has been made, Emergency Response Guidelines (ERGs) DO NOT apply. ERGs may be referred to, but should not be used for Reactor Trip Response.
- A fire in this area will require simultaneous shutdown of both Unit 1 and Unit 2. In this event Unit 1 will control manipulation of system(s)/equipment common to both units unless otherwise directed by the Shift Manager.
- Evaluate the necessity of donning SCBAs, if not already worn, prior to leaving Control Room.
- The symbol [R] has been located throughout this procedure where real or potential radiation hazards are positively identified. This identification technique should not preclude the worker from following good radiation work practices throughout this procedure to ensure his/her occupational exposure is maintained As Low As Reasonably Achievable (ALARA).
- Three two-way radios are maintained at the Remote Shutdown Panel for performance of this procedure.
- This procedure is written assuming minimum staffing requirements. However, should additional personnel be available, consideration should be given to supporting timely completion of Attachments 2, 3, and 4 followed by shutting down secondary plant equipment when conditions permit. IPO-009A may be referred to for general guidance on securing the secondary plant.

☐ 1. REFER to appropriate Fire Preplan Instruction.

Section 2.3

Comments / Reference: ABN-803A	Revision: 16

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-803A
RESPONSE TO A FIRE IN THE CONTROL ROOM OR CABLE SPREADING ROOM	REVISION NO. 16	PAGE 12 OF 68

2.3 Operator Actions

CAUTION: IF either of following conditions exist AND SSPS is subsequently reset, THEN SI Actuation will occur if EITHER of the following occurs:

- Low Main Steam Line pressure at 605 psig
- Low Pressurizer pressure at 1820 psig

NOTE: The RSP controllers for Steam Generator Atmospheric Relief valves may operate differently than Control Room controllers. An Operator may be sent to verify relief valve response.

☐ 16. a. THROTTLE the Steam Generator Atmospheric Relief valves to establish 25°F/hr cooldown rate in the RCS.

- 1-PI-514B, SG 1 PRESS
  1-HC-2325, SG 1 ATMOS RLF VLV CTRL
- 1-PI-524B, SG 2 PRESS
  1-HC-2326, SG 2 ATMOS RLF VLV CTRL

NOTE: Steam tables may be used to determine SG pressures based on loop temperatures.

☐ b. IF any SG exceeds 400 psig above other SGs,
THEN
RELEASE main steam by manually opening affected Steam Generator Atmospheric Relief valves.

- 1-HC-2325, SG 1 ATMOS RLF VLV CTRL
- 1-HC-2326, SG 2 ATMOS RLF VLV CTRL
- 1-HC-2327, SG 3 ATMOS RLF VLV CTRL
- 1-HC-2328, SG 4 ATMOS RLF VLV CTRL

Section 2.3

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			1
	Group			1
	K/A	000007.EA2.04		
Level of Difficulty: 3	Importance Rating			4.6

Reactor Trip, Stabilization, Recovery: Ability to determine or interpret the following as they apply to a reactor trip: If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP

Question # 84

Given the following conditions:

- An ATWT event occurred on Unit 1
- FRS-0.1A, Response to Nuclear Power Generation/ATWT, in progress and the Reactor still NOT tripped
- Boration CANNOT be initiated because of blockage in the Boration flowpaths
- All Power Range Channels indicate 6%
- Startup rate is zero on BOTH Intermediate Range Channels
- Average CET temperature 580°F slowly lowering

Which of the following describes the operator actions under these conditions and the reason for taking these actions?

A. Exit FRS-0.1A because the zero startup rate on the Intermediate Range channels indicate actions taken in FRS-0.1A have been successful.

B. Remain in FRS-0.1A, and allow the RCS to heat up, adding negative reactivity, while continuing efforts to establish Emergency Boration.

C. Exit FRS-0.1A because the lowering CET temperatures indicate actions taken in FRS-0.1A have been successful.

D. Remain in FRS-0.1A, and lower RCS temperature, establishing DNBR margin, while continuing efforts to establish Emergency Boration.

Answer: B

K/A Match: K/A match due to requiring knowledge of the actions taken in response to an ATWT condition.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose and knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures.

Explanation:

A. Incorrect. Plausible since a zero startup rate would not meet the entry conditions for FRS-0.1A if power ranges were below 5% so it could be considered that exit conditions are met.

B. Correct. Exit conditions have not been met in FRS-0.1A, as power range is still above 5%. While attempting establish boration, the RCS is allowed to heat up to add negative reactivity.

C. Incorrect. Plausible since a there is a transition from FRS-0.1 based on CET temperatures, though it is to SAMGs on high temperature.

D. Incorrect. First part is incorrect, but plausible as conditions to exit FRS have not been met. Second part is incorrect, but plausible since it is a true statement but has no bearing on the performance of FRS-0.1A.

Technical Reference(s)	FRS-0.1A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRS-0.1A/B, Response to Nuclear Generation/ATWT in accordance with FRS-0.1. (ERG.FS1.OB04)

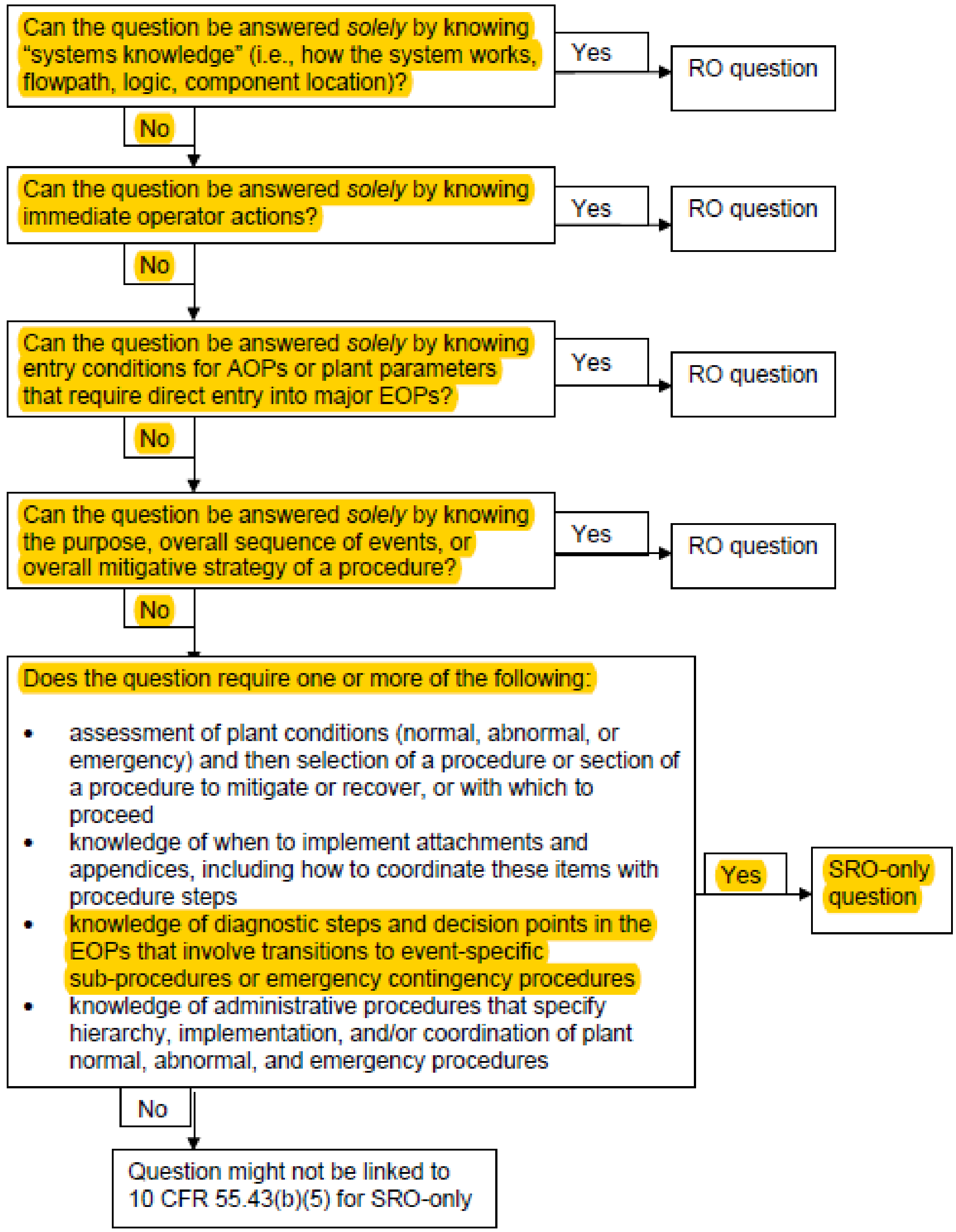
Question Source: Bank # ______
Modified Bank # 33641 (Note changes or attach parent)
New ______

Question History: Last NRC Exam LC20 (Original Modified)

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: Bank 33641	Revision:

Given the following conditions:

- An Anticipated Transient Without Trip (ATWT) event is in progress on Unit 1.
- FRS-0.1A, Response to Nuclear Power Generation/ATWT is in progress and the Reactor is still NOT tripped.
- Boration CANNOT be initiated because of blockage in the Boration flowpaths.
- All Power Range Channels indicate 6%.
- Startup rate is zero on both Intermediate Range Channels.
- Average Core Exit Thermocouple temperature is 580°F and slowly lowering.

Which of the following describes:

1) The operator actions under these conditions; and,

2) The reason for taking these actions?

A. 1) Transition to FRS-0.2A, Response to Loss of Core Shutdown, Step 1, Verify Containment Pressure Less Than 5 PSIG
   2) It is now the procedure and step in effect

B. 1) Remain in FRS-0.1A, Response to Nuclear Power Generation / ATWT, and allow RCS temperature to lower while continuing efforts to establish Emergency Boration.
   2) A lower temperature will maintain an appropriate DNBR margin.

C. 1) Transition to FRS-0.2A, Response to Loss of Core Shutdown.
   2) This is required by the Critical Safety Function SUBCRITICALITY Status Tree based on the current YELLOW path condition.

D. 1) Remain in FRS-0.1A, Response to Nuclear Power Generation / ATWT, and allow the RCS to heat up while continuing efforts to establish emergency boration.
   2) The heatup will insert negative reactivity.

Answer: D

Comments / Reference: Bank 33641	Revision:

A. Plausible because review of the SUBCRITICALITY Status Tree would indicate that this is a possible transition, however, exit conditions for FRS-0.1 have not been met.

B. Plausible because procedure entry is correct, however, actions include allowing the RCS to heat up and add negative reactivity.

C. Plausible because review of the SUBCRITICALITY Status Tree would indicate that this is the current CSFST condition, however, exit conditions for FRS-0.1 have not been met.

D. Remaining in FRS-0.1 and allowing the plant to heat up will insert negative reactivity.

Question 200 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	33641
User-Defined ID:	ILOT7280
Cross Reference Number:	ERG.FS1.OB02.001
Topic:	Given the following conditions: An Anticipated Transient Without Trip (ATWT) event is in progress
K/A:	EPE.029.EK3.12
Question Reference:	
SRO:	Yes
Comments:	R/S18E17; LC17 Audit; LC18 NRC; LC19 Audit; R/S20E16; R/S20E18 (Comp); R/S21E16; LC20 NRC; SD29.SC3.EE1 (2013 SRO Cert); R/S22E16; R/S23E25; R/S24E25 REF: FRS-0.1A, CSFST

Comments / Reference: FRS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRS-0.1A
RESPONSE TO NUCLEAR POWER GENERATION/ATWT	REVISION NO. 9	PAGE 10 OF 33

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17	Verify Reactor Subcritical:	Perform the following:
	a. Power range indication - LESS THAN 5%	1) Continue to borate. IF boration NOT available, THEN allow RCS to heat up.
	b. Intermediate range channels - NEGATIVE STARTUP RATE	2) Perform actions of other FRGs in effect which do not cooldown or otherwise add positive reactivity to the core.
		3) Return to Step 4.
18	Check if RCPs Should be stopped:	
	a. RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)	a. Go to Step 18d.
	b. ECCS pumps - AT LEAST ONE RUNNING • CCP -OR- • SI pump	b. Go to Step 18d.
	c. Stop all RCPs and proceed to Step 19.	
	d. RCP operating parameters - WITHIN LIMITS	d. Stop affected RCP(s).

CAUTION: Boration should continue to obtain adequate shutdown margin during subsequent actions.

19	Return To Procedure And Step In Effect.	

-END-

Comments / Reference: FRS-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRS-0.1A
RESPONSE TO NUCLEAR POWER GENERATION/ATWT	REVISION NO. 9	PAGE 25 OF 33

ATTACHMENT 3
PAGE 9 OF 17

BASES

STEP 16: If the operator enters this step and core exit TC temperatures are greater than 1200°F and increasing, the operator should transition to SAG-1.0. This condition indicates that all attempts to restore core cooling have failed and core damage cannot be prevented.

If the operator enters this step and core exit TC temperatures are less than 1200°F or core exit TC temperatures are greater than 1200°F and decreasing the operator will stay in the loop between Steps 4 and 17 to continue efforts to emergency borate the RCS and check for sources of positive reactivity.

STEP 17: By this time all attempts to identify and isolate the most obvious sources of positive reactivity addition to the RCS have been performed. Furthermore, the boration initiated in Step 4 may already have had some effect in returning the core to a subcritical condition. Hence, a check on subcriticality is in order. This step specifies two conditions which must both be satisfied to verify that the reactor is indeed subcritical. Power range indication below 5% ensures that the heat load to available heat sinks is just the decay heat level normally accommodated with AFW flow. The negative intermediate range startup rate ensures the reactor is subcritical. Notice that no degree of subcriticality is specified and, therefore, any negative startup rate is acceptable.

If neither of the above conditions for subcriticality is satisfied, the operator is directed to continue the boration. If boration is not available, then the RCS should be allowed to heat up in order for the negative reactivity feedback mechanisms (moderator temperature coefficient and Doppler effect) to take effect in reducing nuclear power.

Notice that Power Range indication is available from either post accident qualified instrumentation and the normal instrument which is not post accident qualified. If Containment pressure is greater than or equal to 5 psig, then all normal excore indications are no longer qualified and the Neutron Flux Wide Range instruments should be used.

In addition, actions of other Function Restoration Guidelines in effect can be performed at this time (even though the Subcriticality Status Tree may still indicate a RED or ORANGE priority) as long as they do not cool down or otherwise add positive reactivity to the core. The operator is then returned to Step 4 of FRS-0.1A to continue efforts to emergency borate the RCS and check for sources of positive reactivity.

Other FRGs in effect can mean previous FRGs in effect or lower priority FRGs that may be identified, e.g., high containment pressure.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			1
	Group			1
	K/A	000008.G.2.4.21		
Level of Difficulty: 2	Importance Rating			4.6

Pressurizer Vapor Space Accident: 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.

Question # 85

Given the following on Unit 1:

- A Reactor Trip and SI have occurred
- EOP-1.0A, Loss of Reactor or Secondary Coolant, in progress
- Attachment 2 of EOP-0.0A, Reactor Trip or Safety Injection complete
- All RCPs have been stopped
- RVLIS – All lights are DARK
- CNTMT PRESS (IR) indicating 6.5 psig stable
- PRZR LVL indicating > 100%
- RCS HL PRESS (WR) indicating 1400 psig stable
- CORE EXIT TEMP indicating 765ºF rising
- RCS HL TEMP (WR) indicating 685 ºF rising

A(an) __(1)__ break has occurred.

A transition to __(1)__ is required.

A. (1) RCS cold leg
(2) FRC-0.2A, Response to Degraded Core Cooling

B. (1) RCS cold leg
(2) FRC-0.1A, Response to Inadequate Core Cooling

C. (1) PRZR steam space
(2) FRC-0.2A, Response to Degraded Core Cooling

D. (1) PRZR steam space
(2) FRC-0.1A, Response to Inadequate Core Cooling

Answer: C

K/A Match: K/A match due to requiring knowledge of the parameters used to determine the conditions of the RCS following a PRZR steam space break.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures.

Explanation:

A. Incorrect. First part is incorrect, but plausible since all indication other than PRZR level support indication of a cold leg break on the RCS. Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since an ORANGE path condition exists and for all CSFSTs except FRC, a RED and an ORANGE path condition direct entry into the highest FRG associated with the function.

C. Correct. First part is correct, with RVLIS dark and PRZR level full, this indicates a steam space break. Second part is correct, with RCS less than 1200°F, superheated, No RVLIS indication, and temp >750°F, FRC-0.2 entry conditions have been met.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	FRC-0.1	Attached w/ Revision # See Comments / Reference
	Loss of Coolant Accident Analysis LP	

Proposed references to be provided during examination: ____________________

Learning Objective: **EXPLAIN** the operational implications associated with all cautions, notes and actions of the "Response to Inadequate Core Cooling" procedure in accordance with FRC-0.1 (ERG.FC1.OB03)

Question Source: Bank # 18479

Modified Bank # ____________ (Note changes or attach parent)

New ____________

Question History: Last NRC Exam ____________

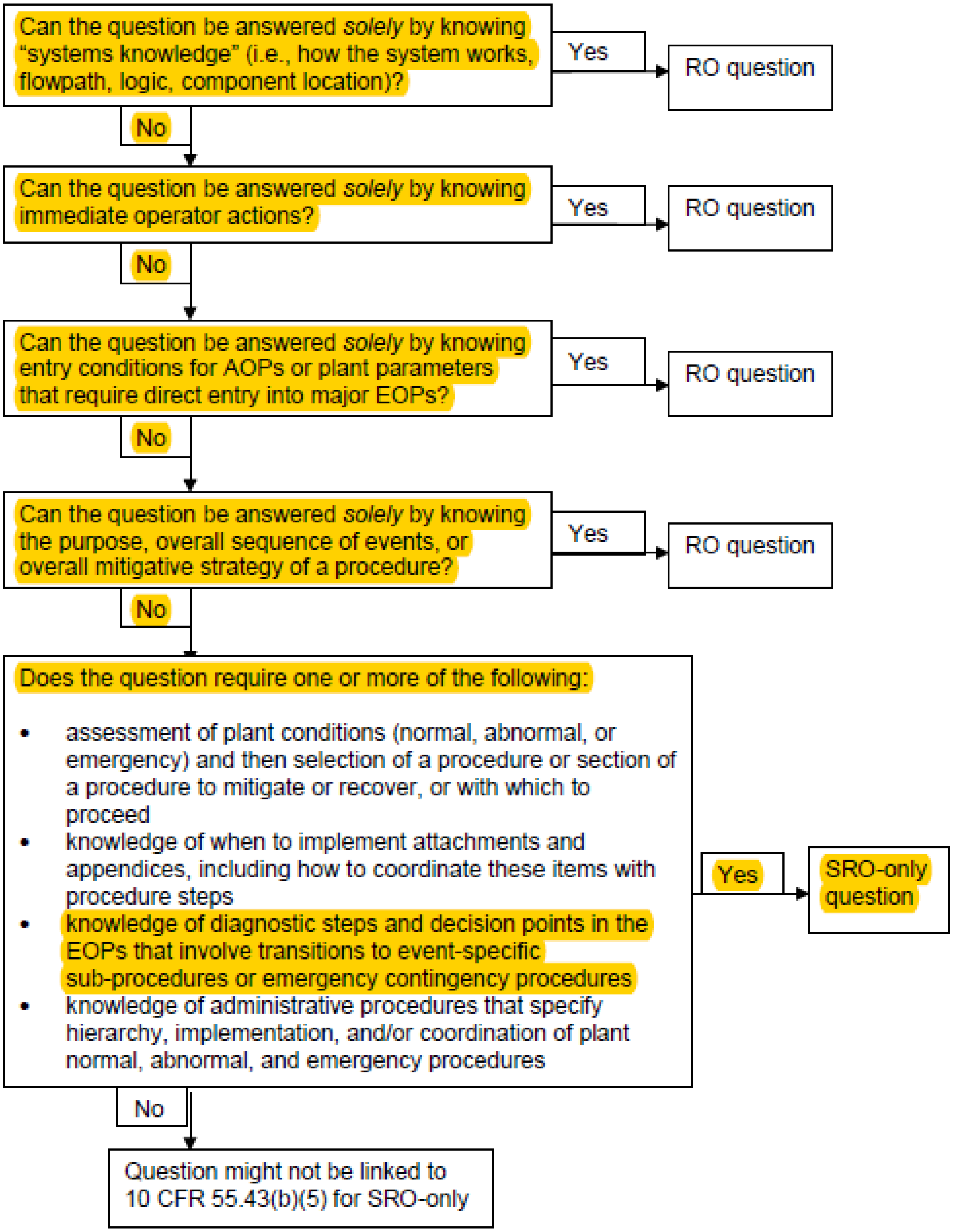
Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____________

55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:

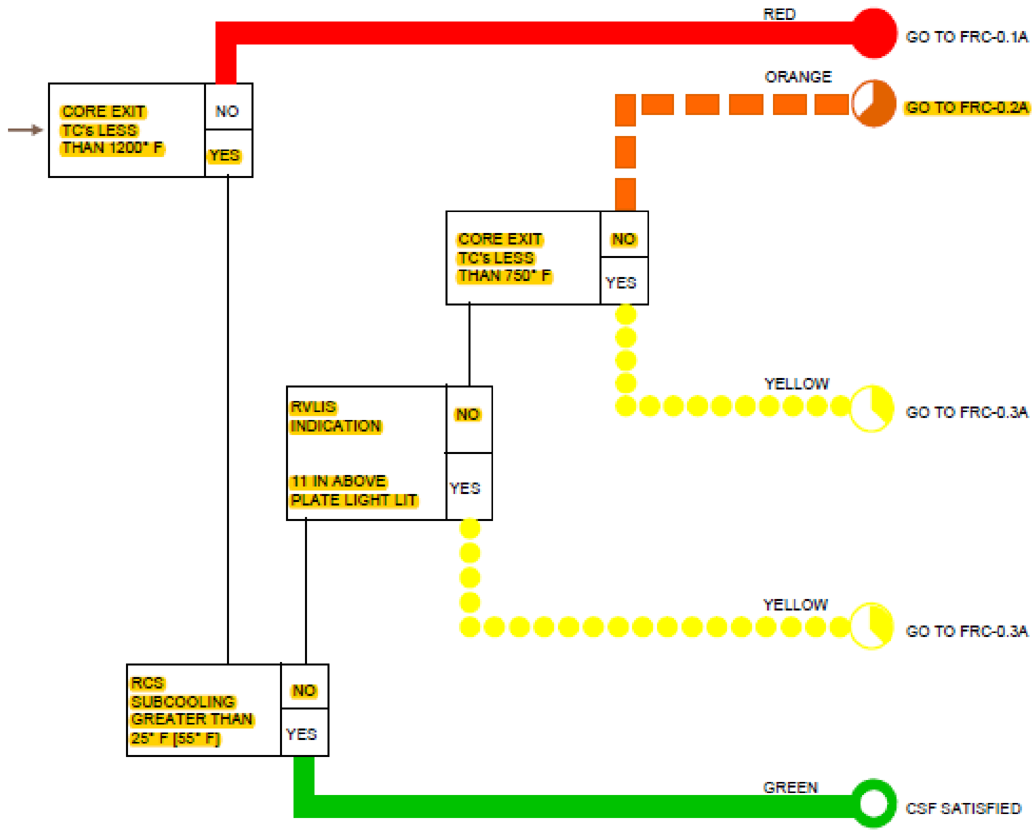


Comments / Reference: FRC-0.1A	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.1A
RESPONSE TO INADEQUATE CORE COOLING	REVISION NO. 9	PAGE 21 OF 46

ATTACHMENT 1.A
PAGE 1 OF 1

CORE COOLING STATUS TREE



Comments / Reference: Loss of Coolant Accident Analysis (PRZR)	Revision: 00-0000

NOTES	LESSON OUTLINE
	b. Single PORV or safety valve fails.
	c. Analysis assumptions minimize the DNBR.
	d. FSAR analysis bounded by small break LOCA analysis.
	2. One safety valve stuck open.
	a. RCS depressurizes throughout transient.
	b. Pressurizer fills as the hot legs and core reach saturation and flash.
	c. Core mixture level reaches top of the hot leg.
	d. Due to mass flow into the pressurizer, steam generator cold side mixture level drains.
	e. Loop seal and downcomer remain full.
	f. RCS inventory is stabilized as safety injection flow matches the break flow.
	g. Core remains covered.
	3. Conclusions
	a. Vapor space breaks are not limiting breaks.
	b. No core uncovery.
(* Important)	**c. *Pressurizer level is not a valid indication of core inventory.**

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier			1
	Group			1
	K/A	000022.AA2.02		
Level of Difficulty: 3	Importance Rating			3.7

Loss of Reactor Coolant Makeup: Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Charging pump problems

Question # 86

Given the following conditions:

- Unit 1 100% power
- CCP 1-01 in service
- VCT level 50%
- 1-FI-121A, CHRG FLO 130 gpm stable
- 1-FI-132, LTDN FLO 120 gpm stable
- 1/1-LCV-112B, VCT TO CHRG PMP SUCT VLV spuriously closes
- The following alarms are received;
  - 1-ALB-6A, Window 1.4 – REGEN HX LTDN OUT TEMP HI
  - 1-ALB-6A, Window 3.4 – CHRG FLO HI/LO

What action must be taken and the procedure used?

A. Stop CCP 1-01 and then isolate letdown per SOP-103A, Chemical and Volume Control System.

B. Stop CCP 1-01 and then isolate letdown per ABN-105, Chemical and Volume Control System Malfunction.

C. Isolate letdown and then stop CCP 1-01 per SOP-103A, Chemical and Volume Control System.

D. Isolate letdown and then stop CCP 1-01 per ABN-105, Chemical and Volume Control System Malfunctions.

Answer: B

K/A Match: K/A match due to requiring diagnostic of charging problems and required actions.

SRO Only: SRO Only due to assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed

Explanation:

A. Incorrect. Plausible because the actions stated are correct, however, ABN-105 is the correct procedure to use.

B. Correct. This is a symptom of pump cavitation due to suction valve being closed. The RNO actions of ABN-105 direct the operator to stop the CCP and isolate Letdown.

C. Incorrect. Plausible because actions stated are correct, however, they must be performed in the opposite order in accordance with ABN-105.

D. Incorrect. Plausible because actions stated are correct, however, they must be performed in the opposite order.

Technical Reference(s)	ABN-105	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **EVALUATE** plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrumentation while responding to a Chemical and Volume Control System malfunction.

Question Source: Bank # 2103 NRC Q77
Modified Bank # ____________ (Note changes or attach parent)
New ____________

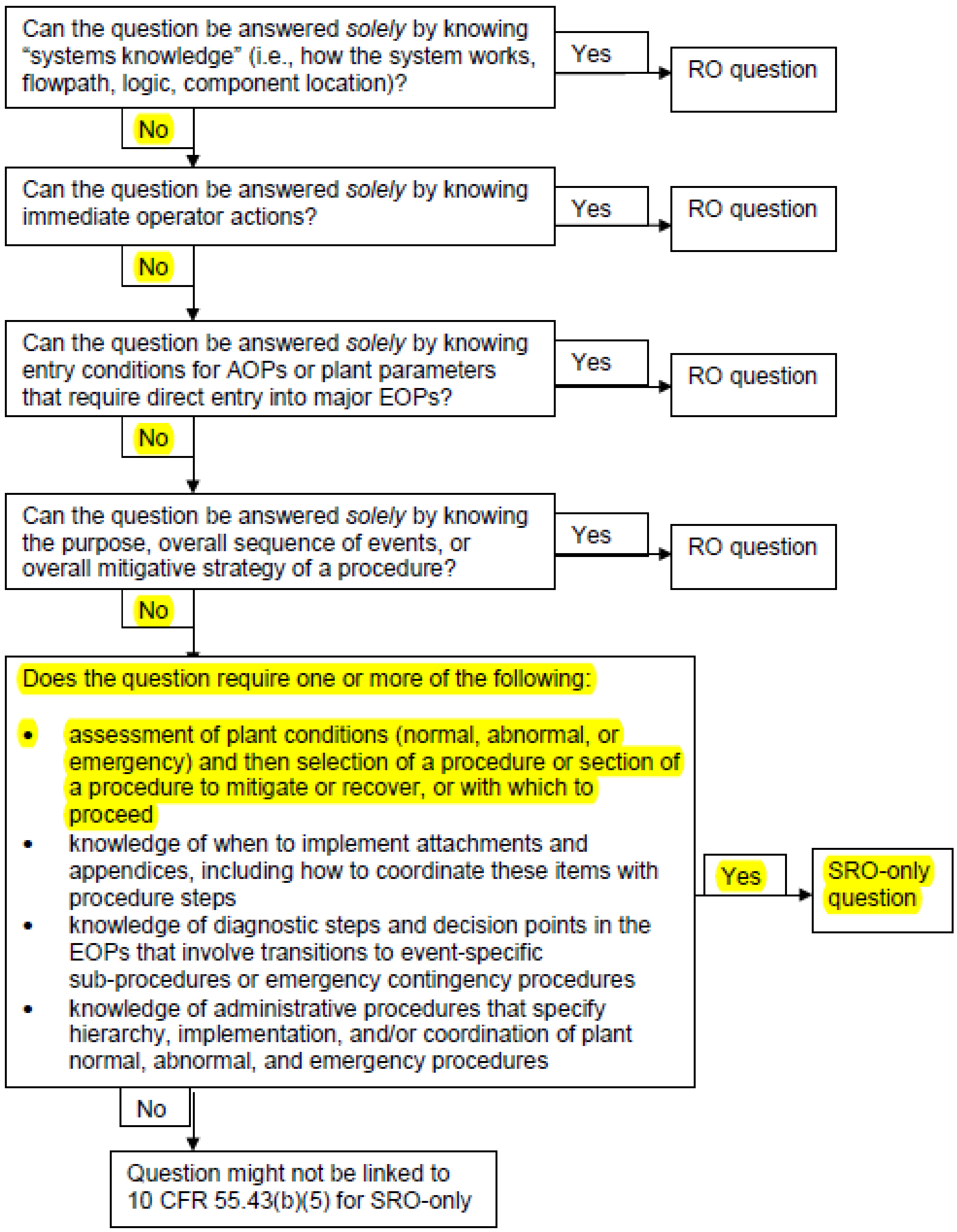
Question History: Last NRC Exam 2013 NRC

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____________
55.43 5

Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Comments / Reference: ABN-105	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-105
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 27 OF 40

7.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Operating a CCP with symptoms of cavitation or gas binding may cause rapid pump failure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 VERIFY VCT conditions - NORMAL	
☐ a. VCT TO CHRG PMP SUCT VLVs - OPEN: • 1/u-LCV-112B • 1/u-LCV-112C	a. PERFORM the following: 1) ENSURE ALL charging pumps STOPPED. 2) ENSURE letdown isolated. [C] 3) OPEN BOTH VCT TO CHRG PMP SUCT VLVs. IF either valve does NOT remain OPEN, AND charging pump is needed immediately, THEN PERFORM the following: A) OPEN RWST TO CHRG PMP SUCT VLVs: • 1/u-LCV-112D • 1/u-LCV-112E B) CLOSE 1/u-LCV-112B AND 1/u-LCV-112C C) VERIFY u-ZL-8220 AND u-ZL-8221, CHRG PMP SUCT HI POINT VENT VLV - CLOSED D) ENSURE 1/u-8202A AND 1/u-8202B, VNT VLV - CLOSED 4) WHEN charging pump suction is aligned, THEN START a centrifugal charging pump per SOP-103A/B.

"Step continued next page"

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier			1
	Group			1
	K/A	000057.G.2.2.42		
Level of Difficulty: 2	Importance Rating			4.6

Loss of Vital AC Instrument Bus: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question # 87

Given the following conditions:

- Unit 2 at 800 MWe
- Loss of 118 VAC Instrument Distribution Inverter IV2EC1 occurs
- ABN-603, Loss of Protection or Instrument Bus, in progress

Per ABN-603, the Unit Supervisor INITIALLY directs energizing 2EC1 by aligning __(1)__ to supply 2EC1.

Per TS 3.8.9, Distribution Systems - Operating, the above action is required to be completed within a MAXIMUM of __(2)__.

A. (1) 120 VAC Bypass Distribution Panel 2EC3
(2) 2 hours

B. (1) 120 VAC Bypass Distribution Panel 2EC3
(2) 8 hours

C. (1) TRN A 118 VAC RPS/SFGD BOP Installed Spare Inverter IV2EC1/3
(2) 2 hours

D. (1) TRN A 118 VAC RPS/SFGD BOP Installed Spare Inverter IV2EC1/3
(2) 8 hours

Answer: A

K/A Match: K/A match due to requiring knowledge of abnormal condition procedures which contain the guidance for restoring power to a Vital AC Instrument Bus and the associated Technical Specification.

SRO Only: SRO Only due to 10 CFR 55.43(b)(2) Analysis requiring application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1).

Explanation:

A. Correct. First part is correct. In accordance with ABN-603, the US will direct an operator in the field to energize 2EC1 via its alternate power supply, 2EC3, by sliding the manual transfer switch to the alternate position at the bottom of the instrument panel. Second part is correct. In accordance with TS 3.8.9, Condition 'B', the AC Vital bus subsystem will be restored to OPERABLE status within 2 hours. Per TS 3.8.9 Bases re-energizing Instrument Panel 2EC1 via its alternate power supply will restore it to OPERABLE status.

B. Incorrect. First part is correct (see A). Second part is incorrect, but plausible as TS 3.8.9 Condition 'A' requires an AC electrical power distribution subsystem to be restored within 8 hours. Condition 'A' applies to 6900V and 480V distribution subsystems and is commonly confused with Condition 'B'.

C. Incorrect. First part is incorrect, but plausible since the next step of ABN 603 is to initiate actions to place the swing inverter, IV2EC1/3, in service. However, this must be performed per SOP-607B and will take some time to perform. Per TS 3.8.9 Bases re-energizing Instrument Panel 2EC1 via the swing inverter will also restore it to OPERABLE status. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	ABN-603	Attached w/ Revision # See Comments / Reference
	TS/TSB 3.8.9	
	208/120 VAC, 118 VAC., Inv. & Lighting LP	

Proposed references to be provided during examination: ____________

Learning Objective: **ANALYZE** the response to Loss of a Protection Bus in accordance with ABN-603, Loss of Protection or Instrument Bus. (ABN.603.OB01)

Question Source: Bank # 75858
Modified Bank # ____________ (Note changes or attach parent)
New ____________

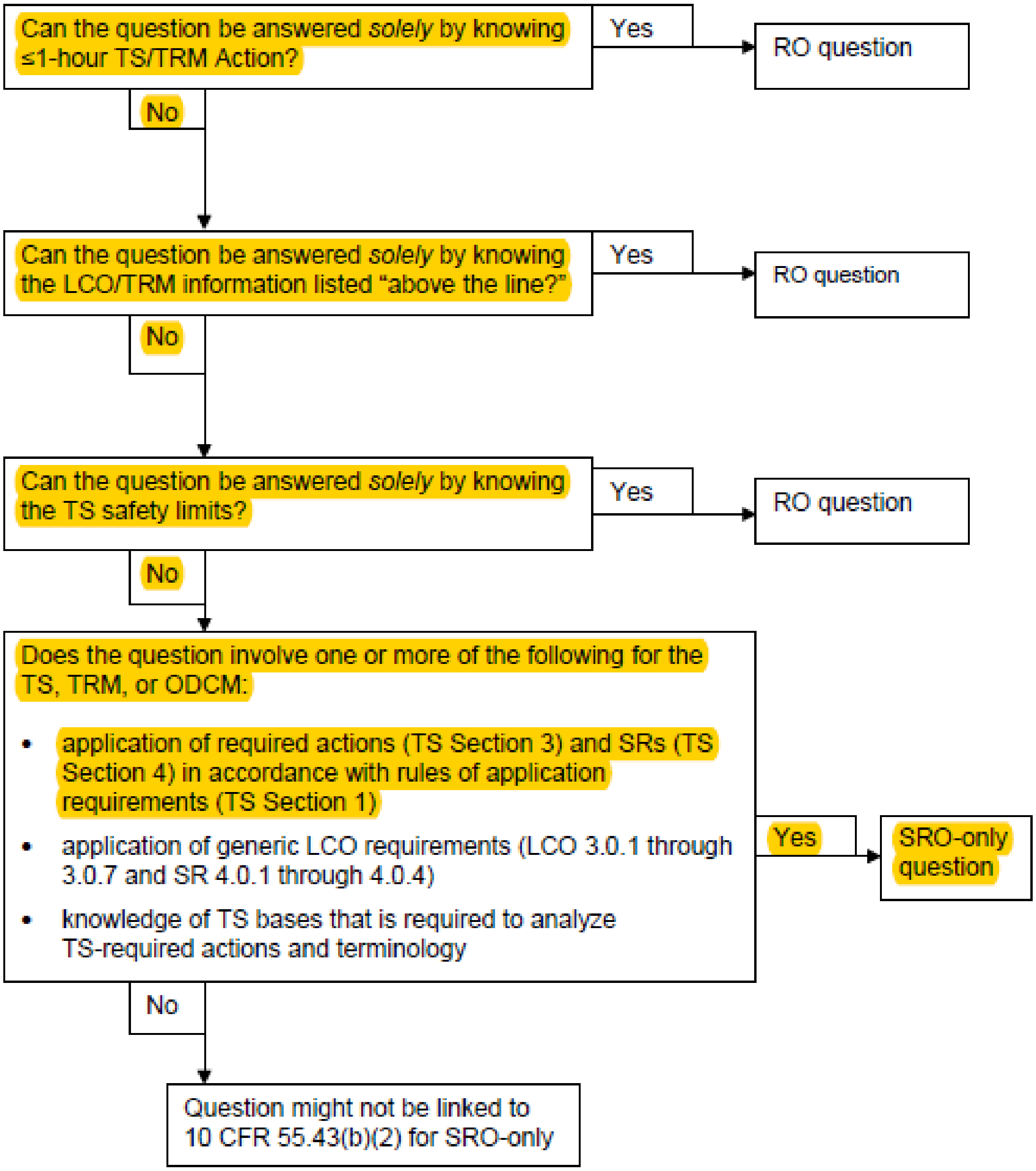
Question History: Last NRC Exam LC24

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____________

55.43 2

Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:



Comments / Reference: Bank 75858	Revision:

- Unit 2 at 800 MWe
- Loss of 118 VAC Instrument Distribution Panel 2EC1 occurs
- ABN-603, Loss of Protection or Instrument Bus in progress

Per ABN-603 the Unit Supervisor first directs energizing 2EC1 by aligning _________________________ to supply 2EC1.

Per Technical Specification 3.8.9, Distribution Systems - Operating, the above action is required to be completed within a MAXIMUM of ____ hours.

A. 120 VAC Bypass Distribution Panel 2EC3
2

B. 120 VAC Bypass Distribution Panel 2EC3
8

C. TRN A 118 VAC RPS/SFGD BOP Installed Spare Inverter IV2EC1/3
2

D. TRN A 118 VAC RPS/SFGD BOP Installed Spare Inverter IV2EC1/3
8

Answer: A

Answer Explanation

A. Correct. Part 1 is correct in accordance with ABN-603 the US will direct an operator in the field to energize 2EC1 via its alternate power supply, 2EC3, by sliding the manual transfer switch to the alternate position at the bottom of the instrument panel. Part 2 is correct in accordance with TS 3.8.9 Condition 'B' the AC Vital bus subsystem will be restored to OPERABLE status within 2 hours. Per TS 3.8.9 Bases re-energizing Instrument Panel 2EC1 via its alternate power supply will restore it to OPERABLE status.

B. Incorrect. Part 1 is correct as described in 'A' above. Part 2 is incorrect but plausible as TS 3.8.9 Condition 'A' requires an AC electrical power distribution subsystem to be restored within 8 hours. Condition 'A' applies to 6900V and 480V distribution subsystems and is commonly confused with Condition 'B'.

C. Incorrect. Part 1 is incorrect but plausible as the next step of ABN 603 is to initiate actions to place the swing inverter, IV2EC1/3, in service. However, this must be performed per SOP-607B and will take some time to perform. Per TS 3.8.9 Bases re-energizing Instrument Panel 2EC1 via the swing inverter will also restore it to OPERABLE status. Part 2 is correct as described in 'A' above.

D. Incorrect. Part 1 is incorrect but plausible as described in 'C' above. Part 2 is incorrect but plausible as described in 'B' above.

Comments / Reference: Bank 75858	Revision:

Question 84 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	75858
User-Defined ID:	ILOT9468
Cross Reference Number:	ABN.603.OB02.007
Topic:	Unit 2 at 800 MWe Loss of 118 VAC Instrument Distribution Panel 2EC1 occurs ABN-603, Loss of Prote
K/A:	057 G.2.4.11
Question Reference:	
SRO:	YES
Comments:	LC25 NRC K/A Match: The question is a K/A match as it requires the applicant to demonstrate knowledge of abnormal condition procedures which contain the guidance for restoring power to a Vital AC Instrument Bus and the associated Technical Specification. SRO Only: The question is SRO only because it requires the applicant to demonstrate knowledge of the application of Required TS actions and assessing plant conditions and then selecting a step of a procedure to recover.

Comments / Reference: ABN-603	Revision: 8

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-603
LOSS OF PROTECTION OR INSTRUMENT BUS	REVISION NO. 8	PAGE 9 OF 34

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Reenergizing the affected protection bus may cause instrumentation spikes on controlling channels which may in turn initiate unwanted actions.

☐ 6 Verify Unit - IN MODE 1	GO TO Step 8.

NOTE: Rod Control should remain in MANUAL until all Tave channels are operable.

☐ a. Place Control Rods in MANUAL

☐ b. Select failed channel on u-TS-412T, T_{AVE} CHAN DEFEAT switch.

☐ c. Dispatch an Operator to reenergize the affected protection bus by moving the manual sliding bar to close the alternate power supply feeder breaker (bottom of protection panel).

☐ d. IF C-7 Armed (PCIP-3.4), THEN select RESET on 43/u-SD, STM DMP MODE SELECT

☐ e. Undefeat affected channel on u-TS-412T

CAUTION: To prevent rods from potentially stepping, allow a minimum of 2 minutes for Tavg circuitry to stabilize following manipulation of u-TS-412T before returning rod control to Auto.

☐ f. Restore Control rods to AUTO.

☐ g. Investigate and initiate corrective action on loss of power to protection bus.

Section 2.3

Comments / Reference: 208/120 VAC, 118 VAC., Inv. & Lighting Lesson	Revision: 01-0001

LO21SYSAC3 Page 8 of 24

LESSON PLAN

NOTES	LESSON OUTLINE
	3. Bus is normally powered from inverters and has bypass power / alternate power supplied from a 480v bypass transformer (referred to as dirty power.)
	4. Inverter takes a 125 VDC input and delivers regulated, filtered, single phase 118 VAC output
	5. Uninterruptible by:
	a. Inverters connected to a DC bus fed from a 480 VAC battery charger if power is lost to the chargers the bus will be fed from a 125 VDC battery.
	b. If inverter fails due to a malfunction or the 125 VDC is lost, then a static switch on the inverter panel will auto swap to an unregulated 120 VAC bypass power source.
Objective 4	**F. Class 1E 118 VAC Vital Instrument Power**
	1. Two trains
	2. Each has four busses (panels) per train – 2 Reactor protection and 2 for BOP systems, each supplied by an inverter.
	3. 1 installed spare inverter can align to any one of the four panels on that train.
	a. DC to the spare needs to be the same as substituted power.
	4. Manual transfer at the bottom of each panel for inverter feed or transformer feed.
	a. Transformer feed is uEC3 or uEC4.
	b. Slide bar Mechanical interlock to prevent closing both supply breakers at the same time.
	c. Bumpless transfer to bypass power accomplished by placing inverter on bypass then removing slide bar, swapping power then replacing the slide bar.
	d. Allowed by procedure because inverter bypass power and distribution panel bypass power are from the same source.
	5. Operated per SOP-607A/B (see current revision.)
	6. 118 VAC Vital distribution panels are uPC1-4, uEC1, 2 & 5, 6.
	a. Located in the Cable Spreading Room (CSR.)
	b. Only operable when supplied from DC bus via inverter.
	c. When uEC1 and uEC2 are on bypass the BO sequencers are not operable because on a loss of offsite power the sequencers will de-energize.

FOR TRAINING USE ONLY Rev. 01 0001

Comments / Reference: TS 3.8.9	Revision: 156

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems – Operating

LCO 3.8.9 Train A and Train B AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution subsystem inoperable.	A.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours
B. One AC vital bus subsystem inoperable.	B.1 Restore AC vital bus subsystem to OPERABLE status.	2 hours
C. One DC electrical power distribution subsystem inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Two trains with inoperable distribution subsystems that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

Comments / Reference: TSB 3.8.9	Revision: 82

BASES

ACTIONS (continued)

B.1

With one AC vital bus inoperable the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital bus must be restored to OPERABLE status within 2 hours by powering the bus from the associated inverter via inverted DC, or alternate bypass power via Class 1E transformers.

Condition B represents one AC vital bus without non-interruptible inverted DC power. In this situation, the unit is significantly more vulnerable to a complete loss of all non-interruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of non-interruptible power to the remaining vital buses and restoring power to the affected vital bus subsystems.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

(continued)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier			1
	Group			1
	K/A	WE04.EA2.01		
Level of Difficulty: 3	Importance Rating			4.3

LOCA Outside Containment: Ability to determine and interpret the following as they apply to the LOCA Outside Containment: Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Question # 88

Given the following conditions:

- Unit 1 Reactor has been tripped due to a lowering RCS pressure
- Indications are that an RCS leak has occurred outside containment
- ECA-1.2A, LOCA Outside Containment, has been entered

In accordance with ECA-1.2A, cycle __(1)__ closed, then open.

If RCS pressure continues to lower, GO TO __(2)__.

A. (1) 1/1-8809A and B, RHR TO CL INJ ISOL VLVS
(2) EOP-1.0A, Loss of Reactor or Secondary Coolant

B. (1) 1/1-8809A and B, RHR TO CL INJ ISOL VLVS
(2) ECA-1.1A, Loss of Emergency Coolant Recirculation

C. (1) 1/1-8840, RHR TO HL 2 & 3 INJ ISOL VLV
(2) EOP-1.0A, Loss of Reactor or Secondary Coolant

D. (1) 1/1-8840, RHR TO HL 2 & 3 INJ ISOL VLV
(2) ECA-1.1A, Loss of Emergency Coolant Recirculation

Answer: B

K/A Match: K/A match due to requiring the ability to determine the selection of appropriate procedures during a LOCA outside containment.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis involving both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed, including knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

Explanation:

A. Incorrect. First part is correct. These valves are normally open, so you are directed to close them and check RCS pressure to see if the leak has stopped. Second part is incorrect, but plausible since a loss of reactor coolant exists. If the leak has not been stopped, you are directed to GO TO ECA-1.1A.

B. Correct. First part is correct (see A). Second part is correct. If the leak still exists, you are directed to GO TO ECA-1.1A.

C. Incorrect. First part is incorrect, but plausible since these valves are verified closed. They are normally closed so it would not be prudent to open them to check for a leak. Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	ECA-1.2	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________________

Learning Objective: **IDENTIFY** the proper transitions out of ECA-1.2 (ERG.C12.OB06)

Question Source: Bank # 72564
Modified Bank # ____________ (Note changes or attach parent)
New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ____________
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Comments / Reference: ECA-1.2	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 9	PAGE 3 OF 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[R] 1	Verify Proper Valve Alignment: a. RHRP 1 & 2 HL RECIRC ISOL VLVS - CLOSED • 1/1-8701A • 1/1-8702A • 1/1-8701B • 1/1-8702B b. RHR TO HL 2 & 3 INJ ISOL VLV - CLOSED • 1/1-8840 c. SI TO HL INJ ISOL VLVS - CLOSED • 1/1-8802A • 1/1-8802B	Manually close valve(s). IF valve(s) can NOT be manually closed, THEN locally close valve(s).
2	Identify And Isolate Break: a. Sequentially close and open the following valves and monitor for an RCS pressure increase: 1) RHR TO CL INJ ISOL VLVS: • 1/1-8809A • 1/1-8809B 2) SI to CL 1•4 INJ ISOL VLV • 1/1-8835	

Comments / Reference: ECA-1.2	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 9	PAGE 4 OF 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	Check If Break Is Isolated:	
	a. RCS pressure - INCREASING	a. Go to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.
	b. Go to EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.	

-END-

-

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier			1
	Group			1
	K/A	WE05.G.2.2.40		
Level of Difficulty: 3	Importance Rating			4.7

Loss of Secondary Heat Sink: Ability to apply Technical Specifications for a system.

Question # 89

Given the following conditions:

- 1/2-PCV-455A, PRZR PORV, is stuck in mid-position and will not cycle
- 1/2-8000A, PRZR PORV BLK VLV, is CLOSED and deenergized

In accordance with TS 3.4.11, Pressurizer Power Operated Relief Valves (PORVs), 1/2-8000A __(1)__.

Approximately twenty-four hours later:

- The Unit has experienced a Loss of Secondary Heat Sink following a reactor trip from 100% power
- FRH-0.1B, Response to Loss of Secondary Heat Sink, in progress

When Bleed and Feed operations are established, 1/2-8000A should __(2)__.

A. (1) can remain in this condition indefinitely
(2) be re-energized and opened to ensure adequate bleed path

B. (1) can remain in this condition indefinitely
(2) remain de-energized and closed to comply with Technical Specifications

C. (1) must be restored to operable within 72 hours
(2) be re-energized and opened to ensure adequate bleed path

D. (1) must be restored to operable within 72 hours
(2) remain de-energized and closed to comply with Technical Specifications

Answer: C

K/A Match: K/A match due to requiring application of TS conditions associated with components used for bleed and feed during a loss of secondary heat sink event.

SRO Only: SRO Only due to 10 CFR 55.43(b)(2) Analysis requiring application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1) and 10 CFR 55.43(b)(5) Analysis requiring assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

Explanation:

A. Incorrect. First part is incorrect, but plausible since PORV block valve does not have to be deenergized for seat leakage and indefinite operation is permitted for these conditions, per condition A of TS. Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since PORV is closed in accordance with Tech Specs and re-opening will violate Tech Specs.

C. Correct. First part is correct, with a PORV stuck and unable to be manually cycled, the PORV block valve is required to be closed and deenergized. The PORV must be restored to an operable condition within 72 hours. These are all part of condition B of TS. Second part is correct. FRH-0.1B requires re-energizing and opening a previously de-energized and closed PORV Block valve regardless of Tech Spec implications to ensure adequate Bleed path.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	TS/B 3.4.11	Attached w/ Revision # See Comments / Reference
	FRH-0.1	

Proposed references to be provided during examination: __________

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRH-0.1 in accordance with FRH-0.1, Loss of Heat Sink. (ERG.FH1.OB04)

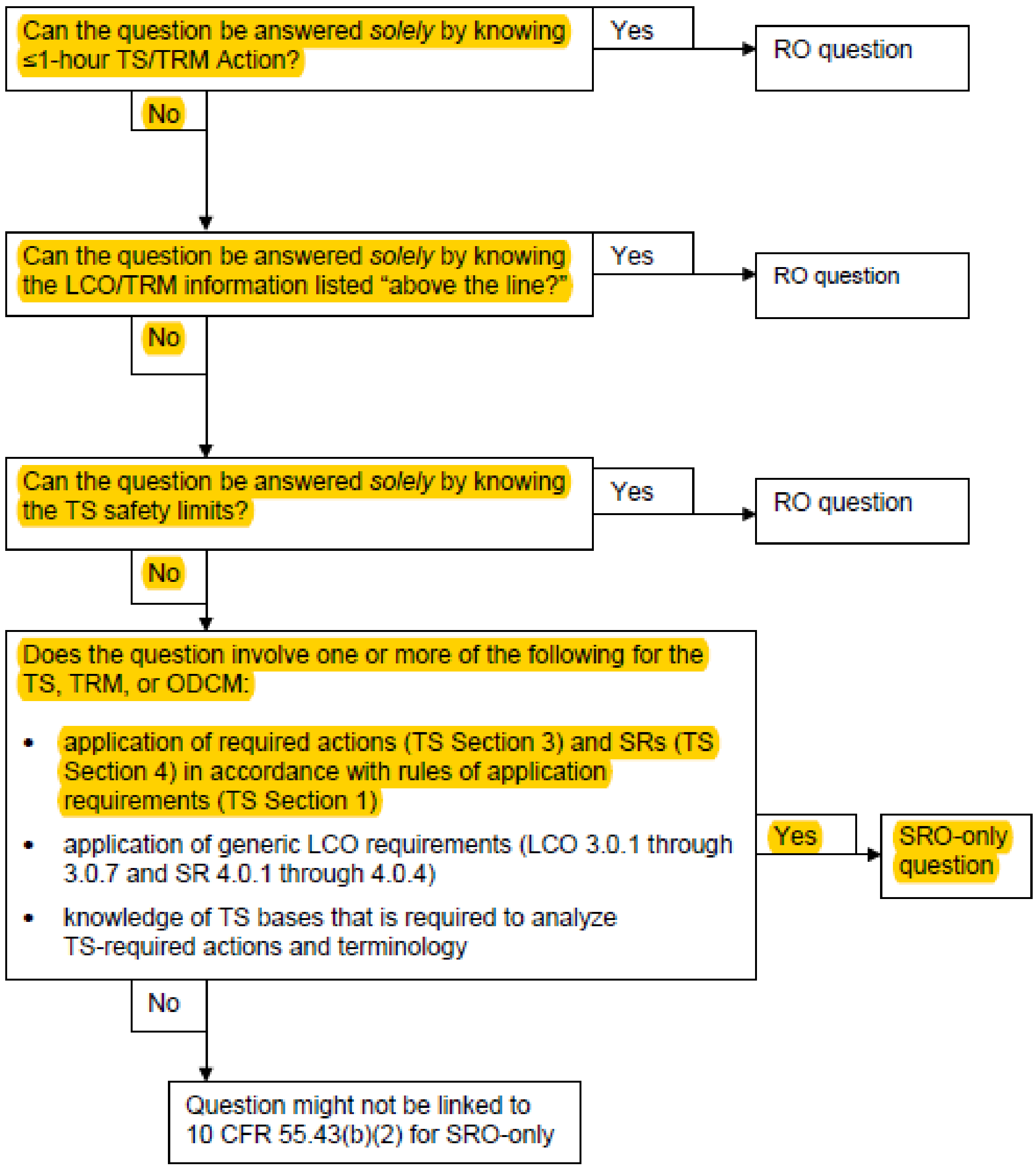
Question Source:
Bank # __________
Modified Bank # __________ (Note changes or attach parent)
New X

Question History: Last NRC Exam __________

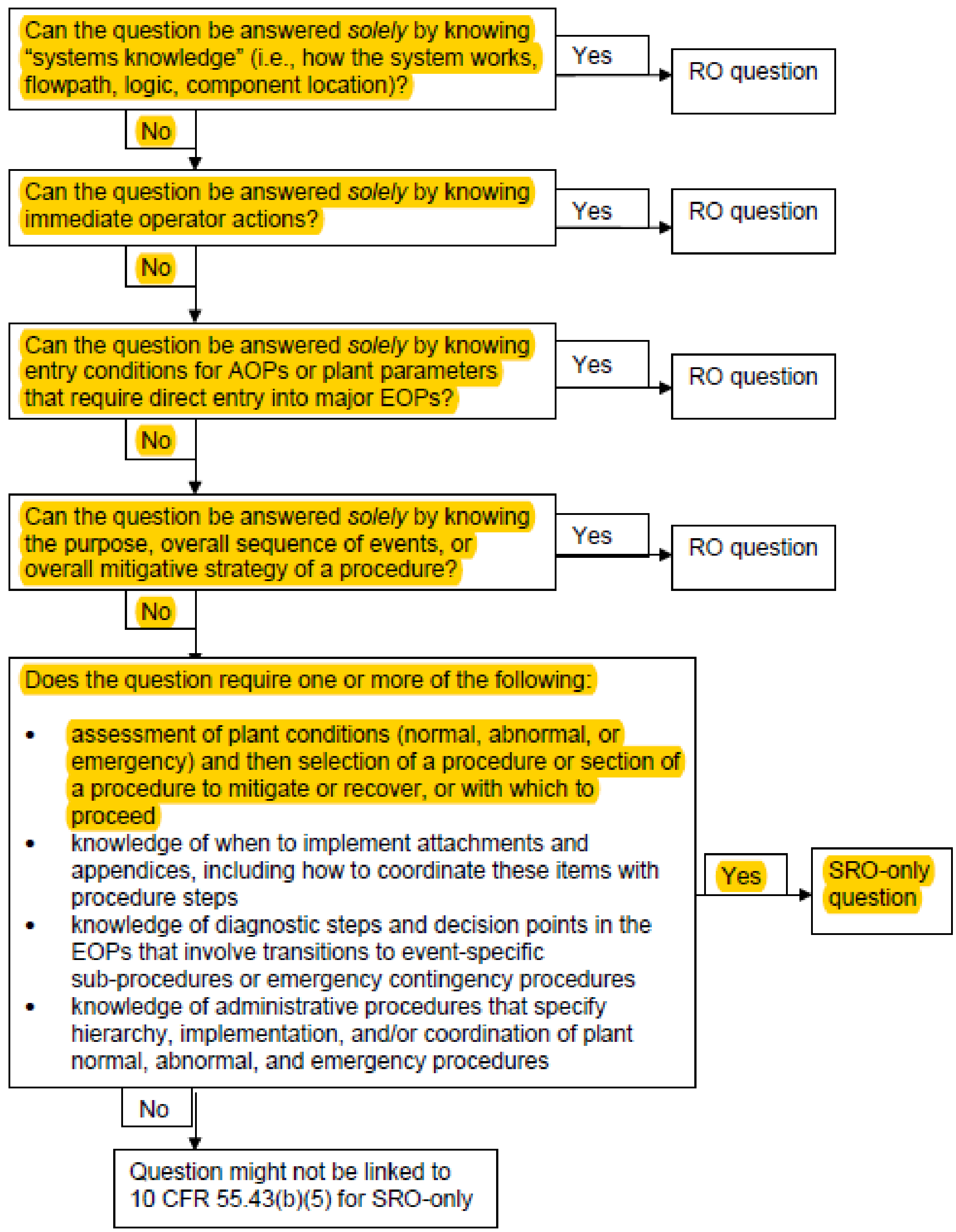
Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 __________
55.43 2/5

Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:



Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: TS 3.4.11	Revision: 156

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each PORV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve. AND B.2 Remove power from associated block valve. AND B.3 Restore PORV to OPERABLE status.	1 hour 1 hour 72 hours

Comments / Reference: TSB 3.4.11	Revision: 80

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are nitrogen operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint up to and including the design step-load decrease. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

-

Comments / Reference: TSB 3.4.11	Revision: 80

BASES

LCO (continued)

b. RCS identified leakage cannot be maintained less than the limits of LCO 3.4.13 without closure of the associated PORV block valve, or

c. PORV tail pipe temperature cannot be restored to or maintained within the limit (Ref. 7).

Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2.

The PORVs and block valves are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event. If a PORV is blocked, the PORV Block Valve is required to be manually opened to enable the PORV. The PORV Block valves are only required to be able to close against the maximum differential pressure associated with a failed open PORV during the SGTR event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODES 4, 5, and 6 with the reactor vessel head in place when both pressure and core energy are decreased and the pressure surges become much less significant. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valve

(continued)

Comments / Reference: TSB 3.4.11	Revision: 80

BASES

ACTIONS (continued)

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

is required to be closed, but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition.

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period. If one PORV is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an event if the inoperable block valve is not fully open. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status. If it cannot be restored within this additional time, the plant must be brought to a MODE in

(continued)

Comments / Reference: FRH-0.1	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 9	PAGE 22 OF 85

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1D] 18	Reset Containment Isolation Phase A And Phase B.	
[1D] 19	Reset Containment Spray Signal.	
[1D] 20	Establish Instrument Air And Nitrogen To Containment:	
	a. Establish instrument air:	
	1) Verify air compressor running.	1) Manually start air compressor and align valve as appropriate.
	-AND-	
	• Establish instrument air to containment.	
	b. Establish nitrogen:	
	1) Verify ACCUM 1•4 VENT CTRL, 1-HC-943 - CLOSED	1) Manually close valve.
	2) Open SI/PORV ACCUM N_2 ISOL VLV, 1/1-8880.	
21	Establish RCS Bleed Path:	
	a. Verify power to PRZR PORV block valves - AVAILABLE	a. Locally restore power to block valve(s).
	b. Verify PRZR PORV block valves - BOTH OPEN	b. Manually open both block valve(s).
	c. Open PRZR PORVs.	
22	Verify Adequate RCS Bleed Path:	Open vents on reactor vessel head and on the PRZR to containment.
	• PRZR PORVs - BOTH OPEN	
	• PRZR PORV block valves- BOTH OPEN	
23	Verify Steps 1 through 8 of EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION, Have Been Completed.	Perform Steps 1 through 8 of EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.

Comments / Reference: FRH-0.1	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. FRH-0.1B
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 9	PAGE 69 OF 86

ATTACHMENT 4
PAGE 16 OF 33

BASES

STEP 20: The restoration of Instrument Air and Nitrogen is necessary to allow the operation of pneumatically operated valves in containment (in this step, nitrogen for the operation of PRZR PORVs is of particular interest). While opening the containment isolation valves is sufficient to restore nitrogen to containment, it might also be necessary to start an air compressor to restore Instrument Air to containment.

STEP 21: The operator ensures that all pressurizer block valves are open and opens all pressurizer PORVs to establish an RCS bleed path. These valves must be maintained in the open position until secondary heat sink is restored.

The Pressurizer PORV block valves are checked to ensure power is available and that the valves are open to confirm the availability of all Pressurizer PORVs. It is preferred to open all PORVs when establishing the RCS bleed path for this procedure to maximize heat removal capability. If a PORV block valve has been closed and deenergized to satisfy Technical Specification requirements, then power should be realigned and the block valve should be opened. If power to a block valve is not available but the block valve is open, the PORV should be open in order to align the bleed path. This step does NOT require power to be available to a block valve prior to opening the PORV (e.g., step does not assume the block valve has to be available in order to isolate the PORV in subsequent recovery actions).

Once the pressurizer PORVs are open, the RCS will depressurize and the CCPs and/or SI pumps will deliver subcooled flow to the RCS. This will provide adequate RCS heat removal until flow can be established to the steam generators to restore secondary heat sink.

The operator may observe increasing pressurizer level after the pressurizer PORVs are opened. Eventually the pressurizer may become water solid with water relief occurring through the pressurizer PORVs.

STEP 22: After manually opening the PRZR PORVs, the operator should check that both PRZR PORVs and block valves are maintained in the open position. If both bleed paths are maintained open, sufficient RCS bleed flow exists to permit RCS heat removal.

If both PRZR PORVs or block valves are not maintained open, the RCS may not depressurize sufficiently to permit adequate feed of subcooled ECCS flow to remove core decay heat. If core decay heat exceeds RCS bleed and feed heat removal capability, the RCS will repressurize rapidly, further reducing the feed of subcooled ECCS flow and resulting in a rapid decrease of RCS inventory.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier			1
	Group			2
	K/A	000032.AA2.09		
Level of Difficulty: 3	Importance Rating			2.9

Loss of Source Range Nuclear Instrumentation: Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Effect of improper HV setting

Question # 90

Given the following conditions:

- Reactor Startup in progress on Unit 1
- Control Rods at 150 steps on Bank A
- SR counts on channels N-31 and N-32 differ by approximately 60 cpm
- Core Performance determined High Voltage adjustment is required on SR channel N-31
- While adjusting SR channel N-31, 1-ALB-6D, Window 1.1 – SR HI VOLT FAIL alarms
- Reactor Startup is suspended in accordance with TS 3.3.1, RPS Instrumentation

The SR HI VOLT FAIL alarm indicates SR Channel N-31 counts will be __(1)__ than actual counts.

If N-31 is NOT restored to OPERABLE within 48 hours, in accordance with TS 3.3.1, __(2)__ must be fully inserted.

A. (1) lower
(2) Control Bank rods ONLY

B. (1) lower
(2) Control and Shutdown Bank rods

C. (1) higher
(2) Control Bank rods ONLY

D. (1) higher
(2) Control and Shutdown Bank rods

Answer: B

K/A Match: K/A match due to requiring knowledge of actions to be taken in response to failed SR high voltage.

SRO Only: SRO Only due to 10 CFR 55.43(b)(2) Analysis requiring application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1).

Explanation:

A. Incorrect. First part is correct (see B). Second part is incorrect, but plausible since actions taken during startup, if the startup is aborted, are to insert control bank rods to CBO and shutdown rods do not have a CBO position.

B. Correct. First part is correct, the improperly set SRHV power supply (110 volts below normal cause the alarm) causes less interactions between the core and the BF3 detector, causing counts to lower. Second part is correct. Per TS 3.3.1, if the inoperable SR is not returned to operable status within 48 hours, all rods, both control bank and shutdown bank rods, must be fully inserted.

C. Incorrect. First part is incorrect, but plausible if thought SRHV supplied compensating voltage, as is done in the IR channels, which would then cause counts to increase upon the loss. Second part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	Excore Nuclear Instrument Study Guide	Attached w/ Revision # See Comments / Reference
	ABN-701	
	TS 3.3.1	

Proposed references to be provided during examination: ______

Learning Objective: Given Excore Instrumentation system operability status or parameter indications, various plant conditions, and a copy of the Technical Specifications or Technical Requirements Manual, **ASSESS** any LCO entries, applicable conditions, and required actions (including Completion Time) in accordance with the associated regulatory requirements and their bases. (SYS.EC1.OB07)

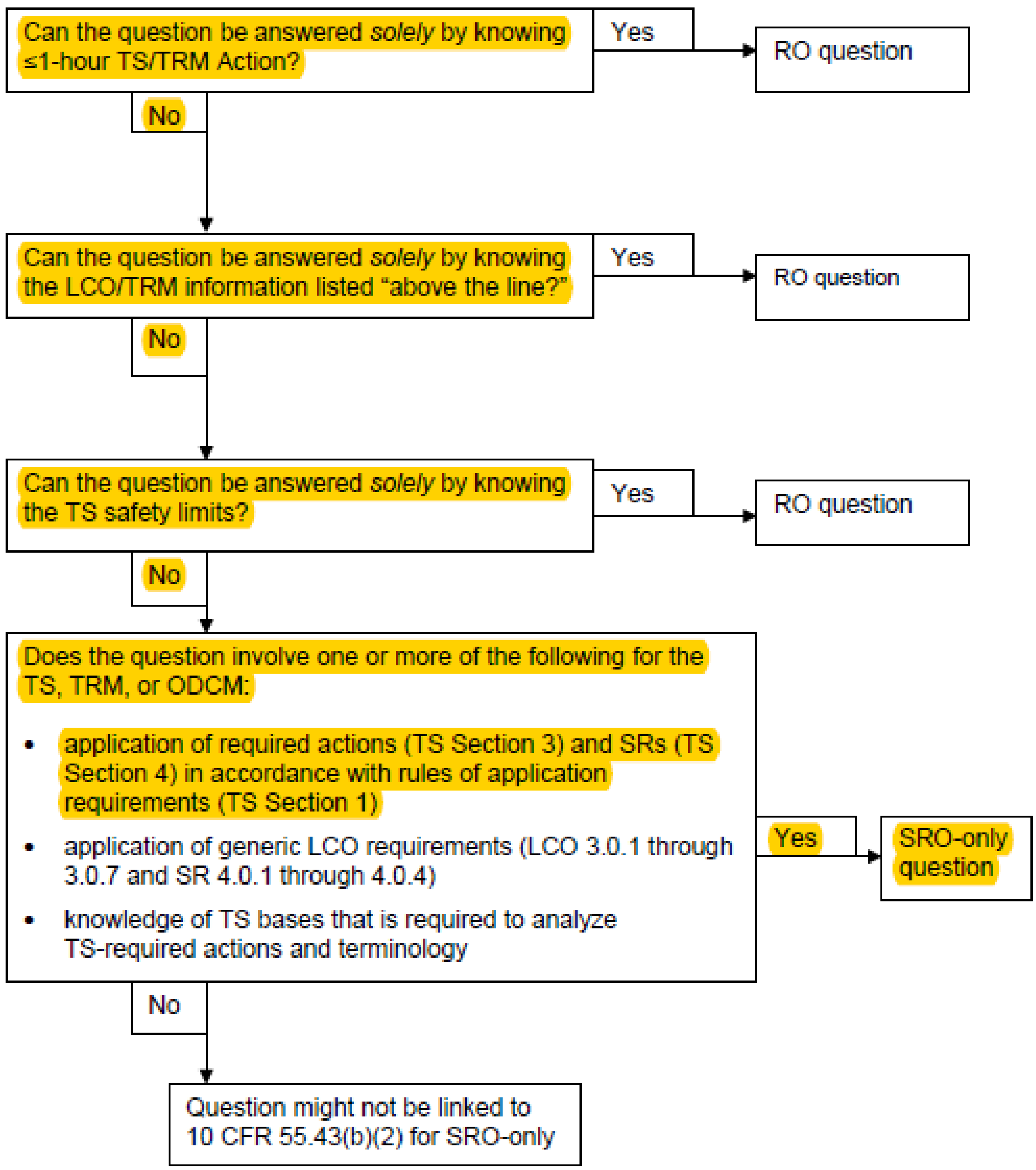
Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:



Comments / Reference: Excore Nuclear Instrumentation Study Guide	Revision: 5-1-2011

Excore Nuclear Instrumentation

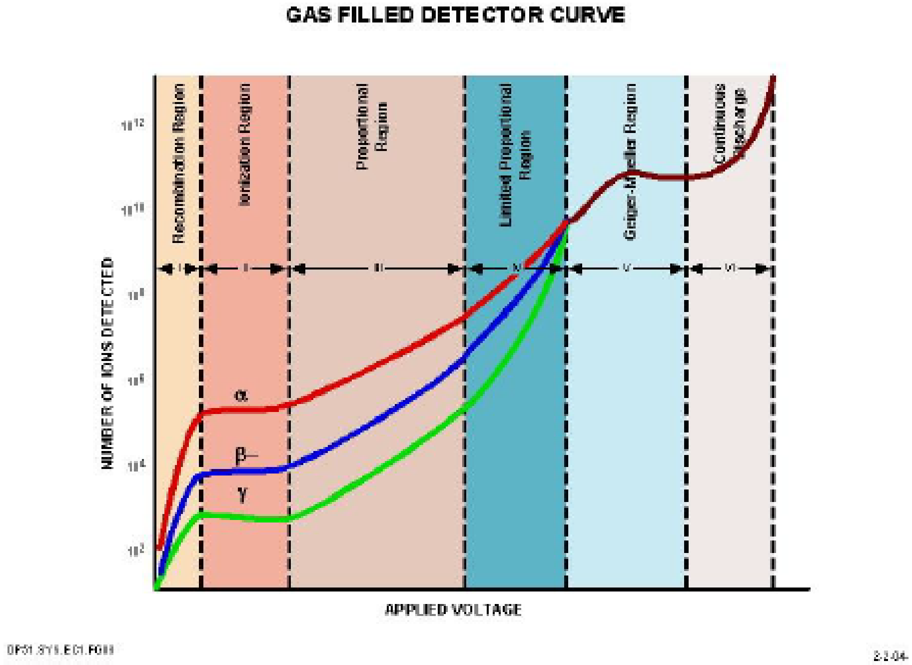


Figure 3 - C

The characteristic curve will vary for each detector design. Similarly the curve will change relative to the field intensity and the ionization level of the incident radiation in the lower regions. No shift will be seen in the Geiger-Mueller region where complete discharge occurs.

Incident charged particles or gamma radiation will cause ionization, but neutrons, uncharged particles, will not. Boron is added in gas or solid form to cause a neutron interaction. This results in charged products, which will then cause the desired ionization.

$$ {}_0n^1 + {}_5B^{10} \rightarrow {}^*[{}_5B^{11}] \rightarrow {}^{+++}{}_3Li^7 + {}^{++}{}_2He^4 + \text{Energy} $$

The boron gas detector (**Figure 4**) contains BF_3 gas and operates in the proportional range. Highly sensitive, it detects both gammas and neutrons thermalized in a polyethylene cover. Gamma signals are smaller in pulse amplitude and therefore can be discriminated (removed) in the electrical circuit (discussed in next section).

Comments / Reference: ABN-701	Revision: 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-701
SOURCE RANGE INSTRUMENT MALFUNCTION	REVISION NO. 12	PAGE 4 OF 12

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION:
- Removing Source Range (SR) control power fuses will result in a Reactor Trip even with the LEVEL TRIP switch in BYPASS, unless above P-6 AND manually blocked OR unless above P-10.
- Removing SR instrument power fuses will result in a Reactor Trip unless LEVEL TRIP switch in BYPASS, OR above P-6 AND manually blocked, OR above P-10.

NOTE:
- The higher count rate indication is assumed to be valid until an ACOT, CHANNEL CALIBRATION, or Intermediate Range cross-reference has confirmed otherwise. With both N-31 and N-32 inoperable, source range neutron flux may be monitored by observing u-NI-50A-2 and u-NI-50B-2.
- See TR 13.3.32 for Gamma-Metrics Flux Monitoring System applicability.

	Step	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐	1	VERIFY NO Core Alterations, RCS Dilution, OR Reactor Startup in progress.	IF less than P-6 AND less than two required SR channels operable, THEN SUSPEND ALL core alterations AND positive reactivity additions.
☐	2	VERIFY at least one required SR channel operable.	REFER to Technical Specifications, listed in Section 5.1 of this procedure. IF below P-6, THEN COMPLETE Attachment 1 within applicable time limits.
☐	3	IF the audible count rate was in service prior to the failure, THEN RESTORE by switching to the other SR channel at the AUDIO COUNT RATE CHANNEL DRAWER.	
☐	4	RESTORE both required SR channels operable within 48 hours OR PERFORM Attachment 1 within the following hour.	

Section 2.3

Comments / Reference: TS 3.3.1	Revision: 156

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
5. Source Range Neutron Flux	2[e]	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.4 E5 cps
	3[b], 4[b], 5[b]	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.4 E5 cps
6. Overtemperature N-16	1,2	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1[q][r]
7. Overpower N-16	1,2	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 112.8% RTP[q][r]
8. Pressurizer Pressure					
a. Low	1[g]	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1863.6 psig (Unit 1) ≥ 1865.2 psig (Unit 2)
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2400.8 psig (Unit 1) ≤ 2401.4 psig (Unit 2)

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 2a, 2b, 6, 7, and 14 (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
(q) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(r) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint or a value that is more conservative than the Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint, the methodology used to determine the as-found tolerance and the methodology used to determine the as-left tolerance shall be specified in the Technical Specification Bases.

Comments / Reference: TS 3.3.1	Revision: 150

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Not used.		
I. One Source Range Neutron Flux channel inoperable.	------------------------NOTE------------------------ Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. --- I.1 Suspend operations involving positive reactivity additions.	 Immediately
J. Two Source Range Neutron Flux channels inoperable.	J.1 Open reactor trip breakers (RTBs).	Immediately
K. One Source Range Neutron Flux channel inoperable.	K.1 Restore channel to OPERABLE status. <u>OR</u> K.2.1 Initiate action to fully insert all rods. <u>AND</u> K.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	48 hours 48 hours 49 hours
L. Not used.		

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			1
	Group			2
	K/A	000036.G.2.4.49		
Level of Difficulty: 2	Importance Rating			4.4

Fuel-Handling Incidents: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Question # 91

Given the following conditions:

- Unit 2 in MODE 6 reloading core
- Control Room notified an irradiated fuel assembly has been dropped into the core
- ABN-908, Fuel Handling Accident, in progress

Per ABN-908, ...

the Fuel Handling Supervisor should ensure transfer cart is in the __(1)__ Building with Fuel Transfer Tube gate valve closed.

Containment Purge may have to be stopped to enable __(2)__.

A. (1) Fuel
(2) closing the Fuel Transfer Tube gate valve

B. (1) Fuel
(2) installation of the equipment hatch

C. (1) Containment
(2) closing the Fuel Transfer Tube gate valve

D. (1) Containment
(2) installation of the equipment hatch

Answer: B

K/A Match: K/A match due to requiring the ability to take action to control the event in accordance with ABN-908.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring knowledge of tasks performed by the Fuel Handling Supervisor which is an SRO position. It also requires knowledge of the content of the procedure and not just the overall mitigation strategy. Also due to 10 CFR 55.43(b)(7) Analysis requiring refueling floor SRO responsibilities.

Explanation:

A. Incorrect. First part is correct (see B). Second part is incorrect, but plausible since it may be thought that the DP between the Containment and Fuel buildings would prevent closing the fuel transfer tube gate valve.

B. Correct. First part is correct. Per ABN-908, Step 2.3.8, the Fuel Handling Supervisor has the specific responsibility to ensure the cart is in the Fuel Buildig. Second part is correct. A note in ABN-908 states it may be necessary to secure Containment Purge to enable installation of the equipment hatch.

C. Incorrect. First part is incorrect, but plausible since it may be thought that the transfer car should be left in Containment in preparation for putting the damaged assembly in the transfer car. Second part is part is incorrect, but plausible (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is correct (see B).

Technical Reference(s)	ABN-908	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: Given that a Fuel Handling accident is in progress, **SUMMARIZE** the expected control room response in accordance with ABN-908, “Fuel Handling Accident.” (OPD1.G16.OB03)

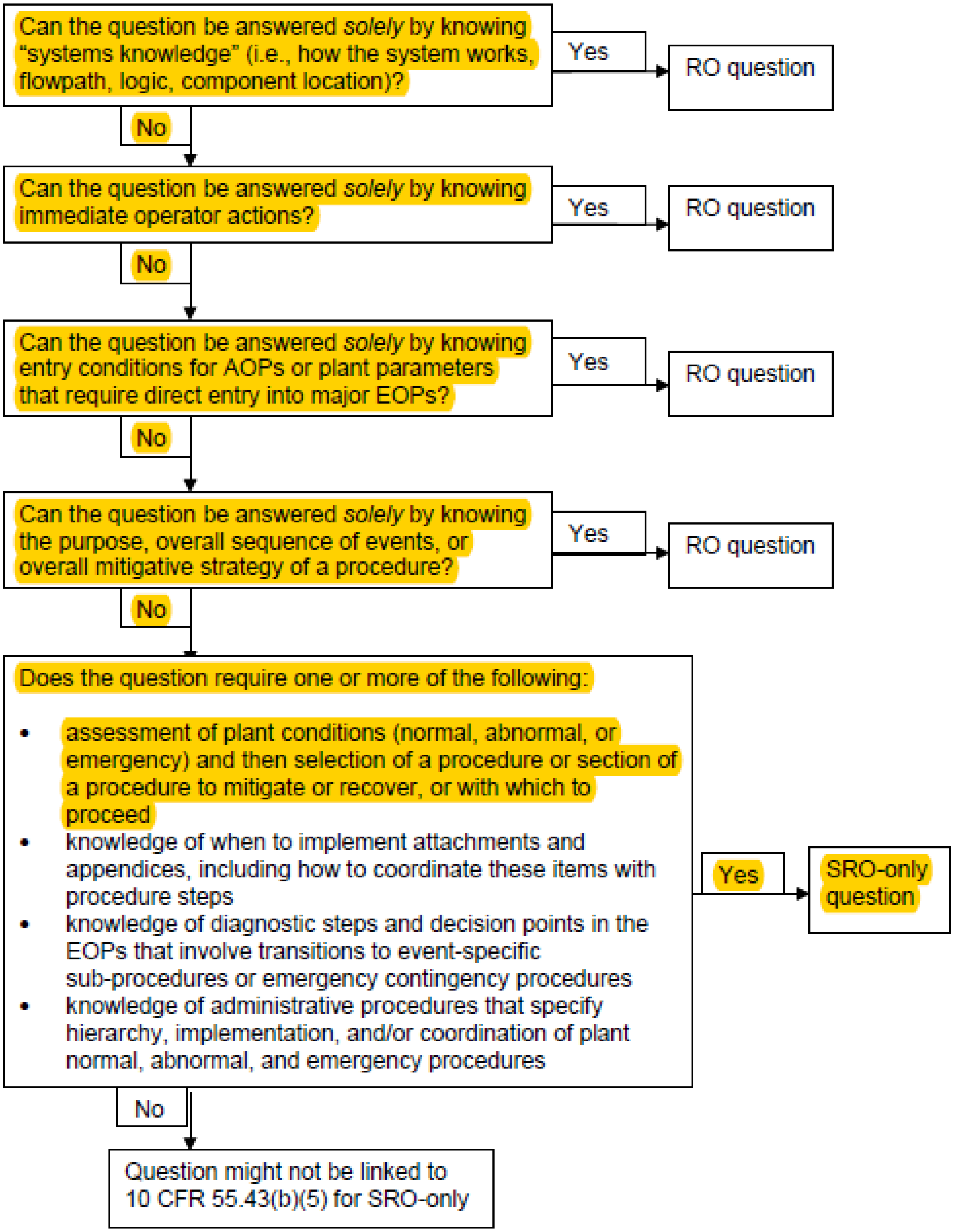
Question Source: Bank # 75869
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam LC24

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 5/7

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: Bank 75869	Revision:

- Unit 2 in MODE 6 reloading core
- Control Room notified an irradiated fuel assembly has been dropped into the core
- ABN-908, Fuel Handling Accident in progress

Per ABN-908 ...

Fuel Handling Supervisor should ensure transfer cart is in the __________ Building with Fuel Transfer Tube gate valve closed.

Containment Purge may have to be stopped to enable ________________.

A. Fuel
closing the Fuel Transfer Tube gate valve

B. Fuel
installation of the equipment hatch

C. Containment
closing the Fuel Transfer Tube gate

D. Containment
installation of the equipment hatch

Answer: B

Answer Explanation

A. Incorrect. First part is correct (See B below). Second part is incorrect but plausible as the applicant may believe that the ?P between the Containment and Fuel buildings would prevent closing the fuel transfer tube gate valve.

B. Correct. First part is correct per ABN-908, Step 2.3.8; the Fuel Handling Supervisor has this specific responsibility. Second part is correct as a note in ABN-908 informs the user that it may be necessary to secure Containment Purge to enable installation of the equipment hatch.

C. Incorrect. First part is incorrect but plausible as the applicant may believe that the transfer car should be left in Containment in preparation for putting the damaged assembly in the transfer car. Second part is part is incorrect but plausible (See A above).

D. Incorrect. First part is incorrect but plausible (See C above). Second part is correct (See B above).

Comments / Reference: Bank 75869	Revision:

Question 90 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	75869
User-Defined ID:	ILOT9479
Cross Reference Number:	RFO.FH2.OB01.009
Topic:	Unit 2 in MODE 6 reloading core Control Room notified an irradiated fuel assembly has been dropped
K/A:	SF8 034 A2.01
Question Reference:	
SRO:	YES
Comments:	LC24 NRC K/A Match: The question is a K/A match as it requires the applicant to demonstrate the ability to take action to control the event in accordance with ABN-908. SRO Only: The question is SRO only in that it requires the applicant to demonstrate knowledge of tasks performed by the Fuel Handling Supervisor which is an SRO position. It also requires knowledge of the content of the procedure and not just the overall mitigation strategy.

Comments / Reference: ABN-908	Revision: 5

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-908
FUEL HANDLING ACCIDENT	REVISION NO. 5	PAGE 5 OF 15

2.3 Operator Actions

NOTE: Containment entry shall require Shift Manager authorization. Security should ensure all personnel have exited containment.

☐ 7 DIRECT Security to control access to containment.

NOTE: Personnel exiting Containment should proceed NO further into Safeguards Building than Containment Control Point UNTIL radiological monitoring is accomplished by Radiation Protection.

8 The Fuel Handling Supervisor in Containment should ensure the following as conditions allow, while taking appropriate precautions for any high radiation:

- ☐ INFORM personnel exiting Containment to assemble in controlled area outside Containment.

NOTE:
- Temporary storage of an assembly against core baffle locations is permissible if no stored assembly is face-adjacent to any other stored assembly and there is at least one open location between the core assemblies and all inward faces and the corners of the stored assembly.(RFO-106, Att. 8.B.)
- It may be necessary to stop Containment Purge to enable installation of the equipment hatch.

- ☐ ENSURE ALL fuel assemblies are stored in core or upender. IF core offload is in progress AND the fuel assembly is being stored temporarily in the core, THEN source range counts should be monitored to ensure counts do not increase.
- ☐ ENSURE NO loads are suspended from the manipulator crane.
- ☐ ENSURE upender is in horizontal position.
- ☐ ENSURE transfer car is in Fuel Building AND Fuel Transfer Tube gate valve is closed.
- ☐ ENSURE Containment Equipment Hatch installed with a minimum of 4 bolts.
- ☐ ENSURE all personnel are exiting Containment to Safeguards Building control point.

"Step continued next page"

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier			1
	Group			2
	K/A	WE14.EA2.01		
Level of Difficulty: 3	Importance Rating			3.8

High Containment Pressure: Ability to determine and interpret the following as they apply to the High Containment Pressure: Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Question # 92

Given the following conditions:

- Unit 1 LBLOCA
- EOS-1.3A, Transfer to Cold Leg Recirculation, entered at RWST LO-LO Level

Subsequently:

- An ECCS flowpath from the containment sump to the RCS could NOT be established
- Efforts to add makeup to the RWST in progress
- RWST level 18% lowering slowly
- Containment pressure 40 psig slowly rising
- CSP suction aligned to the RWST

(1) The US should operate CSPs per __(1)__.

(2) How many Containment Spray Pumps should be run for the stated conditions above?

A. (1) FRZ-0.1A, Response to High Containment Pressure
(2) Operate TWO CSPs until containment pressure less than 18 psig or RWST level is ≤ 9%

B. (1) ECA-1.1A, Loss of Emergency Coolant Recirculation
(2) Operate TWO CSPs until containment pressure less than 18 psig or RWST level is ≤ 9%

C. (1) FRZ-0.1A, Response to High Containment Pressure
(2) Operate NO CSPs to conserve RWST level

D. (1) ECA-1.1A, Loss of Emergency Coolant Recirculation
(2) Operate NO CSPs to conserve RWST level

Answer: B

K/A Match: K/A match due to requiring knowledge of procedures to transition to, as well as implementation of the detailed steps of the procedure.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. Also requires knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures.

Explanation:

A. Incorrect. First part is incorrect, but plausible since an orange path on Containment does exist and FRZ-0.1A implementation may be required (if not already implemented). Second part is correct (see B).

B. Correct. First part is correct, ECA-1.1A would be the procedure governing CSP operation regardless of procedure currently being implemented (FRZ-0.1A or ECA-1.1A). If examinee thought FRZ-0.1A was required due to the orange path, then step 4.d of FRZ-0.1A states to operate CSPs per ECA-1.1A. Second part is correct, with Containment pressure above 18 psig but less than 50 psig, two CSPs would be run until Containment pressure lowered to less than 18 psig or RWST level lowered to ≤ 9% (Empty)

C. Incorrect. First part is incorrect, but plausible (see A). Second part is incorrect, but plausible since ECA-1.1 would require no CSPs be run if RWST level were below 9% or if pressure were below 18 psig.

D. Incorrect. First part is correct (see B). Second part is incorrect, but plausible (see C)

Technical Reference(s)	ECA-1.1	Attached w/ Revision # See Comments / Reference
	FRZ-0.1	

Proposed references to be provided during examination: ____________

Learning Objective: **STATE** the bases for the procedure steps, NOTEs and CAUTIONs contained in FRZ-0.1A/B. (ERG.FZ1.OB04)

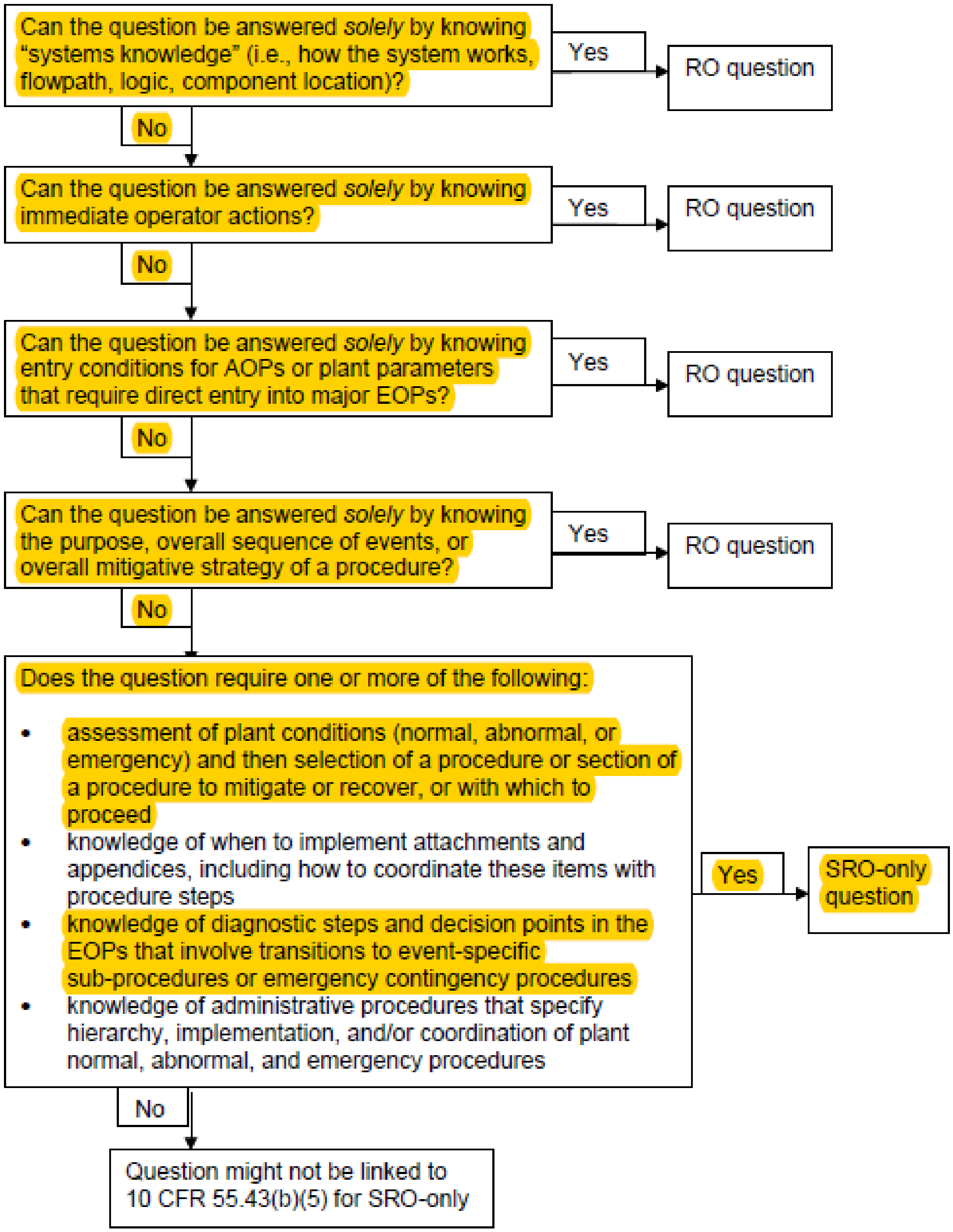
Question Source: Bank # ____
Modified Bank # 52583 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: Bank 52583	Revision:

- Unit 1 LBLOCA
- EOS-1.3A, Transfer to Cold Leg Recirculation was entered at RWST LO-LO Level
- An ECCS flowpath from the containment sump to the RCS could NOT be established
- ECA-1.1A, Loss of Emergency Coolant Recirculation in progress
- Efforts to add makeup to the RWST are in progress
- RWST level is 8% and stable
- Containment pressure has just risen from 38 psig to 52 psig
- CSP suction aligned to the RWST

Which of the following describes the expected procedure use and actions necessary based on the given conditions?

A. Transition to FRZ-0.1A, Response to High Containment Pressure and operate all CSPs until containment pressure is less than 50 psig or RWST level is 6%.

B. Remain in ECA-1.1A, Loss of Emergency Coolant Recirculation and operate all CSPs until containment pressure is less than 50 psig or RWST level is 6%.

C. Transition to FRZ-0.1A, Response to High Containment Pressure and operate NO CSPs in accordance with ECA-1.1A, to conserve RWST level.

D. Remain in ECA-1.1A, Loss of Emergency Coolant Recirculation and operate NO CSPs in accordance with ECA-1.1A, to conserve RWST level.

Answer: C

Answer Explanation

Comments / Reference: Bank 52583	Revision:

I. Incorrect. Plausible because entry into FRZ-0.1A is required but running all 4 containment spray pumps is not IAW with the procedural guidance provided in ECA-1.1A Step 10 or Step 32.
J. Incorrect. Plausible because it could be thought that with RWST level below EMPTY (9%) and the fact that FRZ-0.1A had already been implemented that re-entering FRZ-0.1A is not required however ERG rules of usage do require entering FRZ-0.1A on the RED path and ECA-1.1A Step 10 and Step 32 requires NO containment spray pumps to be in operation when the RWST level is below Empty.
K. Correct. The RED path entry criteria requires re-entering FRZ-0.1A and ECA-1.1A Step 10 and Step 32 dictates running NO containment spray pumps based on RWST level being below Empty.
L. Incorrect. Plausible because NO containment spray pumps in operation is required by ECA-1.1A Step 10 and Step 32 but the transition to FRZ-0.1A is required due to the RED path.

Question 193 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	4.00
System ID:	52583
User-Defined ID:	ILOT1570
Cross Reference Number:	ERG.C11.OB06.003
Topic:	Unit 1 LBLOCA EOS-1.3A, Transfer to Cold Leg Recirculation was entered at RWST LO-LO Level An ECCS
K/A:	4.5E11.EA2.2
Question Reference:	
SRO:	Yes
Comments:	LC22 Audit; S24E24, S25E24, S26E24 Ref: ECA-1.1A Steps 10 & 32; FRZ-0.1A Step 4d; ODA-407 Att. 8A

Comments / Reference: ECA-1.1	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.1A
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 9	PAGE 6 OF 83

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

11 Determine Containment Spray Requirements (Suction From RWST):

a. Containment spray pump suction - ALIGNED TO RWST

a. IF containment spray pump suction aligned to sump, THEN go to Step 13.

b. Determine number of containment spray pumps required from Table 1.

TABLE 1

RWST LEVEL	CONTAINMENT PRESSURE	SPRAY PUMPS REQUIRED
GREATER THAN RWST EMPTY	GREATER THAN 50 PSIG	4
	BETWEEN 18.0 PSIG AND 50 PSIG	2
	LESS THAN 18.0 PSIG	0
LESS THAN RWST EMPTY	-	0

c. Containment spray pumps running - EQUAL TO NUMBER REQUIRED

c. Manually operate containment spray pumps as necessary.

IF containment spray pumps are stopped, THEN close the affected train's containment spray heat exchanger outlet valve.

Comments / Reference: FRZ-0.1	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRZ-0.1A
RESPONSE TO HIGH CONTAINMENT PRESSURE	REVISION NO. 9	PAGE 4 OF 26

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: Component Cooling Water supply to the unit instrument air compressors isolates on a Phase B isolation signal.	
4	Check If Containment Spray Is Required:	
	a. Containment pressure - HAS INCREASED TO GREATER THAN 18.0 PSIG • 1-ALB-2B window 1-8, CS ACT - ILLUMINATED -OR- • 1-ALB-2B window 4-11 CNTMT ISOL PHASE B ACT - ILLUMINATED -OR- • Containment pressure - GREATER THAN 18.0 PSIG	a. Return to procedure and step in effect.
	b. Verify all RCPs - STOPPED	b. Manually stop all RCPs.
	c. Verify Containment Isolation Phase B Valves- CLOSED • Verify 1-MLB-4A3 and 4B3 - ORANGE LIGHTS LIT	c. Manually actuate Phase B. IF valve(s) NOT closed, THEN manually close valve(s). (Refer to Attachment 5)
	d. Verify ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION is NOT in effect.	d. Operate containment spray per ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION. Go to Step 5.

-CONT 4-

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 3	Tier			1
	Group			2
	K/A	WE07.G.2.4.20		
Level of Difficulty: 3	Importance Rating			4.3

Saturated Core Cooling: Knowledge of the operational implications of EOP warnings, cautions, and notes.

Question # 93

Given the following conditions:

- SGTR occurred on Unit 1
- Due to equipment failures, ECA-3.2A, SGTR with a Loss of Reactor Coolant-Saturated Recovery Desired, being performed
- The STA informs you the CSFSTs all GREEN with the exception of the following:
  - CORE COOLING CSFST YELLOW based on Loss of Subcooling
  - INVENTORY CSFST YELLOW based on Reactor Vessel Level

Which of the following describes the implementation of procedures for this event?

A. Address both CSFST YELLOW paths, as desired based on operator judgement.

B. Do NOT address either CSFST YELLOW path as implementation is NOT allowed in the ECA procedures.

C. Address CORE COOLING actions, as desired based on operator judgement, and do NOT perform the actions for INVENTORY due to conflict with ECA-3.2A actions.

D. Address INVENTORY actions, as desired based on operator judgement, and do NOT perform the actions for CORE COOLING due to conflict with ECA-3.2A actions.

Answer: D

K/A Match: K/A match due to requiring knowledge of cautions stated in procedure ECA-3.2.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. Also requires knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures.

Explanation:

A. Incorrect. Plausible because normally YELLOW path procedures are addressed at the discretion of the US, but CORE COOLING actions conflict with ECA-3.2 actions and should not be performed.

B. Incorrect. Plausible because CORE COOLING actions conflict with ECA-3.2 actions and should not be performed, but INVENTORY actions could be taken.

C. Incorrect. Plausible since actions from one of these procedures is not to be performed due to conflicts with ECA-3.2 actions, but the conflicting procedure addresses CORE COOLING.

D. Correct. Per the guidance contained in FRC-0.3A, Response to Saturated Core Cooling. FRC-0.3A directs a reestablishment of RCS subcooling via Safety Injection Flow. This is inconsistent with the actions of ECA-3.2A which reduce RCS subcooling via ECCS flow reduction in order to minimize primary to secondary leakage during a SGTR.

Technical Reference(s)	FRC-0.3	Attached w/ Revision # See Comments / Reference
	OCA-407	

Proposed references to be provided during examination: __________

Learning Objective: **STATE** the bases for operator actions, notes and cautions from ECA-3.2. (ERG.C31.OB12)

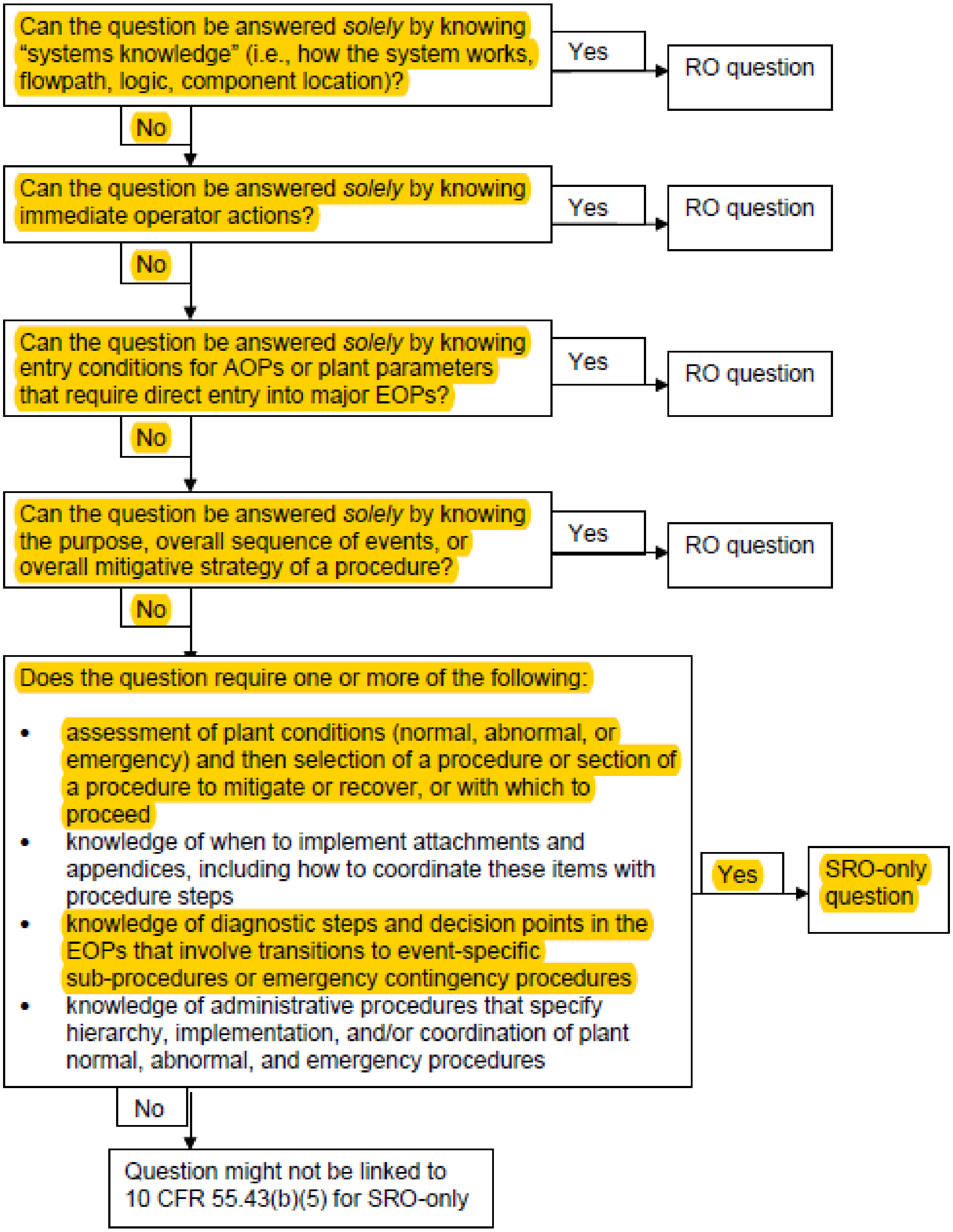
Question Source: Bank # 18539
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: ODA-407	Revision: 17

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 17	PAGE 27 OF 63
	INFORMATION USE	

ATTACHMENT 8.A
PAGE 9 OF 25

ERG RULES OF USAGE

10. • If an FRG is in progress due to an ORANGE priority condition and then the CSFST status for that same procedure goes to a RED priority condition, the operating crew should continue in the procedure from the current step. The procedure actions are the same regardless of color status (e.g., RED or ORANGE priority for FRS-0.1A/B, FRP-0.1A/B, FRZ-0.1A/B based on Containment pressure); therefore, recovery actions should proceed from the current step through completion to the point of a defined transition.

• YELLOW FRG status implementation is based on operator judgement when it is determined that adequate time exists to implement the procedure. The operator does not have to implement a YELLOW condition FRG if a judgement has been made that it is inappropriate based on available time or current plant status; and, if an event of higher priority is in progress, the operator should attend to the more important matters prior to implementing a YELLOW condition FRG. In the prioritization scheme of the ERGs, the ORGs (including applicable foldout pages) have priority over YELLOW path FRG(s). While performing actions of a YELLOW condition, continuous actions or foldout page items of the ORG in effect are still applicable and should be monitored and implemented by the operator. In some cases the YELLOW status might provide an early indication of a developing RED or ORANGE condition.

• In general, performance of the FRGs is dependent on current plant parameters. If a RED or ORANGE priority condition comes in and clears before FRG implementation is initiated, the FRG need not be performed. If conditions degrade, the safety function status will become a continuous RED or ORANGE condition; at which time, the operator will be directed to the appropriate FRG.

  An exception to this rule is made for entry into FRZ-0.1A/B after transition out of EOP-0.0A/B. The corresponding containment pressure for an ORANGE priority condition of FRZ-0.1A/B is also the Containment Spray initiation setpoint; thus, the containment pressure value impacts FRG status and implementation. The following provides a summary of requirements for FRZ-0.1A/B.

Comments / Reference: FRC-0.3	Revision: 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.3A
RESPONSE TO SATURATED CORE COOLING	REVISION NO. 9	PAGE 3 OF 11

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: If ECA-3.2A, SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED, is in effect, this procedure should not be performed.	
* 1	Check RWST Level - GREATER THAN LO-LO LEVEL	Go to EOS-1.3A, TRANSFER TO COLD LEG RECIRCULATION.
2	Check RHR System Status:	
	a. RHR System - HAS BEEN PLACED IN SERVICE FOR COOLDOWN	a. Go to Step 3.
	b. Go to ABN-104, RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION.	
3	Verify ECCS Flow:	
	a. CCP safety injection flow indicator - CHECK FOR FLOW	a. Start pumps and align valves as necessary.
	b. SI pump flow indicators - CHECK FOR FLOW	b. Start pumps and align valves as necessary.
	c. RCS pressure - LESS THAN 325 PSIG (425 PSIG FOR ADVERSE CONTAINMENT)	c. Go to Step 4.
	d. RHR pump flow indicators - CHECK FOR FLOW	d. Start pumps and align valves as necessary.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier			3
	Group			
	K/A	2.1.35		
Level of Difficulty: 3	Importance Rating			3.9

Knowledge of the fuel-handling responsibilities of SROs.

Question # 94

Which of the following is a direct responsibility of the Fuel Handling Supervisor during CORE ALTERATIONS in accordance with RFO-101, Refueling Organization.

A. Performing fuel assembly and insert component inspections.

B. Providing approved Fuel Shuffle Sequence Sheets to the Refueling Crew.

C. Ensuring new fuel assemblies to be loaded into the reactor have been properly processed and inspected and are available for loading.

D. Ensuring direct communications are maintained between the refueling area in Containment, the fuel building, and the Control Room.

Answer: D

K/A Match: K/A match due to requiring knowledge of direct responsibilities of the refueling SRO.

SRO Only: SRO Only due to 10 CFR 55.43(b)(7) Analysis requiring refueling floor SRO responsibilities.

Explanation:

A. Incorrect. Plausible because Performing fuel assembly and insert component inspections is a responsibility contained in RFO-101 but it is for the Core Performance Manager not the Fuel Handling Supervisor.

B. Incorrect. Plausible because providing approved Fuel Shuffle Sequence Sheets to the Refueling Crew is a responsibility contained in RFO-101 but it is for the Core Performance Manager not the Fuel Handling Supervisor.

C. Incorrect. Plausible because ensuring new fuel assemblies to be loaded into the reactor have been properly processed and inspected and are available for loading is a responsibility contained in RFO-101 but it is for the Core Performance Manager not the Fuel Handling Supervisor.

D. Correct. Maintaining direct communication between the refueling area in Containment, the fuel building, and the Control Room during core alterations is a direct responsibility of the Fuel Handling Supervisor per RFO-101.

Technical Reference(s)	RFO-101	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ____________

Learning Objective: Given RFO-101, Refueling Organization, **DISCUSS** the Precautions, Limitations and major procedural actions IAW RFO-101. (OPD1.G16.OB01)

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

Question History: Last NRC Exam ____________

Question Cognitive Level:
- Memory or Fundamental Knowledge X
- Comprehension or Analysis ____

10 CFR Part 55 Content:
- 55.41 ____________
- 55.43 7

Comments / Reference: RFO-101	Revision: 9

CPNPP STATION REFUELING MANUAL		PROCEDURE NO. RFO-101
REFUELING ORGANIZATION	REVISION NO. 9 INFORMATION USE	PAGE 4 OF 11

5.0 Responsibilities During Refueling Operations

5.1 Shift Operations Manager

- Designating qualified individuals to act as Fuel Handling Supervisors during a refueling outage.

5.2 Refuel Work Window Manager

- Coordinating and organizing qualified personnel for the Operations Refueling Crews.
- Coordinate efforts of contract personnel used to support Operations Refueling Crews.
- Provide oversight of the Control Room, Containment, and Fuel Building activities at least once per shift during core alterations.
- During core reload, provide oversight of shoehorn operations.
- Coordinate additional oversight, particularly in the Fuel Building, with vendor supervision, off-duty Fuel Handling Supervisors, and other personnel as necessary.

[C] 5.3 Fuel Handling Supervisor (SRO) [00860]

- Providing direct supervision of fuel handling activities and core alterations in the reactor core and refueling cavity.
- Suspending Core Alterations and/or initiating a Containment evacuation if, in his judgment, any conditions exist which threaten personnel safety or safety of the fuel.
- Ensuring that direct communications are maintained between the refueling area in Containment, the fuel building, and the Control Room when CORE ALTERATIONS are in progress.
- Coordinating fuel movement in the Containment and the Fuel Building in accordance with RFO-106.
- Overall responsibility for fuel movement.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			3
	Group			
	K/A	2.1.36		
Level of Difficulty: 3	Importance Rating			4.1

Knowledge of procedures and limitations involved in core alterations.

Question # 95

Given the following conditions:

- Unit 1 performing an off-load of the core during refueling outage
- ODA-308-3.9.0-S01, Refueling Special Condition Surveillances, Section 2, Surveillances Required for Core Alterations/Movement of Irradiated Fuel being implemented with the following information:
    - One SR detector OPERABLE and the other inoperable
    - One Gamma-Metrics SR monitor OPERABLE and the other inoperable

(1) Which of the following describes the operability status of the Nuclear Instrumentation?

(2) Refueling cavity water level OPERABLITY requires at least 23 feet above the __(2)__.

A. (1) Nuclear Instrumentation OPERABILITY is met
(2) Reactor Vessel flange

B. (1) Nuclear Instrumentation OPERABILITY is met
(2) top of the irradiated fuel assemblies

C. (1) Nuclear Instrumentation OPERABILITY is NOT met
(2) Reactor Vessel flange

D. (1) Nuclear Instrumentation OPERABILITY is NOT met
(2) top of the irradiated fuel assemblies

Answer: A

K/A Match: K/A match due to

SRO Only: SRO Only due to 10 CFR 55.43(b)(2) Analysis requiring knowledge of TS bases that is required to analyze TS-required actions and terminology and 10 CFR 55.43(b)(6) Analysis requiring knowledge of TS bases for reactivity controls.

Explanation:

A. Correct. First part is correct (see B). Second part is correct. Cavity level must be 23 feet above the vessel flange.

B. Incorrect. First part is correct, any 2 of the 4 NIs, consisting of the SR NIs and the Gamma Metrics, are required to be operable. Second part is incorrect, but plausible as level would be required to be 23 feet above the fuel if the requirement was in the fuel building.

C. Incorrect. First part is incorrect, but plausible since until recently the operability of the nuclear instrumentation for refueling was based on either the SR NIs or the Gamma Metrics being operable in pairs, such that both of one set were required. Second part is correct (see A).

D. Incorrect. First part is incorrect, but plausible (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	TS/B 3.9.3	Attached w/ Revision # See Comments / Reference
	TS/B 3.9.7	

Proposed references to be provided during examination: ______

Learning Objective: Given access to Technical Specifications and a refueling activity, **APPLY** the appropriate specification. (OPD1.G16.OB04)

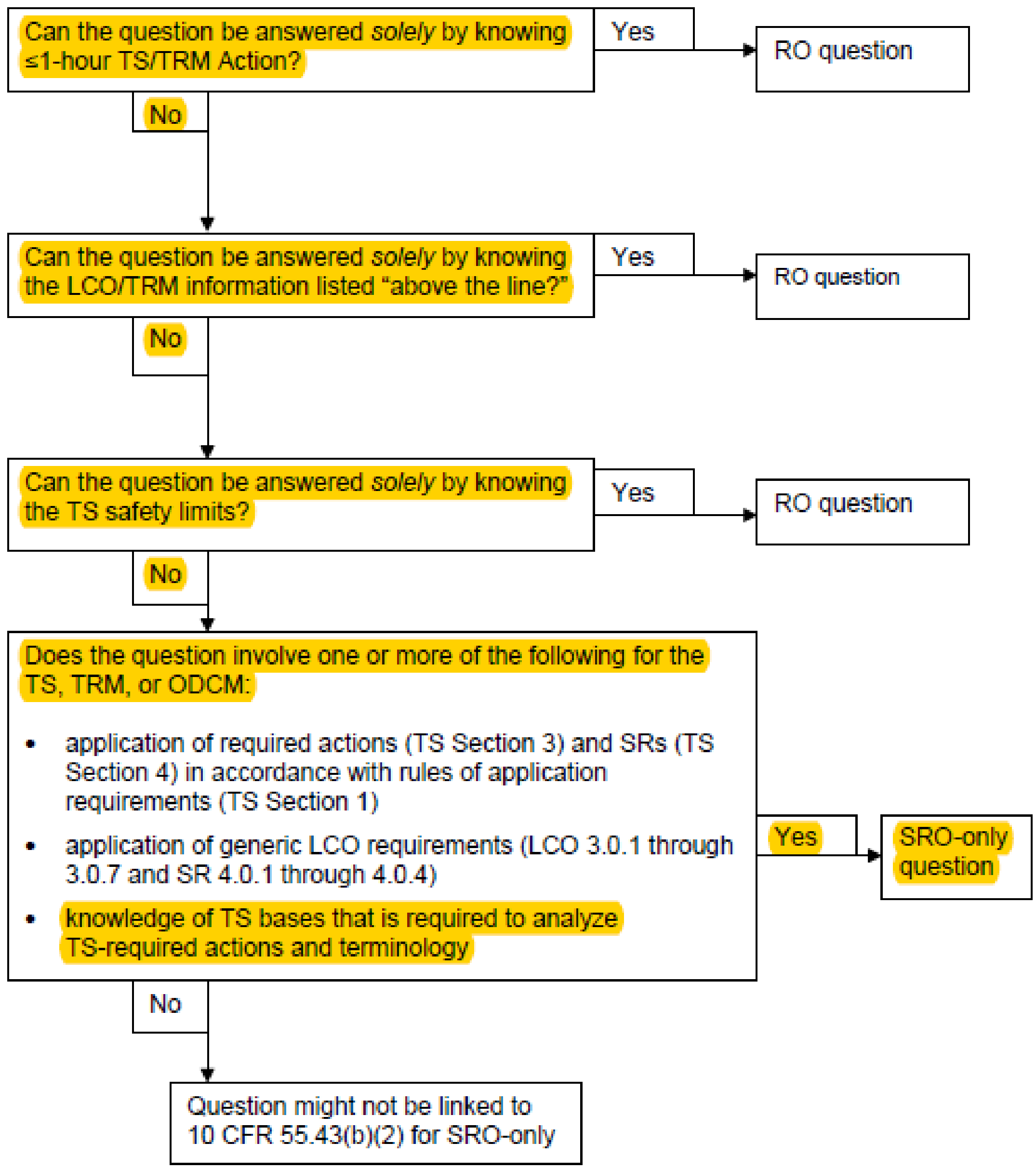
Question Source: Bank # ______
Modified Bank # 52607 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 2/6

Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:



Comments / Reference: Bank 52607	Revision:

- Unit 1 is preparing to commence off-load of the core for a refueling outage.
- ODA-308-3.9.0-S01, Refueling Special Condition Surveillances, Section 2, Surveillances Required for Core Alterations/Movement of Irradiated Fuel is being implemented with the following information:
  - Source Range Neutron Flux Monitor 1-NI-31 is OPERABLE.
  - Source Range Neutron Flux Monitor 1-NI-32 is inoperable.
  - Gamma-Metrics Source Range Flux Monitors 1-NI-50A and 1-NI-50B are OPERABLE.
  - Refueling Cavity water level is 856.5 feet and stable.

Which of the following describes:

Whether or not the core off-load may commence or NOT?

The reason for making this decision?

A. Core off-load may commence
Nuclear Instrumentation OPERABILITY is met

B. Core off-load may NOT commence
Nuclear Instrumentation OPERABILITY is NOT met

C. Core off-load may commence
Refueling cavity water level OPERABILITY is met

D. Core off-load may NOT commence
Refueling cavity water level OPERABILITY is NOT met

Answer: D

Answer Explanation

Comments / Reference: Bank 52607	Revision:

A. Plausible because it could be thought that the Gamma-Metrics pair of monitors may be used for core off-load however refueling cavity water level must be ≥ 857.5 feet.

B. Plausible because it could be thought that both N31 and N32 must be OPERABLE to off-load the core however the Gamma-Metrics pair of monitors may be used for core off-load.

C. Plausible because it could be thought that refueling cavity level only needs to be ≥ 846 feet based on 23 feet above the top of the fuel assemblies however level must be 23 feet above the reactor vessel flange (≥ 857.5 feet) for CORE ALTERATIONS.

D. Core off-load may NOT begin because to move irradiated fuel assemblies in containment (CORE ALTERATIONS) water level must be ≥ 857.5 feet.

Question 541 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	3.00
System ID:	52607
User-Defined ID:	ILOT1586
Cross Reference Number:	ADM.XA1.OB01.057
Topic:	Unit 1 is preparing to commence off-load of the core for a refueling outage. ODA-308-3.9.0-S01, Re
K/A:	G.2.1.36
Question Reference:	
SRO:	Yes
Comments:	LC22 Audit; S23E31 (Admin); S24E31 (Admin); S25E31 (Admin) Ref: TS 3.9.3; TR 13.9.35; ODA-308-3.9.0-S01 p. 3 & 7

Comments / Reference: TS 3.9.3	Revision: 150

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

Comments / Reference: TSB 3.9.3	Revision: 78

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. Any two of four source range neutron flux monitors may be used.

The installed Westinghouse BF_3 source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). The installed source range neutron flux monitors are BF_3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps). The detectors also provide continuous visual indication in the control room. The NIS is designed in accordance with the criteria presented in Reference 1. Each portion of the Westinghouse source range neutron flux monitors has two trains and each is assigned to an independent Class 1E electrical train. These trains are physically and electrically separated in accordance with applicable IEEE Standards.

A separate Gamma-Metrics Neutron Flux Monitoring System (NFMS) is installed to satisfy the requirements of Regulatory Guide 1.97, "Instrumentation For Light-Watered-Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident." The Gamma-Metrics NFMS monitors neutron flux from the source range through 200% Rated Thermal Power (RTP) during all Modes of plant operation. This system utilizes two separate Safety Category I (Class 1E) fission chamber neutron detectors for all ranges of neutron flux indication. Each portion of the Gamma-Metrics instrumentation has two trains and each is assigned to a separate Class 1E electrical train. These trains are physically and electrically separated in accordance with applicable IEEE Standards.

The source range neutron flux monitors do not provide a Reactor Protection System function in Mode 6.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a visual signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly.

The source range neutron flux monitors satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

(continued)

Comments / Reference: TS 3.9.7	Revision: 156

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	In accordance with the Surveillance Frequency Control Program.

Comments / Reference: TSB 3.9.7	Revision: 78

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies, within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3 and acceptance in Reference 6.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.195 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain the following fractions of the total fuel rod inventory (Ref. 1):

0.08 for I-131, 0.10 for Kr-85, 0.05 for all other iodines and noble gases.

The fuel handling accident analysis is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time as described in the Technical Requirements Manual (Ref. 7) prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 4, 5 and 6).

Refueling cavity water level satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.

(continued)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			3
	Group			
	K/A	2.2.11		
Level of Difficulty: 2	Importance Rating			3.3

Knowledge of the process for controlling temporary design changes.

Question # 96

In accordance with STA-602, Temporary Modifications,

(1) Who is responsible for performing walkdowns of active Temporary Modifications, and

(2) How often is the walkdown required to be performed?

A. (1) Work Control Operations Supervisor
(2) Monthly

B. (1) Work Control Operations Supervisor
(2) Quarterly

C. (1) System Engineer
(2) Monthly

D. (1) System Engineer
(2) Quarterly

Answer: D

K/A Match: K/A match due to requiring knowledge of the procedures associated with Temporary Modifications.

SRO Only: SRO Only due to 10 CFR 55.43(b)(3) requiring knowledge of administrative processes for temporary modifications.

Explanation:

A. Incorrect. First part is incorrect, but plausible since this individual maintains and files copies of Temporary Modifications including clearances. Second part is incorrect, but plausible as monthly is a reasonable period of time compared to quarterly.

B. Incorrect. First part is incorrect, but plausible (see A). Second part is correct (see D).

C. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see A).

D. Correct. First part is correct, per STA-602, 5.7.2, the System Engineer is responsible for performance of a quarterly walkdown of TMs associated with their system to ensure proper installation. Second part is correct, the walkdown is required quarterly.

Technical Reference(s)	STA-602	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **DESCRIBE** the administrative requirements for operating plant equipment; performing routine watchstanding evolutions and maintaining system status and plant configuration control in accordance with ODA-106, STI-604.03, OWI-207, ODA-102, ODA-410, ODA-407, OWI-107, STA-694, STA-601 and OWI-409. (ADM.XA1.OB09)

Question Source: Bank # ______
Modified Bank # 35501 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 3

Comments / Reference: Bank 35501	Revision:

In accordance with STA-602, Temporary Modifications, who is responsible for performing walkdowns of active Temporary Modifications, and how often is the walkdown performed?

A. Work Control Operations Supervisor; monthly

B. System Engineer; monthly

C. Temporary Modification Coordinator; quarterly

D. Shift Manager; quarterly

Answer: C

Answer Explanation	
A.	Incorrect. Plausible since this individual maintains and files copies of Temporary Modifications including clearances. Action performed on a quarterly basis.
B.	Incorrect. Plausible since this individual reviews and concurs with Temporary Modification requests. Also coordinates, plans, designs, installs, tests and restores from Temporary Modifications. Action performed on a quarterly basis.
C.	Correct. In accordance with STA-602, Step 6.9.2.
D.	Incorrect. Plausible since this individual maintains active and installed Temporary Modification files. Reviews and approves installation of Temporary Modifications.

Comments / Reference: Bank 35501	Revision:

Question 288 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	35501
User-Defined ID:	ILOT8151
Cross Reference Number:	ADM.XA1.OB02.023
Topic:	In accordance with STA-602, Temporary Modifications, who is responsible for performing walkdowns o
K/A:	G.2.2.11
Question Reference:	
SRO:	Yes
Comments:	LC16 NRC REF: STA-602

Comments / Reference: STA-602	Revision: 20

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-602
TEMPORARY MODIFICATIONS AND TRANSIENT EQUIPMENT PLACEMENTS	REVISION NO. 20 INFORMATION USE	PAGE 10 OF 39

NOTE: Partial restoration is allowed ONLY if the "Partial Restoration Allowed" block of STA-602-16 form is marked "YES", the scope is specifically defined in STA-602-15 form and addressed in the Engineering evaluation. After the scope is identified and the Responsible Engineer block is signed to indicate Engineering approval, the Shift Manager block of STA 602-15 form should be signed to indicate Operations authorization of the partial restoration.

5.6.3 Authorizing restoration or partial restoration of TMs in accordance with STA-606 and Section 6.6 of this procedure and notifying the on-duty Reactor Operator of the TM restoration.

5.6.4 Ensuring marked up copies of drawings affected by the TM are attached to the Temporary Modification drawings sticks maintained in the Control Room.

[C] 5.6.5 The Shift Manager shall ensure that audits of the TM drawing sticks are performed as determined by the Shift Operations Manager, verifying that all affected drawings are attached. [25490]

5.7 The System Engineer is responsible for:

5.7.1 Reviewing and concurring with TM requests per Section 6.1 of this procedure, with the exception of leak repair TMs.

5.7.2 Being a point of contact (TM-owner) for the assigned TM within their system during its life cycle. This includes the performance of a quarterly walkdown to ensure the TM is installed as described in the TM Design Change and this procedure. The SE will have the responsibility to ensure proper coordination of planning, design, installation, testing, and restoration for the assigned TM.

5.7.3 Performing a post installation and pre-restoration review of the TM packages.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 2	Tier			3
	Group			
	K/A	2.2.12		
Level of Difficulty: 3	Importance Rating	4.1		

Knowledge of surveillance procedures.

Question # 97

While operating at 100% power, it is discovered a 31 day Surveillance Requirement on a piece of Tech Spec equipment was overlooked and has NOT been performed for over 6 weeks.

What is the proper course of action?

A. Delay calling the equipment INOPERABLE for up to 24 hours to allow performance of the Surveillance.

B. Declare the equipment INOPERABLE at the time of discovery and delay complying with the LCO ACTION(s) for up to 24 hours.

C. Declare the equipment INOPERABLE and enter LCO 3.0.3.

D. Delay calling the equipment INOPERABLE for up to 31 days to allow performance of the Surveillance.

Answer: D

K/A Match: K/A match due to requiring knowledge of surveillance procedure administrative requirements.

SRO Only: SRO Only due to 10 CFR 55.43(b)(2) Analysis requiring knowledge of generic LCO requirements (LCO 3.0.1 through 3.0.7 and SR 4.0.1 through 4.0.4).

Explanation:

A. Incorrect. Plausible since 24 hours would be the time frame if the surveillance period were less than 24 hours, but the greater interval is permitted.

B. Incorrect. Plausible since the surveillance is used to determine operability, but the LCO does not need to be implemented until the extended time granted by SR 3.0.3 expires.

C. Incorrect. Plausible since the surveillance is used to determine operability, but SR 3.0.3 permits delaying the surveillance.

D. Correct. If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is GREATER. This delay period is permitted to allow performance of the Surveillance.

Technical Reference(s)	TS SR 3.0.3	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **DELINEATE** the role of surveillances into Operability in accordance with the Technical Specifications. (ADM.XA5.OB04)

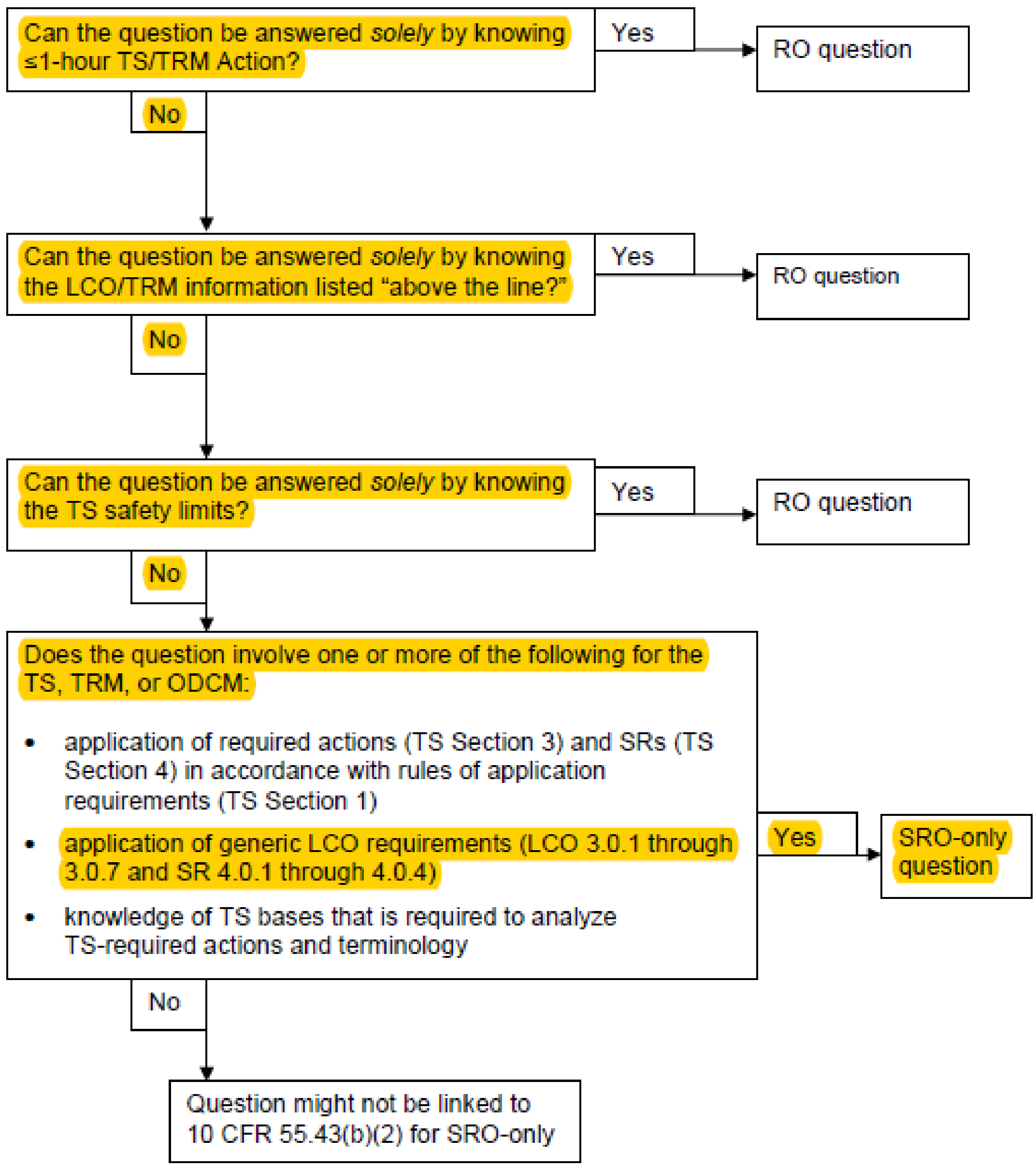
Question Source: Bank # 19387
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments / Reference: 10 CFR 55.43(b)(2) Analysis	Revision:



Comments / Reference: TS SR 3.0.3	Revision: 150

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			3
	Group			
	K/A	2.3.13		
Level of Difficulty: 2	Importance Rating			3.8

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question # 98

A clearance is to be implemented with the following conditions:

- Contact radiation on a piece of equipment is 5 R/hr
- Radiation level is 500 mrem/hr at 1 foot from the piece of equipment
- General area radiation for the room is 125 mrem/hr
- An operator must close and place a clearance tag on a valve which is located 8 feet away from the equipment
- It will take the operator 3 minutes to close the valve and place the clearance tag

The proper classification for the room is __(1)__.

A waiver of the independent verification of the tag placement is __(2)__ by the Shift Manager.

A. (1) High Radiation Area
(2) NOT allowed

B. (1) Radiation Area
(2) NOT allowed

C. (1) Radiation Area
(2) allowed

D. (1) High Radiation Area
(2) allowed

Answer: D

K/A Match: K/A match due to requiring knowledge of radiological safe practices to minimize total exposure to personnel and authority of Shift Manager regarding waivers due to radiation levels.

SRO Only: SRO Only due to 10 CFR 55.43(b)(4) Analysis requiring analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures.

Explanation:

A. Incorrect. First part is correct (see D). Second part is incorrect, but plausible (see B).

B. Incorrect. First part is incorrect, but plausible since a radiation area is defined as an area where an individual can receive a dose equivalent in excess of 5 mrem in 1 hour at 30 cm (12 inches) from the radiation source, but being in excess of 100 mrem in 1 hour upgrades the classification to a High Radiation Area. Second part is incorrect, but plausible since the total dose that the operator will receive is below threshold of 10 mrem to waive (dose received will be approximately 2.6 mrem total), however dose rate is sufficient to allow waiver.

C. Incorrect. First part is incorrect, but plausible (see B). Second part is correct (see D).

D. Correct. First part is correct. A high radiation area is defined as an area where an individual can receive a dose equivalent in excess of 100 mrem in 1 hour at 30 cm (12 inches) from the radiation source. Second part is correct. The dose rate of >100 mrem/hr allows this verification be waived. .halving the distance from a point source causes the intensity to increase by a factor of four. , although total exposure for verification is 8 mrem (160 mrem/hr for 3 minutes, 1/20th of hour), the dose rate of >100 mrem/hr allows this verification be waived.

Technical Reference(s)	Radiation Protection Practices LP	Attached w/ Revision # See Comments / Reference
	STA-694	
	STA-650	

Proposed references to be provided during examination: ____________________

Learning Objective: **DESCRIBE** the administrative requirements for operating plant equipment; performing routine watchstanding evolutions and maintaining system status and plant configuration control in accordance with ODA-106, STI-604.03, OWI-207, ODA-102, ODA-410, ODA-407, OWI-107, STA-694, STA-601 and OWI-409. (ADM.XA1.OB09)

Question Source: Bank # ____________
Modified Bank # 33141 (Note changes or attach parent)
New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____________
55.43 4

Comments / Reference: Bank 33141	Revision:

A surveillance is to be performed on a piece of equipment with the following conditions:

- Contact radiation on equipment is 5 R/hr
- Radiation level is 500 mrem/hr at 12 inches from equipment
- General area radiation for the room is 125 mrem/hr

Which of the following is the proper classification for the room?

A. Radiation Area

B. High Radiation Area

C. Locked High Radiation Area

D. Very High Radiation Area

Answer: B

Answer Explanation
A. Incorrect - Plausible as this is an area with > 5mr/hr
B. Correct - STA-650
C. Incorrect - Plausible as this is an area with > 1R/hr
D. Incorrect - Plausible as this is an area with >500R/hr

Question 57 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	33141
User-Defined ID:	ILOT
Cross Reference Number:	
Topic:	A surveillance is to be performed on a piece of equipment with the following conditions: Contact
K/A:	2.3.13
Question Reference:	STA-650
SRO:	
Comments:	R/S18E16

Comments / Reference: Radiation Protection Practices LP	Revision: 00-0001

NOTES	LESSON OUTLINE
	2) In accordance with the policy of minimizing total dose equivalent, respiratory protection equipment may be used to minimize internal exposure and protective clothing may be used to minimize skin contamination; and
	3) Limit exposures received under these conditions to once in a lifetime.
Objective 7	**E. Heat Stress**
	1. Heat Stress Action Time (AT)
	A time point at which a supervisor or observer initially asks for explicit confirmation from individual workers that they can safely continue to work. This should be repeated at intervals called Check Times (CT). The AT may be compared with Anticipated Work Time for job planning.
	2. Heat Stress Check Time (CT)
	CT will be either five (5) minutes, for AT of 30 minutes or less, or ten (10) minutes for AT greater than 30 minutes.
	3. Heat Stress Stay Time
	a. Two times the Action Time. The maximum time that workers will be allowed to continue working in the permitted work location.
	b. *Example: AT is 27 minutes - Stay Time would be 2×27 = 54 minutes.*
Objective 8	**F. Time/Distance/Shielding Calculations**
	1. Dose: Point source
	D = Distance DR = Dose rate $(DR_1)(D_1)^2 = (DR_2)(D_2)^2$
5000 mrem / 2000 hrs = 2.5 mr/DAC-hr	2. Airborne dose =
	(DACs)(time)(2.5mr/DAC-hr)/(p.f.) or (DAC-hrs)(2.5mr/DAC-hr)/(p.f.) p.f. = respirator protection factor
	3. Stay time =
	(Dose limit)/(Dose Rate)
	4. Shielding
	a. ½" lead blankets will reduce radiation levels by ~24% (RPI-608)

Comments / Reference: STA-650	Revision: 9

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4.3 Annual Limit on Intake (ALI) – means the derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference man that would result in a committed effective dose equivalent of 5 rems (0.05 Sv) or a committed dose equivalent of 50 rems (0.5 Sv) to any individual organ or tissue.

4.4 Contaminated Area – an area having smearable contamination equal to or greater than 1000 dpm/100 cm^2 (100 net counts per minute using a pancake frisker probe) beta-gamma or 20 dpm/100 cm^2 alpha.

4.5 DAC-Hour – the product of the concentration of radioactive material in air (expressed as a fraction or multiple of the derived air concentration for each radionuclide) and the time of exposure to the radionuclide, in hours. 2,000 DAC-hours equals one ALI.

4.6 Derived Air Concentration (DAC) – the concentration of a given radionuclide in air which, if breathed by the reference man for a working year of 2,000 hours under conditions of light work, results in an intake of 1 ALI. DACs are listed in 10CFR20, Appendix B, Table 2.

4.7 High Contamination Area (HCA) – An area where the majority of the area has removable surface contamination equal to or greater than 100,000 dpm/100 cm^2 beta-gamma.

4.8 High Radiation Area (HRA) – any area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem in 1 hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

4.9 Locked High Radiation Area (LHRA) – any area accessible to individuals in which deep dose equivalent rates are greater than or equal to 1 rem per hour (but less than 500 rads in one hour at 1 meter) 30 cm from the source of radiation or from any surface that the radiation penetrates.

4.10 Radiation Area – an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in 1 hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

4.11 Radioactive Material Area – any area in which licensed radioactive material in an amount exceeding 10 times the quantity specified in Appendix C, 10CFR20, is used or stored. This does not apply to radioactive materials contained within process equipment or materials in transport and packaged and labeled in accordance with appropriate regulations.

4.12 Radiologically Controlled Area (RCA) – an area within the restricted area posted in accordance with procedures of protecting individuals against undue risks from exposure to radiation and radioactive materials.

Comments / Reference: STA-694	Revision: 8

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6.6 Exceptions to Verification Performance

6.6.1 The Shift Manager may waive requirements for Independent Verification or Concurrent Verification under any of the following conditions:

- If significant radiation exposures (>10 mrem) are likely
- In areas of high radiation dose rates (>100 mrem/hr)
- In areas where other personnel hazards exist
- In Containment during MODES 1, 2, 3 or 4

6.6.2 Independent Verification and Concurrent Verification activities are not required during performance of, or in response to abnormal operating conditions. Examples include manipulation of plant systems prescribed by the Emergency Operating Procedures (ERGs), Abnormal Condition Procedures (ABNs) or Alarm Response Procedures (ALMs).

6.6.3 Verification activities should not routinely be required on a system components which are in an "out-of-service" condition or are being placed into an "out-of-service" condition.

Step 6.1.5.G describes circumstances that may warrant Independent Verification activities when components are placed in an out of service condition. |

6.6.4 Verification of a component position which has previously been aligned and verified in accordance with another station procedure (e.g., the locked valve program or a clearance) is not required.

The procedure used to establish that the component position has previously been aligned and verified should be reviewed to ensure the controls are in place (e.g., If a locked valve position is verified by the Locked Component List, the Locked Component Deviation Log should be reviewed to ensure the valve is not deviated.)

6.6.5 Vent, drain and test connection valve positions should be Verified on systems listed in Attachment 8.A during the initial lineup or when they are within the Containment isolation valve boundary described in Step 6.1.5.D. |

Subsequent to the initial lineup, Independent Verification activities on vent, drain or test connections are not required unless they are located within the Containment isolation valve boundary described in Step 6.1.5.D. |

6.6.6 Department Directors/Managers should make the final determination whether specific circumstances warrant additional exceptions to the requirements listed in Sections 6.2 or 6.3.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			3
	Group			
	K/A	2.4.22		
Level of Difficulty: 3	Importance Rating			4.4

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Question # 99

Given the following conditions:

- Unit 1 responding to a LBLOCA
- A CORE COOLING status tree ORANGE path causes a transition to FRC-0.2A, Response to Degraded Core Cooling
- During performance of FRC-0.2A, the CORE COOLING status tree changes from ORANGE to YELLOW
- An ORANGE path exists on the CONTAINMENT status tree
- FRZ-0.1A, Response to High Containment Pressure, is the procedure referenced by the CONTAINMENT status tree

Which of the following is the required action?

A. Complete FRC-0.2A and then go to FRZ-0.1A since CONTAINMENT is a lower priority path than CORE COOLING.

B. Go to FRZ-0.1A since an ORANGE path has a higher priority than a YELLOW path. Completion of FRC-0.2A is NOT needed.

C. Complete FRC-0.2A as entry into FRZ-0.1 is NOT needed since it would have been addressed by the performance of steps in EOP-1.0A.

D. Go to FRZ-0.1A since an ORANGE path has a higher priority than a YELLOW path. Return to complete FRC-0.2A after FRZ-0.1A is addressed.

Answer: A

K/A Match: K/A match due to requiring the ability to prioritize safety functions during an event.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Explanation:

A. Correct. Upon entry to a RED or ORANGE path FRG, the procedure is performed to completion unless a higher RED or ORANGE path procedure is encountered, even if the path clears or becomes YELLOW before completion. Since CORE COOLING is a higher priority than CONTAINMENT, FRC-0.2 is completed before FRZ-0.1 is addressed.

B. Incorrect. Plausible since if these conditions had been identified prior to entering FRC-0.2, then FRZ-0.1 would be the higher priority, but incorrect since FRC-0.2 is to be completed even though it has turned YELLOW.

C. Incorrect. Plausible since entry into FRZ-0.1 would not be required if the steps were performed in EOP-1.0 and an ORANGE path did not exist. The crew would enter FRZ-0.1 and immediately transition out at Step 1 in this case.

D. Incorrect. Plausible since if these conditions had been identified prior to entering FRC-0.2, then FRZ-0.1 would be the higher priority, but incorrect since FRC-0.2 is to be completed even though it has turned YELLOW.

Technical Reference(s)	ODA-407	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: __________

Learning Objective: **STATE** when an operator is allowed/required to exit an FRG and the requirements for completion of an FRG exited before it is complete in accordance with ODA-407, Operations Department Procedure Use and Adherence. (ERG.XD2.OB18)

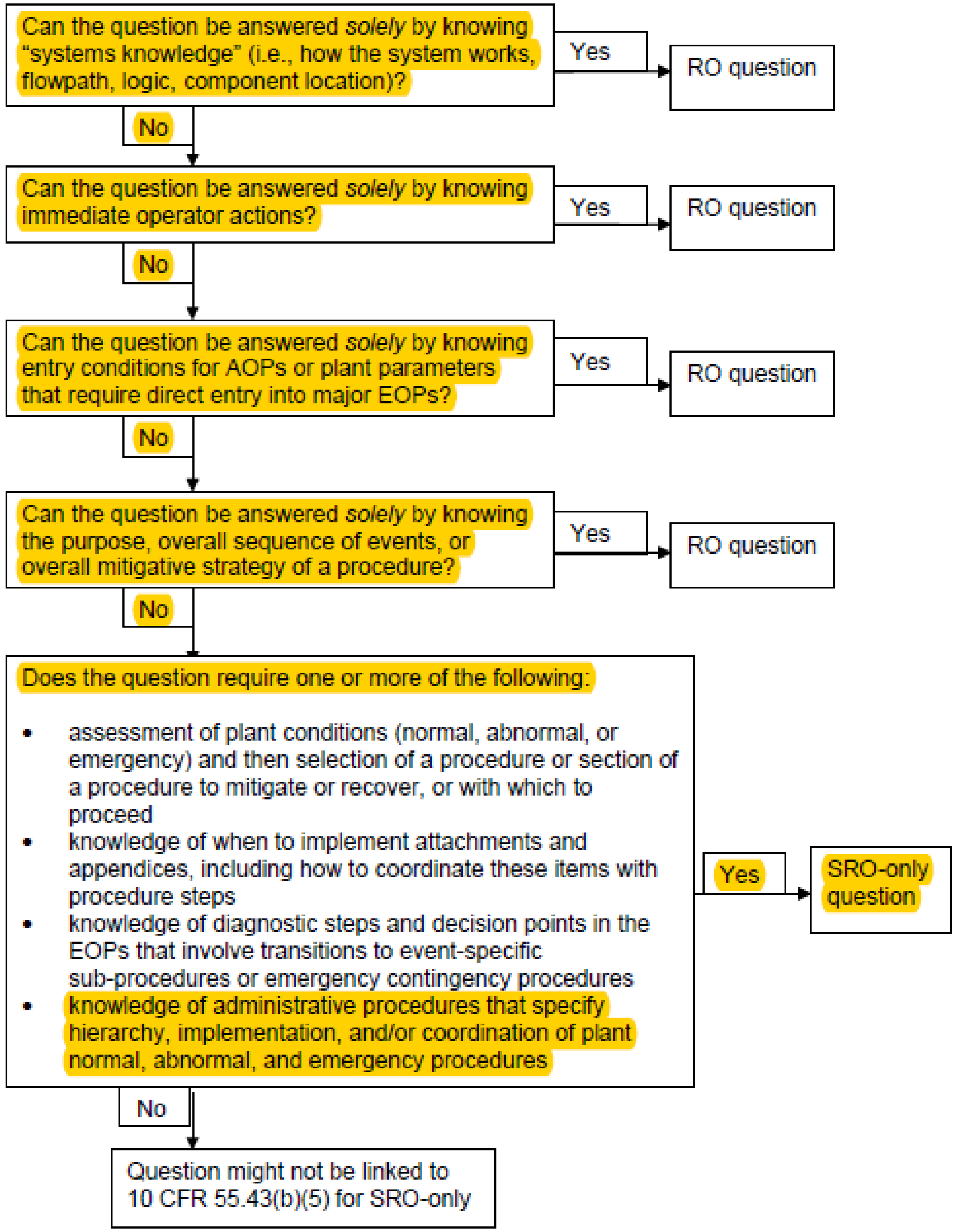
Question Source: Bank # 23165
Modified Bank # __________ (Note changes or attach parent)
New __________

Question History: Last NRC Exam LC26

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 __________
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:



Comments / Reference: Bank 23165	Revision:

Given the following conditions:

- The crew is responding to a large break LOCA
- A CORE COOLING status tree ORANGE path causes a transition to FRC-0.2, Response to Degraded Core Cooling.
- During performance of FRC-0.2, the CORE COOLING status tree changes from ORANGE to YELLOW.
- An ORANGE path exists on the CONTAINMENT status tree.
- FRZ-0.1, Response to High Containment Pressure, is the procedure referenced by the CONTAINMENT status tree.

Which of the following is the required action?

A. Complete FRC-0.2 and then go to FRZ-0.1 since CONTAINMENT is a lower priority path than CORE COOLING

B. Go to FRZ-0.1 since an ORANGE path has a higher priority than a YELLOW path. Completion of FRC-0.2 is NOT needed

C. Complete FRC-0.2 after completing FRZ-0.1, since the CORE COOLING status tree had been in an ORANGE path

D. Perform FRC-0.2 and FRZ-0.1 together, since FR procedures of the same priority can be executed together.

Answer: A

Answer Explanation
Explanation: A is wrong. Must complete the core cooling first, even though its color did change to lower status after you entered it. B is wrong. You must finish the core cooling FRG first. C is correct. Per the ODA-407 you must complete the core cooling FRG and then go back to containment since it is a lower priority D is wrong. Orange on containment does not take priority over orange (when you entered it) for core cooling.

Comments / Reference: Bank 23165	Revision:

Question 55 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	23165
User-Defined ID:	ILOT5977
Cross Reference Number:	ERG.XD2.OB15.001
Topic:	Given the following conditions: The crew is responding to a large break LOCA A CORE COOLING statu
K/A:	G.2.4.16
Question Reference:	
SRO:	YES
Comments:	R21E13RM; R/S22E14; R24E23RM REF: ERG.XD2.LN; ODA-407 **KA Match:** This question matches the KA by requiring the ability to prioritize safety functions during an event. **SRO ONLY:** NUREG 1021, ES-401, Attachment 2 E Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations

Comments / Reference: ODA-407	Revision: 17

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ATTACHMENT 8.A
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ERG RULES OF USAGE

9. The Critical Safety Function Status Trees (CSFSTs) are used to evaluate the current state of pre-defined Critical Safety Functions (Fuel Matrix and Cladding, Reactor Coolant System and Containment Building).

- The CSFSTs are evaluated in the following order of priority: Subcriticality (S), Core Cooling (C), Heat Sink (H), Integrity (P), Containment (Z), Inventory (I).
- The CSFSTs are statused with color-coding and line pattern-coding with the following order of highest to least priority:

  RED (Solid line with a solid shaded pie) - Used to describe the status that represents an extreme challenge of the particular safety function.

  ORANGE (Dashed line with a two-thirds shaded pie) - Used to describe the status that represents a severe challenge of the particular safety function.

  YELLOW (Dotted line with a one-third shaded pie) - Used to describe the status that represents a not satisfied condition of the particular safety function.

  GREEN (Hollow line with a pie not shaded) - Used to describe the status that represents a satisfied condition of the particular safety function.
- The CSFST is entered at the left side of the tree with a branch of questions. Questions are answered based on plant conditions at the time, and the appropriate branch line followed to the next question. The CSFST evaluation is complete when a color-coded or line pattern-coded status is determined.

COLOR	LINE	SYMBOL CODE	STATUS/RESPONSE
RED			The critical safety function is under extreme challenges: immediate operator action is required.
ORANGE			The critical safety function is under severe challenges: prompt operator action is required.
YELLOW			The critical safety function condition is not satisfied. Operator action may be taken.
GREEN			The critical safety function is satisfied. No operator action is needed.

Comments / Reference: ODA-407	Revision: 17

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ATTACHMENT 8.A
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ERG RULES OF USAGE

10. The CSFST evaluation determines the condition of Critical Safety Functions. The following Rules of Priority describe the appropriate operator action based on the CSFST conditions.

- IF a RED status is encountered, THEN the operator is required to immediately stop (do not complete the step in progress) any Optimal Recovery Guideline (ORG) in progress AND perform the required Functional Restoration Guideline (FRG).

- IF during performance of a RED condition FRG, a RED status of higher priority arises, THEN the higher priority condition should be addressed first AND the lower priority RED condition FRG suspended (complete the step in progress). After the higher priority FRG is completed and guidance is given to "return to procedure and step in effect", the previous FRG which was being performed prior to the transition should be re-entered (performed). Performance (re-entry) to the previous FRG being performed is required even if the lower priority condition has cleared in order to complete response and recovery actions that had previously been started.

- IF any ORANGE status is encountered, the operator is expected to monitor all of the remaining CSFSTs, THEN if no RED status is encountered, suspend any ORG in progress (complete the step in progress) AND perform the FRG required by the ORANGE status.

- IF during performance of an ORANGE condition FRG, a RED status or higher priority ORANGE status arises, THEN the RED or higher priority ORANGE condition is to be addressed first AND the original ORANGE condition FRG suspended (complete the step in progress). IF a FRG specifically states that a higher priority condition should NOT be addressed, this requirement does not apply. After the higher priority FRG is completed and guidance is given to "return to procedure and step in effect", the previous FRG which was being performed prior to the transition should be re-entered (performed). Performance (re-entry) to the previous FRG being performed is required even if the lower priority condition has cleared in order to complete response and recovery actions that had previously been started.

- Once a FRG is entered due to a RED or ORANGE condition, that FRG is performed to the point of a defined transition regardless of whether the RED or ORANGE has cleared.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier			3
	Group			
	K/A	2.4.44		
Level of Difficulty: 2	Importance Rating			4.4

Knowledge of emergency plan protective action recommendations.

Question # 100

Per EPP-304, Protective Action Recommendations, initial PARs are issued at the __(1)__ Emergency Classification and are formulated in the Control Room by the Emergency Coordinator and the __(2)__.

A. (1) Site Area
(2) STA

B. (1) Site Area
(2) US

C. (1) General
(2) STA

D. (1) General
(2) US

Answer: C

K/A Match: K/A match due to requiring knowledge of when emergency plan protective action recommendations are implemented and the positions that formulate the PAR.

SRO Only: SRO Only due to 10 CFR 55.43(b)(5) Analysis requiring knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Explanation:

A. Incorrect. First part is incorrect, but plausible (see B). Second part is correct (see C).

B. Incorrect. First part is incorrect, but plausible because a Site Area Emergency is the next lower classification below the General Emergency when PARs are required and it is plausible that PARs would be issued at this EAL Classification. Second part is incorrect, but plausible if it is assumed that the US on the affected unit would function as the second SRO concurrence on PARs as is the case for other activities such as ERG deviations, etc.

C. Correct. First part is correct. In accordance with EPP-304, PARs are issued upon declaration of a GE. Second part is correct. The STA aids the EC in developing PARs from the Control Room.

D. Incorrect. First part is correct (see C). Second part is incorrect, but plausible (see B).

Technical Reference(s)	EPP-304	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **IDENTIFY** the ERO position holders responsible for PARs in each facility, in accordance with EPP-304. (PAR.OW5.OB01)

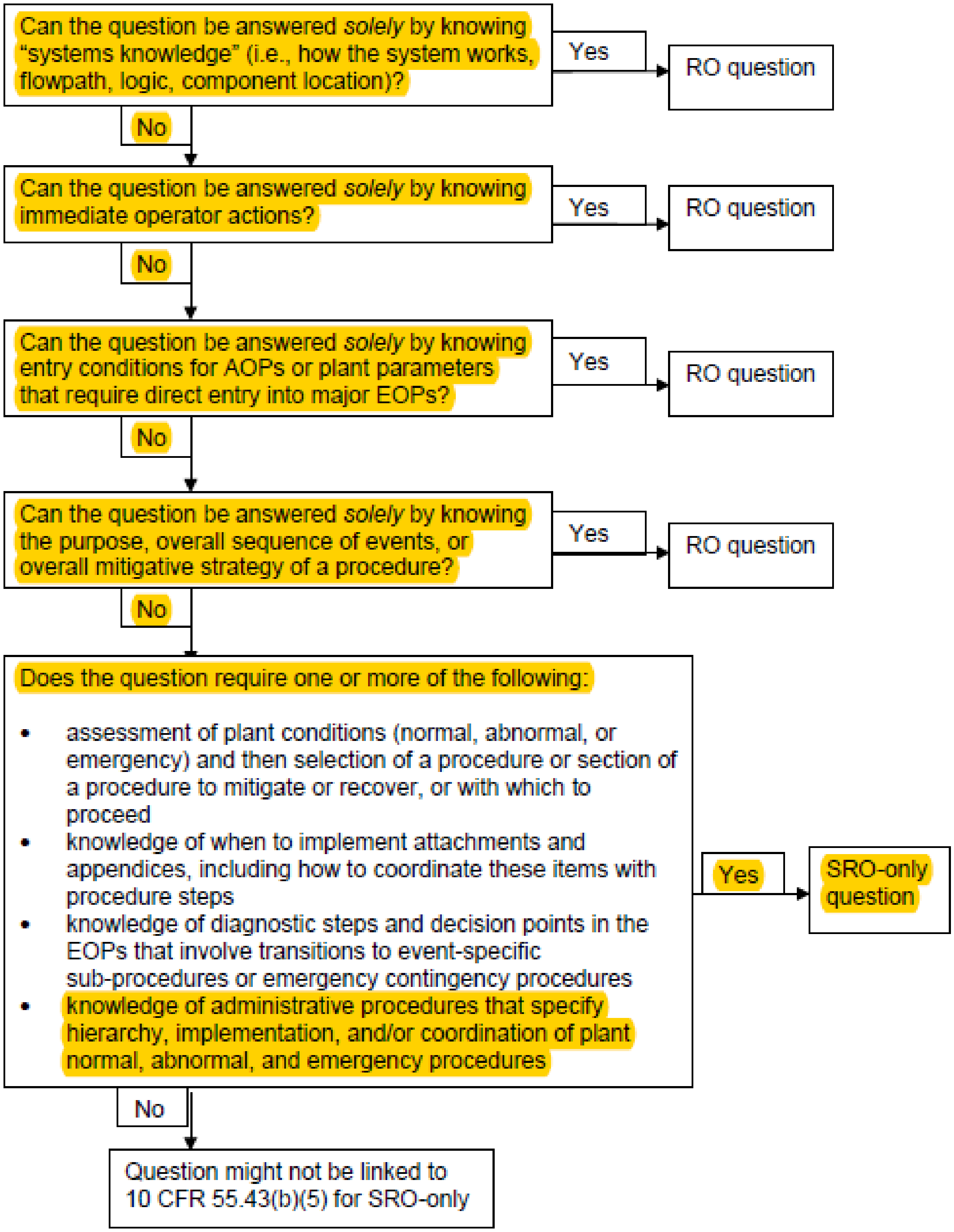
Question Source: Bank # 62572
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: 10 CFR 55.43(b)(5) Analysis	Revision:

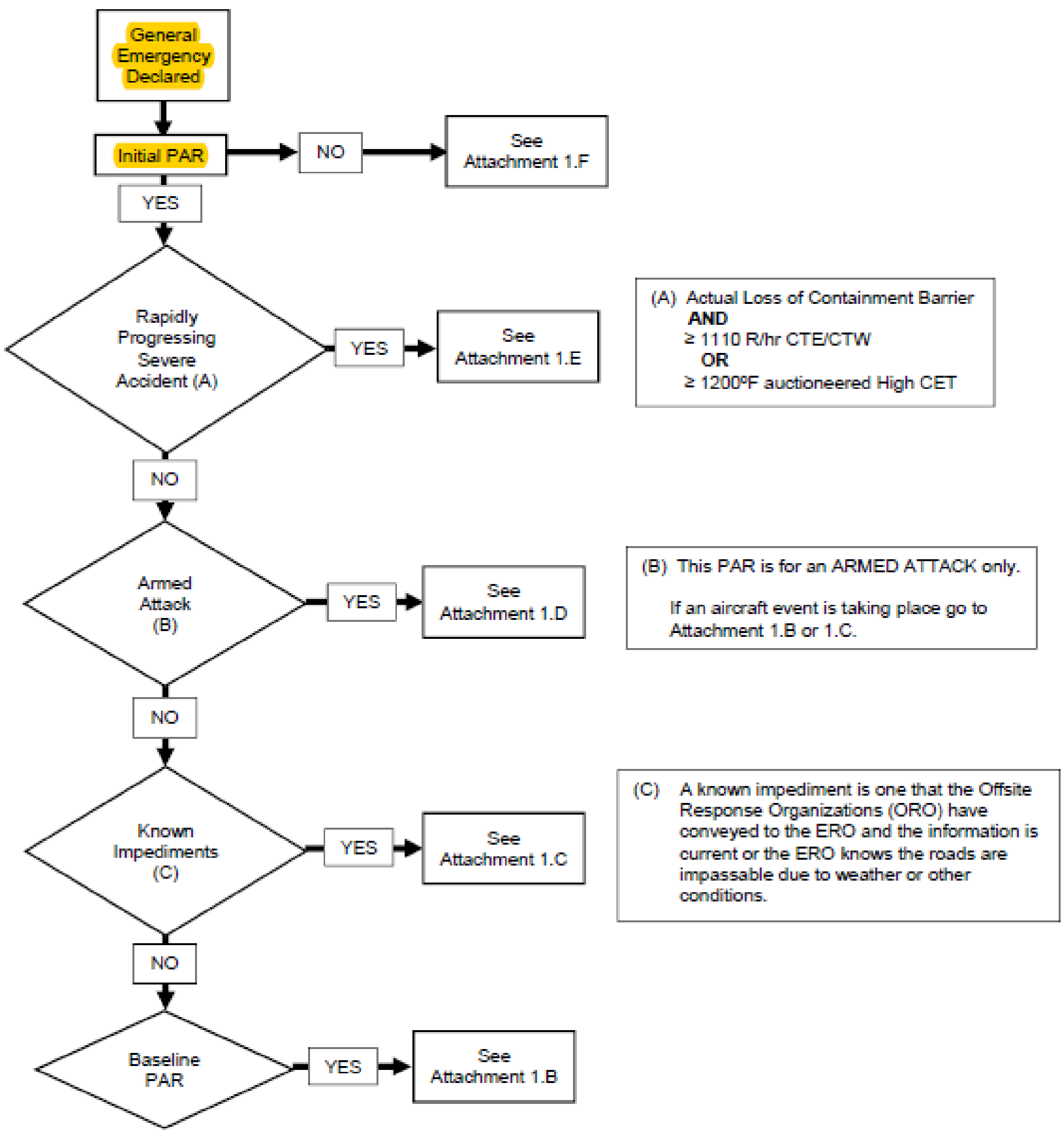


Comments / Reference: EPP-304	Revision: 22

CPNPP EMERGENCY PLAN MANUAL		PROCEDURE NO. EPP-304
PROTECTIVE ACTION RECOMMENDATIONS	REVISION NO. 22 REFERENCE USE	PAGE 13 OF 27

ATTACHMENT 1.A

PAR Flowchart Determination Guideline



Comments / Reference: EPP-304	Revision: 22

CPNPP EMERGENCY PLAN MANUAL		PROCEDURE NO. EPP-304
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4.10.6 A PAR to Shelter-in-Place should be for the shortest time possible and then a new PAR developed per Attachment 1.F (Expanded PAR Flowchart).

4.11 Developing a PAR Exceeding 10-Mile EPZ

4.11.1 If dose projections exceed EPA Protective Action Guides (PAGs) at the outer boundary of the 10-Mile EPZ, then

- Complete the PAR and notify the Department of Public Safety (DPS), Hood County, and Somervell County.
- Once the PAR has been issued, contact the Texas Department of State Health Services / Radiological Control 24-hour emergency number listed in the ERF Phone Directory and notify them of the dose projections outside of the 10-Mile EPZ. The CPNPP ERO, with the Texas Department of State Health Services / Radiological Control, should provide information for a PAR outside of the 10-Mile EPZ.
- Verify dose projection with field team measurements as quickly as possible.
- If field team measurement supports the dose projection, then recommend to the Emergency Coordinator that PARs be expanded to include the 10-Mile radius and the downwind sectors greater than 10-Miles in 2-mile increments until PAGs are not exceeded.

4.12 Responsibilities

4.12.1 The Emergency Planning Manager is responsible for preparing and maintaining this procedure current.

4.12.2 The Emergency Coordinator is responsible to provide Protective Action Recommendations (PARs) for areas within the 10-Mile EPZ to County and State agencies as quickly as possible, but in no case later than fifteen (15) minutes after classifying and/or reclassifying an emergency, or after a change in the PARs.

- In the **Control Room**, the Emergency Coordinator and the Shift Technical Advisor formulate PARs.
- In the **Technical Support Center (TSC)** and the **Emergency Operations Facility (EOF)**, the On-site Radiological Assessment Coordinator (OnRAC), the Off-site Radiological Assessment Coordinator (OffRAC), and the Radiation Protection Coordinator (RPC) respectively are responsible to formulate protective action recommendations for the Emergency Coordinator. They are also responsible to coordinate or designate someone to coordinate protective action recommendations related activities.