Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	CE/E02.EK3.1	
	Importance Rating	3.2	

K/A statement: Knowledge of the reasons for the following responses as they apply to the (Reactor Trip Recovery): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

Proposed Question:

The Plant has been at 100% power for the last 30 days. During the performance of EOP-2.0, "Reactor Trip Recovery," which one of the following describes the expected response of reactor power?

After the initial rapid power reduction, reactor power will stabilize at . . .

A. 10^{-4} % and then slowly lower over a period of hours.

B. The subcritical multiplication level and then slowly lower.

C. The subcritical multiplication level and then remain at that level.

D. 10^{-4} % and then rise slowly over a 24 hour period as Xenon burns out.

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, applicant selects incorrect power level (reactor is still critical at this power level)
B. Correct, EOP-2.0 provides this guidance and trend
C. Incorrect, applicant selects correct power level but incorrect trend
D. Incorrect, applicant selects incorrect power level and misinterprets the significance of Xenon for these conditions

Technical Reference(s): EOP-2.0 Basis
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None .

Learning Objective: ______________ (As available)

Question Source: Bank # X

	Modified Bank #	_______ (Note changes or attach parent)
	New	_______

Question History: Last NRC Exam Palisades 2003____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__5__
	55.43	_____

Comments:

As a result of the Reactor trip initiation, the control rods will be rapidly inserted. Steam flow to the Main Turbine will be terminated and the Main Generator output breakers will open. A rapid decrease in Reactor power and a negative startup rate will be observed. This rapid decrease is followed by a decrease in indicated power (approximately -1/3 decades per minute) until the subcritical multiplication level is reached. Indicated power will stabilize at the subcritical multiplication level and decrease slowly over a period of hours.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	008.AK2.02	
	Importance Rating	2.7	

K/A statement: Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Sensors and detectors

Proposed Question:

The crew is performing AOP-23, "Primary Coolant Leak," to attempt to isolate a PCS leak.

Given the following plant conditions:

- Containment radiation levels are rising
- Pressurizer level is rising
- Charging flow has lowered
- Letdown flow has risen
- Quench Tank level is 72% and stable
- Steam Generator levels are at 66% and stable

Which of the following is the most likely location of the leakage?

A. Charging Line inside of Containment

B. Letdown Line inside Containment

C. Pressurizer Power Operated Relief Valve

D. Pressurizer Vapor Space

Proposed Answer: D

Explanation (Optional):

Due to the increasing Containment Radiation levels and the increasing Pressurizer (PZR) level, choice D is correct. If a PZR Safety Valve were leaking, Quench Tank (QT) level would be increasing rather than stable and containment radiation would be stable unless the rupture disk on the QT blows. If the Charging or Letdown line were leaking, PZR level would not be increasing. However, the decreasing Primary System pressure (swell of the PZR) will cause Charging flow to decrease and Letdown flow to increase.

A. Incorrect, a charging line leak inside containment would not cause PZR level to rise.
B. Incorrect, a letdown line leak inside containment would not cause PZR level to rise.
C. Incorrect, a leak from a PZR PORV would cause QT level to rise, not remain stable.

D. Correct, PZR vapor space is leaking into the containment atmosphere, causing containment radiation to rise and PZR level to rise.

Technical Reference(s): PL-PCS, Primary Coolant System Lesson Plan_________
(Attach if not previously provided, ____
including version/revision number) ____

Proposed references to be provided to applicants during examination: None

Learning Objective: _________________________ (As available)

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

Question History: Last NRC Exam Palisades 2005

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Question modified from Palisades 2005 NRC Exam (Question 2). Edited stem and rearranged distractors. Replaced one distractor with new.

A. Impact on the following systems due to loss or malfunction of Pressurizer Level Control including loss of inputs:

1. CVCS
   a. Charging Pumps will start or stop and/or P-55A speed changes depending on nature of malfunction or loss.
   b. Letdown Orifices open or close depending on nature of malfunction or loss.
2. PCS
   a. A loss or malfunction of PLCS affecting CVCS will cause PCS inventory to be raised or reduced accordingly.
3. Pressurizer Pressure Control
   a. Level channel failure can affect Pressurizer Heaters on lo-lo level cutout at 36% on either Hot Cal channels.
   b. Selector switch can remove affected channel from service to restore heaters.
4. Instrument Line Losses
   a. A failure of the wet leg (High side of DP cell) would result in indicated level to rise.
      1) Minimum Charging and Maximum Letdown.
      2) This is also a Vapor space LOCA
   b. A failure of the Low side results in a PCS leak! Enter Abnormal Operating Procedures. Depending on the size of the failure, this could result in an indicated PZR level of 0%. Maximum Charging and **NO** Letdown.
      1) Zero percent (0%) indicated PZR level trips all PZR heaters and closes Letdown Orifice Stop Valve (CV-2003)

Examination Outline Cross-Reference:

		RO	SRO
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	00009.EA1.13	
	Importance Rating	4.4	

K/A Statement: Ability to operate and monitor the following as they apply to a small break LOCA: ESFAS

Proposed Question:

Given the following conditions:

- The Plant has just tripped from 100% power due to a small break LOCA.
- Preferred AC Bus EY-20 was lost during the reactor trip.
- Containment pressure is 3.4 psig.
- Pressurizer pressure is 1540 psig.

With no operator action, which one of the following describes the response of both the Right and Left Train ESS equipment?

A. NEITHER the Right nor Left Train ESS equipment will be running.

B. BOTH the Right and Left Train ESS equipment are running.

C. ONLY the Right Train ESS equipment is running.

D. ONLY the Left Train ESS equipment is running.

Proposed Answer: **D**

Explanation (Optional):

Preferred AC Bus EY-20 controls the Right Channel ESS equipment. With no control power to actuate the equipment through, the ESS equipment on that Right Channel will not actuate when necessary. Vital AC Bus EY-30 controls the Left Channel ESS equipment. The applicant needs to understand which Vital AC Bus is critical for each Channel and must also understand that a SIAS will occur on low PZR pressure (<1605 psia on 2/4 PZR pressure channels) or on high containment pressure (>3.7 psi to 4.4 psi on 2/4 containment pressure channels).

A. Incorrect, applicant does not believe a valid SIAS will occur under the conditions.
B. Incorrect, applicant believes a loss of EY-20 does not affect Right or Left Channel actuations.
C. Incorrect, applicant believes a loss of EY-20 will only impact the Left Channel.
D. Correct, see explanation

Technical Reference(s): DBD-2.05

(Attach if not previously provided,

including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ___None___________

Learning Objective: __________________________ (As available)

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

Question History: Last NRC Exam __________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 __7__
55.43 ______

Comments:

The SIS Logic is shown on drawings E-17 sheets 3, 4, 5 and 6. The circuits associated with the safety injection function including four Safety Injection Detection circuits for Low Pressurizer Pressure, four Containment High Pressure circuits, SIS block logic, two independent test circuits and two independent actuation channel control circuits (left and right). The two actuation channels initiate operation of two redundant SIS equipment groups. For additional information on these circuits see Tables A-4 through A-7. Components actuated by the SIS are discussed in their respective system DBDs.

The SIS is initiated from two-out-of-four Low-Low Pressurizer Pressure, two-out-of-four Containment High Pressure or manually using either of two push buttons. When pressurizer pressure readings are between the low and low-low pressure setpoints the SIS can be manually blocked. The equipment actuated by the SIS signal will vary depending on whether or not offsite power is available.

The left channel SIS is powered from Vital Bus EY-30 and the right channel SIS is powered from vital bus EY-20. The circuits are functionally identical so only the left channel will be discussed. Upon receipt of a two out of four channel Low Pressurizer Pressure (or Containment High Pressure) SIS initiation signal (and assuming a SIS block signal of the Low Pressurizer Pressure signal is not present) the SIS initiation relays are energized. If offsite power is available the SIS auxiliary relays will initiate the simultaneous start of the Engineered Safeguards equipment. If offsite power fails or is not available, the diesel generators receive an automatic start signal and most loads will be automatically shed in preparation for automatic sequenced loading. Without SIS actuation the normal shutdown (NSD) sequencer logic will operate to load the diesel; with SIS actuation the design basis accident (DBA) sequencer logic will operate (see Sequencer DBD 5.05).

1. Manual Initiation of Safety Injection

   A left-channel (right-channel) SIS can be manually initiated with pushbutton PB1-1 (PB1-2) which is located on the left (right) section of panel EC-13.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000011.EA2.05	
	Importance Rating	3.3	

K/A Statement: Ability to determine or interpret the following as they apply to a Large Break LOCA: Significance of charging pump operation.

Proposed Question:

Given the following:

- The Plant tripped from full power due to a large break LOCA.
- The Crew has implemented EOP-4.0, “LOCA Recovery.”
- PCS temperature is 540°F and stable
- PCS pressure is 1200 PSIA and lowering
- The Crew has determined that PCS cooldown is required
- 2400V Bus 1D is de-energized
- P-55C, Charging Pump, is in service providing 40 GPM charging flow

Based on the given conditions, PCS cooldown __(1)__ commence ____(2)____.

A. (1) may
(2) with no restrictions

B. (1) may
(2) but must be stopped at 490°F to verify shutdown margin

C. (1) cannot
(2) until additional charging pumps are started

D. (1) cannot
(2) until adequate shutdown margin has been verified

Proposed Answer: A

Explanation (Optional):

EOP-4.0 basis document states it is allowable to commence PCS cooldown as long as emergency boration is in progress. Loss of bus 1D results in loss of LC-12 therefore P-55A and P-55B have no power, in addition this also results in loss of MCC-2 and there is no boric acid pump feed available but boric acid gravity feed is in service (activated on SIAS at PCS pressure of 1605 psia). Charging pump P-55C is in-service providing 40 GPM charging line flow (this is greater than the minimum of 33 GPM required for emergency boration).

A. Correct, see explanation.

B. Incorrect, part 1 is correct. Part 2 is incorrect as the applicant may confuse this as being correct, this response actually comes from a requirement in EOP-3 for SBO to verify the Reactor will remain shutdown at 50 degree intervals
C. Incorrect, the applicant incorrectly believes that additional charging pumps must be started to support emergency boration when in fact P-55C is providing 40 GPM charging flow (procedure requires minimum of 33 GPM)
D. Incorrect, the applicant incorrectly believes that SDM must first be verified prior to commencing cooldown

Technical Reference(s): EOP-4.0, EOP-9.0 RA
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: _____None_______

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam ___________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:
Question from Palisades 2014 Audit Exam.

STEP 19

NOTE: IF emergency boration is in progress, THEN cooldown may commence/continue while the required shutdown margin value is calculated.

© 19. **VERIFY** PCS boron concentration greater than or equal to required boron concentration as verified by sample or hand calculation. Refer to EOP Supplement 35.

a. IF Emergency boration is in progress AND PCS boron concentration is greater than or equal to required boron concentration, THEN **SECURE** emergency boration. Refer to EOP Supplement 40.

19.1. IF PCS boron concentration is less than required boron concentration, THEN **PERFORM** BOTH of the following:

a. **ENSURE** emergency boration is in progress.

b. WHEN required boron concentration is reached, THEN **SECURE** emergency boration. Refer to EOP Supplement 40.

CEN-152 LOCA Step:

None

Technical Basis:

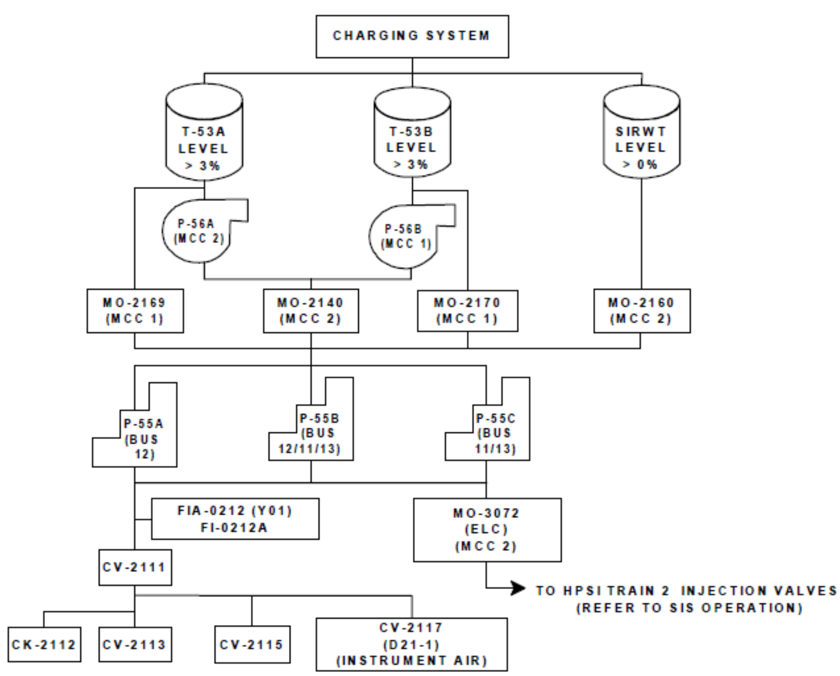
The intent of this step is to maintain the **[reactor shutdown]** throughout the cooldown.

To accomplish this, the operator verifies boron concentration is greater than or equal to the required boron concentration by sample results or hand calculation using EOP Supplement 35. The operator may have to emergency borate to obtain the required PCS boron concentration.

If a PT Curve maximum cooldown rate is maintained, Pressurizer level may not be able to be maintained constant during the initial stages of the cooldown. Therefore, Pressurizer level may lower. This outsurge from the Pressurizer will tend to dilute the PCS boron concentration. The possible dilution effects should be considered in determining cold shutdown boron concentration.

Training Emphasis

Cooldown is allowed prior to establishing the required boron concentration as long as emergency boration is in progress. Emergency boration will maintain the required shutdown margin when the cooldown rate is within the allowed technical specification cooldown rate. If plant conditions do not dictate immediately cooling down, then the boron concentration should be established prior to cooling down to T_{AVE} less than 525°F. This is a judgement call that must be made by the on-shift SROs based on present plant conditions.



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	____
	Group #	1	____
	K/A #	000015/17.AA2.08	
	Importance Rating	3.4	____

K/A Statement: Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on high bearing temperature.

Proposed Question:

The Plant is at 100% power when inadvertently, CV-0940, CCW Return From Containment, closes and cannot be opened.

The crew is performing AOP-29, "Primary Coolant Pump Abnormal Conditions," in conjunction with AOP-36, "Loss of Component Cooling."

Which of the following bearing temperatures would require a manual reactor trip and trip of the affected PCP?

A. Upper Guide bearing temperature 170°F

B. Lower Guide bearing temperature 182°F

C. Upper Thrust bearing temperature 180°F

D. Down Thrust bearing temperature is 171°F

Proposed Answer: **B**

Explanation (Optional):

Based on the spurious Containment Isolation Signal, CCW flow to the PCPs is isolated. Therefore, per AOP-29, the reactor trip criteria are:

- Loss of CCW AND
  - Any PCP bearing temperature exceeding:
    - Upper Guide bearing – 175°F
    - Lower Guide bearing – 175°F
    - Upper Thrust bearing – 185°F
    - Down Thrust bearing -175°F
  - Any PCP Lower Seal temperature exceeding 185°F
  - Any PCP Vapor Seal temperature exceeding185°F

A. Incorrect, below the limit of 175°F
B. Correct, above the limit of 175°F
C. Incorrect, below the limit of 185°F
D. Incorrect, below the limit of 175°F

Technical Reference(s): AOP-29, AOP-31
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments:

REACTOR AND EQUIPMENT TRIP CRITERIA

Reactor Trip

- Loss of CCW <u>AND</u>:
  - Any PCP bearing temperature exceeds setpoint
    - Upper Guide 175°F
    - Lower Guide 175°F
    - Upper Thrust 185°F
    - Down Thrust 175°F
  - Any PCP Lower Seal temperature exceeds 185°F
  - Any PCP Vapor Seal temperature exceeds 185°F
- Any PCP vibration greater than 29 mils
- Imminent PCP failure corroborated by other indications

Equipment Trip

- Affected Operating Primary Coolant Pump(s)
  - Any Reactor Trip criteria met

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000022.G2.4.47	
	Importance Rating	3.4	

K/A Statement: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power.
- An instrument failure caused letdown to isolate and letdown cannot be quickly restored.
- Charging was manually secured per SOP-2A, “Chemical Volume and Control System.”
- PCS T_{AVE} temperature is stable.

With no further operator action, what is the expected effect of the above conditions?

A. Pressurizer level lowers, Volume Control Tank level rises.

B. Pressurizer level is constant, Volume Control Tank level lowers.

C. Pressurizer level lowers, Volume Control Tank level is constant.

D. Pressurizer level is constant, Volume Control Tank level rises.

Proposed Answer: **A**

Explanation (Optional):

A. Correct, Primary Coolant Pumps controlled bleed-off (CBO) is still rejecting back to the VCT, causing level to rise in the VCT. VCT level will have to be maintained manually by cycling the VCT drain valve. Meanwhile, due to a coolant outflow to CVCS via CBO, PZR level will drop due to a lack of makeup (charging was manually secured).
B. Incorrect, PZR level will drop due to CBO back to CVCS (which is isolated). VCT level will rise due to the input from the PCS via CBO.
C. Incorrect, VCT level will rise due to CBO back to CVCS (which is isolated).
D. Incorrect, PZR level will drop due to CBO back to CVCS (which is isolated).

Technical Reference(s): Primary Coolant System Lesson Plan
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: ___None___________

Learning Objective: _________________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

7.3 CHARGING AND LETDOWN SYSTEM OPERATION

7.3.1 To Stop Charging and Letdown at Rated Conditions

a. **PLACE** CV-2023, Ion Exchangers Bypass in BYPASS.

> **CAUTION**
>
> Pressurizer level will lower at 4 gpm without charging system in operation (approximately 16 min per % of level).

b. **PLACE** Charging Pumps Control Select Switch for P-55B and P-55C in MANUAL.

c. **STOP** Charging Pump(s) AND **IMMEDIATELY CLOSE** Letdown Orifice Stop Valves by placing the following handswitches to CLOSE:

- HS-2003, Letdown Orifice Valve Switch
- HS-2004, Letdown Orifice Valve Switch
- HS-2005, Letdown Orifice Valve Switch

> **WARNING**
>
> Unless vented and purged, explosive hydrogen will be released from the VCT if the VCT is allowed to drain completely. Refer to Section 7.4.2.

d. **OPERATE** MV-CVC2086, VCT Drain Valve, as needed to compensate for primary coolant pump controlled bleedoff flow of approximately 4 gpm.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000025.AK1.01	
	Importance Rating	3.9	

K/A statement: Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation

Proposed Question:

Given the following conditions:

- The Plant has tripped due to a large break LOCA.
- The Crew has transitioned to EOP-4.0, "Loss of Coolant Accident Recovery."
- The 1C Bus is ENERGIZED.
- The 1D Bus FAULTED 10 minutes ago and cannot be restored.
- SIRW Tank Level is 2%.

Based on the current plant conditions, which of the following equipment actuations will NOT occur?

A. P-67B LPSI Pump, TRIPS

B. CV-3071, P-66A Subcooling Valve, OPENS

C. CV-3002, Containment Spray Isolation Valve, THROTTLES

D. CV-0826, CCW Heat Exchanger SW Outlet Valve, FULL OPEN

Proposed Answer: **B**

Explanation (Optional):

The Recirculation Actuation Signal (RAS) occurs at 2% SIRWT level. As a result, various actions automatically occur (see reference material).

A. Incorrect, LPSI Pumps receive a trip signal generated by the RAS. P-67B will trip. P-67A tripped on the 1D Bus fault prior to the RAS actuation.
B. Correct, CV-3071 requires P-66A to be running in order to open. With the pump tripped, (loss of bus 1D) the valve will remain closed and not open on the RAS.
C. Incorrect, CV-3002 repositions from fully open to throttled to increase the NPSH for the HPSI and CSS Pumps.
D. Incorrect, the CCW Hx SW Outlet valves fully open to allow maximum SWS flow and thus maximum cooling of the CCW flow.

Technical Reference(s): DBD-2.01
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: ___None______

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam _________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

TABLE 3.4-7
Components Actuated By RAS

Components	Action	Result
P-67A, P-67B	Trip(1)	Trip LPSI pumps
CV-3029, CV-3030	Open	Open suction to containment sump
CV-3031, CV-3057	Close	Remove suction from SIRWT
CV-3027, CV-3056	Close(2)	Stop recirculation to SIRWT
CV-3070(3), CV-3071(4)	Open	Open subcooling, aligns containment spray discharge with HPSI suction
CV-3001, CV-3002	Throttle	Throttle Containment Header Isolation Valves
CV-3001(5)	Close	Close containment header isolation valve
CV-0823, CV-0826	Open	Allow maximum SWS flow through CWHX
CV-0945, CV-0946	Open	Send CCW to CCWHX
CV-0821, CV-0822	Close	Allow SWS flow through CV-0826 and CV-0823 for maximum flow from CCWHX

(1) Trip requires SS143-206 (SS143-111) to be in locked on position.

(2) Action requires HS-3027A and HS-3056A to be in closed position.

(3) Action requires open containment sump valve CV-3030 and running HPSI P-66B.

(4) Action requires open containment sump valve CV-3029 and running HPSI P-66A.

(5) Action requires a failure of CV-3030 sump valve to open on RAS.

Reference drawings: SIS Test and RAS Logic Diagram, E-17, Sheet 5.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000026.AK3.02	
Importance Rating	3.6	

K/A Statement: Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: The automatic actions (alignments) within the CCWS resulting from actuation of the ESFAS.

Proposed Question:

Plant conditions are as follows:

- A Plant startup is in progress with the Plant in Mode 2.
- P-52A, Component Cooling Water (CCW) Pump, is running.
- P-52B and P-52C, CCW Pumps, are in standby.
- A Loss of Offsite Power (LOOP) has just occurred.
- Both Diesel Generator 1-1 and 1-2 are running loaded.
- CCW pump P-52A tripped and would NOT restart following the LOOP.

ONE minute later, a Safety Injection actuation occurred due to a small break LOCA.

With no operator action, which ONE of the following describes the configuration of the CCW system at this time?

A. ONLY CCW pump P-52B supplying all CCW loads due to the failure of CCW pump P-52A.

B. ONLY CCW pump P-52C supplying all CCW loads due to the failure of CCW pump P-52A.

C. TWO CCW pumps supplying all CCW loads due to SIAS start of CCW pump P-52B.

D. TWO CCW pumps supplying all CCW loads due to CCW low pressure start of CCW pump P-52B.

Proposed Answer: **A**

Explanation (Optional):

On a SIAS accompanied by a LOOP, P-52C will start only due to a CCW low pressure signal to prevent unnecessary diesel loading. In this case, P-52A tripped upon the NSD sequencer start. P-52B will start on the NSD sequencer and subsequently again (receive a second start signal) on the DBA sequencer. The DBA sequencer will actuate due to the SIAS that occurred as a result of the LOCA. Since only P-52B will be running at the time of the SIAS, a low pressure signal would not be generated, as the CCW pumps are designed for 100% capacity post-DBA.

A low pressure auto-start condition would be expected upon receipt of a RAS (both CCW HXs inlet valves receive an open signal), which is not present in this case. The applicant must understand the automatic operation of the CCW pumps on a LOOP and a LOCA, as well as the transient operating parameters of the system.

A. Correct, see explanation
B. Incorrect, see explanation.
C. Incorrect, see explanation.
D. Incorrect, see explanation.

Technical Reference(s): DBD-1.01, FSAR Chapter 9 Section 9.3.3 Rev 24
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: __None__________

Learning Objective: ________________________ (As available)

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

Question History: Last NRC Exam ___________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __8__
55.43 _____

Comments:

3.3.6.6 Emergency Actuation - CCW Pumps

The following table shows the actuation of the CCW pumps from emergency signals:

TABLE 3-4
CCW PUMP EMERGENCY ACTUATION

Signal	Pumps	Actuation
SIS[(a)]	P-52A, P-52B, P-52C	Start
Load Shed	P-52A, P-52B, P-52C	Trip
NSD Sequence[(b)]	P-52A, P-52B, P-52C[(d)]	Start
DBA Sequence[(c)]	P-52A, P-52B, P-52C[(d)]	Start

NOTES:

(a) With offsite power available.

(b) With a loss of offsite power.

(c) SIS accompanied with a loss of offsite power.

(d) P-52C starts with the NSD/DBA sequence only in conjunction with a CCW low pressure signal (from PS-0918) to avoid unnecessary diesel loading.

The LOCA response analysis provided a temperature higher than the 140°F system design temperature downstream of the CCW heat exchangers. Reference 63 was completed to revise the design temperature of the portion of the system downstream of the CCW heat exchangers to the shutdown cooling heat exchangers to 165°F

9.3.3 DESIGN ANALYSIS

9.3.3.1 Margins of Safety

1. Any one of three pumps is capable of supplying component cooling requirements during normal Plant operation. During shutdown, one pump can furnish at least 50% of the maximum shutdown cooling water requirements. For post-DBA operation, one pump can furnish 100% of the required capability for cooling the containment spray and safety injection recirculation water. For the Left Channel DBA condition, containment cooling is achieved using two containment spray pumps. For the right channel DBA, containment cooling is achieved using one spray pump and containment air coolers (VHX-1, 2 & 3), which do not require CCW.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000029.EK1.03	
	Importance Rating	3.6	

K/A statement: Knowledge of the operational implications of the following concepts as they apply to the ATWS: Effects of boron on reactivity

Proposed Question:

Given the following conditions:

- The Plant is at 43% power while shutting down for a refueling outage.
- Primary Coolant Pump P-50C spuriously tripped.
- The reactor did not automatically trip.
- Attempts to trip the reactor per EOP-1.0, “Standard Post-Trip Actions,” were unsuccessful.
- A 40 gpm emergency boration was initiated.

Which statement below correctly describes the expected effect on Moderator Temperature Coefficient (MTC) and Axial Shape Index (ASI) due to the emergency boration?

MTC will be __(1)__ and ASI will be __(2)__.

A. (1) Less negative
   (2) More negative (less positive)

B. (1) More negative
   (2) More negative (less positive)

C. (1) Less negative
   (2) More positive (less negative)

D. (1) More negative
   (2) More positive (less negative)

Proposed Answer: **A**

Explanation (Optional):

A. Correct, as negative reactivity is added to the PCS and temperature decreases, MTC will increase (become less negative) and ASI will decrease (become more negative). Due to the boration, MTC will add more positive reactivity to the top half of the core than the bottom half of the core and ASI will shift negative.
B. Incorrect, part 1 is incorrect. The applicant believes the value of MTC will become more negative as boron concentration rises.

C. Incorrect, part 2 is incorrect. The applicant believes that ASI will become more positive (less negative) as boron concentration rises.
D. Incorrect, both parts are incorrect.

Technical Reference(s): EM-04-17, General Physics course 192004
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ___None___________

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __1__
55.43 _____

Comments:

CORE PARAMETER CHANGE AND CORRESPONDING ASI EFFECT GUIDELINES

PARAMETER BEING CHANGED	CAUSE	CORE POWER WILL BE PUSHED TOWARDS	ASI WILL BECOME
PCS Temperature (with negative MTC)	Power Reduction	Top Half of Core	More negative or less positive
	Power Escalation	Bottom	More positive or less negative
Rod Position (ARO to Midpoint)	Withdrawal	Top	More negative or less positive
	Insertion	Bottom	More positive or less negative
PCS Boron	Boration	Top	More negative or less positive
	Dilution	Bottom	More positive or less negative

NOTE:	As the Moderator Temperature Coefficient (MTC) becomes more negative near EOC conditions, maintaining ASI within Target ASI ± 0.05 band may not be practical due to PDIL limitations above 75% power.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000038.G2.2.44	
	Importance Rating	4.2	

K/A Statement: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question:

Given the following conditions:

- The Plant was tripped due to a Steam Generator Tube Rupture.
- PCS pressure is 895 psia.
- PCS subcooling is 55°F.
- Primary Coolant Pumps P-50B and P-50C are running.
- 'A' S/G pressure is 790 psia.
- 'A' S/G has been isolated per EOP Supplement 12.
- 'A' S/G level is 73% NR and rising slowly.
- 'B' S/G level is 48% NR and stable.

Which ONE of the following is the preferred method to control level in the isolated steam generator and minimize the spread of contamination?

A. Steam the 'A' S/G to atmosphere via the Atmospheric Dump Valves.

B. Steam the 'A' S/G to the condenser via the Turbine Bypass Valve.

C. Restore S/G blowdown to the condenser from 'A' S/G.

D. Lower PCS pressure below "A" S/G pressure and allow backflow to the PCS.

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, this will lower S/G pressure to further below PCS pressure which will increase S/G level and increase the spread of contamination.
B. Incorrect, this will lower SG pressure to further below PCS pressure which will increase S/G level. Steaming to the condenser would minimize the chance of release to the environment, but still spread the contamination to the secondary.
C. Incorrect, re-establishing S/G blowdown will lower S/G level, but will spread contamination to the secondary.
D. Correct, this will lower PCS pressure and reduce SG level by moving water into the PCS. Contamination will be limited by putting the contaminated water back in the PCS. With PCPs running, the potential for boron dilution is reduced.

Technical Reference(s): EOP-5.0 Bases
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: _____None________

Learning Objective: ________________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _10__
55.43 _____

Comments:

d. **STEAM** the isolated S/G to atmosphere using the ASDVs.

CEN-152 SGTR Step 27:

*27. Maintain the isolated steam generator level less than [top of the indicated range] by **ANY** of the following methods:

- Lowering pressurizer pressure to below isolated steam generator pressure.
- Blowing down the isolated steam generator to the condenser.
- Steaming the isolated steam generator to the condenser.
- Steaming the isolated steam generator to atmosphere.

Technical Basis:

The intent of this step is to prevent overfilling of the isolated S/G.

The potential exists for primary coolant to flow through the tube rupture into the isolated S/G. This could fill the S/G steam space and the main steam piping to the MSIV and result in the inadvertent opening of the MSSVs. This would present an undesirable spread of contamination.

If the isolated S/G level is rising due to PCS in-leakage, it should be allowed to continue rising to greater than the top of the tube bundle (30% **[top of the tube bundle]**, then maintained less than 140% (110% for degraded containment conditions) **[top of indicating range]**. The intent of this step is to maintain the isolated S/G level to prevent over filling and keep the tubes covered.

Reducing PCS pressure below the isolated S/G pressure (ie, backflow to the PCS) can lower S/G level. Using S/G blowdown to the Radwaste System or the Main Condenser can also be done to reduce level and minimize the spread of contamination. If S/G draining is

not feasible or is insufficient, then steaming the S/G to the Main Condenser will also reduce level and minimize radioactivity release. As a last resort, the S/G may be steamed to the atmosphere. Before doing so, the TSC should assess the radioactive releases to the environment.

Backflow to the PCS can reduce the boron concentration of the PCS. This is a concern if PCPs are not running. The circumstance in which PCP operation has been lost, due to a loss of offsite power or other reasons, has been analyzed in CE-NPSD-990. This analysis considered the effects of back flow of S/G water while in natural circulation and the effects of restarting a PCP. As part of the conclusions, it was considered highly desirable to minimize any S/G back flow while in natural circulation operation. The sole exception to this was to allow back flow as a means of preventing the S/G water level from exceeding the high end of the indicating range.

This step which initiates backflow is different from subsequent EOP steps which initiate backflow. In the subsequent step where backflow is used to cool the most affected S/G, larger quantities of water will be transferred. Therefore, an additional substep was added to calculate the PCS boron change. Because of the smaller quantities of water being transferred, no calculation is necessary in this step.

The second option to blowdown the S/G provides two flow paths. The first flow path is preferred since it minimizes the contamination of the secondary systems. This method requires available space in the radwaste system. The second flow path directs blowdowns to either the hotwell or T-2 as specified by the SOP.

The third and fourth options provide for reducing level by steaming the generator. These options are least preferred since it results in a release to the environment. The use of the condenser is preferred since this will be a monitored release path and some scrubbing of radioactivity will occur in the condenser. Some radioactivity will be retained in the condenser. The fourth option provides a direct, unmonitored release path to the environment.

Training Emphasis

The delta P between S/G and pressurizer will not read zero when the S/G and pressure are at equal pressure as read by the control room instrumentation. Since the instruments read pressure on the steam space, no accounting is made for the differences in the water levels between the two vessels. For example if the pressurizer reads 0% level (628' 5") and the S/G is at 0% level (648') there is a difference of 20' 5" of head. Therefore the PCS pressure would have to read approximately 9 psia higher than the S/G pressure to have a zero d/p. This assumes equal densities. This means that the transfer of water will occur even when the PCS pressure is higher than the S/G pressure.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	CE/E05.EA1.03	
Importance Rating	3.4	

K/A Statement: Ability to operate and/or monitor the following as they apply to the (Excess Steam Demand): Desired operating results during abnormal and emergency situations

Proposed Question:

The following temperatures exist after an Excess Steam Demand Event in which the 'B' S/G has blown dry:

- Loop 1A T_{cold}: 432ºF
- Loop 1A T_{hot}: 464ºF
- Loop 2B T_{cold}: 424ºF
- Loop 2B T_{hot}: 456ºF

The CRS has directed the NCO to stabilize PCS pressure and temperature. Which one of the temperature indications given above should be used to manually control the Atmospheric Dump Valve in order to stabilize temperature with the least amount of PCS heatup?

A. Loop 1A T_{cold}

B. Loop 1A T_{hot}

C. Loop 2B T_{cold}

D. Loop 2B T_{hot}

Proposed Answer: A

Explanation (Optional):

A. Correct, the unaffected S/G T_{cold} would be used to minimize the heatup.
B. Incorrect, corresponds to the T_{cold} of the unaffected S/G. Plausible, as the applicant could not understand the least amount of PCS heatup requirement.
C. Incorrect, corresponds to the T_{hot} of the affected S/G. Plausible, as the applicant could chose to use the affected S/G. This could be true in a dual event scenario.
D. Incorrect, corresponds to the T_{hot} of the unaffected S/G. Plausible, as the applicant could chose to use the affected S/G. This could be true in a dual event scenario.

Technical Reference(s): EOP-6.0, Steam Tables
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam St Lucie 2008
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments: Yes, Steam Tables are normally provided as a reference.

STEP 16

CAUTION

When ALL PCPs are stopped, steaming the least affected S/G must occur prior to dryout of the most affected S/G to prevent lifting PZR Code Safety Valves or Pressurized Thermal Shock rupture of the PCS.

© 16. **STABILIZE** PCS temperature as follows:

a. **MAINTAIN** level in the least affected S/G between 60% and 70%.

b. IF the steam leak is isolated, THEN **ESTABLISH** steam flow from BOTH S/Gs using the Atmospheric Steam Dump Valves.

WARNING

IF Containment pressure is higher than the most affected S/G pressure AND the ESDE is inside of containment, THEN opening of the ASDVs on the most affected S/G will provide a direct release path to the environment.

NOTE: Steaming BOTH S/Gs using ASDVs is permitted prior to isolation of the most affected S/G if necessary to control temperature /pressure of the least affected S/G.

c. IF the steam leak is NOT isolated,
THEN **STEAM** the least affected S/G as necessary to maintain the following, as applicable:

- WHEN T_cs in the affected loop are lowering,
THEN **MAINTAIN** the least affected S/G pressure within 50 psid above the most affected S/G pressure

- WHEN T_cs in the affected loop are NOT lowering,
THEN **STABILIZE** PCS T_cs using the least affected S/G

CEN-152 ESDE Step 13:

*13. Stabilize RCS temperature by controlled steaming of the least affected steam generator using the atmospheric dump valves

Technical Basis:

The intent of this step is to direct the operator to control and stabilize PCS temperature following dry out of the most affected S/G during an ESDE.

PCS temperature will rise after the most affected S/G dries out unless a means of controlling PCS heat removal is established. The rise in PCS temperature may result in a water-solid condition due to inventory added from Safety Injection and Charging operation during the blowdown phase of the event. The post dryout heatup and repressurization also presents a PTS concern.

In order to mitigate PCS heatup, a controllable heat removal method must be established prior to dryout of the most affected S/G. The intent is to regain PCS temperature control and stabilize T_cs, thus preventing an uncontrolled PCS heatup and repressurization. Cooling down the least affected S/G by controlled steaming will aid in minimizing the amount of heatup that would occur following a loss of heat removal on the most affected S/G. By maintaining the least affected S/G pressure within 0 to 50 psig of the most affected S/G will keep the least affected S/G nearly coupled with the remainder of the PCS and yet

prevent adding to the cooldown of the PCS. As the most affected S/G approaches a dryout condition its S/G pressure will drop more rapidly. Attempting to keep the least affected S/G within 50 psi at this point would result in excessive cooldown of the PCS. When it is noted that the loop T colds on the most affected loop rate of decline is slowing, then control of the least affected S/G must be controlled by comparisons of the loop temperatures.

Feedwater temperature and flow rate can have a significant affect on PCS temperatures. Therefore, the affect of feedwater should be taken into account when attempting to stabilize PCS temperatures.

Training Emphasis

Controlling of the cooldown of the least affected S/G must be performed with care. It is important to maintain the least affected S/G as close to the most affected S/G as possible until the most affected S/G starts to loose the ability to remove heat. Use of pressure indications is acceptable until the most affected S/G starts to lose the ability to remove heat. At that point the operator must use the temperature indications and stabilize PCS temperatures.

Associated Notes, Cautions, Warnings:

The cautions alerts the operator of the importance of maintaining the least affected S/G close to the most affected S/G. Failure to do so significantly raises the probability of exceeding PTS conditions.

The warning alerts the operator that if Containment pressure is higher than the affected S/G pressure and the ESDE is inside of Containment, then opening of the ASDVs on the affected S/G will provide a direct release path from containment to the environment. This release path needs to be accounted for by the TSC.

The note on ASDV isolation informs the operator that if unable to isolate ASDVs on the most affected S/G, then steaming both S/Gs using the ASDVs is acceptable to control the temperature and pressure of the least affected S/G until the most affected S/G can be isolated.

Deviations from EPG:

Maintaining the least affected S/G pressure within 50 psi above the most affected S/G pressure is an addition to the CEN-152 requirements.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	CE/E06.EK2.2	
	Importance Rating	3.5	

K/A Statement: Knowledge of the interrelations between the (Loss of Feedwater) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question:

A loss of feedwater accident has occurred at the end of core cycle. When the NCO attempted to trip the Reactor per EOP-1.0, "Standard Post-Trip Actions," all Control Rods remained fully withdrawn.

Which of the following describes the initial response of the following plant parameters in response to this event, with no operator action?

	Reactor Power	Pressurizer Pressure	S/G Pressure
A.	rising	rising	rising
B.	rising	lowering	lowering
C.	lowering	rising	rising
D.	lowering	lowering	lowering

Proposed Answer: **C**

Explanation (Optional):

In an ATWS scenario with a loss of all feedwater, the reduction of secondary system heat removal capability will cause PCS temperature to rise, along with an increase in PCS pressure. Reactor power will decrease. On the secondary, a loss of feedwater to the S/Gs will cause S/G pressure to increase

A. Incorrect, reactor power will decrease due to the addition of negative reactivity via rising PCS temperature.
B. Incorrect, see choice A. Pressurizer pressure and S/G pressure will rise (see explanation).
C. Correct, see explanation.
D. Incorrect, pressurizer pressure and S/G pressure will rise (see explanation).

Technical Reference(s): FSAR Chapter 14 Section 14.13 Rev 24

(Attach if not previously provided, ______

including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ___None___________

Learning Objective: __________________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

14.13 LOSS OF NORMAL FEEDWATER

14.13.1 EVENT DESCRIPTION

A Loss of Normal Feedwater Flow (LNFF) event is initiated by the trip of the Main Feedwater (MFW) pumps or a malfunction in the feedwater control valves. The loss of MFW flow decreases the amount of subcooling in the secondary side downcomer leading to an increase in the primary coolant system (PCS) temperature. As the PCS temperature increases, the coolant expands into the pressurizer which increases the pressure by compressing the steam volume.

Steam generator liquid levels, which steadily drop after termination of MFW flow, soon reach the low steam generator level reactor trip setpoint and the low steam generator level Auxiliary Feedwater (AFW) actuation setpoint. This initiates the starting sequence for the AFW pumps and initiates a reactor scram, which ends the short-term-heatup phase of the event. When the delivery of AFW begins, the rate of level decrease in the steam generators slows.

The automatic turbine trip at reactor scram and the continuing primary-to-secondary transfer of the decaying core power and the reactor coolant pump heat cause steam generator pressures to rapidly increase. When steam generator pressures and coolant temperatures have increased to the appropriate values, the steam dump system and/or the Main Steam Safety Valves (MSSVs) serve to limit the increase in steam generator pressures.

Eventually, a long-term-heatup phase of the event may begin if primary-to-secondary heat transfer degrades as a result of steam generator tube uncovery.

As the decay heat level drops, liquid levels in the fed steam generators stabilize and then begin to rise. Also, reactor coolant temperatures stabilize and then begin to decrease. These conditions mark the end of the challenge to the event acceptance criteria.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000056.AK1.03	
Importance Rating	3.1	

K/A statement: Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Definition of subcooling: use of steam tables to determine it.

Proposed Question:

Given the following conditions:

- The reactor tripped from 100% full power conditions.
- The Plant experienced a Loss of Offsite Power and has entered EOP-8.0, "Loss of Offsite Power / Forced Circulation Recovery."

Which of the following sets of indications would verify natural circulation flow in at least one Primary Coolant System loop in accordance with EOP-8.0?

	PZR Pressure	Average CET	Loop T_{hot}(s) (^oF)	Loop T_{cold}(s) (^oF)
A.	1500 psig, stable	571°F, stable	566°F, lowering	530°F, stable
B.	1550 psig, stable	587°F, stable	576°F, lowering	540°F, lowering
C.	1600 psig, stable	578°F, lowering	561°F, stable	530°F, lowering
D.	1650 psig, stable	582°F, stable	575°F, lowering	525°F, stable

Proposed Answer: **A**

Explanation (Optional):

15 minutes following the SBO, Natural Circulation is developing. T_{hot} and T_{cold} separate, but the delta T between should be no more than 50°F per Natural Circulation criteria.

The Natural Circulation criteria, per EOP-8.0, are:

1) Core delta T less than 50°F (Average CET minus T_c)
2) Loop T_{hot}(s) and Loop T_{cold}(s) stable or lowering
3) Average CET at least 25°F subcooled
4) Difference between Loop T_{hot} and Average CET is less than or equal to 15°F

A. Correct, all criteria are met (T_{sat} = 597°F)
B. Incorrect, criteria 3 not met, subcooling = 15°F (T_{sat} = 602°F)
C. Incorrect, criteria 4 not met, CET – T_{hot} = 17°F (T_{sat} = 606°F)
D. Incorrect, criteria 1 not met, loop delta T = 57°F (T_{sat} = 610°F)

Technical Reference(s):	EOP-8.0, steam tables		
(Attach if not previously provided,	________		
including version/revision number)	________		

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective: ________ (As available)

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

Question History: Last NRC Exam ____

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	5
	55.43	____

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS
19. IF ALL PCPs are stopped, THEN **VERIFY** natural circulation flow in at least one PCS loop by ALL of the following: • Core ΔT less than 50°F (Average of Qualified CETs minus T_C) • Loop T_Hs and Loop T_Cs constant or lowering • Average of Qualified CETs at least 25°F subcooled • Difference between Loop T_H and Average of Qualified CETs is less than or equal to 15°F	19.1 **ENSURE** proper control of S/G feeding and steaming rates.
20. IF ANY 4160 VAC or 2400 VAC buses are NOT energized, THEN **ENSURE OPEN** ALL feeder and supply breakers on the affected buses. Refer to EOP Supplement 34.	

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000057. G2.4.03	
Importance Rating	3.7	

K/A Statement: Ability to identify post-accident instrumentation.

Proposed Question:

A loss of Preferred AC Bus EY-10 has caused the loss of multiple indications. Which of the following indications will remain available for post-accident use with no operator action?

A. LIA-0102A, PZR Level (Cold Calibrated).

B. LTIR-0101A, Reactor Vessel Level/Qualified Core Exit Thermocouples.

C. SMM-0114, Subcooling Margin Monitor.

D. RIA-2321, Refueling Containment Area Monitor

Proposed Answer: C

Explanation (Optional):

A. Incorrect, LIA-0102A will fail low on a loss of EY-10. (AOP-12, Att. 1 pg 2 of 5)
B. Incorrect, LTIR-0101A will lose power with a loss of EY-10. (AOP-12, Att. 1 pg 2 of 5)
C. Correct, SMM-0114 is powered by Preferred AC Bus EY-30 and would remain unaffected by a loss of EY-10.
D. Incorrect, RIA-2321 will lose power with a loss of EY-10. (AOP-12, Att. 1 pg 4 of 5)

Technical Reference(s): AOP-12, LCO 3.3.7
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Pal 2012 audit exam question used for reference, but question is New.

LOSS OF PREFERRED AC BUS EY-10

LOST AND REDUNDANT INSTRUMENTATION

COMPONENT	SERVICE	AVAILABLE REDUNDANT COMPONENT(S)	ALTERNATE COMPONENTS/COMMENTS
FI-0308A & B	HPSI Flow to Loop 1A (EC-13 & EC-33)	None	Monitor Flow on other HPSI Loop FIs (EC-13 & EC-33)
HIC-0780A, 0780B, 0781B	Steam Dump Control (EC-01 & EC-33)	None	Use controller PIC-0511, Turbine Bypass CV-0511. **REFER TO** Technical Specifications LCO 3.7.4.
LI-1107A & B	West ESS Sump Level (EC-03 & EC-33)	None	None
LIA-0102A	PZR Level (Cold Calibrated) (EC-02)	LI-0103A	**REFER TO** Technical Specifications LCO 3.3.7.
LIA-0332A	SIRWT Level (EC-13)	LIA-0331 (EC-13)	LIA-0332B (EC-150). **REFER TO** Technical Specifications LCO 3.5.4.
LIA-0365	T-82A	None	High/Low Level Switch Alarms Available
LIA-0920	CCW Surge Tank Level (EC-08)	LIA-0917	**MONITOR** CCW Surge Tank Local Site Glass.
LPIR-0383	Containment Wide Range Pressure/ Level Recorder (EC-13)	LPIR-0382	None. **REFER TO** Technical Specifications LCO 3.3.7 & LCO 3.4.15.
LS-0327	SIRW Tank Low Level	None	Level switch assumes tripped condition and satisfies one of the two logic inputs for RAS actuation.
LTIR-0101A	Reactor Vessel Level/Qualified Core Exit Thermocouples	LTIR-0101B	Non-Qualified Core Exit Thermocouples. **REFER TO** Technical Specifications LCO 3.3.7.
NI-5	Power Range NI	NI-6, NI-7, NI-8	NI-1/3A, NI-2/4A. **REFER TO** Technical Specifications LCO 3.2.1, LCO 3.2.3, LCO 3.2.4, LCO 3.3.1 and ORM 3.11.2, ORM 3.17.6.

LOSS OF PREFERRED AC BUS EY-30

LOST AND REDUNDANT INSTRUMENTATION

COMPONENT	SERVICE	AVAILABLE REDUNDANT COMPONENT(S)	ALTERNATE COMPONENTS/COMMENTS
PTR-0112	PCS Loop 1 Wide Range - T_C PCS Loop 1 Wide Range - T_H PCS Wide Range Pressure	PTR-0122	Recorder fails as is. Palisades Plant Computer (PPC). **REFER TO** Technical Specifications LCO 3.3.7.
RIA-1807	Containment Area Monitor	RIA-1805, RIA-1806, and RIA-1808	Affected RIA assumes tripped condition. **REFER TO** Technical Specifications LCO 3.3.3.
SIAS-LEFT Initiation	Left Channel of SIAS Initiation	Right Channel of SIAS Initiation	**REFER TO** Technical Specifications LCO 3.3.4.
SIS Det Ckt 3	SIS Detection Circuit	SIS Det Ckt 1, 2, and 4	Affected SIS detection circuit assumes tripped condition. **REFER TO** Technical Specifications LCO 3.3.3.
SMM-0114	Subcooling Margin Monitor	SMM-0124	Manually Calculate Subcooling Margin. **REFER TO** Technical Specifications LCO 3.3.7.

Table 3.3.7-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION E.1
1.	Primary Coolant System Hot Leg Temperature (wide range)	2	F
2.	Primary Coolant System Cold Leg Temperature (wide range)	2	F
3.	Wide Range Neutron Flux	2	F
4.	Containment Floor Water Level (wide range)	2	F
5.	Subcooled Margin Monitor	2	F
6.	Pressurizer Level (wide range)	2	F
7.	(Deleted)		
8.	Condensate Storage Tank Level	2	F
9.	Primary Coolant System Pressure (wide range)	2	F
10.	Containment Pressure (wide range)	2	F
11.	Steam Generator A Water Level (wide range)	2	F
12.	Steam Generator B Water Level (wide range)	2	F
13.	Steam Generator A Pressure	2	F
14.	Steam Generator B Pressure	2	F
15.	Containment Isolation Valve Position	1 per valve[a]	F
16.	Core Exit Temperature - Quadrant 1	4	F
17.	Core Exit Temperature - Quadrant 2	4	F
18.	Core Exit Temperature - Quadrant 3	4	F
19.	Core Exit Temperature - Quadrant 4	4	F
20.	Reactor Vessel Water Level	2	G
21.	Containment Area Radiation (high range)	2	G

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000058.AK3.01	
	Importance Rating	3.4	

K/A Statement: Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of DC control power by D/Gs.

Proposed Question:

Diesel Generator (DG) 1-2 is running partially loaded during a surveillance test run when DC breaker 72-401 (DG 1-2 Field Flashing) on D-21A trips.

Which one of the following describes the effect, if any, on DG 1-2 and why?

A. DG 1-2 output breaker, 152-213, trips on overcurrent to protect the generator from the loss of voltage condition.

B. DG 1-2 output breaker, 152-213, trips on a loss of excitation due to a loss of generator load, to protect the engine from an overspeed condition.

C. No effect on DG 1-2. Field current supplied by the exciter is controlled by the generator voltage regulator automatically after engine startup.

D. No effect on DG 1-2. Field current is not required after the generator develops sufficient voltage upon startup.

Proposed Answer: **C**

Explanation (Optional):

While the DG output breaker will trip on a loss of excitation (as well as an overcurrent condition), a loss of excitation or overcurrent will not occur due to a loss of DC field flashing power when the DG has reached rated voltage as it has in this case. The DC bus supplies the current for the field flashing as the DG starts, and upon reaching 70% of its rated voltage, the DG exciter will supply the field current. The voltage regulator will maintain this such that there would be no overall impact to the DG.

A. Incorrect, the DG will remain running with the output breaker closed. See explanation.
B. Incorrect, the DG will remain running with the output breaker closed. See explanation.
C. Correct, there is no effect to the DG since the DG is running at rated speed and voltage.
D. Incorrect, field current is required, but the field current is supplied by the DG's exciter.

Technical Reference(s): DBD-5.06, E-8 Sheet 2
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: ____None_________

Learning Objective: _________________________ (As available)

Question Source: Bank # _______

Modified Bank # ___X___ (Note changes or attach parent)

New _______

Question History: Last NRC Exam __________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __8__

55.43 _____

Comments:

Modified from 2007 NRC Exam. Stem of question changed to use D/G 1-2 only partially loaded during the surveillance run. Rearranged and reworded distractors, and changed one distractor (choice B).

Engine Started/Generator Field Flashing

Since the shaft of the generator is driven by the crankshaft of its associated diesel engine, the generator is "started" mechanically whenever the diesel engine is started. When the engine control logic determines that the engine has in fact started, an "engine started" interlock is provided to the generator control system.

In response to this "engine started" interlock, the generator field current shorting signal is removed, and field flashing current from the external 125V DC power source is admitted to the generator. The selected generator voltage control mode (normally the automatic mode, as when the system is aligned for automatic response to emergencies) is activated. Furthermore, the "generator-field-noT-activated" standing-trip interlock is removed from the generator breaker to permit later connection of the generator to its associated 2400V AC load bus.

Generator Acceleration/Output Voltage Buildup

As the engine/generator is accelerated by the mechanical speed control portion of the engine governor, the external field flashing source supplies the generator field current necessary to obtain quick output voltage buildup. Once the generator output voltage has reached a level for sufficient self-excitation, the external 125V DC generator field flashing source is shut off automatically. Since the field flashing command signal is always present under "engine started" conditions, and is subsequently defeated only when generator output voltage is sufficient for self-excitation, the Palisades design is not susceptible to the loss of generator output on an emergency start shortly following a normal shutdown. This was a problem noted elsewhere in the industry (Reference 227).

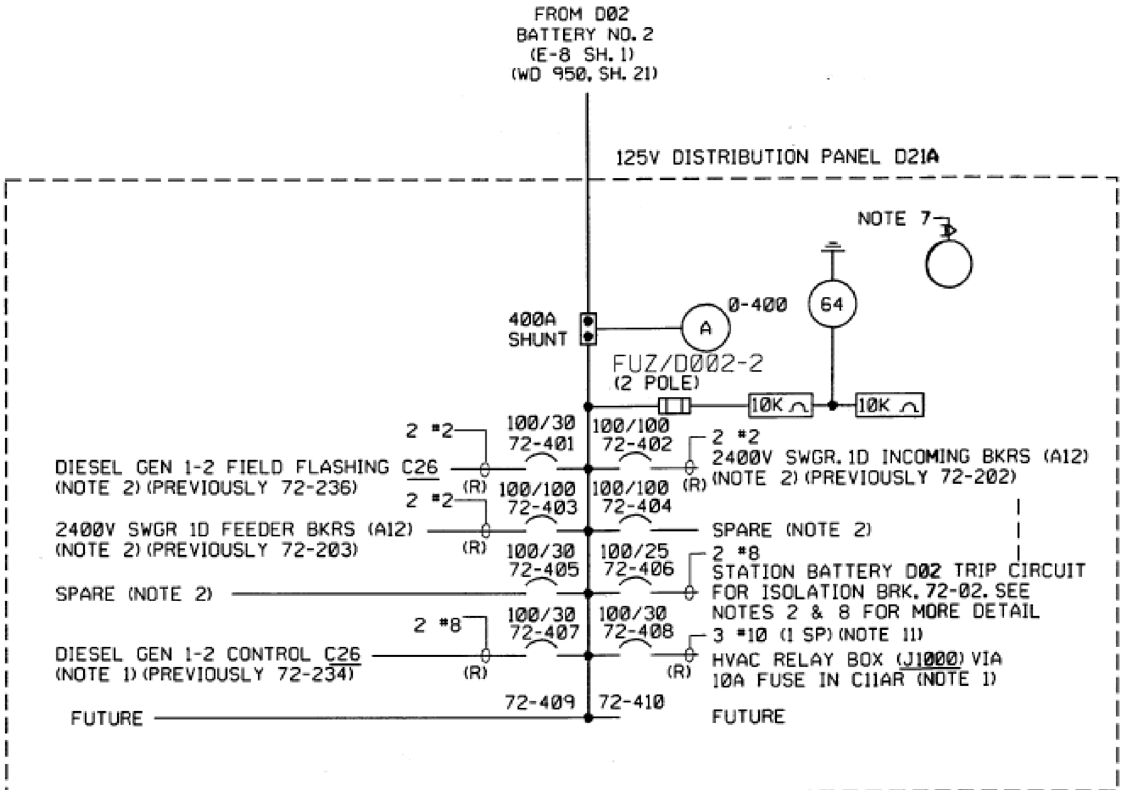
The generator field current supplied by the exciter is controlled by the generator voltage regulator, which will provide automatic generator output voltage control in the automatic mode. In this mode, the exciter and voltage regulator work together to raise the generator output voltage to the required setting during startup, and also to maintain the output voltage at the required level, once attained.

EDra

2) Automatic field flashing is provided from the 125V DC system at 18 Amperes for rapid voltage buildup upon starting.

   a) It is automatically removed when the regulator commences operation (when generator voltage is approximately 70% of nominal).

P&ID E-8 Sheet 2



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000062.AA1.06	
	Importance Rating	2.9	

K/A Statement: Ability to operate and/or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Control of flow rates to components cooled by SWS

Proposed Question:

Given the following conditions:

- The Plant was manually tripped from 80% power due to a small break LOCA.
- A valid Safety Injection Actuation Signal (SIAS) was received.
- A Loss of Offsite Power (LOOP) occurred
- Diesel Generator (DG) 1-1 started and loaded normally
- DG 1-2 started and tripped on overspeed

Service Water cooling demands can be met with __(1)__ Service Water pump(s) if only DG 1-1 is operating, provided manual operator action is taken to isolate service water to __(2)__.

A. (1) ONE
   (2) non-critical service water header

B. (1) ONE
   (2) containment

C. (1) TWO
   (2) non-critical service water header

D. (1) TWO
   (2) containment

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, while only P-7B will be operating due to the failure of DG 1-2 to energize 1D Bus, the non-critical service water header is isolated automatically on the SIAS. Manual operator action is required to isolate service water cooling flow from containment. Containment air coolers VHX-1, VHX-2, VHX-3 fans are powered from Bus 1D (as Service Water pump P-7A and P-7C are), water not required for non-operating fans is necessary for other equipment. VHX-4 fan is powered from Bus 1C, however, the service water supply valve to VHX-4 is closed upon the SIAS.

B. Correct, see choice A explanation.
C. Incorrect, see choice A explanation.
D. Incorrect, see choice A explanation

Technical Reference(s): EOP-9.0, DBD-1.02
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:
Procedural guidance to support required actions are found in AOP-35.

SAFETY FUNCTION STATUS CHECK SHEET

SAFETY FUNCTION
MAINTENANCE OF VITAL AUXILIARIES - WATER

RESOURCE TREE
H

SUCCESS PATH	ACCEPTANCE CRITERIA	CRITERIA SATISFIED
MVAW-1: Service Water and Component Cooling Water	a. Equipment in operation per the following table:	___ ___ ___ ___

<table>
<tr><th colspan="2" rowspan="2">Components</th><th colspan="3">Electrical Buses Energized</th></tr>
<tr><th>'C' & 'D' Buses</th><th>'C' Bus</th><th>'D' Bus</th></tr>
<tr><td colspan="2">P-7A & P-7C</td><td rowspan="2">≥ 1 NO SIAS
≥ 2 with SIAS</td><td></td><td>≥ 1 NO SIAS
2 with SIAS</td></tr>
<tr><td colspan="2">P-7B</td><td></td><td rowspan="2"></td></tr>
<tr><td colspan="2">P-52A & P-52C</td><td rowspan="2">≥1 NO SIAS
≥ 2 with SIAS</td><td>≥ 1 NO SIAS
2 with SIAS</td></tr>
<tr><td colspan="2">P-52B</td><td></td><td></td></tr>
<tr><td rowspan="2"><u>IF</u> RAS actuation</td><td>Service Water Return from Containment CV-0824 closed</td><td></td><td></td><td></td></tr>
<tr><td>Both CCW Hxs in operation</td><td></td><td></td><td></td></tr>
</table>

b. BOTH Critical SW headers in operation with pressures greater than or equal to 36 psig. ___ ___ ___ ___

c. NO high temperature conditions on operating equipment cooled by the Critical SW headers in operation. ___ ___ ___ ___

b. If the SIS is accompanied by loss of offsite power, the service water pumps are sequenced onto the DG buses. A 1E power source provides power to the SWS instrumentation and valves required for operation during an accident.

c. If the SIS is accompanied by loss of offsite power with a failure of DG 1-2 to start, only service water pump P-7B is started by DG 1-1. When the operator recognizes that DG 1-2 Bus 1D is not energized, containment air coolers service water return valve CV-0824 must be shut from the control room to isolate service water to containment since containment air coolers VHX-1, VHX-2, and VHX-3 fans are powered from the same bus (1D) as service water pumps P-7A and P-7C. With only one service water pump running and the resulting reduced capacity, water not required for non-operating fans VHX-1, VHX-2, and VHX-3 is provided to other equipment.
(Reference EOP-9.0, MVAW-2 and AOP-35).

EOP-1 adds a new option for isolating SW by closing CAC High Capacity Valves. A service water pressure of 42 psig is used as the low pressure limit.

When the water inventory in the safety injection and refueling water (SIRW) tank reaches the low level point, a recirculation actuation signal (RAS) occurs which realigns the safety injection pump suctions from the SIRW tank to the containment sump. The RAS effect on the SWS is to close the temperature control valves on the service water side of the component cooling water heat exchangers (CCWHXs) and open the CCWHX service water high capacity outlet valves, CV-0823, CV-0826. This results in maximum service water flow to the CCWHXs.

See Section 5.0 for other equipment which continues to be cooled by service water.

d. References 68 and 84 describe the proposed duration of station blackout (total loss of ac power) as four hours. Service water is not available during this time to mitigate containment heatup (Reference 69) or to mitigate control room heatup (Reference 70).

9.1.2.3 System Operation

1. Normal Operation

   Two service water pumps are required to furnish the normal cooling water demand; the third pump will normally be on standby. Two pressure switches are provided in the discharge of each pump connecting to the starting circuits of the remaining two pumps. If the service water pressure falls below a preset value, one of the switches initiates automatic starting. The auto-start feature is automatically reset on bus undervoltage to prevent cycling the pump breaker onto a dead bus.

2. Shutdown Operation

   Service water flow requirements during shutdown cooling will remain essentially the same as for normal operation. This is due to the fact that there are no significant heat loads on the non-critical header, but service water flow to the CCW heat exchanger increases significantly while on shutdown cooling. Both component cooling water heat exchangers are used to cool the primary coolant from 300°F to the refueling temperature. Service water flow is maintained to equipment on an as-needed basis (eg, FWP air compressors, containment air coolers.)

3. Post-DBA Operation

   Either one or two service water pumps are required to provide cooling in the event of a DBA, depending on the accident events. If Plant offsite power sources are lost, all pump motors are automatically supplied with power from the emergency diesel generators with one pump on Diesel 1-1 and two pumps on Diesel 1-2. Cooling water demands can be met with one pump if only Diesel 1-1 is operating provided service water to containment is isolated, and with two pumps if only Diesel 1-2 is operating.

   Service water through most noncritical systems is terminated by automatic closure of the noncritical header shutoff valve on a Safety Injection Signal (SIS), thus ensuring that all available service water is routed to the critical systems. The automatic shutoff valve can also be actuated remotely from the main control room or by a local handwheel. This does not isolate all non-critical loads; instrument air compressors C-2A and C-2C aftercoolers are on critical service water headers B and A, respectively, and are not isolated because their service water requirements are not significant (see Table 9-1).

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000065.AA2.07	
Importance Rating	2.8	

K/A statement: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Whether backup nitrogen supply is controlling valve position.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power.
- A transient occurred that resulted in Instrument Air header pressure indicating 62 psig and slowly lowering.
- The crew is implementing AOP-37, "Loss of Instrument Air."

Which of the following valves will remain unaffected as a result of the transient?

A. CV-2099, PCP Controlled Bleedoff Containment Isolation

B. CV-0847, SW Supply to Containment

C. CV-1212, Service Air Header Isolation

D. CV-0909, CCW Outlet from Letdown Heat Exchanger

Proposed Answer: B

Explanation (Optional):

A. Incorrect, CV-2099 fails closed on a loss of instrument air pressure.
B. Correct, CV-0847 uses nitrogen backup from nitrogen backup station 1A and >80psig nitrogen will passively allow operation of the valve for 24 hours following the loss of compressor operability.
C. Incorrect, CV-1212 fails closed on a loss of instrument air pressure to isolate Service Air from Instrument Air.
D. Incorrect, CV-0909 will fail open on a loss of instrument air pressure to ensure letdown is maintained adequately cooled.

Technical Reference(s): AOP-37, DBD-1.05
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source: Bank # ________

Modified Bank # ________ (Note changes or attach parent)

New __X___

Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __8__

55.43 ______

Comments:

BOTTLED GAS BACKUP STATIONS - GENERAL

a. Provide a backup pressure source for operation of vital valves in the event of a DBA with loss of offsite power, or a Station Blackout (SBO), or a fire that destroys critical circuits.

b. Under Appendix R the plant was required to be on shutdown cooling within 72 hrs to achieve safe and stable conditions. Under NFPA 805 safe and stable conditions of hot standby and Keff <0.99 must be achieved and maintained with no specified time period. Once PNPs license amendment request is approved, NFPA 805 requirements will apply. Until then both Appendix R and NFPA requirements must be considered. Nitrogen backup station design and operating requirements established under Appendix R may have changed under NFPA 805 fire risk evaluations. Refer to section 3.4.3.5 for more details on NFPA 805 design requirements and references to evaluations justifying transition from Appendix R.

NITROGEN BACKUP STATION 1

a. Provide motive pressure >60 psig to allow operation of the auxiliary feedwater regulating valves CV-0727 & CV-0749 for 8 hours (Reference 78) following the loss of compressor operability.

b. Maintain valve operability for Appendix R and SBO requirements. SOP-19 identifies restrictions that are applicable if Station 1 is inoperable.

c. Used for SBO (4 hours required), Appendix R (12 hours required). Local handwheel meets the last 4 hours of Appendix R requirement (Reference 78).

d. Bottle pressure maintained >2000 psig.

e. Required to meet and maintain NFPA safe and stable conditions. Continued use beyond design ability occurs through bottle replacement or a switch to manual valve operation.

NITROGEN BACKUP STATION 1A

a. Provide motive pressure >80 psig to allow operation of valve CV-0847 SW header B to containment for 24 hours following loss of compressor operability.

b. Required for post fire safe shutdown (Appendix R).

c. If Station 1A is inoperable at the same time that Station 3B (CV-0824) is inoperable, then Technical Specification LCO 3.7.8 is entered for Service Water Pump P-7B.

ACTIONS\EXPECTED RESPONSE | **RESPONSE NOT OBTAINED**

NOTE: Erratic equipment operation may occur whenever Instrument Air pressure is less than 50 psig.

NOTE: At approximately 50 psig Instrument Air pressure, CV-0823 and CV-0826, CCW Hx SW Outlet CVs, will fail open. If the plant is on Shutdown Cooling, this will affect Shutdown Cooling return temperature. The Control Operator may have to take prompt action to prevent overcooling of the PCS.

NOTE: At approximately 25 psig Instrument Air pressure, CV-3006, SDC Hx Bypass, and CV-3025, SDC to LPSI, will be affected, and control of Shutdown Cooling will be lost.

5. IF erratic equipment behavior observed on equipment required for safe plant operation, THEN **TRIP** the Reactor if reset. Refer to EOP-1.0, "Standard Post-Trip Actions."

6. IF directed by the Shift Manager, THEN **COMMENCE** a power reduction.

   a. **REFER TO** GOP-8, "Power Reduction and Plant Shutdown to Mode 2 or Mode 3 ≥ 525°F."

LOSS OF INSTRUMENT AIR

VALVES WHICH FAIL CLOSED

VALVE	DESCRIPTION	REQUIRED ACTION
CV-1212	Service Air Header Isol	IF the 1-3 Station Power Transformer Deluge System or Track Alley Sprinkler System have been activated, AND fire is NOT evident, THEN **PERFORM** the following as applicable:
		CLOSE MV-FP204, S/P Xfmr 1-3 Deluge Isolation, to isolate Deluge to Station Power Transformer 1-3.
		CLOSE MV-FP256, Track Alley Sprinkler Sys Isolation, to isolate the Track Alley Sprinkler System.
		CLOSE MV-FP652, VRS Area Sprinkler System Isolation, to isolate VRS Sprinkler System.
		NOTIFY Security to commence compensatory measures.
CV-0730	Condensate Pumps P-2A/B Recirc	**STOP** one Condensate Pump. • P-2A • P-2B
		BLEED air from Main Feed Pump Recirc Valve CV-0710 or CV-0711 by closing the air supply to its solenoid valve and removing the air line from the top of the valve operator.
		ENSURE Aux Oil Pump running in HAND on Main Feed Pump whose Recirc CV is open.

LOSS OF INSTRUMENT AIR

VALVES WHICH FAIL CLOSED

NOTE: Valves identified with an * are affected by isolating air to Containment.

VALVE	DESCRIPTION	REQUIRED ACTION
*CV-0148	Quench Tank Drain Control Valve	NONE
*CV-0150	Quench Tank Nitrogen Supply	
*CV-0152	Quench Tank T-73 Vent Valve	
*CV-0155	Penet MZ-42 Quench Tank T-73 Spray Valve	
*CV-2003	Letdown Orifice Stop Valve	**MAINTAIN** Pressurizer level by securing P-55A and allowing P-55B or P-55C to operate intermittently in automatic
*CV-2004	Letdown Orifice Stop Valve	
*CV-2005	Letdown Orifice Stop Valve	
*CV-2202	Letdown Orifice Bypass Control	
*CV-2002	Letdown Orifice Bypass Valve	
CV-2155	Make-up Stop	**OPEN** MO-2160, SIRWT T-58 Outlet to Charging PP P-55A, B, C. **CLOSE** MO-2087, VCT T-54 Outlet
CV-2083	PCP P-50A, B, C &D Controlled Bleedoff	**NOTE**: An accumulator maintains CV-2191 open on loss of air. IF desired to maintain CV-2191, PCP Controlled Bleedoff Stop open prior to accumulator bleeding off, THEN **MANUALLY OPEN** CV-2191 (if Containment entry is possible). IF CV-2191 closes, THEN **SECURE** all Primary Coolant Pumps.
CV-2099	PCP Controlled Bleedoff Containment Isolation	
*CV-1057	Pressurizer Spray Valve from Loop 1B	**MANUALLY CONTROL** Pressurizer Heaters to maintain pressure as desired.
*CV-1059	Pressurizer Spray Valve from Loop 2A	
*CV-2117	Pressurizer Auxiliary Spray Valve	

LOSS OF INSTRUMENT AIR

VALVES WHICH FAIL OPEN

NOTE: Valves identified with an * are affected by isolating air to Containment.

VALVE	DESCRIPTION	REQUIRED ACTION
*CV-2111	Charging Line Stop Valve	NONE
*CV-2113	Charging Line Stop Valve	
*CV-2115	Charging Line Stop Valve	
	Feedwater Heater High Level Dump Valves	NONE
		NOTE: CV-0911 and CV-0940 have accumulators.
CV-0910	CCW Containment Isolation Valve	IF valves are required to be closed, THEN **REFER TO** Step 13 of this procedure.
CV-0911	CCW Containment Isolation Valve	
CV-0940	CCW Containment Isolation Valve	
CV-0909	CCW outlet from Letdown Heat Exchanger	**REFER TO** ARP-4, EK-0706, LETDOWN HX COOLING EXCESS FLOW.
CV-3006	Shutdown Cooling Hx Bypass	**REFER TO** AOP-30, "Loss of Shutdown Cooling."

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000077.AK2.07	
	Importance Rating	3.6	

K/A statement: Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine / generator control

Proposed Question:

Given the following conditions:

- A grid disturbance occurred due to severe weather in the area.
- The Plant is at 825 MWe.
- Generator reactive load is 225 MVAR OUT.
- Generator Hydrogen pressure is 75 psig.
- The following annunciators are LIT.
  - EK-0303, “VOLTAGE REGULATOR LIMITER OPERATION.”
  - EK-0310, “GENERATOR VOLTAGE REG TRIP.”

To restore Generator parameters, the Control Room Supervisor (or CRS) should direct the NCO-T to __(1)__ reactive load by using the __(2)__.

A. (1) RAISE
(2) AC Adjuster

B. (1) RAISE
(2) DC Adjuster

C. (1) LOWER
(2) AC Adjuster

D. (1) LOWER
(2) DC Adjuster

Proposed Answer: **D**

Explanation (Optional):

The Generator has three protective functions, the Maximum Excitation Limiter (MXL), the On-line Field Forcing (FF) relay, and the Over Excitation Protection (OXP) relay. The MXL will require the Voltage Regulator to attempt to limit field current to 273 amps (as evident by alarm EK-0303) while in auto (i.e. AC Regulate Mode using the AC Adjuster). However, during the transient, the Voltage Regulator tripped (alarm EK-0310), preventing further control in auto. At this point, any control must be made in manual (i.e. DC Regulate Mode using the DC Adjuster). The unit is in a stable condition, however, based on the information provided, the unit is

operating on the verge of overexcitation. To restore reactive load and maintain a safe operating condition within the bounds of the Generator Capability Curve, the NCO-T must LOWER reactive load using the DC Adjuster. The first part of this question requires the applicant to interpret the generator capability curve and understand that reactive load must be lowered, rather than raised. The second part of the question requires the applicant to realize that, based on the alarms in, the voltage regulator has tripped and action must be taken using the DC adjuster to control reactive load.

A. Incorrect, see explanation
B. Incorrect, see explanation
C. Incorrect, see explanation
D. Correct

Technical Reference(s): SOP-8, ARP-2
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: SOP-8 Attachment 4 (Generator Capability Curve Only)

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

TITLE: GENERATOR SCHEME EK-03 (EC-11)

1	7	13	19	25	31
2	8	14	20	26	32
3	9	15	21	27	33
4	10	16	22	28	34
5	11	17	23	29	35
6	12	18	24	30	36

GENERATOR VOLTAGE REG TRIP	
Sensor:	94R8
Trip Setpoints:	Voltage Regulator tripped
Alternate Indication:	None

AUTOMATIC FUNCTION:
- None

OPERATOR ACTION:
- IF Generator has NOT tripped, THEN **CHECK** Generator Terminal Voltage normal.
- IF Generator **Terminal** Voltage is NOT normal, THEN **ADJUST** with DC Adjuster by performing the following:
  - **VERIFY** Regulator Balance Meter indicates approximately zero.
  - **PLACE** 390CS, Voltage Regulator Control Switch to OFF or TEST position.
  - **ADJUST** 370DC/CS, Voltage Regulator Manual Control Switch to control Generator Terminal Voltage between 21kV and 23kV.

FOLLOWUP ACTION:
- CORRECT cause of Voltage Regulator trip.
- **RESTORE** Voltage Regulator to normal.
- Within 30 minutes **NOTIFY** the System Reliability Controller (SRO) that the Voltage Regulator is in MANUAL. Provide an estimated time for restoring to AUTO control.

REFERENCES:
- None

TITLE: GENERATOR SCHEME EK-03 (EC-11)

1	7	13	19	25	31
2	8	14	20	26	32
3	9	15	21	27	33
4	10	16	22	28	34
5	11	17	23	29	35
6	12	18	24	30	36

VOLTAGE REGULATOR LIMITER OPERATION	
Sensor:	MXT, MXL, VHL, MEL, Voltage Regulator Minimum Excitation Limiter
Trip Setpoints:	MEL: see Generator Capability Curve VHL: 23.1KV
Alternate Indication:	None.

AUTOMATIC FUNCTION:

- Automatic regulator limits prevent operation outside allowable parameters.

OPERATOR ACTION:

NOTE:	Volts/hertz limiting condition is indicated by alarm on the volts/hertz module inside C-281, Voltage Regulator Panel. 105% on the programmable overexcitation relay (395) digital display mounted on the door of C-281 is the volts/hertz limiter alarm setpoint.

- **ADJUST** Generator Terminal Voltage using AC Adjuster to clear alarm.

FOLLOWUP ACTION:

- IF alarm cannot be cleared using AC Adjuster, THEN **REQUEST** Electrical Engineering assistance to troubleshoot Voltage Regulator.

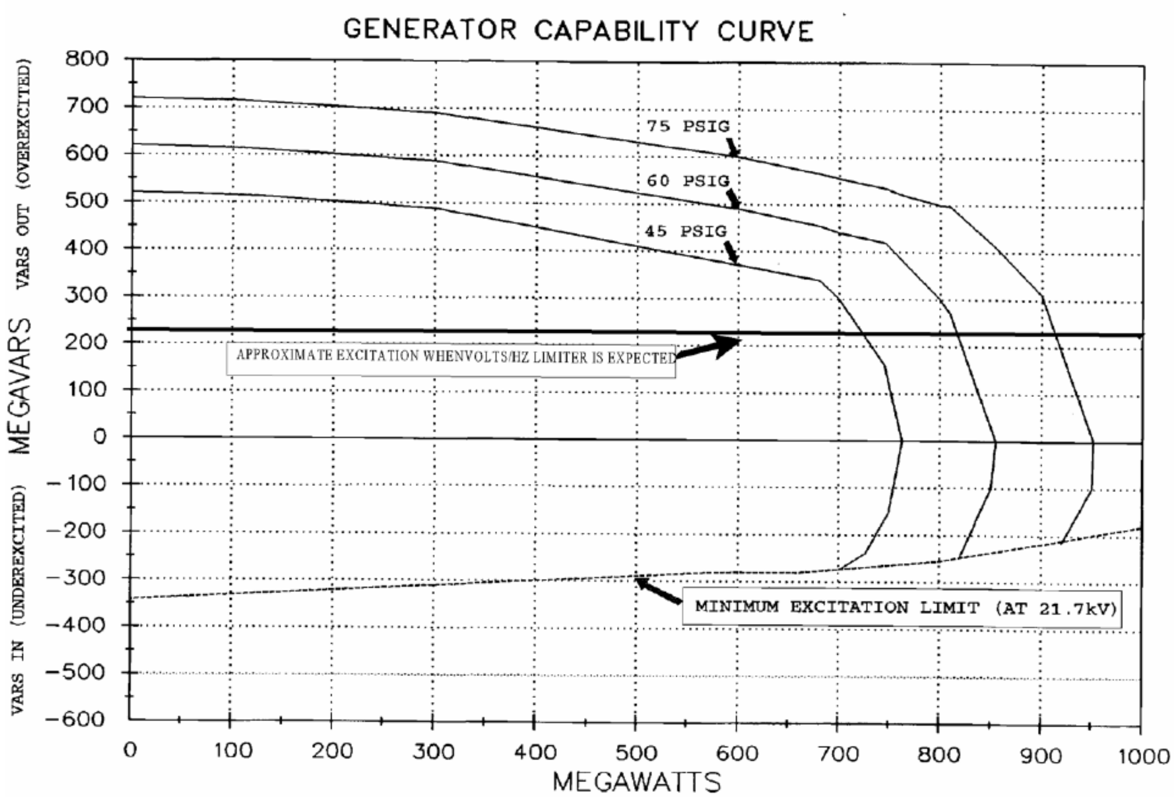
NOTE:	Automatic regulator limits, which prevent operation outside allowable parameters, will not function when controlling with the DC adjuster.

- IF the AC Adjuster has failed, THEN **PERFORM** the following to control Generator Terminal Voltage using the DC Adjuster.
  - **VERIFY** Regulator Balance Meter indicates approximately zero.
  - **PLACE** 390CS, Voltage Regulator Control Switch to OFF or TEST position.
  - **ADJUST** 370DC/CS, Voltage Regulator Manual Control Switch to control Generator Terminal Voltage between 21kV and 23kV.

REFERENCES:

- SOP-8, "Main Turbine and Generating Systems," Attachment titled "Turbine Generator Calculated Capability Curve"

TURBINE GENERATOR CALCULATED CAPABILITY CURVES



TURBINE GENERATOR CALCULATED CAPABILITY CURVES

GENERAL NOTES ON THE GENERATOR CAPABILITY CURVE

- The Minimum Excitation Limit line on the Capability Curve (page 1 of this attachment) is shown for a specific generator terminal voltage of 21.7 kV. The Limiter will limit at lower reactive power values when the terminal voltage is less than 21.7 kV and will limit at higher reactive power values when the terminal voltage is greater than 21.7 kV.

- The horizontal line at 225 MVARs OUT described as "Approximate Excitation When Volts/Hz Limiter is Expected" represents the amount of field current that would normally cause the terminal voltage to reach 23.1 kV at nominal system voltage of approximately 357 kV.

- There are three protective devices for the generator; the Maximum Excitation Limiter (MXL), the On-line Field Forcing (FF) relay, and the Over Excitation Protection (OXP) relay.

- The MXL will alarm EK-0303, "VOLTAGE REGULATOR LIMITER OPERATION," and automatically limit field current to 273 amps (114% of the rated 236 amps).

- The FF alarms only at the same 273 amps and will alarm EK-0316, "GEN FIELD FORCING/OVER EXCITATION."

- The OXP will pick up and alarm EK-0316, "GEN FIELD FORCING/OVER EXCITATION," at 283 amps assuming the MXL has failed to limit field current. If the field current persists at 283 amps for approximately 33 seconds, the OXP relay will trip the voltage regulator to manual. If that does not correct the situation, the OXP relay will trip the generator after an additional three seconds.

- Be aware that the above protective devices are arranged to function outside of the normal operating bands for the machine. During normal operation, the only device that may occasionally be challenged would be the volts/Hz limiter. Operation of the machine within the capability curve prevents the necessity of these protective devices from coming into play.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000003.AK3.08	
	Importance Rating	3.1	

K/A Statement: Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod: Criteria for inoperable control rods.

Proposed Question:

In which of the following conditions would a Control Rod be required to be declared inoperable? (Assume initial plant conditions at full power.)

A. Rod 39 indicates 7" from the other rods within its group.

B. Rod 16 drops to 126" withdrawn.

C. Rod 6 seal leak off high temperature alarm is locked in.

D. CRD matrix is lost due to a failure of Instrument AC Bus EY-01.

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, the control rod operability misalignment limit is greater than 8".
B. Correct, Rod 16 is a shutdown group B rod. Shutdown group rods are approximately 131" withdrawn at normal full power conditions. Shutdown rods must be $\geq$ 128" withdrawn to be considered operable.
C. Incorrect, high seal leakoff temperature could be indicative of exceeding the PCS leakage Tech Spec for identified leakage. However, this condition does not cause the control rod to be considered inoperable per LCO 3.1.4 or 3.1.5
D. Incorrect, with a loss of Instrument Bus EY-01, the CRD matrix and CRD LED displays on EC-02 are lost. For the primary rod position indication system to be operable, the digital position readout or the PPC display must provide valid rod position indication, or for regulating and part-length rods, the cam operated red matrix light gives positive indication of rod position. The PPC will remain unaffected as a result of this transient.

Technical Reference(s): Tech Spec 3.1.4 bases, SOP-6, DBD-2.06, ARP-5, PL-CRD "Control Rod Drive System Lesson Plan"

(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source: Bank # ___X___

Modified Bank # _______ (Note changes or attach parent)

New _______

Question History: Last NRC Exam Palisades 2005

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__

55.43 _____

Comments:

Modified one distractor, and replaced one distractor with new.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The most limiting static misalignment occurs when Bank 4 is fully inserted with one rod fully withdrawn ([Bank 4 is 99 inches out of alignment with the rated Power Dependent Insertion Limit [PDIL].) This event was bounded by the dropped full-length control rod event (Ref. 4). Since the control rod drop incidents result in the most rapid approach to SAFDLs caused by a control rod misoperation, the accident analysis analyzed a single full-length control rod drop. The above control rod misoperations may or may not result in an automatic reactor trip. In the case of the full-length rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters. The results of the control rod misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or PCS pressure occur. Control rod alignment satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).
LCO	The limits on shutdown, regulating, and part-length rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the full-length control rods will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the control rod banks maintain the correct alignment and that each full-length control rod is capable of being moved by its CRDM. The OPERABILITY requirement for the part-length rods is that they are fully withdrawn. The requirement is to maintain the control rod alignment to within 8 inches between any control rod and all other rods in its group. To help ensure this requirement is met, the control rod position deviation alarm generated by either the PIP node or the SPI system, must be OPERABLE and provide an alarm when any control rod becomes misaligned > 8 inches from any other rod in its group. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. This case bounds the safety analysis for a single rod in any intermediate position.

BASES

LCO (continued)

The primary rod position indication system is considered OPERABLE, for purposes of this specification, if the digital position readout or the PPC display provides valid rod position indication, or if the cam operated red matrix light (regulating and part-length rods only) gives positive (ON) indication of rod position. The secondary rod position indication system is considered OPERABLE, for purposes of this specification, if the magnetically operated reed switches are providing valid indication of rod position either via the plant process computer or by taking direct readings of the output from the magnetic reed switches or if the reed switch operated red matrix light (shutdown rods only) gives positive (ON) indication of rod position.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDM, any of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on control rod OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of control rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and regulating rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the PCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

LCOs 3.1.4, 3.1.5, and 3.1.6, and their ACTIONS were written to support each other. The combined intent is to assure the following:

1. There is adequate SDM available in withdrawn control rods to assure the reactor is shutdown by, and remains shutdown following, a reactor trip,

2. The control rod positioning does not cause unacceptable axial or radial flux peaking, and

3. The programmed rod withdrawal sequence and group overlap result in reactivity insertion rates within the assumptions of the Inadvertent Control Rod Bank Withdrawal Analyses.

BASES

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown rod groups are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained; and

b. The potential effects of a control rod ejection accident are limited to acceptable limits.

Control rods are considered fully withdrawn at 128 inches, since this position places them in an insignificant reactivity worth region of the integral worth curve for each bank.

On a reactor trip, all full-length control rods (shutdown and regulating), except the most reactive rod, are assumed to insert into the core. The shutdown and regulating rod groups shall be at or above their insertion limits and available to insert the required amount of negative reactivity on a reactor trip signal. The regulating rods may be partially inserted in the core as allowed by LCO 3.1.6, "Regulating Rod Group Position Limits." The shutdown rod group insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)) following a reactor trip from full power. The combination of regulating rod and shutdown rods (less the most reactive rod, which is assumed to remain fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 2). The shutdown rod group insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown rods as well as regulating rod insertion limits and inoperability or misalignment are that:

a. There be no violation of:

   1. Specified acceptable fuel design limits, or

   2. Primary Coolant System pressure boundary damage; and

b. The core remains subcritical after accident transients.

Shutdown requirements include allowances for power defect, flux redistribution, power dependent insertion limit, Group 4 rod insertion and void effects. Excess shutdown margin is defined as the shutdown margin minus the required shutdown margin. The required shutdown margin is equal to the reactivity insertion following a small steam line break accident. The value used for the required shutdown margin is 2.0% $\Delta\rho$ at both the beginning and end of core life.

3.2 SYSTEM DESIGN REQUIREMENTS

3.2.1 General System Design

The Palisades plant is equipped with 45 cruciform shaped control rods, 41 full-length rods and 4 part-length rods. Each control rod is connected to a control rod drive mechanism (CRDM). The CRDM is an electro mechanical unit which utilizes rack and pinion gearing to achieve controlled linear motion of the control rod in response to operating signals. Synchro transmitters, limit switches and reed switches provide rod position signals for rod position indication and for control rod drive interlocks and prohibits.

The control rods are assigned to one of three group types as shown in Table 3.2-1. Shutdown rods provide definite shutdown margin at all times. Part-length rods were originally provided for xenon tilt and oscillation control. The part-length rods currently perform no function, as discussed in Section 3.2.2.1. Regulating rods can provide short term reactivity control and xenon tilt and oscillation control. All control rods are positioned by one of the following methods: 1) Individual rod movement (one rod is selected and moved); 2) Individual group movement (one group is selected and moved); and 3) Sequential group movement.

TABLE 3.2-1
Control Rod Grouping

Type of Group	Group Number	Rod Numbers	Number of Rods
Shutdown Groups	A	1 - 12	12
	B	13 - 20	8
Regulating Groups	1	21 - 28	8
	2	29 - 32	4
	3	33 - 37	5
	4	38 - 41	4
Power Shaping Group (part-length)	1	42 - 45	4

TITLE: PRIMARY COOLANT PUMP STEAM GENERATOR AND ROD DRIVES SCHEME EK-09 (C-12)

37	43	49	55	61	67
38	44	50	56	62	68
39	45	51	57	63	69
40	46	52	58	64	70
41	47	53	59	65	71
42	48	54	60	66	72

ROD DRIVE SEAL LEAK OFF HI TEMP	
Sensor:	TRA-0150, CRDM Seal Leakoff Temp Alarm Recorder
Trip Setpoints:	200°F
Alternate Indication:	CRDM Seal Leakage Temperature Recorder

AUTOMATIC FUNCTION:
- None

OPERATOR ACTION:
- IF concurrent symptoms of loss of Component Cooling exist, THEN GO TO AOP-36.
- CHECK Rod Drive temperatures on TRA-0150.
- CHECK at least one CRDM Cooling Fan, V-49A or V-49B, in service.
- IF all or most CRDM temperatures are above 200°F or neither V-49A nor V-49B is available, THEN **CONSIDER** Plant shutdown to preclude CRDM damage.

FOLLOW UP ACTION:
- IF alarm appears to be valid as indicated by any Rod Drive seal leakoff temperature between 200°F to 230°F, THEN **PERFORM** a primary system leakage calculation per DWO-1.
- IF alarm is spurious as indicated by an open channel or any Rod Drive seal leakoff temperature above 250°F, indicating a failed or failing thermocouple, THEN **INITIATE** Work Request for repairs.
- **ACKNOWLEDGE** alarm at TRA-0150 to restore reflash as follows:
  - **OPEN** front door
  - **DEPRESS** FUNC key
  - **DEPRESS** ALARM ACK soft key

REFERENCES:
- AOP-36, "Loss of Component Cooling"
- DWO-1, "Operator's Daily/Weekly Item Modes 1, 2, 3, and 4"

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000024.AA1.17	
	Importance Rating	3.9	

K/A Statement: Ability to operate and/or monitor the following as they apply to Emergency Boration: Emergency borate control valve and indicators

Proposed Question:

Given the following conditions:

- The Reactor was tripped from 100% power.
- Three (3) Control Rods remain fully withdrawn.
- P-66B HPSI Pump is running.
- P-56A Boric Acid Pump is running.
- P-55A Charging Pump is running.
- EOP-1.0, "Standard Post-Trip Actions," is in progress.
- The crew is initiating Emergency Manual Boration per SOP-2A, "Chemical Volume and Control System."

Which of the following Emergency Boration flowpaths should be selected if VCT outlet valve (MO-2087) is open and will NOT close from the Main Control Board?

A. Open MO-2169 and MO-2170, Gravity Feed Valves.

B. Open MO-2160, SIRWT to Charging Pump Suction.

C. Open MO-2140, Pumped Feed Valve.

D. Open MO-3072, CVCS to HPSI Train 2.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, with the gravity feed valves MO-2169/2170 open, the VCT outlet (MO-2087) must be closed to ensure adequate boration capability is maintained. If the lineup cannot be satisfied, another lineup must be pursued.
B. Incorrect, while borating from the SIRWT is an acceptable emergency boration flowpath, it is not the correct flowpath to use with the given conditions. Per SOP-2A, the pumped feed or gravity feed lineup should be used for performing the emergency boration.
C. Correct, at least one borated flowpath must be lined up. Opening MO-2140 to allow pumped feed to the suction of the charging pumps satisfies that lineup.
D. Incorrect, incorrect procedure adherence. While SOP-2A does allow for alternate PCS injection using HPSI, it is primarily intended to accommodate isolation of the normal

charging path to the PCS when it is anticipated to be of short duration and actual makeup to the PCS will not be required.

Technical Reference(s): SOP-2A
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ______None________

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam Palisades 2003__________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __8__
55.43 _____

Comments:

Modified stem only. All distractors remain unchanged.

NOTES:

1. 'Y' or specified position indicates that valves are required to be available or in the specified position. 'N' indicates that valves are NOT required to be available or in the specified position.

2. Heat trace for Paths 1 and 2 are available as long as temperature is maintained 25°F above the respective boric acid solubility temperature on the flow path.

3. Use of Path 1 renders Path 2 unavailable due to charging pump suction header pressurization (possibly as high as 110 psig) heavily seating the gravity flow check valves. Availability is restored by Attachment 6, "Concentrated Boric Acid Flow Path Test."

Allowable Boric Acid Flow Paths

Path	T-53A	T-53B	P-56A	P-56B	MO-2169	MO-2170	MO-2140	MO-2087	MO-2160
1	> 94.4%	N	Y	N	N	N	Y or Open	N	N
	N	> 94.4%	N	Y	N	N	Y or Open	N	N
	Both < 94.4% But sum > 94.4%		Y	Y	N	N	Y or Open	N	N
2	> 94.4%	N	N	N	Y	N	N	Y or Closed	N
	N	> 94.4%	N	N	N	Y	N	Y or Closed	N
	Both < 94.4% But sum > 94.4%		N	N	Y	Y	N	Y or Closed	N
3	N	N	N	N	N	N	N	Y or Closed	Y

EMERGENCY MANUAL BORATION

1.0 **ENSURE** charging flow greater than 33 gpm.

2.0 **ESTABLISH** at least one (both preferred) boric acid flow path(s) as follows:

a. IF Bus 1D energized, THEN **ESTABLISH** pumped feed:

1. **START** at least one (both preferred) Boric Acid Pump(s).

   - P-56A, Boric Acid Pump
   - P-56B, Boric Acid Pump

2. **OPEN** MO-2140, Boric Acid Pump P-56A/B Feed Isolation

3. **VERIFY** Charging Flow greater than 33 gpm.

b. IF Bus 1C energized, THEN **ESTABLISH** Gravity Feed:

1. **OPEN** Boric Acid Tank Gravity Feed Isol Valves.

   - MO-2169, BAST T-53A Gravity Feed Isolation
   - MO-2170, BAST T-53B Gravity Feed Isolation

2. **CLOSE** CV-2155, Make-Up Stop.

CAUTION
If CK-CV-2171, Boric Acid Gravity Feed Check, sticks closed during the next step, Charging Pumps may trip on low suction pressure.

3. **CLOSE** MO-2087, VCT T-54 Outlet Valve.

4. **ENSURE CLOSED** MO-2160, SIRWT T-58 Outlet To Charging Pp P-55A,B,C.

EMERGENCY MANUAL BORATION

5. IF EY01 is not energized, THEN **PERFORM** the following:

    a. **ENSURE OPEN** breaker 52-207, Refueling Wtr Chrg PP Valve 2160. Refer to SOP-30, "Station Power."

    b. **MANUALLY CLOSE** MO-2160, SIRWT T-58 Outlet To Charging Pp P-55A,B,C.

6. **VERIFY** charging flow greater than 33 gpm as indicated by FIA-0212, Charging Line Flow Indicator Alarm.

3.0 WHEN desired to secure from Emergency Boration, THEN **REFER TO** EOP Supplement 40, "Charging Pump Suction Alignment."

7.3.10 To Establish Alternate Charging Path to the PCS Via HPSI

NOTE: This section is primarily intended to accommodate isolation of the normal charging path to the PCS when it is anticipated to be of short duration and actual makeup to the PCS will not be required. Use of this path will result in the following:

- Thermal transient and boron concentration dilution for in use HPSI Train 2 line.
- Higher Concentration Boron addition to PCS during initial use of HPSI Train 2 line(s).
- Injection of relatively cool water into safety injection lines. Rapid injection of cool water may cause a water hammer if a steam vapor void exists in the lines.

a. **MAINTAIN** the requirements of Sections 4.3 and 4.4.

b. IF necessary to compensate for the expected duration that charging flow will be secured, THEN take manual pressurizer level control AND **RAISE** pressurizer level to less than or equal to 60% on LIC-0101A or LIC-0101B. Refer to SOP-1A, "Primary Coolant System," Section 7.2.1, "Pressurizer Level Control."

c. **STOP** Charging and Letdown flow. Refer to Section 7.3.1.

CAUTION

Pressurizer level must remain greater than or equal to 40% on LIC-0101A or LIC-0101B to ensure that assumptions made in accident analyses remain valid.

d. WHEN it is desired to commence charging to the PCS, THEN **PERFORM** the following:

1. **OPEN** one HPSI Train 2 Injection Valve:

VALVE	LOOP	DESCRIPTION
MO-3062	2B	Redundant HPSI to Reactor Coolant Loop 2B
MO-3064	2A	Redundant HPSI to Reactor Coolant Loop 2A
MO-3066	1B	Redundant HPSI to Reactor Coolant Loop 1B
MO-3068	1A	Redundant HPSI to Reactor Coolant Loop 1A

2. **ENSURE CLOSED** CV-3018, P-66B Disch to HPSI Train 2.

3. **OPEN** MO-3072, HPSI Train2/Charging X-Conn (MZ-22).

4. **ENSURE IN MANUAL** AND **CLOSE** SIT Pressure Indicating Controller associated with valve opened in Step 7.3.10d.1:

VALVE	PIC	DESCRIPTION
MO-3062	PIC-0338	Safety Inject Tank T-82D Pressure Control
MO-3064	PIC-0347	Safety Inject Tank T-82C Pressure Control
MO-3066	PIC-0346	Safety Inject Tank T-82B Pressure Control
MO-3068	PIC-0342	Safety Inject Tank T-82A Pressure Control

5. **CLOSE** CV-2111, Charging Line Containment Isolation (MZ-45) Valve.

6. **OPERATE** a Charging Pump with suction from VCT.

(a) **MONITOR** Pressurizer level and pressure closely.

(b) **MONITOR** associated HPSI Flow Indicator for flow:

VALVE	FI	DESCRIPTION
MO-3062	FI-0313A	High Pressure Safety Inject Flow - Loop 2B
MO-3064	FI-0312A	High Pressure Safety Inject Flow To Loop 2A
MO-3066	FI-0310A	High Pressure Safety Inject Flow To Loop 1B
MO-3068	FI-0308A	High Pressure Safety Inject Flow To Loop 1A

(c) WHEN desired Pressurizer level is reached (less than or equal to 60% on LIC-0101A or LIC-0101B), THEN **STOP** Charging Pump.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000028.G2.4.31	
	Importance Rating	4.2	

K/A Statement: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power.
- Pressurizer level control is selected to Channel B.
- Pressurizer Heater Control is selected to Channel A & B.

A transient occurred which resulted in:

- Pressurizer level is 51% and lowering.
- Annunciator EK-0761 “PRESSURIZER LEVEL HI-LO” alarming.

Which of the following actions would NOT automatically occur as a result of this transient?

A. P-55A Charging Pump operates at maximum speed.

B. All Pressurizer Heaters de-energize.

C. P-55B and P-55C Charging Pumps start.

D. CV-2004 and CV-2005, Letdown Orifice Stop Valves, close.

Proposed Answer: B

Explanation (Optional):

A. Incorrect, P-55A will increase speed to maximize charging flow.
B. Correct, pressurizer heaters only de-energize on a low-low level (36%).
C. Incorrect, both P-55B and P-55C and constant 40gpm pumps when operating and are cycled on/off as necessary to maintain level. Both pumps will start on a low pressurizer level.
D. Incorrect, letdown orifice isolation valves will close in an attempt to maintain maximum charging flow (i.e. maximize the input, minimize the output).

Technical Reference(s): ARP-4, PL-PLCS Pressurizer Level Control System Lesson Plan

(Attach if not previously provided, including version/revision number) ____________________

Proposed references to be provided to applicants during examination: ___None___

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam __________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ___7___
55.43 ______

Comments:

TITLE: PRIMARY SYSTEM VOLUME LEVEL PRESSURE SCHEME EK-07 (C-12)

37	43	49	55	61	67
38	44	50	56	62	68
39	45	51	57	63	69
40	46	52	58	64	70
41	47	53	59	65	71
42	48	54	60	66	72

PRESSURIZER LEVEL HI-LO	
Sensor:	LIC-0101A, Pressurizer Level Control Channel 1, or LIC-0101B, Pressurizer Level Control Channel 2, whichever is selected
Trip Setpoints:	5.75% Deviation from Setpoint
Alternate Indication:	None

AUTOMATIC FUNCTION:

- High Level Alarm:
  - o CV-2004 and CV-2005, Letdown Orifice Stop Valves, open.
  - o P-55B, 'B' Charging Pump, and P-55C, 'C' Charging Pump, stop.
  - o P-55A, 'A' Charging Pump, operates at minimum speed (33 gpm).
  - o Pressurizer Backup Heaters energize (in AUTO position only).
- Low Level Alarm:
  - o CV-2004 and CV-2005 close.
  - o P-55B and P-55C start.
  - o P-55A operates at maximum speed (53 gpm).

OPERATOR ACTION:

- **CHECK** Charging and Letdown response correct.
- IF Charging and Letdown response NOT correct, THEN **REFER TO** AOP-22 for applicable actions .
- IF selected T_{AVE} input is inaccurate, THEN:
  - o **REFER TO** AOP-27 for Tave/Tref Controller failure actions.

FOLLOW UP ACTION:

- **IMPLEMENT** any applicable Technical Specifications LCO 3.4.9 actions.

REFERENCES:

- Technical Specifications LCO 3.4.9
- AOP-22, "Pressurizer Level Control Malfunctions"
- AOP-27, "T_{AVE}/T_{REF} Controller Failure"

A. Alarms (ARP-4)

1. EK-0761, "PRESSURIZER LEVEL HI-LO"
    a. Sensor:
        1) LIC-0101A or LIC-0101B, whichever is selected by HS 1/LRC-0101
    b. Setpoint:
        1) 5.75% level deviation from setpoint.
    c. Alarm impact on PZR Level Control:
        1) AUTO: IF high alarm, THEN:
            a) CV-2004 and CV-2005 Orifice Stop Valves open.
            b) Charging Pumps P-55B and P-55C stop.
            c) Charging Pump P-55A operates at minimum speed (33 gpm).
            d) Pressurizer Backup Heaters energize (in "AUTO" position only).
        2) AUTO: IF low alarm, THEN:
            a) CV-2004 and CV-2005 Orifice Stop Valves close.
            b) Charging Pumps P-55B and P-55C start.
            c) Charging Pump P-55A operates at maximum speed (53 gpm).
        3) Indications to validate alarm:
            a) LIC-0101A or LIC-0101B which ever is selected by HS 1/LRC-0101.
            b) Check Charging and Letdown response correct; IF not, THEN shift selected level controllers LIC 0101A or LIC 0101B.
2. EK-0763, "PRESSURIZER LEVEL CH 'A' LO-LO"
    a. Sensor:
        1) LIC-0101AL (Hot Cal)
    b. Setpoint:
        1) 36%
    c. Alarm impact on PZR Level Control:
        1) AUTO: IF "A" or "A & B" PZR heater control is selected, THEN all PZR heaters are tripped. Letdown Orifice Stop Valve CV-2003 closes.
    d. Indications to validate alarm:
        1) All available Hot Cal level instruments in the control room
3. EK-0764, "PRESSURIZER LEVEL CH 'B' LO-LO"
    a. Sensor:

1) LIC-0101BL (Hot Cal)

b. Setpoint:

1) 36%

c. Alarm impact on PZR Level Control:

1) AUTO: IF "B" or "A" & "B" PZR heater control is selected, THEN all PZR heaters are tripped. Letdown Orifice Stop Valve CV-2003 closes.

d. Indications to validate alarm:

1) All available Hot Cal level instruments in the control room

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000032.AK3.02	
	Importance Rating	3.7	

K/A Statement: Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Guidance contained in EOP for loss of source-range nuclear instrumentation.

Proposed Question:

Given the following conditions:

- The Plant was at 89% power when a transient occurred which required a manual reactor trip.
- Three (3) control rods did NOT insert into the core.
- The crew has just entered EOP-01, "Standard Post-Trip Actions."

If both Source Range Nuclear Instruments have become inoperable, what is the effect, if any, on the Reactor Operator's ability to check the status of the Reactivity Control safety function?

A. No effect, since Reactivity Control is satisfied due to Xenon building in for the next approximately 10-12 hours.

B. Reactivity Control must be satisfied by manually driving down ONE of the stuck control rods.

C. Will need to check Reactor power at less than 1E-4 % and constant or lowering on Wide Range Excore indication.

D. Will need to check Reactor power at less than 2% using delta T power indication.

Proposed Answer: C

Explanation (Optional):

A. Incorrect, although Xenon will behave this way, this is not an approved methodology for satisfying the Reactivity Control safety function.
B. Incorrect, as, per the stem conditions, this would still result in having more than one control rod stuck fully withdrawn.
C. Correct, per EOP-1.0 Event Diagnostic Flowchart Reactivity Control Requirements.
D. Incorrect, again, this is not an approved methodology for satisfying the Reactivity Control safety function.

Technical Reference(s): EOP-1.0
(Attach if not previously provided,

including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ___None___________

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam Palisades 2003

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _10__
55.43 _____

Comments:

Modified question from Palisades 2003 NRC Exam. Modified stem such that Source Range instrumentation is lost rather than Wide Range and changed correct answer to new answer based upon second conditional statement for verification of safety function. NOTE: 3 Stuck control Rods does not necessarily drive the CRS to transition to EOP 9.0. In EOP 9.0, RC-2 would be the applicable Success Path.

SAFETY FUNCTION: Reactivity Control
SUCCESS PATH: Control Rod Insertion: RC-1
RESOURCE TREE: Tree A

INSTRUCTIONS	CONTINGENCY ACTIONS
© 3. **VERIFY** RC-1 (Control Rod insertion) is satisfied by ANY of the following conditions: Condition 1: a. Maximum of one full-length control rod NOT fully inserted b. Rx power lowering and a negative Startup Rate Condition 2: a. Rx power is less than 10^{-6}% and is constant or lowering on Wide Range Excore indication OR Rx power is less than 100 cps and is constant or lowering on Source Range excore indication	3.1. IF the Reactivity Control safety function is still in jeopardy, THEN **GO TO** the next appropriate Reactivity Control Success path.

© = Continuously applicable step

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000051.AA2.02	
	Importance Rating	3.9	

K/A Statement: Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip.

Proposed Question:

Given the following conditions:

- A degrading condenser vacuum condition has caused the Crew to enter AOP-6, “Loss of Condenser Vacuum.”
- Condenser vacuum has degraded to 21.7”Hg and is slowly lowering.
- Main Generator output is approximately 465 MWe.

Based on these conditions, the crew should…

A. Trip the reactor and enter EOP-1.0, “Standard Post-Trip Actions.”

B. Trip the turbine and continue efforts to correct the problem in AOP-6, “Loss of Condenser Vacuum.”

C. Commence a Power reduction, per GOP-8, “POWER REDUCTION AND PLANT SHUTDOWN TO MODE 2 OR MODE 3 > 525 °F,” to stabilize condenser vacuum.

D. Commence a Rapid Power Reduction, per AOP-7, “RAPID POWER REDUCTION,” to stabilize condenser vacuum.

Proposed Answer: A

Explanation (Optional):

A. Correct, per AOP-6. The reactor should be tripped if condenser vacuum lowers to less than 22” Hg.
B. Incorrect, while the turbine does need to be tripped under these conditions, the reactor must also be tripped since reactor power is greater than 15%. If reactor power was less than 15%, this would be true.
C. Incorrect, under these conditions, a reactor trip is required. If condenser vacuum were less than 24”Hg, this would be a suitable action.
D. Incorrect, under these conditions, a reactor trip is required. If condenser vacuum were less than 24”Hg, this would be a suitable action.

Technical Reference(s): AOP-6
(Attach if not previously provided,

including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: _____None_________

Learning Objective: __________________________ (As available)

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

Question History: Last NRC Exam _____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:
Pal 2010 question 25 was used for reference; question as written is new.

REACTOR AND EQUIPMENT TRIP CRITERIA

Reactor Trip

- **Reactor Power greater than or equal to 15%**
    - Main Condenser vacuum lowers to less than 22" Hg
    - Loss of both cooling tower pumps

Turbine Trip

- **Reactor Power less than 15%**
    - Main Condenser vacuum lowers to less than 22" Hg
    - Loss of both cooling tower pumps

6.0 OPERATOR ACTIONS

ACTIONS\EXPECTED RESPONSE	RESPONSE NOT OBTAINED
© 1. **VERIFY** the following: • At least one cooling tower pump is in service • Main Condenser vacuum is greater than 22" Hg	1.1 IF Reactor Power is greater than or equal to 15%, THEN **TRIP** the Reactor. a. GO TO EOP-1, "Standard Post-Trip Actions." 1.2 IF Reactor Power is less than 15%, THEN **TRIP** the Turbine. a. GO TO Step 3. **NOTE**: Power reduction at lower rates allows the use of boron and may prevent entry into Technical Specifications LCO 3.1.6 (PDIL).
© 2. **VERIFY** Main Condenser vacuum is greater than 24" Hg AND NOT lowering.	2.1 **PERFORM** one of the following as necessary to stabilize and/or restore Main Condenser vacuum greater than 24 inches Hg. a. **COMMENCE** a rapid Reactor Power reduction in accordance with AOP-7, "Rapid Power Reduction." b. **COMMENCE** Reactor Power reduction in accordance with GOP-8, "Power Reduction and Plant Shutdown To MODE 2 or MODE 3 ≥ 525°F." **NOTE**: With a loss of a cooling tower pump, Main Condenser vacuum will remain stable until water drains from the cooling tower then expect Main Condenser vacuum to lower. 2.2 WHEN Main Condenser vacuum has stabilized, THEN **STABILIZE** Reactor Power at a level to maintain Main Condenser vacuum greater than 24" Hg.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000067.AK1.01	
	Importance Rating	2.9	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site: Fire classifications, by type.

Proposed Question:

Nuclear Plant Operators have just completed racking in breaker 52-1111, Main Exhaust Fan V-6B. Upon remote closure of the breaker, an arc flash occurred and a sustainable fire followed.

What is the class of fire and what type of extinguisher shall be utilized, per fire protection procedures?

A. Class B fire; use water OR Yellow Foray dry chemical fire extinguisher.

B. Class B fire; use any type of dry chemical OR CO_2 fire extinguisher.

C. Class C fire; use ONLY a Purple K type dry chemical fire extinguisher.

D. Class C fire; use any type of dry chemical OR CO_2 fire extinguisher.

Proposed Answer: **D**

Explanation (Optional):

Class A fires are "ordinary combustible" fires. Water extinguishers are used on Class A fires.

Class B fires are "flammable and combustible liquid and gas." ABC dry chemical extinguishers are used on Class A, Class B, or Class C fires and are filled with Monoammonium Phosphate, a yellow-colored dry chemical (yellow foray). BC dry chemical extinguishers are used on Class B or Class C fires and are filled with Purple K Potassium Bicarbonate, a purple-colored dry chemical.

Class C fires are "electrical" fires. Carbon Dioxide (CO2) extinguishers are used on Class B or Class C fires. This is an electrical fire that should be extinguished using CO2 and/or dry chemical extinguishers, per FPIP-4 "Fire Protection Systems and Fire Protection Equipment."

A. Incorrect, applicant does not understand the class of fire or the type of extinguisher to use. See explanation.
B. Incorrect, applicant understands the type of extinguisher to use, but not the class of fire. See explanation.
C. Incorrect, applicant understand the class of fire, but not type of extinguisher to use. See

explanation.

D. Correct, any type of dry chemical (ABC or BC) or CO2 extinguisher is adequate. See explanation.

Technical Reference(s): FPIP-4
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: OB 44825(1101) (As available)

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

Question History: Last NRC Exam ____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments:

Question modified from Palisades 2003 Audit Exam. Modified stem to change question from Class A fire to a Class C fire. Two distractors were changed, including changing the answer.

7.2 WATER FIRE EXTINGUISHERS

7.2.1 Water fire extinguishers are used on Class "A" (Ordinary Combustible) fires.

7.2.2 The individual using the fire extinguisher is responsible for:

a. Reporting its use to the Shift Manager.

b. Transporting used fire extinguisher(s) to Fire Depot in the Feedwater Purity Building.

7.2.3 Water fire extinguishers are maintained and charged onsite by Operations Department personnel,

OR

Tagged and delivered to the Fire Depot in the Feedwater Purity Building if offsite maintenance is required.

7.3 DRY CHEMICAL FIRE EXTINGUISHERS

7.3.1 The two types of dry chemical fire extinguishers used onsite are as follows:

a. ABC Dry Chemical Fire Extinguishers are:

1. Used on Class A (Ordinary Combustible) fires.
2. Used on Class B (Flammable and Combustible Liquid and Gas) fires.
3. Used on Class C (Electrical) fires.
4. Filled with Monoammonium Phosphate, a yellow-colored dry chemical.

b. BC Dry Chemical Fire Extinguishers are:

1. Used on Class B (Flammable and Combustible Liquid and Gas) fires.
2. Used on Class C (Electrical) fires.
3. Filled with Purple K Potassium Bicarbonate, a purple-colored dry chemical.

7.3.2 The individual using the fire extinguisher is responsible for:

a. Reporting its use to the Shift Manager.

b. Transporting used fire extinguisher(s) to the Fire Depot in the Feedwater Purity Building.

7.3.3 Dry chemical fire extinguishers are maintained and filled onsite by Operations Department personnel,

OR

Tagged and delivered to the Fire Depot in the Feedwater Purity Building if offsite maintenance is required.

WARNING

Mixing dry chemicals when charging fire extinguishers may create a chemical reaction within the fire extinguisher which could cause injury to personnel.

7.3.4 **NEVER MIX DRY CHEMICALS.** If other than dry chemicals designated for their respective units are used:

a. Improper dry chemicals shall void the underwriters listing on the fire extinguisher; and

b. May create a chemical reaction within the fire extinguisher which could result in injury to the operator.

7.4 CARBON DIOXIDE (CO_2) FIRE EXTINGUISHERS

7.4.1 Carbon Dioxide (CO_2) fire extinguishers are used on:

a. Class "B" (Flammable and Combustible Liquid and Gas) fires.

b. Class "C" (Electrical) fires.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	069.AK2.03	
	Importance Rating	2.8	

K/A statement: Knowledge of the interrelations between the Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch

Proposed Question:

Given the following conditions:

- The Plant is in Mode 5 on Shutdown Cooling (SDC) preparing to heat up the Primary Coolant System (PCS).
- Both Personnel airlock doors are open.
- A loss of SDC occurs and PCS temperature rises to 214°F.

Which one of the following statements describes the current status of containment integrity?

Containment integrity:

A. Is not required in the current Mode.

B. Is not required until all OPERABLE PCS T_{cold} instruments read greater than 220 °F.

C. Has been lost. At least one OPERABLE airlock door must be closed within 1 hour.

D. Has been lost. Both OPERABLE airlock doors must be maintained closed at all times.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, containment integrity is required for current plant conditions. LCO 3.6.2 is applicable in Modes 1-4. The applicant must understand that due to PCS temperature rising > 200°F, the plant is now in Mode 4, where LCO 3.6.2 is applicable.
B. Incorrect, containment integrity must be restored within 1 hour. This is incorrect as the applicant chose to maintain the doors open and place the plant into a Mode where LCO 3.6.2 does not apply (Mode 5).
C. Correct, at least one airlock door must be closed within 1 hour IAW LCO 3.6.2.
D. Incorrect, only one airlock door is required to be closed IAW LCO 3.6.2.B

Technical Reference(s): LCO 3.6.2.A, AOP-32
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: ___None__________

Learning Objective: _________________________ (As available)

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

Question History: Last NRC Exam _______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _10___
55.43 _____

Comments:

Based upon the initial conditions given in the stem, both airlock doors are still OPERABLE, but the interlock mechanism is not, since both door are open simultaneously. This means that TS 3.6.2.B must be entered and required actions taken upon entering Mode 4.

STEP 8

Step Text

8. **VERIFY** the interlock mechanism is OPERABLE in each Containment Air Lock.

8.1 **ENSURE** an OPERABLE door is closed on the affected air lock within 1 hour.

NOTE: The following actions are to ensure only one door on the affected air lock is opened at a time.

8.2 **CAUTION TAG** the outer door handwheel of the affected air lock stating:

a. Condition of the interlock.

b. Shift Manager approval required prior to use.

© 8.3 Shift Manager shall **CONDUCT** a pre-entry briefing to anyone needing to use the affected air lock.

8.4 **UPDATE** LCO status board with the above information.

8.5 **LOCK CLOSED** an operable door on the affected air lock within 24 hours.

8.6 **REFER TO** Technical Specifications LCO 3.6.2.B.

Technical Basis

Intent of this step is to ensure LCO 3.6.2.B.1 and 3.6.2.B.2 Required Actions are taken when appropriate. Steps 8.2.a and 8.2.b address Operations expectations for controlling access into an air lock that has an inoperable interlock.

Associated Notes, Cautions, and Warnings:

A note precedes Step 8.2 to inform the operators that they are now initiating actions that will physically control door opening and take the place of the broken interlock mechanism.

Training Emphasis:

NONE

3.6 CONTAINMENT SYSTEMS

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 through a "locked" air lock door 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 leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	----------------NOTES--------------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. ---	
	A.1 Verify the OPERABLE door is closed in the affected air lock. <u>AND</u>	1 hour
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed in the affected air lock. AND -----NOTE----- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- A.3 Verify the OPERABLE door is locked closed in the affected air lock.	24 hours Once per 31 days
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	-----NOTES----- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. ---------- B.1 Verify an OPERABLE door is closed in the affected air lock. AND	 1 hour

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND		
	---NOTE--- Air lock doors in high radiation areas may be verified locked closed by administrative means.		
	B.3	Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

Examination Outline Cross-Reference:

	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	CE/A11.AK1.2	
	Importance Rating	3.0	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to the (RCS Overcooling): Normal, abnormal and emergency operating procedures associated with (RCS Overcooling).

Proposed Question:

Given the following:

- A Main Steam Line Break has occurred upstream of the 'B' Steam Generator MSIV.
- Main Steam Line Isolation has automatically actuated.

Which one of the following is of concern if a steaming path from the unaffected Steam Generator is not established immediately following dryout of the affected Steam Generator?

A. Void formation in the Reactor Vessel upper head region.

B. Rapid rise in Average QCET temperatures causing a loss of natural circulation.

C. Rapid repressurization of the PCS and subsequent pressurized thermal shock.

D. Rapid rise in T_{cold} of the unaffected loop which would result in a loss of natural circulation.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, void formation is an undesirable condition, but the concern does not apply here.
B. Incorrect, loss of natural circulation is a concern, but does not apply for given conditions.
C. Correct, prevents an uncontrolled heatup and repressurization due to loss of decay heat removal.
D. Incorrect, loss of natural circulation is a concern, but does not apply for given conditions.

Technical Reference(s): EOP-6.0 Basis
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________________________ (As available)

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

Question History: Last NRC Exam Palisades 2003____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:

Rearranged distractors ‘C’ & ‘D’ based upon length of distractors. ‘C’ is now correct answer.

INSTRUCTIONS | CONTINGENCY ACTIONS

CAUTION

When ALL PCPs are stopped, steaming the least affected S/G must occur prior to dryout of the most affected S/G to prevent lifting PZR Code Safety Valves or Pressurized Thermal Shock rupture of the PCS.

© 16. **STABILIZE** PCS temperature as follows:

a. **MAINTAIN** level in the least affected S/G between 60% and 70%.

b. IF the steam leak is isolated, THEN **ESTABLISH** steam flow from BOTH S/Gs using the Atmospheric Steam Dump Valves.

c. IF the steam leak is NOT isolated, THEN **STEAM** the least affected S/G as necessary to maintain the following, as applicable:

- WHEN T_cs in the affected loop are lowering, THEN **MAINTAIN** the least affected S/G pressure within 50 psid above the most affected S/G pressure

- WHEN T_cs in the affected loop are NOT lowering, THEN **STABILIZE** PCS T_cs using the least affected S/G

CEN-152 ESDE Step 13:

*13. Stabilize RCS temperature by controlled steaming of the least affected steam generator using the atmospheric dump valves

Technical Basis:

The intent of this step is to direct the operator to control and stabilize PCS temperature following dry out of the most affected S/G during an ESDE.

PCS temperature will rise after the most affected S/G dries out unless a means of controlling PCS heat removal is established. The rise in PCS temperature may result in a water-solid condition due to inventory added from Safety Injection and Charging operation during the blowdown phase of the event. The post dryout heatup and repressurization also presents a PTS concern.

In order to mitigate PCS heatup, a controllable heat removal method must be established prior to dryout of the most affected S/G. The intent is to regain PCS temperature control and stabilize T_cs, thus preventing an uncontrolled PCS heatup and repressurization. Cooling down the least affected S/G by controlled steaming will aid in minimizing the amount of heatup that would occur following a loss of heat removal on the most affected S/G. By maintaining the least affected S/G pressure within 0 to 50 psig of the most affected S/G will keep the least affected S/G nearly coupled with the remainder of the PCS and yet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	CE/E09.EA2.1	
	Importance Rating	3.2	

K/A statement: Ability to determine and interpret the following as they apply to the (Functional Recovery): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question:

The CRS has implemented EOP-9.0, “Functional Recovery,” due to a Steam Generator (S/G) Tube Rupture on the ‘A’ S/G and a stuck open Main Steam Safety Valve (MSSV) on the ‘B’ S/G.

Which of the following describes the required operator actions?

A. Isolate the ‘A’ S/G. Use the ‘B’ S/G for heat removal.

B. Isolate the ‘B’ S/G. Use the ‘A’ S/G for heat removal.

C. Isolate both S/G’s. Use Once Through Cooling for heat removal.

D. Isolate the ‘B’ S/G MSSV by gagging it closed, then use the ‘B’ S/G for heat removal.

Proposed Answer: **B**

Explanation (Optional):

In a dual event scenario, the applicant must determine which S/G is considered the most affected S/G and isolate that S/G. With a dual event scenario, the applicant must assess the current plant conditions and make that choice. With a stuck open MSSV, the S/G will inevitably blow dry and there is little the operator can do to mitigate that. Isolating that S/G will not isolate the steam path past the MSSV. However, once the S/G is blown dry, the operator must use the ruptured S/G to cooldown in order to maintain heat sink availability. In this case, the operator can control the amount of radioactivity being released via the ruptured S/G and minimizing off site dose should be the primary concern in this case.

A. Incorrect, the ‘B’ S/G is considered the most affected S/G and the ruptured S/G (‘A’) must be used for cooldown.
B. Correct, the ‘B’ S/G is considered the most affected S/G and the ruptured S/G (‘A’) must be used for cooldown.
C. Incorrect, isolating both S/G’s and utilizing Once Through Cooling for heat removal is a last resort method, even during a dual event scenario.
D. Incorrect, incorrect procedure compliance.

Technical Reference(s): EOP-9 HR-2 and Bases
(Attach if not previously provided,

including version/revision number) ____

Proposed references to be provided to applicants during examination: ___None___________

Learning Objective: __________________________ (As available)

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

Question History: Last NRC Exam _St Lucie 2000__________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 __10_
55.43 ______

Comments:

Modified question stem and changed distractor D. Used different S/Gs to change answer.

INSTRUCTIONS

CONTINGENCY ACTIONS

CAUTION

Each D/G is limited to a 2500 KW continuous load rating and a 2750 KW two-hour load rating. Operation of VC-10 (VC-11) will draw approximately 44 KW.

12. **ENSURE** at least one train of CR HVAC in Emergency Mode. Refer to SOP-24, "Ventilation and Air Conditioning System."

© 13. **DETERMINE** the most affected S/G by considering ALL of the following:

- High steam flow from S/G
- Lowering S/G pressure
- Lowering S/G level
- Lowering Loop T_c temperature

INSTRUCTIONS

CONTINGENCY ACTIONS

NOTE: Maintenance of heat removal via the least affected S/G during dual events (SGTR/SGTR, ESD/ESD, or SGTR/ESD combinations) is preferable to isolation of both S/Gs and going to once-through-cooling.

14. IF MSIS has NOT isolated the leak, THEN **ISOLATE** the most affected steam generator. Refer to the following applicable EOP supplement:

    - EOP Supplement 17 ('A' S/G)
    - EOP Supplement 18 ('B' S/G)

INSTRUCTIONS

15. **VERIFY** the correct S/G is isolated by comparing ALL of the following:
    - S/G pressures
    - S/G levels
    - PCS Loop T_c temperatures

CONTINGENCY ACTIONS

15.1 IF the wrong S/G was isolated, THEN **PERFORM** ALL of the following on the least affected S/G:

a. **OPEN** the Atmospheric Steam Dump Valve air supply valves and manual isolation valves.
Refer to the following applicable EOP Supplement:
   - EOP Supplement 17 ('A' S/G)
   - EOP Supplement 18 ('B' S/G)

b. **ESTABLISH** Auxiliary Feedwater flow through ANY associated AFW valve:

'A' S/G	'B' S/G
CV-0737A CV-0749	CV-0736A CV-0727

15.2 **GO TO** Step 14 to isolate the affected S/G.

SAFETY FUNCTION: **PCS And Core Heat Removal**
SUCCESS PATH: **Via S/G with SIS in operation; HR-2**
RESOURCE TREE: **Tree E**

INSTRUCTIONS	CONTINGENCY ACTIONS
c. Close the MSIV bypass valve.	
d. Close the associated main feedwater isolation valve.	
e. Isolate blowdown for the affected SG.	
f. Close MS line drains located upstream of the associated MSIVs.	
g. Verify SG safety valves are closed.	
h. [Close the auxiliary feedwater isolation valve(s) including the steam driven pump steam supply valve associated with the steam generator being isolated].	

Technical Basis:

The intent of this step is to isolate the most affected S/G from the environment and the rest of the secondary system.

Prior to the most affected S/G isolation, PCS temperature (T_H) must be lowered to less than the **{[MSSV lift prevent temperature] minus 15°F}** which gives the value of 524°F. when corrected for instrument inaccuracies. This action is necessary to prevent inadvertently opening a MSSV after the S/G is isolated. S/G isolation is intended to re-establish the Containment isolation safety function breached by the SGTR.

The most affected S/G is isolated using EOP Supplement 12 or 13. These supplements provide the direction for containing the contaminated secondary fluid within the affected S/G. S/G isolation does not address the control of secondary system contamination which occurred prior to isolation.

TITLE: FUNCTIONAL RECOVERY PROCEDURE BASIS

SAFETY FUNCTION: PCS and Core Heat Removal
SUCCESS PATH: Via S/G with SIS in operation; HR-2
RESOURCE TREE: Tree E

INSTRUCTIONS CONTINGENCY ACTIONS

Training Emphasis:

There could be SGTR/ESDE combinations. Therefore, the operator should refer to the applicable steps of this success path to aid in identifying that a dual event is in progress.

If there is a conflict between isolating a steam generator and maintaining adequate heat removal via at least one S/G, then maintaining PCS heat removal via the least affected steam generator has precedence. At lease one S/G should always be available for heat removal, if at all possible. Isolation of both S/Gs and going to once-through-cooling is a last resort option.

Dual faulted S/G events (SGTR in one S/G, ESDE in the other) are beyond the design basis. Therefore, prescriptive guidance concerning which S/G is the least affected is not possible to provide. Success is defined as establishing the heat removal safety function using a S/G in lieu of Once Through Cooling. The following guidance is provided as an aid to operators, training instructors, and ERO staff in response to these events.

Isolating the SGTR S/G in order to keep dose to the public ALARA is preferable. This is especially important when fuel failures (as indicated by any relevant process or area monitor) exist concurrently in this event. The following strategies are available to the operating crew.

Isolating the SGTR S/G and Steaming the ESDE S/G

This strategy would be preferable if the ESDE is not inside containment or inside the CCW Room (due to CCW room EQ concerns.) This strategy is also preferable when fuel failures exist. However, if the ESDE inside containment is of a small enough magnitude such that the Containment Atmospere and Core/PCS Heat Removal safety functions can be maintained, then steaming to the containment is still preferrable to either steaming the SGTR S/G or OTC.

Exceeding prescribed cooldown rate limits can be tolerated. The Palisades PRA model is discussed in an NRC letter report “Palisades Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA)”, ADAMS Accession number ML042880473, March 2005. It was concluded that ignoring uncontrolled RV cool down as a result of an excessive steam demand event(s) in the development of plant procedures posses little change in the plants operational risk profile.

The varying of the rate of feedwater addition will be the predominant method of PCS temperature control. Operators will have to use their skills and knowledge in order to determine an appropriate feed rate. It should be noted that S/G level may continue to lower even though the appropriate feed rate is established. As long as the S/G continues to remove heat, feeding a S/G with level lower than minus 84% is acceptable. Should the ability to control feedwater be lost, then consideration should be given to un-isolating the S/G faulted by the SGTR and steaming it. If fuel failures are present which result in measured releases are greater than 10 CFR 100 for the only generator that can be employed for cool down, then isolate both S/G's and commence once-through-cooling.

Isolating the ESDE S/G and Steaming the SGTR S/G

This strategy would be preferable to going to OTC unless fuel failures exists that would result in exceeding 10CFR100 dose limits at the site boundary. It would also be preferable if the steaming rate from the ESDE required feedwater feed rates that exceed what can be provided or when the ESDE is inside containment or the CCW room.

Use of either of the above strategies is acceptable if the necessity to go to OTC is avoided. If either S/G cannot be used, then the only available option is to go to OTC.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003.K1.01	
	Importance Rating	2.6	

K/A statement: Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems: RCP lube oil

Proposed Question:

The Plant is on Shutdown Cooling. You have been directed to start Primary Coolant Pump P-50B during plant startup. After starting the AC Oil Lift Pump P-80B, you note that the WHITE "PUMP START OIL PERMISSIVE" light just above the P-50B handswitch does NOT illuminate. Which of the following is the alternate method of satisfying the required oil permissive interlock?

A. Start P-50B, Primary Coolant Pump, without delay.

B. Start the DC Oil Lift Pump.

C. Notify Maintenance to prime the Oil Lift Pumps.

D. Wait two minutes and attempt to start P-50B, Primary Coolant Pump.

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, the breaker will not close without the Pump Start Oil Permissive satisfied
B. Correct.
C. Incorrect, this is done if the Pump Start Oil Permissive is not met with both AC and DC Oil Lift Pumps running.
D. Incorrect, the two minute duration for starting the PCP is met after the Lift Oil Pump(s) have run with the Pump Start Oil Permissive satisfied.

Technical Reference(s): SOP-1A
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam Palisades 2001___________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__

55.43 _____

Comments:

NOTE:	Running more than one DC lift pump on a DC bus for extended time periods will cause a draw on the battery and result in annunciation of EK-0548, "125 VOLT DC BUS TROUBLE/UNDERVOLTAGE."

NOTE:	In an odd year, the AC Oil Lift Pump should be used if available for the PCP(s) to be started. In an even year, the DC Oil Lift Pump should be used if available for the PCP(s) to be started.

p. **START** the AC or DC Oil Lift Pump for the PCP(s) to be started as directed by CRS:

PCP	Oil Lift Pump	Description
P-50A	P-80A	AC Primary Coolant Pump Oil Lift Pump
	P-81A	PCP P-50A DC Oil Lift Pump
P-50B	P-80B	PCP P-50B AC Lift Pump
	P-81B	PCP P-50B DC Lift Pump
P-50C	P-80C	PCP P-50C AC Lift Pump
	P-81C	P-50C DC Oil Lift Pump
P-50D	P-80D	AC Primary Coolant Pump Oil Lift Pump
	P-81D	DC Primary Coolant Pump Oil Lift Pump

NOTE:	Both AC and DC Oil Lift Pumps may be required to be in operation to satisfy the lift oil pressure interlock.

q. <u>IF</u> lift oil pressure interlock is <u>NOT</u> satisfied with one lift pump operating, <u>THEN</u> **START** second Oil Lift Pump for PCP to be started.

r. <u>IF</u> lift oil pressure interlock is <u>NOT</u> satisfied with both Lift Pumps in operation, <u>THEN</u> **NOTIFY** Maintenance to prime Lift Pumps.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004.K2.05	
	Importance Rating	2.7	

K/A Statement: Knowledge of bus power supplies to the following: MOVs

Proposed Question:

Given the following conditions:

- The Plant tripped from 100% power.
- Emergency boration requirements are met.
- Bus 19 is faulted and cannot be re-energized.

An emergency boration can be performed from the Control Room using which of the following valves?

A. MO-2170, Boric Acid Tank T-53B Gravity Feed Isolation.

B. MO-2169, Boric Acid Tank T-53A Gravity Feed Isolation.

C. MO-2140, Boric Acid Pump Feed Isolation Valve.

D. MO-2087, VCT Outlet Isolation Valve.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, MO-2170 power supply is MCC-1 (fed from Bus 19)
B. Incorrect, MO-2169 power supply is MCC-1 (fed from Bus 19)
C. Correct, MO-2140 power supply is MCC-2 (fed from Bus 20). BA Pump P-56A remains energized to allow for pumped feed boration capability.
D. Incorrect, MO-2087 power supply is MCC-1 (fed from Bus 19)

Technical Reference(s): SOP-2A, PL-CVCS Chemical and Volume Control System Lesson Plan, E-1 Sheet 1, E-4 Sheet 1 & 2

(Attach if not previously provided, including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective:	______________________ (As available)		
Question Source:	Bank #	_______	
	Modified Bank #	_______	(Note changes or attach parent)
	New	___X___	
Question History:	Last NRC Exam	____________	

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

EMERGENCY MANUAL BORATION

1.0 **ENSURE** charging flow greater than 33 gpm.

2.0 **ESTABLISH** at least one (both preferred) boric acid flow path(s) as follows:

a. IF Bus 1D energized, THEN **ESTABLISH** pumped feed:

1. **START** at least one (both preferred) Boric Acid Pump(s).

   - P-56A, Boric Acid Pump
   - P-56B, Boric Acid Pump

2. **OPEN** MO-2140, Boric Acid Pump P-56A/B Feed Isolation

3. **VERIFY** Charging Flow greater than 33 gpm.

b. IF Bus 1C energized, THEN **ESTABLISH** Gravity Feed:

1. **OPEN** Boric Acid Tank Gravity Feed Isol Valves.

   - MO-2169, BAST T-53A Gravity Feed Isolation
   - MO-2170, BAST T-53B Gravity Feed Isolation

2. **CLOSE** CV-2155, Make-Up Stop.

CAUTION
If CK-CV-2171, Boric Acid Gravity Feed Check, sticks closed during the next step, Charging Pumps may trip on low suction pressure.

3. **CLOSE** MO-2087, VCT T-54 Outlet Valve.

4. **ENSURE CLOSED** MO-2160, SIRWT T-58 Outlet To Charging Pp P-55A,B,C.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	004.A1.09	
Importance Rating	3.6	

K/A Statement: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: RCS Pressure and Temperature

Proposed Question:

Given the following conditions:

- The Plant is solid in Mode 5.
- One train of Shutdown Cooling (SDC) is in service.
- PIC-0202, "Intermediate Pressure Letdown Controller," is in MANUAL for control of Intermediate Letdown Regulating Valve, CV-2012.

If CV-3025, SDC Hx Outlet, is throttled CLOSED, what is the effect on PCS temperature AND how would CV-2012 be operated to maintain letdown pressure on setpoint?

PCS temperature will:

A. RISE requiring CV-2012 to be throttled OPEN.

B. LOWER requiring CV-2012 to be throttled CLOSED.

C. RISE requiring CV-2012 to be throttled CLOSED.

D. LOWER requiring CV-2012 to be throttled OPEN.

Proposed Answer: A

Explanation (Optional):

If CV-3025 was throttled closed, PCS temperature would rise. With PCS temperature rising, PCS pressure will rise. Letdown regulating valve CV-2012 will auto open to lower pressure back to setpoint. PIC-0202 output raises causing CV-2012 to OPEN to lower letdown pressure.

A. Correct. See explanation.
B. Incorrect. See explanation. CV-3025 operates to control PCS temperature. This would be the response if the valve were throttled closed
C. Incorrect. See explanation.
D. Incorrect. See explanation. CV-3025 operates to control PCS temperature. This would be the response if the valve were throttled closed

Technical Reference(s): SOP-1B, PL-CVCS CVCS Lesson Plan Rev 7

(Attach if not previously provided, including version/revision number) ______________________

Proposed references to be provided to applicants during examination: ____None__________

Learning Objective: __________________________ (As available)

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

Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

e. CVs-2012, 2122 (Intermediate Letdown Regulating valves)

1) 2 valves in parallel

2) Maintains letdown pressure above saturation (and less than RV-2006 setpoint) pressure between the letdown heat exchanger and pressure reduction orifices. Prevents flashing (boiling) of letdown fluid prior to being cooled by the letdown heat exchanger.

3) Reduces letdown pressure to design limits of demins

4) Controls PCS pressure during solid plant operations. Varies letdown flow for fixed manual charging rate. Controls PCS pressure by controlling the PCS volume. Selected Intermediate Pressure Letdown Control Valve will control PCS pressure by adjusting letdown flow. This pressure can be set or adjusted via the valve controller on C-02. A change in temperature will cause either a contraction of the water molecules in the PCS (lower temperature) or expansion of the water molecules in the PCS (higher temperature) and Letdown flow will need to be adjusted accordingly.

5) 2 controllers

a) PIC-0202 (Intermediate Pressure Letdown Controller) on C-02

Can control in either:

(1) Automatic

(a) Auto controls via input from PT-0202 (Low Pressure Letdown Pressure Transmitter) and also receives an anticipatory signal from the Letdown Stop Valves to prevent/-reduce pressure transients and reduce cycling of RV-2006.

(b) PIC-0202 "remembers" what its output should be for 1, 2 and 3 orifices open. When an orifice either opens or closes, a constant is added to the controller output, which will cause a step change in the controller output. Once the constant has been added, the controller feedback will take over and control at its setpoint.

(c) Auto pushbutton will be lit

(d) Red pointer shows Letdown pressure

(e) Blue pointer shows Letdown pressure setpoint

(f) Meter shows output to CV

(g) Lever has no function in Auto

(h) Fail light lit, call I&C

(i) Alarm light lit, call I&C

(2) Manual

(a) Manual pushbutton will be lit,

(b) Red pointer shows Letdown pressure

(c) Blue pointer shows Letdown pressure setpoint, has no function in Manual

(d) Meter shows output to CV

(e) Lever will open and close CV

(f) Fail light lit, call I&C

(g) Alarm light lit, call I&C

b) HIC-2122 (Intermediate Pressure Letdown Controller) on C-12

Manual control only

(1) Red pointer shows Letdown pressure

(2) Blue pointer shows signal to CV, corresponds to position

(3) Knob opens and closes CV

c) HS-0202 (on C-12)

(1) Determines which valve is controlled by which controller.

(2) 2 positions

(a) CV-2122 Manual / CV-2012 Auto

CV-2122 is controlled by HIC-2122 and CV-2012 is controlled by PIC-0202

(b) CV-2122 Auto / CV-2012 Manual

CV-2122 is controlled by PIC-0202 and CV-2012 is controlled by HIC-2122

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005.K3.07	
	Importance Rating	3.2	

K/A Statement: Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: Refueling operations

Proposed Question:

Given the following conditions:

- The Plant is shut down for a refueling outage
- Core reload is in progress
- LPSI Pump P-67B is in service providing Shutdown Cooling
- Qualified CETs indicate 105°F
- Reactor cavity level is 647 feet
- The following alarms have just alarmed:
  - EK-1101, "Containment Instr Air Lo Press"
  - EK-1102, "Instrument Air Lo Press"
  - EK-1103, "Service Air Lo Press"
- Instrument Air header pressure is lowering at the rate of 15 psig per minute

Complete the following statements:

After 5 MINUTES, the Primary Coolant System cooldown rate will __(1)__ and core reload must be suspended __(2)__.

A. (1) Rise
(2) immediately

B. (1) Rise
(2) within 1 hour

C. (1) Lower
(2) immediately

D. (1) Lower
(2) within 1 hour

Proposed Answer: **C**

Explanation (Optional):

A loss of Instrument Air (~25 psig after the 5 minute duration) will result in the SDC Hx bypass valve to fail open (CV-3006) and the SDC Hx outlet valve (CV-3025) to fail closed. This will result in the PCS cooldown rate to lower (i.e. PCS temperature will rise as a result of the loss of SDC). Per AOP-30 Step 9, fuel movements are to be stopped upon a loss of shutdown cooling, with no allowable duration.

A. Incorrect, the applicant does not understand the relationship between a loss of instrument air and shutdown cooling, specifically the impact on the PCS cooling RATE. While PCS temperature will RISE, the cooldown rate will LOWER.
B. Incorrect, see choice A. Additionally, the applicant could not understand the procedural requirements for a loss of shutdown cooling.
C. Correct, see explanation.
D. Incorrect, the applicant does not understand the procedural requirements for a loss of shutdown cooling.

Technical Reference(s): DBD-2.01, AOP-37
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: ___None______

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam __________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _7__
55.43 _____

Comments:

In its 24 hour mission mode, the LPSI System unavailability is dominated by the failure to provide core cooling due to failure to establish flow through the SDCHX. The important failures are;

1. Failure of CV-3025 or CV-3055 to open (no flow to the heat exchanger).

2. Failure of CV-3006 to close (diversion of flow away from the heat exchanger).

These failures represent approximately 85% of the system unavailability. Other minor contributions involve loss of air to the valves due to filter plugging, failure of the solenoid valves to CV-3055 to actuate or malfunction of pressure regulation.

In the shutdown cooling mode of operation, the dominant failures of the LPSI System involve single failures of the air-operated and motor-operated valves common to both pump suctions and discharges. The dominant failure modes involve failure of system actuation. These failures include;

1. Failure of the air-operated valves (CV-3025 and CV-3055) to open (loss of cooling).

2. Failure of the motor-operated valves (MO-3015 and MO-3016) to open (loss of pump suction).

3. Failure of CV-3006 to close (loss of cooling - diversion of flow away from the shutdown heat exchanger).

Other minor contributions include failure of the limit switches on the motor-operated valves, failure of I/I-0306A/B and possible system misalignment.

LOSS OF INSTRUMENT AIR

VALVES WHICH FAIL CLOSED

VALVE	DESCRIPTION	REQUIRED ACTION
CV-3025	SDC Heat Exchanges E-60A/B Outlet (MZ-32)	REFER TO AOP-30, "Loss of Shutdown Cooling."
CV-1002	Primary System Drain Tank T-74 Outlet Isol	NONE
CV-1007	Primary System Drain Tank T-74 Outlet Isol	
	All normal Feedwater Heater Drain Valves	NONE
		CAUTION Hogging Air Ejector suction should be opened only on orders of Shift Manager. Pulling vacuum on hot condenser will cause acceleration of Condenser flashing. This flashing can result in Condenser damage including Circulating Water leakage/ruptures.
CV-0511	Turbine Bypass to Condenser	<u>IF</u> MSIVs are closed, <u>THEN</u> REFER TO EOP Supplement 23, "Align Hogging Air Ejector for PCS Cooldown," to supply steam to the Hogging Air Ejector.

LOSS OF INSTRUMENT AIR

VALVES WHICH FAIL OPEN

NOTE: Valves identified with an * are affected by isolating air to Containment.

VALVE	DESCRIPTION	REQUIRED ACTION
*CV-2111	Charging Line Stop Valve	NONE
*CV-2113	Charging Line Stop Valve	
*CV-2115	Charging Line Stop Valve	
	Feedwater Heater High Level Dump Valves	NONE
		NOTE: CV-0911 and CV-0940 have accumulators.
CV-0910	CCW Containment Isolation Valve	IF valves are required to be closed, THEN REFER TO Step 13 of this procedure.
CV-0911	CCW Containment Isolation Valve	
CV-0940	CCW Containment Isolation Valve	
CV-0909	CCW outlet from Letdown Heat Exchanger	REFER TO ARP-4, EK-0706, LETDOWN HX COOLING EXCESS FLOW.
CV-3006	Shutdown Cooling Hx Bypass	REFER TO AOP-30, "Loss of Shutdown Cooling."

ACTIONS\EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. **ENSURE** PCS level is restored as high as possible via existing PCS Inventory Addition Flow Path(s) to extend the time to 200°F. Refer to GOP-14, "Shutdown Cooling Operations," attachment titled "Shutdown Cooling Equipment Availability."	
9. **STOP** movement of irradiated fuel.	

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005.A2.03	
Importance Rating	2.9	

K/A Statement: Ability to (a) predict the impacts of a RHR pump/motor malfunction, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

Proposed Question:

The following conditions exist.

- The Plant is in Mode 5.
- Shutdown cooling is in-service using LPSI Pump P-67A.
- Primary Coolant Pumps P-50B and P-50D are in-service.
- All PCS and SDC temperatures are slowly lowering.

The control room receives annunciator EK-1162, “LPSI Pump Low Discharge Pressure.”

Following the alarm’s receipt, the control room operators observe the following indications:

- PCS Pressure slowly rising.
- PCS Temperatures slowly rising.
- Pressurizer Level Off-Scale High.
- SDC temperatures are stable.
- Red indicating light for P-67A is LIT.
- SDC Hx valves CV-3006 and CV-3025 position indication remain unchanged.
- SDC Flow is zero.

Which of the following is a possible explanation of the indications provided and what would be the appropriate course of action per the applicable procedure?

A. Problems with the SDC Hx valves CV-3006 and/or CV-3025; dispatch an NLO to investigate.

B. Problems with CCW Cooling flow to the SDC Hx; dispatch an NLO to investigate.

C. Problems with LPSI Pump P-67A; trip LPSI Pump P-67A.

D. Problems with LPSI Pump P-67A; start LPSI Pump P-67B.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, CV-3006 and CV-3025 positions indications remain unaffected and SDC flow is zero. Both SDC Hx control valves would have to close to lose flow indication.
B. Incorrect, problems with CCW flow to the SDC Hx would not explain a loss of SDC flow and stable SDC temperatures.
C. Correct, per AOP-30 reactor and equipment trip criteria, if shutdown cooling flow is less than 170 gpm with an operating LPSI pump, that pump shall be tripped.
D. Incorrect, incorrect procedure adherence. LPSI Pump P-67A must be tripped and the low shutdown cooling flow conditions resolved prior to starting P-67B.

Technical Reference(s): P&ID 204 sheet A, ARP-7, AOP-30
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: ____None_________

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:
Question modified from Palisades 2006 Audit Exam. Modified question stem, changed correct answer, replaced one distractor.

ACTIONS\EXPECTED RESPONSE	RESPONSE NOT OBTAINED
LPSI Pump Tripped (Step 23 through Step 28)	
23. **VERIFY** the following conditions exist: • PCS level greater than or equal to 617'8" • Shutdown Cooling flowrate in compliance with GOP-14 prior to LPSI pump trip • Shutdown Cooling has been lost for less than 10 minutes • LPSI pump to be started was not tripped due to cavitation • Available LPSI pump has power: o P-67A (Bus 1D) o P-67B (Bus 1C) • EDG Load Sequencing completed, if operating	23.1 GO TO Step 24.
a. **START** an available LPSI pump. 1) Standby pump (preferred) 2) Tripped pump	a.1 GO TO Step 24.
b. **VERIFY** in service LPSI Pump flow rate greater than 170 gpm.	b.1 **SECURE** affected LPSI Pump. b.2 **DETERMINE** <u>AND</u> **CORRECT** problem prior to starting either LPSI pump. b.3 GO TO Step 24.
c. **VERIFY** SDC flowrate requirements are met.	c.1 **ADJUST** Shutdown Cooling Valves to establish adequate flow.
d. GO TO Step 92.	

REACTOR AND EQUIPMENT TRIP CRITERIA

Equipment Trip

Emergency Diesel Generator 1-1, using K-6A/MOS, D/G 1-1 Mechanical Overspeed Trip/Reset

- Jacket water temperature greater than or equal to 195°F on TI-1482, Diesel Generator K-6A Jacket Water Temp
- Lube oil temperature greater than or equal to 200°F on TI-1478, Diesel Generator K-6A Lube Oil Temperature Indicator

Emergency Diesel Generator 1-2, using K-6B/MOS, D/G 1-2 Mechanical Overspeed Trip/Reset

- Jacket water temperature greater than or equal to 195°F on TI-1492, Diesel Generator K-6B Jacket Water Temp
- Lube oil temperature greater than or equal to 200°F on TI-1488, Diesel Generator K-6B Lube Oil Temperature Indicator

Operating LPSI Pump, P-67A or P-67B

- Severe LPSI Pump cavitation not corrected by throttling Shutdown Cooling flow
- Shutdown Cooling flow less than 170 gpm

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	006.K4.11	
	Importance Rating	3.9	_____

K/A Statement – Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Reset of SIS

Proposed Question:

Given the following conditions:

- The Plant is being cooled down and depressurized in preparation for a refueling outage.
- Safety Injection Signal (SIS) has been BLOCKED.
- A failure of the Pressurizer pressure controller causes PCS pressure to rise from 1550 psia to the following:
    - A Channel - 1700 psia
    - B Channel - 1685 psia
    - C Channel - 1695 psia
    - D Channel - 1705 psia

Based on the above conditions, the Safety Injection Signal is:

A. No longer blocked since 3/4 pressure channels have increased above the reset setpoint. Safety Injection WILL actuate when pressure is lowered to <1605 psia.

B. No longer blocked since 3/4 pressure channels have increased above the reset setpoint. Safety Injection WILL actuate when pressure is lowered to <1690 psia.

C. Still blocked since not all of the pressure channels have increased above the reset setpoint. Safety Injection WILL NOT actuate when pressure is lowered.

D. Still blocked since the block switches have not been placed to RESET. Safety Injection WILL NOT actuate when pressure is lowered.

Proposed Answer: **A**

Explanation (Optional):

A. Correct, when 3 of 4 pressure channels increase to >1690 psia, the SIS rearms itself (unblocks). When 2/4 pressure channels lower to <1605 psia, SIS will re-actuate.
B. Incorrect, applicant is correct that the SIS is no longer blocked but misapplies the setpoint of initiation.
C. Incorrect, applicant misapplies the reset logic, in that all 4 do not have to rise above 1690.

D. Incorrect, applicant believes that the SIS block must be reset with the switch but it automatically resets on pressure.

Technical Reference(s): E-17 Sheet 3, PL-SIS Safety Injection System Lesson Plan
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam Palisades 2009
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

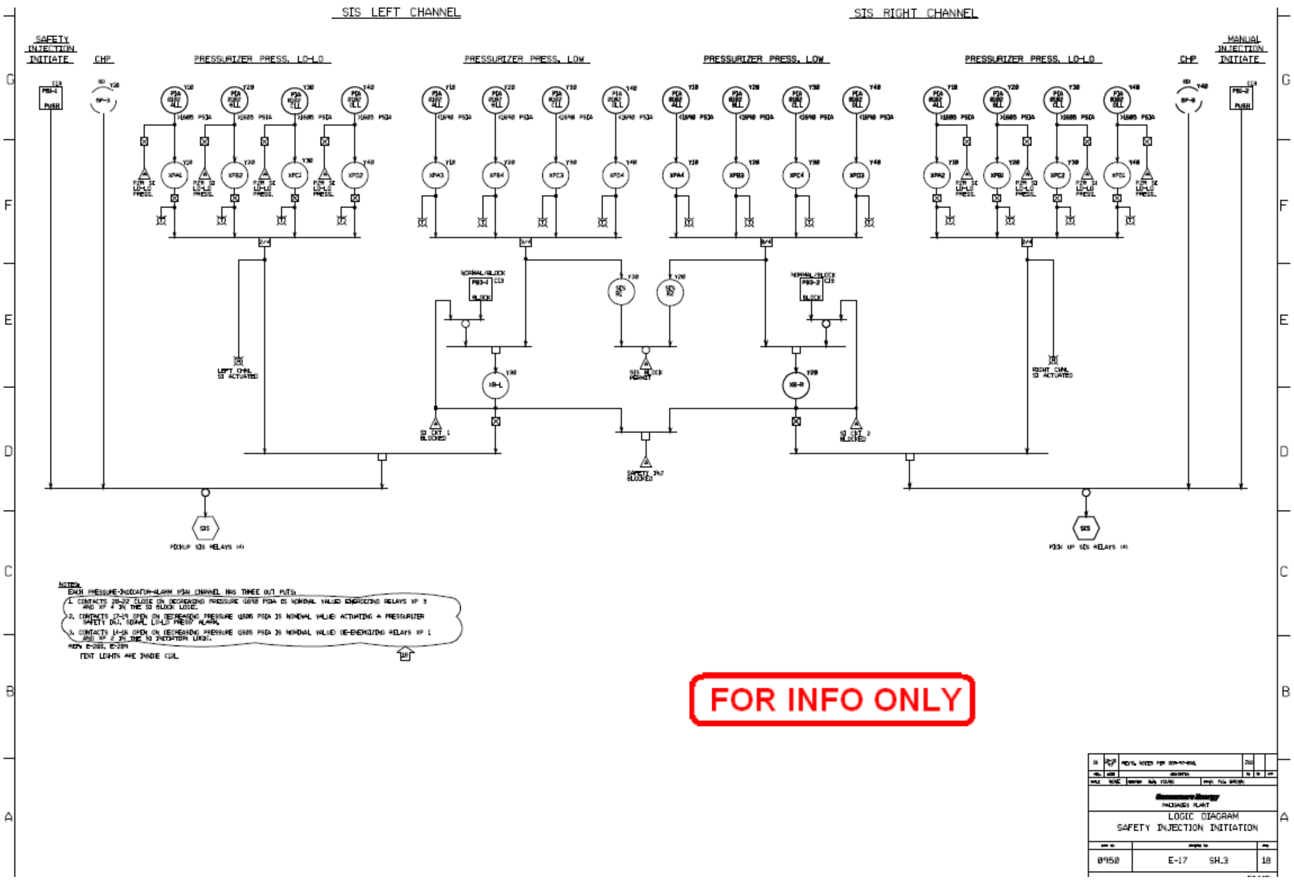
10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

A. Blocking SI Actuation

1. What conditions are necessary to block SIS?

A: 3 of 4 pressures (PIA-0102ALL, 0102BLL, 0102CLL, 0102DLL) less than 1690 psia. Then PB 3-1 and PB 3-2, “SIS Block Switches” are each turned and momentarily held to physically block each train of SIAS.



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006.A1.13	
	Importance Rating	3.5	

K/A Statement: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration).

Proposed Question:

Given the following conditions:

- The Plant is at 100% power.
- Safety Injection Tank (SIT) T-82B has a high level alarm in.
- SOP-3 Attachment 3 is being performed to lower level in T-82B.
- Chemistry has provided an SIT T-82B boron concentration, at the beginning of the shift, of 2520 ppm
- SIT T-82B level is 190”. (~38% narrow range)
- SIT T-82B pressure is 210 psig.

With the given conditions, what is the current operability condition of SIT T-82B and why?

A. Operable, all SIT parameters are within LCO 3.5.1 limits.

B. Inoperable, the boron concentration is not within LCO 3.5.1 limits.

C. Inoperable, the level is not within LCO 3.5.1 limits.

D. Inoperable, the pressure is not within LCO 3.5.1 limits.

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, boron concentration is out of spec high.
B. Correct, boron concentration must be 1720-2500 ppm.
C. Incorrect, level is within the volumetric requirement of 1040-1176 ft^3 (174-200”).
D. Incorrect, pressure is >200 psig, as required per LCO 3.5.1.

Technical Reference(s): LCO 3.5.1 and Bases, ARP-8, SOP-3
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source: Bank # _______

Modified Bank # _______ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam ___________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__

55.43 _____

Comments:

Is there a learning objective which states the ROs must know these TS surveillance limits?

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each SIT isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each SIT is ≥ 1040 ft^3 and ≤ 1176 ft^3.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is ≥ 200 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each SIT is ≥ 1720 ppm and ≤ 2500 ppm.	31 days
SR 3.5.1.5	Verify power is removed from each SIT isolation valve operator.	31 days

#

BASES

APPLICABLE SAFETY ANALYSES (continued)

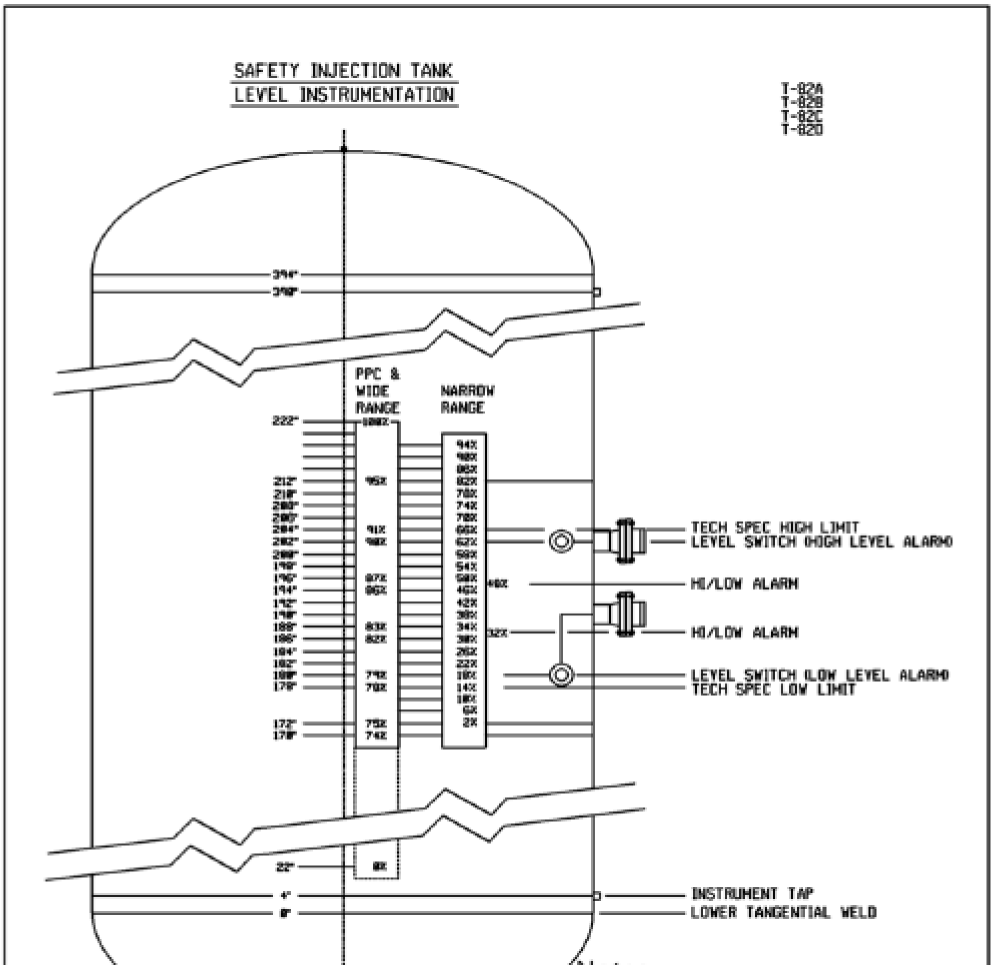
The minimum SIT volume of 1040 ft^3 and the maximum SIT volume of 1176 ft^3 correspond to a level of 174 inches and 200 inches, respectively. Each SIT is equipped with two float type level switches which activate control room alarms on high and low level. To allow for instrument inaccuracy, the low SIT level switch alarm is set at 176 inches and the high SIT alarm is set at 198 inches. As a backup to the SIT level switches and to facilitate operator use, level indication is also provided by a differential pressure transmitter which displays in percent tank level. The narrow indicating range of the differential pressure transmitter contains high and low alarms. The high level alarm trips at a slightly lower level than the high level switch and the low level alarm trips at a slightly higher level than the low level switch to alert the operator they are approaching the technical specification values.

The minimum nitrogen cover pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analyses.

A minimum pressure of 200 psig is used in the analyses. Each of the four SITs is equipped with two pressure switches and one pressure transmitter. The pressure switches activate separate control room alarms. One pressure switch provides a high pressure alarm and the other provides a low pressure alarm. The pressure transmitter provides a display of tank pressure and a common high/low pressure alarm. The low pressure alarms from the pressure switch and pressure transmitter are set sufficiently above the 200 psig value used in the safety analysis to provide margin for instrument inaccuracies. The high pressure alarms from the pressure switch and pressure transmitter are set well below the 250 psig tank design pressure and sufficiently above the normal operating pressure to avoid nuisance alarms.

The 1720 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the SITs, the reactor will remain subcritical in the cold condition following mixing of the SITs, Safety Injection Refueling Water Tank and PCS water volumes. Small break LOCAs assume that all full-length control rods are inserted, except for the control rod of highest worth, which is withdrawn from the core. Large break LOCA analyses assume that all full-length control rods remain withdrawn until the blowdown phase is over. For large break LOCAs, the initial reactor shutdown is accomplished by void formation. The most limiting case occurs at beginning of core life.

SAFETY INJECTION TANK LEVEL INSTRUMENTATION



Notes

1. TECHNICAL SPECIFICATION REFERENCE TANK LEVELS ARE REFERENCED IN INCHES ABOVE LOWER INSTRUMENT TAP. (LCO 3.5.1)
2. 178" (T.S. 174") IS EQUIVALENT TO A WATER VOLUME OF 1040 CUBIC FEET
3. 204" (T.S. 200") IS EQUIVALENT TO A WATER VOLUME OF 1176 CUBIC FEET.
4. 1 INCH CHANGE IN TANK LEVEL EQUALS 39.7 GALLONS

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007.K5.02	
	Importance Rating	3.1	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to PRTS: Method of forming a steam bubble in the PZR.

Proposed Question:

Given the following conditions:

- The Plant has just completed drawing a bubble from a solid plant condition.
- Quench Tank, T-73 pressure was noted to be 1 psig.
- Pressurizer (PZR) is being maintained at 225 psig by cycling backup PZR heaters.
- PZR temperature is 397°F.

If a PZR PORV is venting fluid to the Quench Tank, which of the following states what the expected PORV tailpipe temperature would be AND the Technical Specification LCO leakage limit for this type of leakage?

A. ~320ºF; 1 gpm

B. ~320ºF; 10 gpm

C. ~387ºF; 1 gpm

D. ~387ºF; 10 gpm

Proposed Answer: **A**

Explanation (Optional):

By TS definition, Identified leakage is leakage, such that from pump seals or valve packing (except Primary Coolant Pump seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank. The limit for Identified leakage is 10 gpm (see TS definitions). Quench Tank leakage is not “Identified Leakage” as the Plant has no accepted method to quantify it per the Leakrate Program.

A. Correct, Find the 240 psia (225 psig) constant pressure line and where it intersects with the saturation line. Draw a straight line (throttling process) from the 240 psig mark on the saturation line to the 16 psia (Quench Tank is at 1 psig) pressure line. The intersecting temperature line is approximately 320ºF. Part 2 correct, 1 gpm is the limit for unidentified leakage.

B. Incorrect, Part 1 is correct. See explanation above for Part 2.

C. Incorrect, applicant uses the saturated temperature (387ºF) for 225 psig. 1 gpm is the limit for unidentified leakage.

D. Incorrect, see explanation in choice C. Part 2 correct.

Technical Reference(s): Steam Tables, Tech Specs Section 1.1 (definitions)______
(Attach if not previously provided, ____
including version/revision number) ____

Proposed references to be provided to applicants during examination: Steam Tables______

Learning Objective: ___________________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

1.1 Definitions

CHANNEL FUNCTIONAL TEST (continued)	b. Digital channels - the use of diagnostic programs to test digital hardware and the injection of simulated process data into the channel to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or control rods within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except Primary Coolant Pump seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

1.1 Definitions

LEAKAGE

a. Identified LEAKAGE (continued)

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known not to interfere with the operation of leakage detection systems and not to be pressure boundary LEAKAGE; and

3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE (except Primary Coolant Pump seal leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an PCS component body, pipe wall, or vessel wall.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008.K3.01	
	Importance Rating	3.4	

K/A Statement: Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Proposed Question:

The Plant is at 100% power with the following conditions:

- Component Cooling Water (CCW) Pump P-52A out of service for maintenance.
- CCW Pump P-52B is running
- CCW Pump P-52C is in standby

If the handswitch for CV-0944A, CCW to SFPHXs & RW Evaps, was inadvertently placed in the BYPASS position, which one of the following describes an expected consequence?

A. CV-0944A fails open; CCW Pump P-52C auto-starts on low discharge header pressure.

B. CV-0944 fails closed; the Radwaste Evaporators would lose CCW cooling.

C. Upon receipt of a SIAS, CCW flow to required components would be diverted.

D. Upon receipt of a SIAS, the SFP HXs would lose CCW cooling.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, CV-0944A will remain open with the switch in bypass and will not close, as expected, upon receipt of a SIS
B. Incorrect, CV-0944 will remain open with the switch in bypass. This allows continued cooling to the Radwaste Evaporators.
C. Correct, upon the SIAS, CV-0944A would remain open, diverting required CCW cooling flow from DBA loads.
D. Incorrect, CV-0944A would remain open and SFP HX cooling would continue.

Technical Reference(s): DBD-1.01
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source: Bank # _______

Modified Bank # ___X___ (Note changes or attach parent)

New _______

Question History: Last NRC Exam __________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__

55.43 _____

Comments:

Palisades 2010 Audit Exam. Modified stem, modified (one new) and rearranged distractors.

Specification/Field Change (SFC) 75-011 (Reference 115) physically moved the monitor to a different location in the room, reoriented the monitor to direct sample flow upwards through the sample chamber, and added a drain valve.

SIS Actuated Valve Controls

CV-0944A (CCW supply to the SFPHX), isolates upon receipt of a SIS in order to "free" cooling water for emergency equipment. However, capability for the operator to override the SIS later in order to maintain fuel pool temperatures within required limits is important to safety. Previous design of the controls included "normal" and "bypass" hand switch positions only. While the switch was in "normal," the associated valve was open, but would close upon receipt of a Safety Injection Signal (SIS). The operator had to place the hand switch in "bypass" in order to close the associated valve during normal operation; however, this had the undesirable side effect of setting up logic for the valve to open upon the receipt of a SIS.

In response to Deviation Report D-PAL-87-200A, FC-869 (Reference 102) was implemented to revise the control logic for CV-0944A. A new hand switch was installed with three positions: "Close," "Bypass" and "Open," thus eliminating the identified operational problem.

Diversion of CCW to the SFPHX to prevent overheating following a DBA has been evaluated (Reference 86). The capability of the CCW System to provide adequate flow during normal shutdown was considered to be bounding for the DBA case with SFPHX flow.

Engineering Analysis EA-A-PAL-92-105 (Reference 178) evaluated a situation in which one channel of Safety Injection, rather than one Emergency Diesel Generator, fails during a LOCA. The analysis determined that if right channel SIS failed, CV-0944A would not close on a Safety Injection Signal. This would cause a diversion of CCW flow from the CCWHX and a consequential reduction in the amount of containment cooling provided by the ECCS. Reference 178 determined that the resulting CCW flow rates still met the minimum flow requirements for all essential loads, with the exception of Charging Pumps P-55B and P-55C, and the shutdown cooling heat exchangers. The reduction in flow to these components was analyzed and found to be acceptable. References 178, 179, 180, 222 and 298 analyzed the containment response assuming this CCW flow diversion due to the SIS channel failure. Reference 250 subsequently evaluated the reduction of CCW flow to the ESS pumps and charging pumps along with the elevated CCW temperature caused by the flow diversion due to the SIS channel failure. Also see Sect 3.2.9 of this document.

SYSTEM DESIGN - COMPONENT COOLING WATER

The Component Cooling Water System is designed to cool components carrying radioactive and potentially radioactive fluids. It provides a monitored intermediate barrier between these fluids and the Service Water System. Thus, the probability of leakage of contaminated fluid into the lake is greatly reduced.

The system is a closed loop consisting of three motor-driven centrifugal pumps (6,000 gpm @ 164 ft head), two heat exchangers, and a surge tank. During normal operation, one or two of the CCW pumps and the two CCW heat exchangers will be in service. The pump(s) run continuously with the other pump(s) in "STANDBY", which would get a start signal if CCW header pressure lowers to a preset value.

Both Component Cooling heat exchangers are required to be in service during all plant operating modes above cold shut down. The temperature of the Component Cooling Water at the heat exchanger discharge is controlled between 72° and 90° F by regulation of the Service Water flow to the heat exchangers.

A radiation monitor in the pump discharge header monitors radioactive inleakage into the CCW system from components being cooled. The CCW surge tank is normally vented to room atmosphere. If the CCW radiation monitor senses high radiation, the surge tank vent is automatically swapped to the Vent Gas collection header. Make-up to the CCW surge tank is automatic from the Level Control System, with make-up from primary make-up utility water.

During shutdown cooling operations, two or all three CCW pumps will be in operation with Component Cooling Water being supplied to the Shutdown Cooling heat exchangers.

CCW SYSTEM RESPONCE POST DBA (SIS)

The CCW pumps will receive starts following a Safety Injection. If offsite power is available following the SIS, all three CCW pumps get a start signal. Following a SIS without-offsite power available, two of the CCW pumps will get a start signal from the DBA Sequencers, the third CCW pump is sequenced to "STANDBY", and will only start on CCW system low pressure. If offsite power to the CCW pumps is lost without a SIS present, two pumps are given a start signal from the normal shutdown (NSD) sequencer, and the third pump is sequenced on to start if a low discharge pressure is present, indicating that the other pumps are not operating. With a SIS, the supply of CCW to the Spent Fuel Cooling System and to the Radwaste Evaporators is automatically isolated by closing valves in the supply and return lines. The control valve in the CCW supply line to the Spent Fuel Cooling System can be opened from the control room during the event to prevent overheating of the spent fuel pool. The isolation of cooling assures additional cooling capability to the shutdown cooling heat exchangers.

With a receipt of a low-level in the SIRW tank, the CCW inlet valves to the CCW heat exchangers receive an open signal. If a containment high pressure occurs, the CCW supply and return header control valves for containment will close.

For post-DBA CCW operation, one pump can furnish 100% of the required capability for cooling the containment spray and safety injection recirculation water.

AUTOMATIC STARTS OF CCW PUMPS

CCW P-52A or P-52C (P-52B), whichever is selected for "STANDBY" is started by PS-0918 (PS-0919) on low CCW header pressure of 80 psig. Disabling the low pressure "STANDBY" start of the CCW pumps does not render them inoperable.

COMPONENT COOLING WATER ALARMS

ALARM		SENSOR	SETPOINT	AUTOMATIC ACTION
EK-1155 (1156)	WEST (EAST) ROOM SAFEGUARD PPS CLG WTR LO FLOW	FS-0954 AND POS-0879, POS-0880, OR POS-0913 FS-0958 AND POS-0879, POS-0880, OR POS-0913	9 TO 11 GPM 4 TO 6 GPM	NONE
EK-1167	COMPONENT CLG PUMPS P-82A, P-52B, P-52C TRIP	$\dfrac{152\text{-}109}{b}$, $\dfrac{152\text{-}116}{b}$ or $\dfrac{152\text{-}208}{b}$	BREAKER OPEN WITH CONTROL SWITCH IN "AFTER-CLOSE" POSITION	NONE
EK-1168	COMPONENT CLG PUMPS STANDBY PUMP RUNNING	144-109+$\dfrac{152\text{-}109}{a}$, 144-116+$\dfrac{152\text{-}116}{a}$, OR 144-208+$\dfrac{152\text{-}208}{a}$	BREAKER CLOSED ON A PUMP SELECTED FOR "STANDBY"	COMPONENT COOLING WATER PUMP P-52A OR P-52C (P-52B), WHICH- EVER IS SELECTED FOR "STANDBY", IS STARTED BY PS-0918 (PS-0919) ON LOW CCW PRESSURE (80 PSIG)
EK-1169	COMPONENT CLG PUMP DISCHARGE LO PRESS	PIA-0918	80 PSIG	CCW P-52A OR P-52C (P-52B), WHICHEVER IS SELECTED FOR "STANDBY", IS STARTED BY PS-0918 (PS-0919) ON LOW CCW HEADER PRESSURE (80 PSIG)
EK-1170 (1171)	COMPONENT CLG EX-E54A (EX-E54B) HI-LO TEMP	TIA-0914 TIA-0916	HIGH: 89° F LOW: 71° F	NONE
EK-1172	COMPONENT CLG SURGE TANK T-3 HI-LO LEVEL	LIA 0917, OR LIA 0920	HIGH: 93% LOW: 35%	CCW SURGE TANK T-3 MAKEUP CV-0918 IS OPENED BY LS-0918 AT 47% AND IS CLOSED AT 74%
EK-0706	LETDOWN Hx COOLING EXCESS FLOW	DPS-0909	14 PSID	NONE NOTE: DPM-0909 SHOULD HAVE OVERRIDDEN LOW PRESSURE LETDOWN TEMPERATURE CONTROLLER TIC-0203 AT 12 PSID.

TECHNICAL SPECIFICATIONS

3.7.7 Component Cooling Water (CCW) System
Two CCW trains shall be OPERABLE.
Applicability: Modes 1, 2, 3, and 4

SOURCE DOCUMENTS	
FSAR	9-3
P&ID	M-209 SH 3
ARP	7
ARP	4

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DRAWING TITLE
COMPONENT COOLING WATER

FILE NAME	DATE	REV NO.
SOD_CCW_02_R0	2.28.2002	0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010.K6.03	
	Importance Rating	3.2	

K/A Statement: Knowledge of the effect of a loss or malfunction of the following will have on the PZR: PZR sprays and heaters.

Proposed Question:

Given the following conditions:

- The Plant is operating at 92% power during a power ascension.
- Pressurizer Pressure Control is selected to Channel B.
- Pressurizer Level Control is selected to Channel B.
- All Backup heaters are ON.
- All Proportional heaters are energized.
- Preferred AC Bus EY-20 is lost.

Assuming NO Operator action, which ONE of the following states the response of the Proportional and Backup heaters?

	Proportional Heaters	Backup Heaters	Spray Valves
A.	De-energized	De-energized	Open
B.	De-energized	De-energized	Closed
C.	MINIMUM output	Energized	Open
D.	MINIMUM output	Energized	Closed

Proposed Answer: **B**

Explanation (Optional):

On a loss of AC Bus EY-20, with the Pressurizer (PZR) pressure control selector switch in the Channel B position, all pressurizer heaters are de-energized. Spray valves will also close. A loss of EY-20 will also impact the pressurizer level controller, LIC-0101BL (fail LOW on loss of power), resulting in a loss of ALL heaters. The effected Pressurizer pressure indicating controller (PIC) will fail to 0 output, in this case there will be NO PZR heaters energized (due to LIC loss of power) AND PZR Spray valves will be closed (due to PIC loss of power).

A. Incorrect, the applicant understands that pressurizer heaters will de-energize, but believes the spray valves will remain unaffected.
B. Correct, see explanation.

C. Incorrect, applicant believes the PZR pressure and level control channels are selected such that a loss of preferred AC bus EY-20 will not impact them.
D. Incorrect, the applicant understands the impact of the loss of power to the PIC (spray valves close), but does not understand the heaters de-energize due to a loss of power to the LIC.

Technical Reference(s): PL-PPCS Pressurizer Pressure Control Lesson Plan, AOP-13

(Attach if not previously provided, including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

2. Loss of any Preferred AC Bus

   a. Y10/Y20 provides power to Pressurizer Pressure Control circuitry, Channel A/B. Effected Pressurizer PIC will fail to 0 output.

EO 10a

REF ARP-4 windows 63/64

      1) Heater Control Selector switch 1/LIC-101 is normally in the A&B position, therefore a loss of Y-10/20 will also impact LIC-0101AL/ LIC-0101BL (fail LOW on loss of power), resulting in a loss of ALL heaters. The Effected Pressurizer PIC will fail to 0 output, in this case there will be NO PZR heaters energized (due to LIC loss of power) AND PZR Spray valves will be closed (due to PIC loss of power)

      2) Effect on the PCS -PCS pressure could rise due to loss of power impacting the PLCS (PZR Level Control System) resulting in maximum charging and minmum letdown. LPZR would rise compressing the bubble in the PZR.

      3) Without operator action pressure could rise and reach the high PCS pressure Reactor trip setpoint.

   b. Y30/Y40 provides power to Channel A/B LTOP circuitry. Effected LTOP circuitry is lost therefore losing overpressure protection from the effected channel.

      As long as the other LTOP channel is operable the PCS will be protected from overpressure with one PORV still able to operate.

STEP #7

Step Text

7. **MANUALLY OPERATE** Charging System to maintain Pressurizer level between 42% and 57%.

Technical Basis

Heater Select Switch, HS1/LIC-0101, is normally in the "A+B" position (per SOP-2A "Chemical and Volume Control System"), Letdown flow will be completely isolated (reference E-252, Sheets 1 and 2). All PZR Heaters will also be deenergized. Assuming no other transients or malfunctions are in progress when the loss of EY-20 occurs (with 'B' Channel PPCS and PLCS in service), Charging Pumps P-55B and P-55C will start and P-55A will go to maximum speed and Letdown will be isolated. This results in PZR level rising ~ 1.95 %/min. (133 gpm - 4 gpm) / 66 gal/% = 1.95 %/min. PZR Spray Valves, CV-1057 and CV-1059 close and all PZR Heaters are deenergized. PZR pressure will rise due to the PZR level rise. Assuming normal PZR level at 57%, about 3 minutes is available to restore Letdown prior to violating LCO 3.4.9 PZR level limit of 62.8% if Charging is not manually controlled. Manually controlling Charging (stopping P-55B and P-55C and slowing P-55A to minimum speed) slows the PZR level and pressure rise to ~ 0.44%/minute. (33 gpm - 4 gpm) / 66 gal/% = 0.44 %/min.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	012.K5.01	
	Importance Rating	3.3	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to the RPS: DNB

Proposed Question:

Which Reactor Protection System protective function provides Departure from Nucleate Boiling protection?

A. Low Primary Coolant System Flow Trip

B. Low Steam Generator Level Trip

C. Variable High Power Trip

D. High Pressurizer Pressure Trip

Proposed Answer: A

Explanation (Optional):

A. Correct, the low PCS flow trip provides DNB protection during events which suddenly reduce PCS flow rate during power operation.
B. Incorrect, the low S/G level trips provide protection against PCS overcooling (excessive steam demand event) and PCS overpressurization (loss of feedwater event)
C. Incorrect, the variable high power trip provides protection against positive reactivity excursions.
D. Incorrect, the high pressurizer pressure trip provides protection against PCS overpressure at operating temperature

Technical Reference(s): LCO 3.3.1 Bases
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __5__

55.43 _____

Comments:

Is there a learning objective that supports this as required knowledge for ROs?

BASES

BACKGROUND (continued)

Trip Channel Bypass

A Trip Channel Bypass is used when it is desired to physically remove an individual trip unit from the system, or when calibration or servicing of a trip channel could cause an inadvertent trip. A trip Channel Bypass may be manually initiated or removed at any time by actuation of a key-lock switch. A Trip Channel Bypass prevents the trip unit output from affecting the RPS logic matrix. A light above the bypass switch indicates that the trip channel has been bypassed. Each RPS trip unit has an associated trip channel bypass:

The key-lock trip channel bypass switch is located above each trip unit. The key cannot be removed when in the bypass position. Only one key for each trip parameter is provided, therefore the operator can bypass only one channel of a given parameter at a time. During the bypass condition, system logic changes from two-out-of-four to two-out-of-three channels required for trip.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference 4 takes credit for most RPS trip Functions. The High Startup Rate and Loss of Load Functions, which are not specifically credited in the accident analysis, are part of the NRC approved licensing basis for the plant, and are required to be operable in accordance with their respective LCO. The High Startup Rate and Loss of Load trips are purely equipment protective, and their use minimizes the potential for equipment damage.

The specific safety analyses applicable to each protective Function are identified below.

1. Variable High Power Trip (VHPT)

   The VHPT provides reactor core protection against positive reactivity excursions.

   The safety analysis assumes that this trip is OPERABLE to terminate excessive positive reactivity insertions during power operation and while shut down.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

2. High Startup Rate Trip

There are no safety analyses which take credit for functioning of the High Startup Rate Trip. The High Startup Rate trip is used to trip the reactor when excore wide range power indicates an excessive rate of change. The High Startup Rate trip minimizes transients for events such as a continuous control rod withdrawal or a boron dilution event from low power levels. The trip may be operationally bypassed when THERMAL POWER is < 1E-4% RTP, when poor counting statistics may lead to erroneous indication. It may also be operationally bypassed at > 13% RTP, where moderator temperature coefficient and fuel temperature coefficient make high rate of change of power unlikely.

There are only two wide range drawers, with each supplying contact input to auxiliary trip units in two RPS channels.

3. Low Primary Coolant System Flow Trip

The Low PCS Flow trip provides DNB protection during events which suddenly reduce the PCS flow rate during power operation, such as loss of power to, or seizure of, a primary coolant pump.

Flow in each of the four PCS loops is determined from pressure drop from inlet to outlet of the SGs. The total PCS flow is determined, for the RPS flow channels, by summing the loop pressure drops across the SGs and correlating this pressure sum with the sum of SG differential pressures which exist at 100% flow (four pump operation at full power T_{ave}). Full PCS flow is that flow which exists at RTP, at full power T_{ave}, with four pumps operating.

4, 5. Low Steam Generator Level Trip

The Low Steam Generator Level trips are provided to trip the reactor in the event of excessive steam demand (to prevent overcooling the PCS) and loss of feedwater events (to prevent overpressurization of the PCS).

The Allowable Value assures that there will be sufficient water inventory in the SG at the time of trip to allow a safe and orderly plant shutdown and to prevent SG dryout assuming minimum AFW capacity.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

4, 5. Low Steam Generator Level Trip (continued)

Each SG level is sensed by measuring the differential pressure in the upper portion of the downcomer annulus in the SG. These trips share four level sensing channels on each SG with the AFW actuation signal.

6, 7. Low Steam Generator Pressure Trip

The Low Steam Generator Pressure trip provides protection against an excessive rate of heat extraction from the steam generators, which would result in a rapid uncontrolled cooldown of the PCS. This trip provides a mitigation function in the event of an MSLB.

The Low SG Pressure channels are shared with the Low SG Pressure signals which isolate the steam and feedwater lines.

8. High Pressurizer Pressure Trip

The High Pressurizer Pressure trip, in conjunction with pressurizer safety valves and Main Steam Safety Valves (MSSVs), provides protection against overpressure conditions in the PCS when at operating temperature. The safety analyses assume the High Pressurizer Pressure trip is OPERABLE during accidents and transients which suddenly reduce PCS cooling (e.g., Loss of Load, Main Steam Isolation Valve (MSIV) closure, etc.) or which suddenly increase reactor power (e.g., rod ejection accident).

The High Pressurizer Pressure trip shares four safety grade instrument channels with the TM/LP trip, Anticipated Transient Without Scram (ATWS) and PORV circuits, and the Pressurizer Low Pressure Safety Injection Signal.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	013.G2.4.50	
	Importance Rating	4.2	_____

K/A Statement: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question:

Given the following conditions:

- The reactor has just tripped from full power
- 'A' S/G level is 20% and lowering
- 'B' S/G level is 22% and lowering
- The following annunciator has just LIT:
  - EK-16 'ARRAY A' Annunciator 3-4, "P-8A Tripped"

Under these conditions, the Auxiliary Feedwater Actuation Signal (AFAS) will first start the __(1)__ pump to deliver a minimum of 100 gpm to __(2)__ S/G(s) to prevent further system actuation(s).

A. (1) P-8B, Steam Driven Auxiliary Feedwater Pump
(2) at least ONE

B. (1) P-8B, Steam Driven Auxiliary Feedwater Pump
(2) BOTH

C. (1) P-8C, Auxiliary Feedwater Pump
(2) at least ONE

D. (1) P-8C, Auxiliary Feedwater Pump
(2) BOTH

Proposed Answer: C

Explanation (Optional):

EK-16 'ARRAY A' Annunciator 3-4 will alarm if the P-8A breaker opens after receiving a valid auto start signal. The AFAS (Aux Feedwater Actuation Signal) will auto-start the P-8A pump first (in this case, the pump is tripped and will not start). Therefore, if P-8A were to start and establishes flow (100 gpm minimum) to one S/G before Pump P-8C timer times out, the starting of Pump P-8C is blocked. Low flow (less than 100 gpm) to both S/Gs from Pump P-8A will not trip Pump P-8A, but will remove the block for starting of Pump P-8C and then Pump P-8B if low flow exists in Pump P-8C. The pumps continue to operate if low flow is detected but the pumps will not be tripped. If Pump P-8A fails to operate for any reason, the AFAS system will initiate a

start signal for P-8C and then P-8B, should Pump P-8C trip. Therefore, since P-8A is tripped with no flow being provided to either S/G, pump P-8C will start and establish flow to at least one S/G.

A. Incorrect, Pump P-8C will start prior to Pump P-8B.
B. Incorrect, Pump P-8C will start prior to Pump P-8B. Flow > 100 gpm to at least ONE S/G will prevent a low flow condition.
C. Correct.
D. Incorrect, Flow > 100 gpm to at least ONE S/G will prevent a low flow condition.

Technical Reference(s): ARP-36,DBD-1.03
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________________ (As available)

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

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

Pump Start Sequence Logic

During normal plant operations, the AFW system is on standby in the automatic mode. When the level of water in either Steam Generator A or B drops below a predetermined level, the AFAS signal will start Pump P-8A as explained in the previous section. The automatic start sequence for the system has time delay relays which have been set for the appropriate times established by SC-87-156. If Pump P-8A starts and establishes flow to one steam generator before Pump P-8C timer times out, the starting of Pump P-8C is blocked. Low flow to both steam generators from Pump P-8A will not trip Pump P-8A but will remove the block for starting of Pump P-8C and then Pump P-8B if low flow exists in Pump P-8C. The pumps continue to operate if low flow is detected but the pumps will not be tripped. If Pump P-8A fails to operate for any reason such as a breaker trip or the pump trips because of bus undervoltage, or low suction pressure, an alarm (C-O1) is provided to indicate that the pump has been tripped. When the pump has been tripped the system will initiate a start signal for Pump P-8C and then Pump P-8B should Pump P-8C trip. When Pump P-8A is tripped, valves CV-0727 and CV-0749 will be closed to eliminate suction pressure problems on pump start. Electrical interlocks to the controllers for CV-0727 and CV-0749 are provided using a set of relay contacts. The logic used for Pump P-8C is the same as Pump P-8A (E-196, Sh 1-14). Pump P-8B is started if Pump P-8C fails to provide flow or is tripped due to low suction pressure or bus undervoltage. The valve alignment and controls are shown on E-238, Sh 13. The auxiliary feedwater controls become class 1E and automatic in response to NUREG 0578/0737.

AFW Flow Control

Four air operated flow control valves (CV-0727, CV-0749, CV-0736A, and CV-0737A) are located in the separate discharge lines to each steam generator. Two additional valves (CV-0736, CV-0737) were added in parallel to CV-0736A and CV-0737A in the discharge line from Pump P-8C. The flow indicating controllers are used to regulate flow of water to the steam generator requiring makeup inventory (FIC-0727, FIC-0749, FIC-0736A and FIC-0737A). The flow indicating controllers are set at 165 gpm (SC-87-156) flow during normal operation so that when a AFAS is actuated, the operator must take action to dial in a reduced flow to match boiloff once normal level has been restored. The AFW flow control valve logic is as shown on E-17, Sh 22. Additional flow control is provided in panel C-33R and C-33L. Hand indicating controllers (HIC-0727, HIC-0736A, HIC-0737A, and HIC-0749) provide control of water to both A and B steam generators. HIC-0727C and HIC-0749C located in Panel EC-150 provide additional control for Pump P-8B. Indication of AFW isolation to each steam generator is provided at EC-33R and EC-33L via panel lights L-E50A and L-E50B.

TITLE: AUXILIARY FEEDWATER SYSTEM STATUS ARRAYS SCHEME EK-16 (C-11)

1-1	1-2	1-3	1-4	1-5	1-6	1-7	1-8	1-9	1-10
2-1	2-2	2-3	2-4	2-5	2-6	2-7	2-8	2-9	2-10
3-1	3-2	3-3	3-4	3-5	3-6	3-7	3-8	3-9	3-10

ARRAY "A"

P-8A TRIPPED

Sensor:	3X-5/P8A
Trip Setpoints:	Breaker 152-104 OPEN after receiving an Auto Start signal (AFAS or Test), or after control switch taken to CLOSE.

AUTOMATIC FUNCTION:

- If AFAS signal is present and less than 100 gpm flow is being delivered to either Steam Generator, then P-8C, Auxiliary Feedwater Pump will start. If P-8C fails to develop 100 gpm flow to at least one Steam Generator then P-8B, Steam Driven Auxiliary Feedwater Pump will start.

OPERATOR ACTION:

- IF AFAS signal is present, THEN **VERIFY STARTED** P-8C or P-8B.
- IF neither Auxiliary Feedwater Pump started or additional Auxiliary Feedwater is needed, THEN **REFER TO** EOP Supplement 19.

FOLLOWUP ACTION:

- IF P-8A tripped on low suction pressure, THEN **REFER TO** SOP-12 to reset low suction pressure trip.
- **INITIATE** Work Request for troubleshooting/repair as required.
- **REFER TO** Technical Specifications LCO 3.7.5.

REFERENCES:

- Technical Specifications LCO 3.7.5
- EOP Supplement 19, "Alternate Auxiliary Feedwater Methods"
- SOP-12, "Feedwater System"

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	013.A3.02	
	Importance Rating	4.1	_____

K/A Statement: Ability to monitor automatic operation of the ESFAS including: Operation of actuated equipment.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power.
- Train B Control Room HVAC is in operation.
- A spurious Containment High Radiation (CHR) signal was just received.
- No further operator actions have been taken.

Which of the following equipment actuations and/or indications is expected given the CHR signal?

A. ES Room Sump Pump West P-73B red light is LIT.

B. Control Room Condensing Unit VC-11 red light is LIT.

C. Control Room Air Filter Unit Fan V-26B red light is LIT.

D. PCP Controlled Bleedoff Valve CV-2083 red light is LIT.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, the ESS Room sump pumps auto-start feature is disabled in order to prevent highly radioactive waste from ESS Rooms being transferred to the Dirty Waste Drain Tank during an accident. The pump will be off (green light lit).
B. Incorrect, the control room condensing unit trips (green/off light will be lit) on the CHR signal and are manually restarted.
C. Correct, with the B Train Control Room HVAC in normal operation, the V-26B fan is lined up in AUTO. With the control switch in auto, the fan will start on a CHR signal (red/run light will be lit).
D. Incorrect, PCP CBO valve CV-2083 will isolate (green/closed light will be lit).

Technical Reference(s): PL-CTMT Containment Building Lesson Plan, FSAR Section 9.8.10.b Rev 31, E-17 Sheet 7, AOP-31

(Attach if not previously provided, including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ____None_________

Learning Objective: _________________________ (As available)

Question Source: Bank # _______

Modified Bank # _______ (Note changes or attach parent)

New __X___

Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__

55.43 _____

Comments:

1) *Equipment Actuation due to CHR Right & Left Channel*

   a) All automatic containment isolation valves close except:

      1. CCW Containment Supply and Returns CV-0910, CV-0911, CV-0940
      2. Main Steam Isolations CV-0501, CV-0510
      3. Main Feedwater Isolations CV-0701, CV-0703, CV-0734, CV-0735

3) *Actuation of CHR also causes the AUTO start feature of the ES Sump Pumps to be <u>disabled</u>.*

   a) The auto feature is disabled in order to prevent highly radioactive waste from ESS Rooms being transferred to the Dirty Waste Drain Tank and possibly beyond during an accident.

   b) Any system leakage will be retained in the sump.

      1. Pumps can still be operated in manual.

   c) Right Channel – ESS Pumps P-73A, P-72A

      1. Left Channel – ESS Pumps P-73B, P-72B

   d) *Right or Left Channel also results in tripping Air Room Purge Fan V-46.*

   e) *CHR also places the Control Room HVAC in EMERGENCY MODE.*

      1. Right Channel - B Train
      2. Left Channel - A Train

   f) CHR Reset Pushbuttons

      1. Left channel RESET pushbutton ONLY resets Left Channel.
      2. Right channel RESET pushbutton ONLY resets Right Channel.
      3. When CHR is RESET, Containment Isolation Valves will NOT re-open. Valve Hand Switches must be placed to CLOSE, they may then be opened.

b. Emergency Mode

The emergency mode of operation is actuated either by a containment high-radiation or a containment high-pressure signal (Section 7.3), or manually from the control room. During emergency mode operation, the air handling units and the charcoal filter units of both Train A and Train B operate. The refrigerant Condensing Units VC-10 and VC-11 shut down and are manually started by the operator. The control room operator has the option to turn off either Train A or Train B. During an emergency signal, operation of Purge Fan V-94 and Isolation Dampers D-15 and D-16 is blocked. The toilet exhaust fan in the viewing gallery is shut off, and Fan Isolation Dampers D-17 and D-18 close. A manual switch to override each outside air duct damper (D-7 and D-14) is provided to isolate the control room from outside air and to allow 100% air recirculation. Humidifiers VH-12 and VH-13 are shutdown to isolate domestic water vapor from affecting HEPA filtration as well as charcoal filter absorbtion. VHX-26A and VHX-26B, electric heaters upstream of the emergency HEPA and Charcoal filters, are placed in service to reduce relative humidity of incoming air.

SPURIOUS CONTAINMENT ISOLATION

CONTAINMENT ISOLATION FROM CONTAINMENT HIGH RADIATON

✓	VALVE	SWITCH	DESCRIPTION	PANEL
	CV-1065	HS-1065	Clean Waste Receiver Tank Cover Gas Vent	EC-13
	CV-0770	HS-0770	S/G E-50B Bottom Blowdown Valve	EC-13
	CV-0771	HS-0771	S/G E-50A Bottom Blowdown Valve	EC-13
	CV-0767	HS-0767	Steam Gen E-50A Bottom Blowdown Isol Valve	EC-13
	CV-0768	HS-0768	Steam Gen E-50B Bottom Blowdown Isol Valve	EC-13
	CV-0738	HS-0738	Steam Gen E-50B Top Blowdown Isol Valve	EC-13
	CV-0739	HS-0739	Steam Gen E-50A Top Blowdown Isol Valve	EC-13
	CV-0939	HS-0939	Shield Cooling Surge Tank Inlet Valve	EC-13
	CV-1004	HS-1004	Discharge Line From Degasifier Pumps	EC-13
	CV-1037	HS-1037	P-70 Discharge CV-1037	EC-13
	CV-1358	HS-1358	Nitrogen Isolation Valve	EC-13
	CV-1001	HS-1001	Primary System Drain Tank T-74 Isolation	EC-13
	CV-1910	HS-1910	Pri Sys Sample Flow Isolation Valve Switch	EC-13
	CV-1911	HS-1911	Primary System Sample Flow Isolation	EC-13
	CV-3001	HS-3001A	Containment Spray Flow Control Valve	EC-03
	CV-3002	HS-3002A	Containment Spray Flow Control Valve	EC-03
	CV-2009	HS-2009	Low Press Letdown Containment Isol Switch	EC-02
	CV-2099	HS-2099	Low Press Letdown Containment Isol Switch	EC-02
	CV-2083	HS-2083	Pri Cool Pump Controlled Bleedoff Switch	EC-02
	CV-0155	HS-0155	Demin Water to Quench Tank T-73 Switch	EC-02

8. **PLACE** the following handswitches to OPEN to restore Controlled Bleedoff:

- HS-2083, Primary Coolant Pumps Controlled Bleedoff
- HS-2099, Primary Coolant Pumps Controlled Bleedoff

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022.G2.1.27	
	Importance Rating	3.9	

K/A Statement: Knowledge of system purpose and/or function.

Proposed Question:

Per the Design Bases Documents DBD-2.03, "Containment Spray System," and DBD-2.08, "Containment Air Coolers," which of the following design functions are NOT shared by both the Containment Air Cooling system and the Containment Spray system?

A. Act as a barrier to limit radioactive releases from containment.

B. Provide post-accident cooling capability to limit containment pressure to within containment structure design value of 55 psig.

C. Provide post-accident cooling capability to achieve within 24 hours a containment pressure which is 50% or less the design pressure.

D. Remove energy from the atmosphere within the Containment Building during normal operation to keep the air temperature below 140°F.

Proposed Answer: D

Explanation (Optional):
The CAC and CS systems share multiple design functions: to act as a barrier to limit radiological releases, to limit containment pressure below the design value during an accident, and to provide sufficient cooling to lower containment pressure to 50% the design value within 24 hours post-accident. The CS system does not have a normal operating condition function, as the CAC system does. The CAC system is designed to remove heat from containment to maintain containment temperature during normal operations to less than 140°F.

A. Incorrect, see explanation.
B. Incorrect, see explanation.
C. Incorrect, see explanation.
D. Correct, see explanation.

Technical Reference(s): DBD-2.03, DBD-2.08
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History:	Last NRC Exam	_____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

NSPC	Nuclear Safety Performance Criteria
PPAC	Periodic and Predetermined Activity Control
SEP	Systematic Evaluation Program
SIS	Safety Injection Signal
SSA	Safe Shutdown Analysis (See NSCA)
SWS	Service Water System
WG	Water Gauge

2.0 LICENSING BASIS

2.1 SYSTEM REQUIREMENTS

2.1.1 System Functional Requirements

The following table provides the Functional Requirements for the CACs and the engineering design basis calculations which support them. The calculations are referenced by number to those calculations listed in Appendix D, Table D-1, "Engineering Design Basis Calculations."

TABLE 2.1-1
FUNCTIONAL REQUIREMENTS

Description of Requirement	Table D-1 Calculation No
1. The CACs shall remove energy from the atmosphere within the Containment Building during normal operation to keep the air temperature below 140°F.	4
2. The CACs shall provide post-accident cooling capability to limit containment pressure to within containment structure design value of 55 psig.	1, 2, 3
3. The CACs shall provide post-accident cooling capability to achieve within 24 hours a containment pressure which is 50% or less of the design pressure.	1, 2, 3
4. The CACs shall act as a barrier to limit radioactive releases from containment.	(a)

(a) Verification that the CACs act as a barrier to radioactive releases is demonstrated by compliance with the structural analysis codes and standards given in Sections 3.4.2 and 3.4.3 of this document.

Final Safety Analysis Report (FSAR) Chapter 14 events which address the operation of the CACs are discussed in Section 4.0 of this document.

2.0 LICENSING BASIS

2.1 SYSTEM REQUIREMENTS

2.1.1 System Functional Requirements

The following table provides the functional requirements for the Containment Spray System (CSS) and the engineering design basis calculations which support them. The calculations are referenced by number to the calculations listed in Appendix D, Table D-1, "Engineering Design Basis Calculations." These are not licensing basis requirements, but are functional provisions of the system. Final Safety Analysis Report Chapter 14 transient analyses which address CSS operation are discussed in Section 4 of this document.

	Description of Requirement	Calculation No
1.	The CSS shall provide post-accident cooling capability to limit containment pressure to within containment structure design value of 55 psig.	1, 2, 3, 5, 6, 12, 13, 14
2.	The CSS shall provide post-accident cooling capability to achieve within 24 hours a containment pressure which is 50% or less of the design pressure.	1, 2, 3, 5, 6, 12, 13, 14
3.	The CSS shall act as a barrier to limit radioactive releases from containment.	(a), 8, 9, 10, 11
4.	The CSS shall boost HPSI discharge pressure to enable safe shutdown following a fire in the charging pump area.	4
5.	The CSS shall boost HPSI suction pressure during the recirculation mode following a Loss of Coolant Accident.	12

(a) Verification that the CSS acts as a barrier to radioactive releases is demonstrated by compliance with the structural analysis codes and standards given in Sections 3.4.2 and 3.4.3 of this document.

Final Safety Analysis Report (FSAR) Chapter 14 events which address CSS operation are discussed in Section 4 of this document.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	026.K2.01	
Importance Rating	3.4	

K/A Statement: Knowledge of bus power supplies to the following: Containment spray pumps.

Proposed Question:

Complete the following statements:

Power to Containment Spray pump P-54C is normally supplied from __(1)__ and the pump will start on the DBA sequencer, if necessary, __(2)__ after the Diesel Generator output breaker closes.

A. (1) Bus 1C
(2) 2 seconds

B. (1) Bus 1C
(2) 19 seconds

C. (1) Bus 1D
(2) 2 seconds

D. (1) Bus 1D
(2) 19 seconds

Proposed Answer: **B**

Explanation (Optional):

P-54A is energized from Bus 1D (DG 1-2 on a LOOP). P-54B and P-54C are energized from Bus 1C (DG 1-1 on a LOOP). P-54A and P-54B start 2 seconds after DG output breaker closure. P-54C will not start until 19 seconds after breaker closure. A nominal 15 second auto-start time delay was added to P-54C control circuit to mitigate potential DG overload concerns.

A. Incorrect, see explanation.
B. Correct, see explanation.
C. Incorrect, see explanation.
D. Incorrect, see explanation.

Technical Reference(s): DBD-2.03, DBD-5.05
(Attach if not previously provided, ______
including version/revision number) ______
Proposed references to be provided to applicants during examination: None

Learning Objective:	____________________________ (As available)		
Question Source:	Bank #	_______	
	Modified Bank #	_______	(Note changes or attach parent)
	New	__X___	
Question History:	Last NRC Exam	____________	

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

3.2.7 Failure Modes and Effects Analysis

The CSS design is channelized so that a failure in one train will not affect the operation of the other train. All the components for CSS pumps P-54B and P-54C are supplied and controlled from left channel sources which are independent from right channel sources. The same is true for right channel pump P-54A which is supplied and controlled from right channel power sources which are independent from left-channel sources.

The CSS was designed with redundant active components, but not redundant passive components. In the interest of completeness, the System Failure Analysis (Reference 72) included failure of check valves which were passive components in 1971 but are considered active components by current standards. The analysis identified the following single failure concerns.

1. Should Check Valve CK-ES3208 on the discharge line of P-54C fail to close, backflow through the pump would occur for a limited time upon a DBA. The DBA sequencer starts P-54A and P-54B, two seconds after it is energized (pumps will not start until CHP is present). P-54C is started 19 seconds after sequencer energization. During the time P-54C is idle, backflow through the pump would reduce spray flow to the containment atmosphere potentially to a level below that assumed in the DBA analyses. Failure of CK-ES3208 as evaluated under CCP Discrepancy Report F-CG-92-119 (Reference 225) was determined to be bounded by the current FSAR Chapter 14 DBA Analyses.

Non-Safety Related Power Source

Lighting Panel L-25 (Motor Space Heaters)

The following list shows the CSS pump, breaker and panel relationships:

TABLE 3-5
CSS Pump/Breaker/Panel Relationships

		2400V AC		Panel		
CSS Pump	Motor	Bus	Breaker	Swgr	Contl Rm	Sfgd
P-54A	EMA-1210	1D	152-210	EA-12	EC-03R	EC-33R
P-54B	EMA-1112	1C	152-112	EA-11	EC-03L	EC-33L
P-54C	EMA-1114	1C	152-114	EA-11	EC-03L	EC-33L

TABLE 3.1-6
EMERGENCY DIESEL GENERATOR 1-1
DBA SEQUENCE

Load	DBA[1] Sequence-Seconds	Margins[1] (Sec)
Misc 480V[2]	On Bkr Closure	-
MOV-3009}		14.7
MOV-3011}		14.7
MOV-3010}		13.5
MOV-2087}		N/A
MOV-3007}	0(+0.3,-0)	14.7
MOV-3013}		14.7
MOV-3008}		13.5
MOV-2169}		N/A
MOV-2170}		N/A
BA P-56B	2(±0.3)	N/A
CHG P-55C	2(±0.3)	N/A
CCF V-4A	2(±0.3)	28.7
CS P-54B	2(±0.3)	3.0
HPSI P-66B	6(±0.3)	19.5
SWS P-7B	10(±0.3)	21.3
LPSI P-67B	13(±0.3)	14.3
CS P-54C	19(±0.3)[5]	3.5
CCW P-52A	23(±0.3)	24.5
CCW P-52C	40(±0.3)	7.5
AFW P-8A	45(±0.3)[4]	33.7
AHU V-95	55(±0.3)[3]	21.2

(1) See References 26, 29, 50, 154, 155 and 194.
(2) Not sequenced. Energizes when DG breaker closes, since load centers remain connected to the bus.
(3) See DBD-1.06 (Reference 52), as well as References 40 and 63.
(4) Additional 2 to 5 second time delay in starting incorporated per SC-87-156 (Reference 23). See also Reference 63.
(5) Nominal additional 15 second auto-start time delay added to P-54C control circuit per SC-92-099 (Reference 119).

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	039.K1.02	
Importance Rating	3.3	

K/A Statement: Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: Atmospheric relief dump valves.

Proposed Question:

Given the following conditions:

- Reactor power is 80%
- T_{ave} is 554°F

Which one of the following statements describes the expected response of the Atmospheric Steam Dump Valves (ADVs) immediately following a reactor trip from the above initial conditions?

A. ADVs initially modulate open, then modulate closed and are full closed when T_{AVE} is 535°F.

B. ADVs initially modulate open, then modulate closed and are full closed when T_{AVE} is 540°F.

C. ADVs initially quick open, then modulate closed and are full closed when T_{AVE} is 535°F.

D. ADVs initially quick open, then modulate closed and are full closed when T_{AVE} is 540°F.

Proposed Answer: A

Explanation (Optional):

The ADVs and TBV modulate on temperature and pressure. The temperature input is from T_{ave}-T_{ref} calculators (TYT-0100 and TYT-0200) providing inputs to the steam dump controller, HIC-0780A. The ADVs and TBV will modulate from full open at 25°F error (T_{ave} minus no load T_{ave}) to full closed at 3°F error. In this scenario, the error signal seen is 554°F-532°F (no load T_{ave}) = 22°F error. At an error of > 25°F, the ADVs and TBV will quick open, upon actuation of the SDCR (steam dump control relay). The 3°F error is used on a decreasing T_{ave} signal. The TBV will cycle with the ADVs, as discussed, but will also cycle to maintain steam header pressure at 900 psia (saturation pressure at no load T_{ave} of 532°F), with a +/- 5 psia band. Therefore, the TBV will be full closed at 895 psia and full open at 905 psia. The TBV does not receive a quick open signal from the TBV pressure controller. PIC-0511.

A. Correct, see explanation.
B. Incorrect, the ADVs and TBV modulate open on increasing Tave at an 8°F error signal

and are full open at 25°F error. However, in this case, a decreasing Tave signal is observed, from 554°F to no load Tave (532°F)

C. Incorrect, the ADVs and TBV receive modulate open signals. See explanation.

D. Incorrect, the ADVs and TBV modulate open on increasing Tave at an 8°F error signal and are full open at 25°F error. However, in this case, a decreasing Tave signal is observed, from 554°F to no load Tave (532°F). See explanation.

Technical Reference(s): DBD-1.09, PL-MSS Main Steam System
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

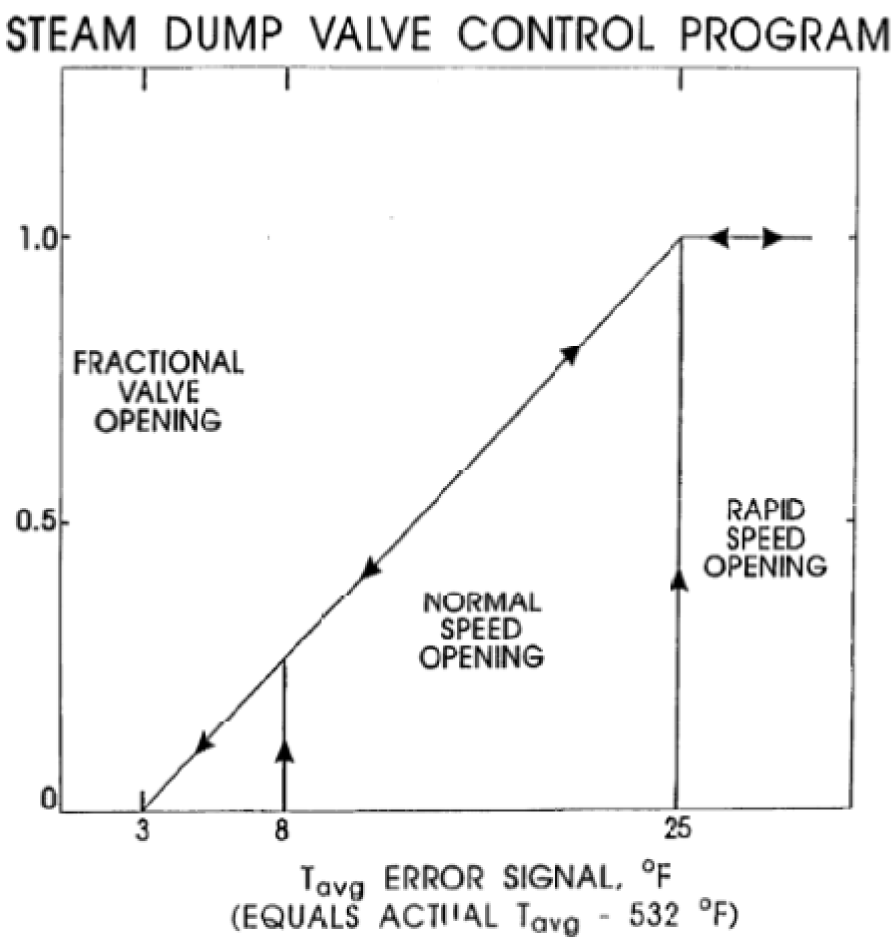
Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

FIGURE 3.2-2


STEAM DUMP VALVE CONTROL PROGRAM
1.0
0.5
0
FRACTIONAL VALVE OPENING
NORMAL SPEED OPENING
RAPID SPEED OPENING
3
8
25
Tavg ERROR SIGNAL, °F
(EQUALS ACTUAL Tavg - 532 °F)

3.2.2.2 Plant Emergency Operations

Steam Dump and Bypass System

The steam dump and bypass system is composed of five control valves and associated instrumentation and controls. The functions of the steam dump and bypass system are to provide a path for removing the stored energy and decay heat within the Primary Coolant System during cool down and to preclude actuation of the primary and secondary safety valves in the event of reactor trip with a subsequent turbine trip.

Steam is discharged from the main steam lines to the atmosphere via four steam dump valves and to the condenser via the turbine bypass valve.

The four steam dump valves are each sized to pass 841,875 lbm/hr for a total of 3,367,500 lbm/hr steam flow (Reference 77). The turbine bypass valve is sized to pass 528,000 lbm/hr of the flow (Reference 77). The valves are designed to open in 10 seconds when the T_{avg} error signal (T_{ave} minus 532°F) is less than 25°F. The valves will open in 5 seconds when the T_{ave} error signal exceeds 25°F. Reference 7 contains background information on these features.

The steam dump and turbine bypass valves have no safety related function. The steam dump valves are prevented from opening in automatic unless a turbine trip has occurred. The valves regulate the flow in response to the T_{avg} error signal (T_{avg} minus the zero load reference temperature, 532°F). See Figure 3.2-2 for the steam dump valve control program. An error signal of 8°F (minimum) is required to start the valves open to prevent accidental opening, hunting and short cycling and an error signal of 3°F is required to assure complete closure when approaching hot standby conditions. Rapid opening is used at higher T_{ave} because more energy has to be dissipated. These settings were established by Combustion Engineering (Reference 7).

The turbine bypass valve receives the higher of the dump control temperature error signal or a secondary pressure signal, whichever results in the maximum bypass valve opening. Following turbine trip, the bypass valve also opens rapidly to shave the peak pressure, and then modulates in response to the primary coolant temperature error signal to avoid over-cooling, see Figure 3.2-3.

For transients in which an associated major secondary system disturbance occurs that would cause the MSIV's to close, the turbine bypass valve will not be available as a steam release path without operator action to manually reopen the MSIV's or MSIV bypass valves.

The turbine bypass valve is positioned to maintain secondary pressure at 900 ± 5 psia and maintain the primary coolant temperature near the 532°F zero load value as hot standby condition is approached. The bypass valve is prevented from opening on loss of condenser vacuum to prevent damage to the condenser.

The operator may control primary coolant temperature during plant cool down with turbine bypass valve by manually changing the set point of the turbine bypass pressure controller. When in the automatic mode, once the turbine trips and the steam dump permissive switch contacts close, the operator may manually control the steam dump valves as well as the turbine bypass valve.

The emergency procedure (EOP Supplement 23) provide a means to remove PCS heat with the hogging air ejector, but this option may be insufficient to remove decay heat unless the turbine driven auxiliary feedwater pump is in service to provide additional steam load (Reference 36).

3. Quick Open Mode

   a. When a turbine trip causes the 386 AST relay to energize, a quick open signal is generated.

   b. If T_{AVE} is ≥ 556.9°F, the steam dump control relay (SDCR) is energized and closes contacts to align the quick open air supply solenoids to the ADV valve actuators and the TBV to open the valves fully.

   c. The ADVs and TBV will stay full open until T_{AVE} is less than 556.9°F.

   d. When T_{AVE} lowers to less than 556.9°F the SDCR will de-energize and remove the quick open function.

   e. The modulating mode will then control the ADVs.

4. Modulate Mode

   a. When a turbine trip causes the 386X1 AST relay to energize, a contact is closed to arm Steam Dump Controller HIC-0780A.

   b. Steam Dump Controller HIC-0780A will modulate the ADVs and TBV based on a T_{AVE} – T_{REF} Error Signal.

c. The error signal is developed by comparing actual T_{AVE} to 532°F (Reference No-load Value).

d. The control system will modulate the ADVs and TBV from full open when T_{AVE} is at 556.9°F (25°F error) to full closed at 535°F (3°F error).

e. For increasing T_{AVE} , the control system will modulate the ADVs and TBV from full closed when T_{AVE} is at 540°F (8°F error) to full open at 556.9°F (25°F error).

5. PIC-0511 controls CV-0511 to maintain the steam pressure setpoint.

a. Normally 900 psia (T_{AVE} at 532°F)

b. At 5 psi greater than the setpoint (905 psia, 532.6°F), CV-0511 will be full open.

c. CV-0511 will be fully closed at 5 psi less than the setpoint (895 psia, 531.3°F) on PIC-0511.

d. The TBV scale is 800 psia to 1000 psia.

Since S/G pressure is approximately 770 psia at full power, I&C has set the out of range alarm function (Yellow alarm light in solid) below expected values for full power operation to prevent the yellow light from being illuminated all the time at full power conditions.

e. The TBV pressure control function <u>DOES NOT</u> require a turbine trip (e.g. does not require 386AST relay actuation).

6. PM-0511 auctioneers the signals from PIC-0511 and HIC-0780A, taking the larger of the two signals.

a. Therefore, in addition to the pressure control input, CV-0511 receives a modulate open signal from steam dump controller HIC-0780A through PM-0511.

Example: If the Atmospheric Dump Valves are being operated in manual using HIC-0780A, a signal will also be sent to PM-0711. If this signal is greater than the signal from PT-0510, the TBV will open.

b. This is the same as the signal received by the ADVs from the TAVE Computer. When the TBV opens as a result of input from HIC-0780A, the output meter on PIC-0511 will show “zero” output. Note also that the TBV can open from HIC-0780A when PIC-0511 is in the manual mode.

c. The 386X1 AST turbine trip relay must be energized to receive this signal.

d. Modulate signal is removed when Tave is lowered to 535°F (+3°F) and will be provided when Tave is greater than 540°F (+8°F). The TBV should already be full open due to the pressure signal if Tave is at 540°F (962.8 psia).

e. CV-0511 also receives the same 'quick opening' signal as the ADVs.

1) Tave at 556.9°F and turbine trip via the 386 AST turbine trip relay

2) Opens SV-0589B and closes SV-0589C to align the ‘quick open’ air supply and close the modulate air supply.

f. CV-0511 is interlocked to prevent opening and will close if there is less than 5 inches of vacuum in the main condenser.

1) SV-0509A is closed to stop the air supply and SV-0509B is opened to vent the air off CV-0511 actuator

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059.A4.12	
	Importance Rating	3.4	

K/A Statement: Ability to manually operate and monitor in the control room: Initiation of automatic feedwater isolation.

Proposed Question:

Given the following conditions:

- An Excess Steam Demand Event occurred, resulting in a reactor trip.
- The turbine had to be manually tripped.
- Containment pressure is 3.8 psig and rising.
- ‘A’ S/G pressure is 495 psia and lowering.
- ‘B’ S/G pressure is 595 psia and rising.

Assuming NO operator actions, what is the status of each S/G’s Feed Reg Valve and Feed Reg Bypass valve?

	‘A’ S/G	‘B’ S/G
A.	open	open
B.	open	closed
C.	closed	open
D.	closed	closed

Proposed Answer: **C**

Explanation (Optional):

Feedwater is isolated from either a CHP signal or low S/G pressure. A CHP signal will close FRV and FRV bypass valves for both S/Gs. The setpoint for containment isolation on containment high pressure is 4 psig, so CHP will not cause the FW isolation in this case. Feedwater will isolate to the ‘A’ S/G as pressure is less than 512 psia. Feedwater will be unaffected to the ‘B’ S/G as pressure is > 512 psia.

A. Incorrect, ‘A’ S/G FW will isolate on low S/G pressure. ‘B’ S/G FW will remain unaffected.
B. Incorrect, ‘A’ S/G FW will isolate on low S/G pressure. ‘B’ S/G FW will remain unaffected.
C. Correct, see explanation.

D. Incorrect, ‘B’ S/G FW will remain unaffected.

Technical Reference(s): FSAR 7.5.1.3, ARP-8, ARP-21, PL-MSS Main Steam System Lesson Plan

(Attach if not previously provided, including version/revision number) ____________

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

Manual control of feedwater flow may be assumed by the operator at any time. In the event of a reactor or turbine trip and if the feedwater pump turbine drivers are in the automatic control mode, the speeds of the feed pumps are automatically ramped down to a lower value. Following or during the ramp-down, operators can manually control main feedwater flow or initiate auxiliary feedwater flow as necessary to restore and maintain desired steam generator levels.

In the event of low steam generator pressure < 500 psia or containment high pressure (CHP), the main feedwater regulating and regulating bypass valves are closed to prevent excessive flow into the steam generators. This ensures containment pressure is not exceeded during a main steam line break inside containment (refer to Subsections 7.2.3.8 and 7.3.3.3). The valve closing on CHP was added by FC-906 in 1990 when analysis disclosed that for a small steam line break, low steam generator pressure would not occur fast enough to prevent exceeding containment design pressure. Control of the bypass of the steam generator pressure signal to close the main steam isolation valves, the main feedwater regulating and regulating bypass valves is facilitated by using push buttons on the panel to override the signal to allow manual take-over of the controls. In addition to the push buttons, manual take-over of the feedwater regulating bypass valves can be accomplished by using key-operated switches.

TITLE: REACTOR PROTECTIVE SYSTEM SCHEME EK-06 (C-06)

RACK "B"			
1	2	3	4
5	6	7	8

LO PRESSURE SG1 CHANNEL TRIP	
Sensor:	Bistable Trip Unit (Any 1 of 4)
Trip Setpoints:	512 psia
Alternate Indication:	Steam Generator E-50A Pressure Indicators

AUTOMATIC FUNCTION:

- Reactor Trip on 2 of 4 coincidence logic.
- Both MSIVs (CV-0501 and CV-0510) close at 512 psia on 2 of 4 coincidence logic in the MSIV closing circuitry.
- 'A' S/G Main Feedwater Regulation and Bypass Valves (CV-0701 and CV-0735) close at 512 psia on 2 of 4 coincidence logic in the MSIV closing circuitry.

OPERATOR ACTION:

- **CHECK** Steam Generator E-50A pressure indicators:
    - PIC-0751A
    - PIC-0751B
    - PIC-0751C
    - PIC-0751D

FOLLOW UP ACTION:

- IF faulty instrument, THEN:
    - **BYPASS** Low Pressure SG1 Trip unit for affected channel per SOP-36.
    - **REFER TO** Technical Specifications LCO 3.3.1, LCO 3.3.3, LCO 3.3.4, LCO 3.3.7.

REFERENCES:

- Technical Specifications LCO 3.3.1, LCO 3.3.3, LCO 3.3.4, LCO 3.3.7
- SOP-36, "Reactor Protective System and Anticipated Transient Without Scram (ATWS) System"

TITLE: SAFEGUARDS SAFETY INJECTION AND ISOLATION SCHEME EK-13 (EC-13)

37	43	49	55	61	67	73
38	44	50	56	62	68	74
39	45	51	57	63	69	75
40	46	52	58	64	70	76
41	47	53	59	65	71	77
42	48	54	60	66	72	78

CONTAINMENT HI PRESS	
Sensor:	PSX-1801, PSX-1802, PSX-1803, or PSX-1804
Trip Setpoints:	4 psig
Alternate Indication:	Containment pressure indications (multiple)

AUTOMATIC FUNCTION:
- SIAS/CIAS on 2 of 4 coincidence.

OPERATOR ACTION:
- **REFER TO** appropriate EOP based on additional indications.

FOLLOW UP ACTION:
- IF inadvertent Containment Isolation results, THEN **GO TO** AOP-31.
- IF alarm caused by a single inoperable channel, THEN:
  - **INITIATE** Work Request.
  - **IMPLEMENT** any applicable Technical Specifications LCO 3.3.1, LCO 3.3.3, LCO 3.3.4, LCO 3.6.4 actions.

REFERENCES:
- AOP-31, "Spurious Containment Isolation"
- Technical Specifications LCO 3.3.1, LCO 3.3.3, LCO 3.3.4, LCO 3.6.4

TITLE: REACTOR PROTECTIVE SYSTEM SCHEME EK-06 (C-06)

RACK "C"			
1	2	3	4
5	6	7	8

CONTAINMENT HI PRESSURE TRIP	
Sensor:	Bistable Trip Unit (Any 1 of 4)
Trip Setpoints:	3.7 psig
Alternate Indication:	Containment pressure indicators

AUTOMATIC FUNCTION:

- Reactor Trip on 2 of 4 coincidence logic.

OPERATOR ACTION:

- **CHECK** Containment pressure indicators:
  - PIA-1814
  - PIA-1815
- IF high pressure indicated, THEN **REFER TO** in-use EOP.

FOLLOW UP ACTION:

- IF faulty instrument, THEN:
  - **BYPASS** Containment High Pressure Trip unit for affected channel per SOP-36.
  - **REFER TO** Technical Specifications LCO 3.3.1, LCO 3.3.3.

REFERENCES:

- Technical Specifications LCO 3.3.1, LCO 3.3.3
- SOP-36, "Reactor Protective System and Anticipated Transient Without Scram (ATWS) System"

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061.K6.01	
	Importance Rating	2.5	

K/A Statement: Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Controllers and positioners.

Proposed Question:

Given the following conditions:

- The Control Room has been evacuated due to a fire
- C-150, Auxiliary Hot Shutdown Panel, has been placed in service

As a result of placing C-150 in service, AFW Pump P-8B:

A. Will NOT automatically trip on low suction pressure.

B. Will NOT be available as a source of feedwater.

C. Automatic speed control is disabled.

D. Overspeed trip protection is disabled.

Proposed Answer: A

Explanation (Optional):

A. Correct, pump P-8B will not trip on low suction pressure. SV-0522C (C-150) and SV-0522H/G (ATWS) bypass low suction pressure trip, allowing SV-5022B (steam supply valve) to be cycled from C-150.
B. Incorrect, the applicant believes that P-8B cannot be controlled from C-150.
C. Incorrect, the applicant believes that electrical power is required for speed control of P-8B.
D. Incorrect, the applicant believes that electrical power is required for overspeed protection.

Technical Reference(s): PL-AFW Auxiliary Feedwater System Lesson Plan, E-17 Sheet 21A

(Attach if not previously provided, including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam Palisades 2009
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

16. Local Controls and Indications

g. Panel C-150, Auxiliary Hot Shutdown Panel

1) Enabling instrumentation on Auxiliary Hot Shutdown Panel EC-150/EC-150A disables the corresponding instrumentation in the Control Room and on Redundant Safety Injection Panel EC-33.

2) P-8B – can open and close CV-0522B via HS-0522C

   a) P-8B will NOT trip on low suction pressure when CV-0522B is opened from C-150A

3) CV-0749 – Can operate HIC-0749C, P-8A/B AFW flow to 'A' S/G if HS-0102A is placed in the "C-150" position

4) CV-0727 – Can operate HIC-0727C, P-8AB AFW flow to 'B' S/G if HS-0102B is placed in the "C-150" position

5) FI-0727B – Flow to 'B' S/G from P-8A/B

6) FI-0749B – Flow to 'A' S/G from P-8A/B

17.a. Control Room Controls

7) Low suction pressure prevents SV-0522B from operating

   a) SV-0522C (C-150) and SV-0522H/G (ATWS) bypass low suction pressure trip.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022.G2.1.27	
	Importance Rating	3.9	

K/A Statement: Knowledge of system purpose and/or function.

Proposed Question:

Per the Design Bases Documents DBD-2.03, “Containment Spray System,” and DBD-2.08, “Containment Air Coolers,” which of the following design functions are shared by both the Containment Air Cooling system and the Containment Spray system?

1. Remove energy from the atmosphere within the Containment Building during normal operation to keep the air temperature below 140°F.
2. Act as a barrier to limit radioactive releases from containment.
3. Provide post-accident cooling capability to limit containment pressure to within containment structure design value of 55 psig.
4. Provide post-accident cooling capability to achieve within 24 hours a containment pressure which is 50% or less the design pressure.

E. 1, 2, and 3

F. 1, 2, and 4

G. 1, 3, and 4

H. 2, 3, and 4

Proposed Answer: **D**

Explanation (Optional):
The CAC and CS systems share multiple design functions: to act as a barrier to limit radiological releases, to limit containment pressure below the design value during an accident, and to provide sufficient cooling to lower containment pressure to 50% the design value within 24 hours post-accident. The CS system does not have a normal operating condition function, as the CAC system does. The CAC system is designed to remove heat from containment to maintain containment temperature during normal operations to less than 140°F.

E. Incorrect, see explanation.
F. Incorrect, see explanation.
G. Incorrect, see explanation.
H. Correct, see explanation.

Technical Reference(s): DBD-2.03, DBD-2.08

(Attach if not previously provided, including version/revision number) ______________________

Proposed references to be provided to applicants during examination: ____None**?**________

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

NSPC	Nuclear Safety Performance Criteria
PPAC	Periodic and Predetermined Activity Control
SEP	Systematic Evaluation Program
SIS	Safety Injection Signal
SSA	Safe Shutdown Analysis (See NSCA)
SWS	Service Water System
WG	Water Gauge

2.0 LICENSING BASIS

2.1 SYSTEM REQUIREMENTS

2.1.1 System Functional Requirements

The following table provides the Functional Requirements for the CACs and the engineering design basis calculations which support them. The calculations are referenced by number to those calculations listed in Appendix D, Table D-1, "Engineering Design Basis Calculations."

TABLE 2.1-1
FUNCTIONAL REQUIREMENTS

	Description of Requirement	Table D-1 Calculation No
1.	The CACs shall remove energy from the atmosphere within the Containment Building during normal operation to keep the air temperature below 140°F.	4
2.	The CACs shall provide post-accident cooling capability to limit containment pressure to within containment structure design value of 55 psig.	1, 2, 3
3.	The CACs shall provide post-accident cooling capability to achieve within 24 hours a containment pressure which is 50% or less of the design pressure.	1, 2, 3
4.	The CACs shall act as a barrier to limit radioactive releases from containment.	(a)

(a) Verification that the CACs act as a barrier to radioactive releases is demonstrated by compliance with the structural analysis codes and standards given in Sections 3.4.2 and 3.4.3 of this document.

Final Safety Analysis Report (FSAR) Chapter 14 events which address the operation of the CACs are discussed in Section 4.0 of this document.

2.0 LICENSING BASIS

2.1 SYSTEM REQUIREMENTS

2.1.1 System Functional Requirements

The following table provides the functional requirements for the Containment Spray System (CSS) and the engineering design basis calculations which support them. The calculations are referenced by number to the calculations listed in Appendix D, Table D-1, "Engineering Design Basis Calculations." These are not licensing basis requirements, but are functional provisions of the system. Final Safety Analysis Report Chapter 14 transient analyses which address CSS operation are discussed in Section 4 of this document.

	Description of Requirement	Calculation No
1.	The CSS shall provide post-accident cooling capability to limit containment pressure to within containment structure design value of 55 psig.	1, 2, 3, 5, 6, 12, 13, 14
2.	The CSS shall provide post-accident cooling capability to achieve within 24 hours a containment pressure which is 50% or less of the design pressure.	1, 2, 3, 5, 6, 12, 13, 14
3.	The CSS shall act as a barrier to limit radioactive releases from containment.	(a), 8, 9, 10, 11
4.	The CSS shall boost HPSI discharge pressure to enable safe shutdown following a fire in the charging pump area.	4
5.	The CSS shall boost HPSI suction pressure during the recirculation mode following a Loss of Coolant Accident.	12

(a) Verification that the CSS acts as a barrier to radioactive releases is demonstrated by compliance with the structural analysis codes and standards given in Sections 3.4.2 and 3.4.3 of this document.

Final Safety Analysis Report (FSAR) Chapter 14 events which address CSS operation are discussed in Section 4 of this document.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061.A3.03	
	Importance Rating	3.9	

K/A Statement: Ability to monitor automatic operation of the AFW, including: AFW S/G level control on automatic start.

Proposed Question:

Given the following sequence of events:

- The reactor trips from full power due to a spurious turbine trip
- S/G levels lower to 24%
- AFW Pump P-8A does not auto start
- AFW Pump P-8C auto starts and supplies Auxiliary Feedwater (AFW) to both S/Gs
- After one hour, S/G levels are restored to 60-70%
- AFAS is NOT reset

The NCO then places AFW Pump P-8C in MANUAL and stops AFW Pump P-8C.

Which of the following best describes the AFW system response?

A. AFW Pump P-8B will start approximately 112.5 seconds later.

B. No AFW pumps will automatically start.

C. AFW Pump P-8C will start after 30.5 seconds.

D. AFW Pump P-8B will automatically start immediately.

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, plausible if the operator believes the timer starts when the running pump is secured.
B. Incorrect, plausible if operator fails to recognize P-8B is still in auto with the timer timed out and a standing AFAS signal.
C. Incorrect, plausible if the operator believes that the pump will start based solely on the timer and the AFAS signal.
D. Correct, P-8B is in automatic, AFAS is present, and the timer has timed out.

Technical Reference(s): PL-AFW Auxiliary Feedwater System Lesson Plan
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: ____None________

Learning Objective: ________________________ (As available)

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

Question History: Last NRC Exam Palisades 2001
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

13.a.8)

Each Steam Generator supply line for each AFW train contains a flow control valve. These valves will automatically open and control flow at a pre-set value (**165 gpm**) when the associated AFW pump(s) start.

17. Control Room Controls

c. Operation of FIC-0727/0749

1) Control flow from P-8A/B through 4" lines.

2) FIC and HIC powered from EY-10. Auto swaps to EY-30 if EY-10 lost.

3) Controller response when P-8A and P-8B OFF.

a) If FIC placed to AUTO, BLUE pen drops to 0%. Operator can then adjust setpoint as required.

4) Controller response when P-8A or P-8B started.

a) If FIC is in AUTO, valve opens to BLUE setpoint flow rate.

b) If FIC is in CASCADE, valve opens to obtain 165 gpm. Adequate flow for removing decay heat.

5) Controller response when P-8A or P-8B running

a) If FIC is transferred from CASCADE to AUTO, setpoint remains at 165 gpm.

6) FIC will transfer to AUTO and track the HIC if:

a) FIC was originally in MANUAL

b) HIC is then swapped from AUTO to MANUAL

7) FIC-0727/0749 Operability Requirements. Considered operable if:

a) In CASCADE, or

b) In AUTO AND no MFW in service AND S/G levels and PCS temps maintained within bands required for decay heat removal.

c) NOT operable if FIC or HIC in MANUAL.

b. Operation of FIC-0737A/0736A

1) This controller can control either the 1 ½" bypass valves (CV-0736/CV-0737) or the main 4" valves (CV-0736A/CV-0737A).

2) HIC (on panel C-33) and FIC powered from EY-20. No auto swap to alternate source. Loss of EY-20 also TRIPS P-8C since the pressure switches lose power for the low suction pressure trip.

3) P-8C is preferred for plant startups and shutdowns. WHY?

a) PF Mode (Program Function) controls the small bypass valves for finer control of AFW flow.

b) If in AUTO or PF Mode, and an AFAS occurs, the controller auto swaps to CASCADE, and the bypass valves close.

4) P-8C must be operating before FIC can be swapped to AUTO. If you try to select AUTO with P-8C OFF, the FIC swaps to CASCADE!

17.7)

Q: Why are FIC-0736A/0737A controllers operable in AUTO always but FIC-0727/0749 controllers have restrictions?

A: FIC-0736A/0737A controllers will automatically swap to CASCADE upon an AFAS and FIC-0727/0749 controllers will not.

ATTACHMENT 3: Worksheet and Answer Key

WORKSHEET AND ANSWER KEY

8. Given the following conditions:
   - All AFW HICs are in the Auto position
   - All FICs are in Auto Mode
   - AFW Pump P-8A and P-8C are operating

A. What controller mode would FIC-0736A and FIC-0737A transfer to following an AFAS?
Cascade

B. What Controller Mode would FIC-0727 and FIC-0749 transfer to following an AFAS?
They would remain in Auto Mode at the current set point

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	____
	Group #	1	____
	K/A #	062.K4.02	
	Importance Rating	2.5	____

K/A Statement: Knowledge of the AC distribution system design feature(s) and/or interlock(s) which provide for the following: Circuit breaker automatic trips.

Proposed Question:

Given the following conditions:

- The Plant is at 35% power.
- GOP-5, "Power Escalation in Mode 1," is in progress.
- The Crew is preparing to start the second Feedwater Pump.
- The following annunciators have just alarmed:
  - EK-0334, "Switchyard Critical Trouble"
  - EK-5004, "Bkr 29R8 Fail to Trip Operated"

Given the conditions noted above, what is the expected status of the Switchyard Rear "R" Bus and what is the expected status of the Plant?

A. De-energized; Plant remains online.

B. De-energized; Plant is tripped.

C. Energized, Plant remains online.

D. Energized, Plant is tripped.

Proposed Answer: **A**

Explanation (Optional):

The rear bus is de-energized due to the 486BF relay tripping 29H9, actuation of a transfer trip on the Vergennes Line, and actuation of both the 486-P/R (Bus Protection Lockout Primary) and 486-B/R (Bus Protection Lockout Backup) relays tripping 25R8, 27R8, and 31R8. The Plant will remain online as the 4160VAC busses are on Station Power transformers and not the Start-Up transformers (the swap is performed at 20% power, per GOP-5).

A. Correct, see explanation.
B. Incorrect, the applicant incorrectly believes that the Start-Up transformers are supplying power.
C. Incorrect, the applicant does not understand the impacts of the lockout relay actuation.

D. Incorrect, the applicant does not understand the impacts of the lockout relay actuation and incorrectly believes that the Start-Up transformers are supplying power.

Technical Reference(s): ARP-13, DBD-6.02
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

Question modified from 2010 Palisades NRC Exam. Modified question to utilize switchyard breaker failure to trip in lieu of a Start-up transformer sudden pressure relay. Significantly modified question stem. Reworded and reorganized distractors.

TITLE: GENERATOR SCHEME EK-03 (EC-11)

1	7	13	19	25	31
2	8	14	20	26	32
3	9	15	21	27	33
4	10	16	22	28	34
5	11	17	23	29	35
6	12	18	24	30	36

SWITCHYARD CRITICAL TROUBLE

Sensor:	Contact
Trip Setpoints:	Annunciation at Switchyard Panels C-53 and C-54 (Annunciator # 3, 4, 10, 17, 21, 22, 23, 25, 29, 30, 31, 33, 37, 38, 39, 41, 45, 46, 47, 50, 51, 52, 53, 55, 57, 60, 61, 62, 63, 65, 69, 70, 71, 73, 77, 78, 79, 81, 85, 86, 87, 89, 91, 93, 95)
Alternate Indication:	None

AUTOMATIC FUNCTION:
- None

OPERATOR ACTION:

NOTE:	If an individual switchyard breaker alarm is actuated, at the Local Breaker Cabinet, then the Control Room "Switchyard Critical Trouble" alarm will not clear even if the Relay House Alarm Panel is reset. The local alarms require ITC to unlock the Local Breaker Cabinet so the alarm can be reset.

- REFER TO ARP-13.

FOLLOWUP ACTION:
- None

REFERENCES:
- ARP-13, "345 kV Switchyard Scheme EK-50 (C-53, C-54)"

TITLE: 345 kV SWITCHYARD SCHEME EK-50 (C-53, C-54)

1	2	3	4	5	6	7	8
9	10	11	12	13	14	15	16
17	18	19	20	21	22	23	24
25	26	27	28	29	30	31	32
33	34	35	36	37	38	39	40
41	42	43	44	45	46	47	48

BKR 29R8 FAIL TO TRIP OPERATED

Sensor: 486BF/474

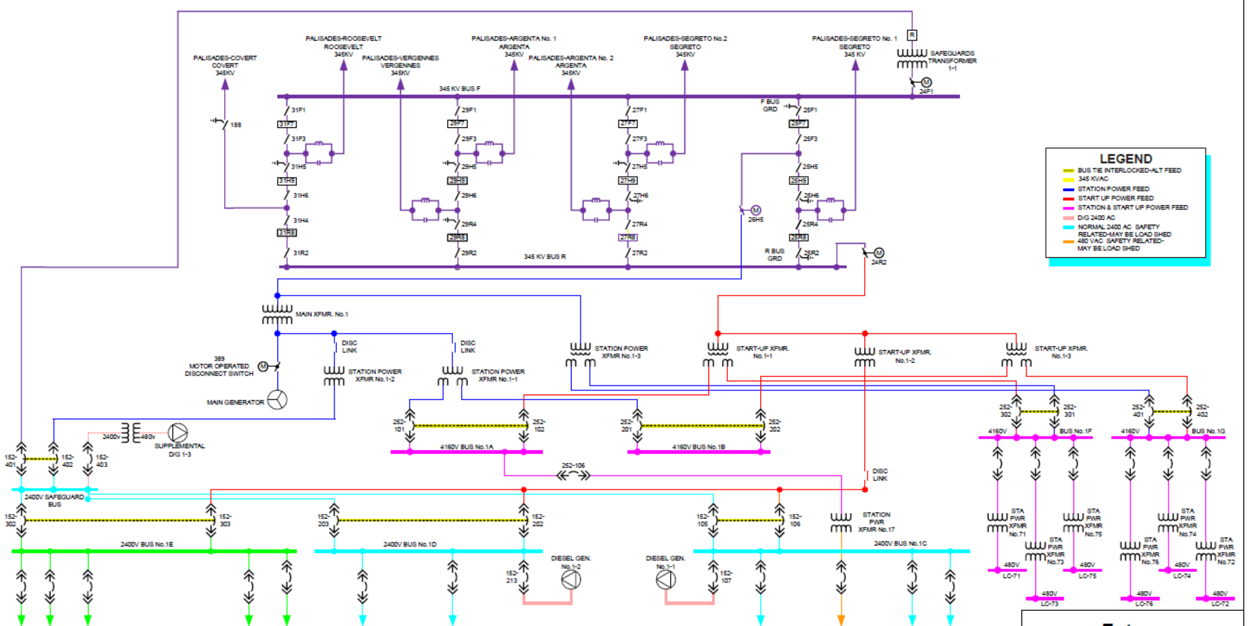
Trip Setpoints: 'b' contact

AUTOMATIC FUNCTION:
- Annunciates EK-0334, Switchyard Critical Trouble.
- 486BF trips 29H9.
- Initiates Transfer Trip on Vergennes Line.
- Trips 486B-P/R, Bus Protection Lockout Primary and 486B-B/R, Bus Protection Lockout Backup relays which trips 25R8, 27R8, and 31R8.

OPERATOR ACTION:
- **VERIFY** the following breakers OPEN
  - 25R8
  - 27R8
  - 29H9
  - 31R8
- **NOTIFY** System Control.

FOLLOW UP ACTION:
- IF in Mode 1, 2, 3, or 4, THEN **REFER TO** Technical Specifications LCO 3.8.1.
- IF in Mode 5 or 6, THEN **REFER TO** Technical Specifications LCO 3.8.2.
- **REQUEST** System Control to restore the "R" Bus.
- WHEN "R" Bus is restored, THEN **RESTORE** Station Power.

REFERENCES:
- Technical Specifications LCO 3.8.1, LCO 3.8.2



SOURCE DOCUMENTS	
P&ID	WD 950 SH. A WD 950 SH. 1 WD 1421 SH. 31

Entergy

PALISADES NUCLEAR PLANT

FOR TRAINING USE ONLY

DRAWING TITLE: SINGLE LINE METER & RELAY DIAGRAM

FILE NAME: SOD_EPS_01_r04 | DATE: 08/01/16 | REV NO. 4

The sudden pressure relays for Startup Transformers 1-1, 1-2 and 1-3 were disconnected from tripping in 1978 per FC-414 (Reference 116) as part of the R Bus security ("security" as defined in Section 3.2.11) upgrade after a series of mistrips (Reference 115). It was later discovered that water intrusion had been the cause of all the "R" bus trips, and the equipment sealing was corrected. The trip function of the sudden pressure relays was subsequently restored during the 1990 Refueling Outage per SC-90-041 (Reference 117). Operation of the relays will also sound an alarm and initiate the deluge system.

Automatic reclosing is not provided for the 25F7 generator breaker. Automatic reclosing is provided for the 25H9 breaker by synchronism check. However, this reclosing is cut out of service and will not be used. Present CPCo practices prohibit reclosing on Air Circuit Breakers located adjacent to a generator.

Automatic reclosing for all other 345 KV breakers is provided by high speed, synchronism check, through the reclosing relays associated with each breaker.

3.2.11.5 Breaker Failure

Each 345 KV breaker is equipped with a breaker failure relay. The breaker failure relay is a complex relay having current fault detectors, two timers (one fast and one slow) a tripping auxiliary relay, several inputs, and miscellaneous logic to provide breaker failure protection for various applications. Each relay is fed by a seal-in cut-out. The seal-in cut-out when used, seals in the initiation circuit if the relay has been initiated and the fault detectors have operated. Operation of the relay is as follows: If the breaker's trip coils are energized via its protective relays and the fault detectors have operated (current greater than their set value) both timers will start timing. Opening of the breakers 52a contact (in less than 1.5 cycles) will stop the fast timer. Interruption of the fault by the breaker will drop out the fault detectors which will stop the slow timer. If the breaker fails to operate, the fast timer will operate the breaker failure tripping auxiliary relay. If the breaker operates but fails to interrupt the fault, the slow timer will operate the breaker failure tripping auxiliary relay.

R Bus

Operation of a breaker failure tripping auxiliary relay will trip and block closing of all adjacent breakers, initiate the direct transfer trip transmitter to trip the remote terminals breaker (when adjacent), initiate both direct trip systems for a unit shutdown for either adjacent unit breaker, or initiate the tripping of the three startup transformer bank low side breakers for failure of a breaker adjacent to the R Bus.

GCL 5.1

POWER ESCALATION IN MODE 1

	Time	Date	Initial
CAUTION VHPT setpoints may be nonconservative if reset prior to performance of a heat balance.			
2.11 WHEN indicated power is between 15% to 28.5%, THEN **PERFORM** a heat balance. Refer to DWO-1, "Operator's Daily/Weekly Items Modes 1, 2, 3, or 4."	______	______	______
2.12 **PERFORM** the following at approximately 20% power:			
a. **TRANSFER** 4160V Buses to Station Power Transformers. Refer to SOP-30, "Station Power."	______	______	______
b. **CLOSE** all Main Turbine drains, Main Steam drains, and Reheater Drain Tank startup vents. Refer to SOP-8, "Main Turbine and Generating Systems."	______	______	______
c. **PLACE** Heater Drain System in operation. Refer to SOP-10, "Extraction and Heater Drain Systems."	______	______	______
d. **ALIGN** Feedwater Heater Vents. Refer to SOP-10, "Extraction and Heater Drain System," Attachment 2, "Feedwater Heater Vent Alignment for Operation."	______	______	______

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062.A4.07	
	Importance Rating	3.1	

K/A Statement: Ability to manually operate and/or monitor in the control room: synchronizing and paralleling of different AC supplies.

Proposed Question:

An operator is attempting to parallel an available offsite source with a loaded Diesel Generator per SOP-22, "Emergency Diesel Generators." The operator notices the synchroscope rotating in the fast direction at 10 seconds per revolution.

What should the operator do in order to perform the synchronization?

A. Raise DG speed until synchroscope turns slowly in the slow direction.

B. Lower DG speed until synchroscope turns slowly in the fast direction.

C. Raise DG speed until synchroscope turns slowly in the fast direction.

D. Lower DG speed until synchroscope turns slowly in the slow direction.

Proposed Answer: **D**

Explanation (Optional):
The synchroscope must be rotating slowly in the slow direction, per SOP-22, in order to synch a loaded DG to an offsite source. Since the synch scope is rotating fast in the fast direction (as evidenced by 10 second per revolution on the synchroscope), the DG speed must be lowered. SOP-22 requires 45 seconds (or more) per revolution to ensure adequate operation of the protective synch relays. The applicant needs to interpret the 10 seconds per revolution and apply that to being "fast in the fast direction."

A. Incorrect, see explanation. The applicant misunderstands the operation of adjusting DG speed in relation to the synchroscope. If speed is raised, the synchroscope will rotate faster in the fast direction.
B. Incorrect, see explanation. The applicant is confusing the synchronization of an unloaded DG onto an offsite source, rather than a loaded DG onto an offsite source.
C. Incorrect, both parts are incorrect, see explanation.
D. Correct, see explanation.

Technical Reference(s): SOP-22
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: ______None________

Learning Objective: ________________________ (As available)

Question Source: Bank # _______

Modified Bank # _______ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__

55.43 _____

Comments:

7.5.4 To Parallel Available Offsite Power Source to Loaded Diesel Generator

a. Both Diesel Generators shall not be connected to the same 2.4KV transformer at the same time.

b. IF loss of 1C or 1D bus has previously occurred, THEN **REFER TO** Abnormal Operating Procedure AOP-8, " Loss of EA-11 (Bus 1C)," or Abnormal Operating Procedure AOP-9, " Loss of EA-12 (Bus 1D)," to parallel offsite power source to a loaded diesel generator.

c. The Diesel Generator is inoperable when operated loaded in parallel with offsite power. **REFER TO** Technical Specifications:

- Modes 1, 2, 3 and 4 - LCO 3.8.1
- Modes 5, 6, and during movement of irradiated fuel assemblies - LCO 3.8.2

d. **RECORD** the completion of the initial offsite source checks with the Diesel Generator LCO log entry.

e. **PLACE** Parallel/Unit selector switch for affected D/G to PARALLEL position:

D/G 1-1
G1-1/DSR, Parallel/Unit Selector

D/G 1-2
G1-2/DSR, Parallel/Unit Selector

f. IF paralleling to startup power, THEN **ENSURE RESET** Startup Transformers Undervoltage Auxiliary Relay.

1C Bus	1D Bus
127X-5	127X-6

LOCATION: EC-04

g. **TURN ON** synchroscope for applicable offsite power source.

h. **ADJUST** RUNNING voltage (D/G) to match INCOMING voltage (offsite power source) (± 1 volt on synchronizing voltmeter) by adjusting voltage with the Field Rheostat switch.

NOTE:	The speed of rotation of the synchroscope should be no faster than 45 seconds per revolution to allow for operation of the associated protective relay interlocks to support breaker closure.

i. **ADJUST** D/G speed with the Governor Set Point switch until synchroscope turns slowly in the counter-clockwise (**SLOW**) direction.

CAUTION

Failure to ensure that the D/G is loaded to a minimum of 50 KW may result in reverse powering the D/G.

j. WHEN synchroscope nears 1200 hours on meter AND the synchronizing lights are OFF, THEN **CLOSE** desired offsite power supply breaker AND **ENSURE** D/G load is at least 50 KW.

OFFSITE POWER SOURCE	BUS 1C BREAKER	BUS 1D BREAKER
Safeguards/Station Power	152-105	152-203
Startup Power	152-106	152-202

k. **TURN OFF** synchroscope.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	063.K1.03	
Importance Rating	2.9	

K/A Statement: Knowledge of the physical connections and/or cause effect relationships between the DC electrical system and the following systems: Battery charger and battery

Proposed Question:

The Plant was at 100% power with ED-16, Station Battery Charger No 2, in service supplying the No 2 DC Bus.

FIVE minutes ago, the Plant experienced:

- A Loss of Offsite Power
- The AC supply breaker (52-225) for ED-16, Station Battery Charger No 2, tripped and cannot be reclosed.

Given the plant conditions noted above, DC Bus No 1 is currently powered from __(1)__ and DC Bus No 2 is currently powered from __(2)__.

A. (1) ED-15, Station Battery Charger No 1 and Station Battery # 1.
(2) ED-18, Station Battery Charger No 4 and Station Battery # 2.

B. (1) ONLY Station Battery # 1.
(2) ONLY Station Battery # 2.

C. (1) ED-17, Station Battery Charger No 3 and Station Battery # 1.
(2) ED-18, Station Battery Charger No 4 and Station Battery # 2.

D. (1) ED-15, Station Battery Charger No 1 and Station Battery # 1.
(2) ONLY Station Battery # 2.

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, see explanation for choice D
B. Incorrect, see explanation for choice D
C. Incorrect, see explanation for choice D
D. Correct, the DC bus will remain energized due to power being supplied from Station Battery No 2. Battery Chargers ED-15 and ED-17 are capable of powering DC Bus No 1, with only one charger normally lined up to the bus. Battery chargers ED-16 and ED-18 are capable of powering DC Bus No 2, with only one charger normally lined up to the bus. DG 1-1 supplies chargers ED-15 and ED-18 while DG 1-2 supplies chargers ED-17 and ED-16. Power to ED-18, Battery charger No 4, will need to be restored from a

backup power source (DG 1-1).

Technical Reference(s): SOP-30, DBD-4.02________________________________
(Attach if not previously provided, ________________________________
including version/revision number) ________________________________

Proposed references to be provided to applicants during examination: ______None_______

Learning Objective: __________________________ (As available)

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

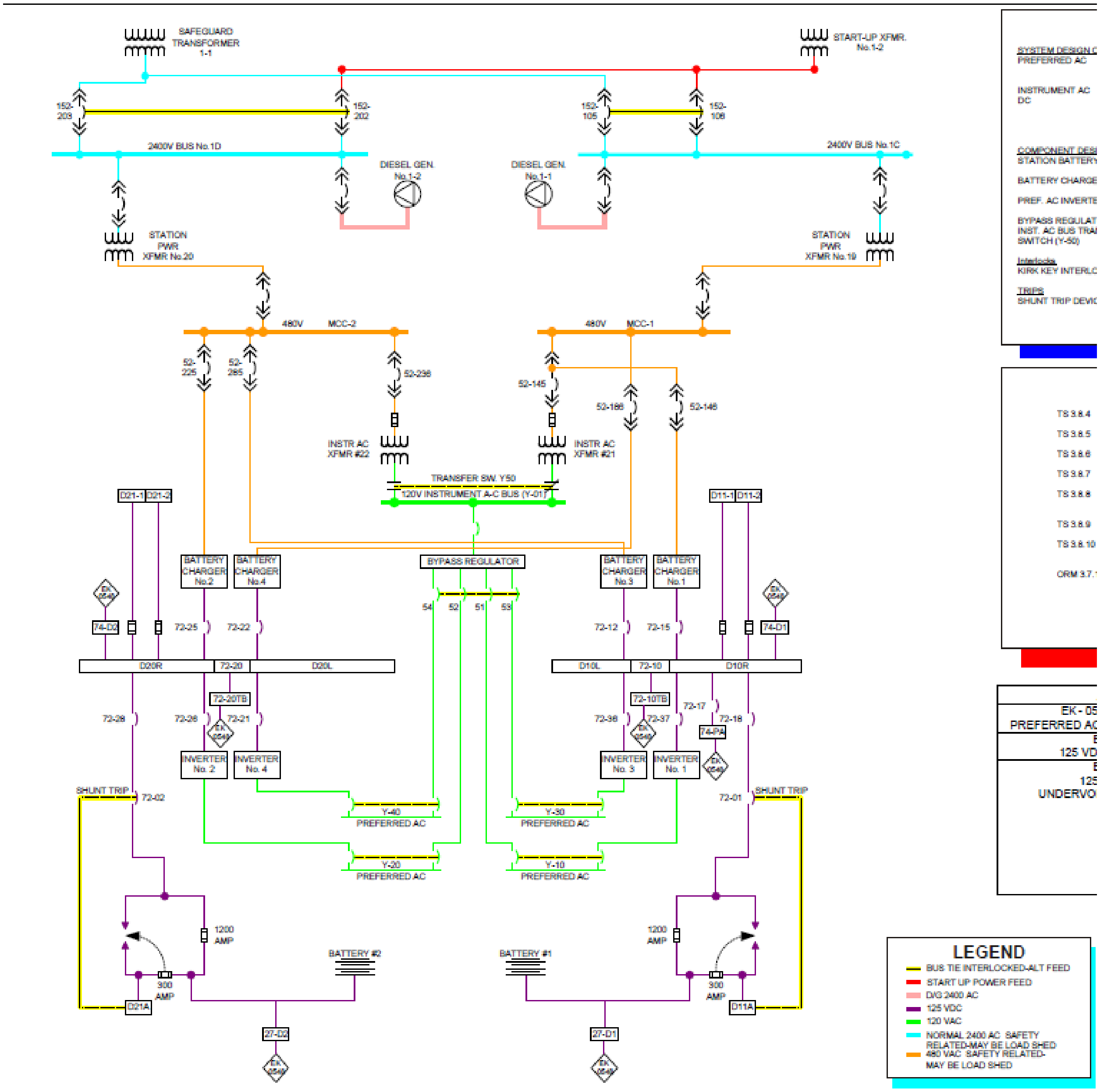
Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:



SAFEGUARD TRANSFORMER 1-1
START-UP XFMR. No.1-2
152-203
152-202
152-105
152-106
2400V BUS No.1D
2400V BUS No.1C
DIESEL GEN. No.1-2
DIESEL GEN. No.1-1
STATION PWR XFMR No.20
STATION PWR XFMR No.19
480V MCC-2
480V MCC-1
52-225
52-285
52-236
52-145
52-186
52-146
INSTR AC XFMR #22
INSTR AC XFMR #21
TRANSFER SW. Y50
120V INSTRUMENT A-C BUS (Y-01)
D21-1
D21-2
D11-1
D11-2
BATTERY CHARGER No.2
BATTERY CHARGER No.4
BYPASS REGULATOR
BATTERY CHARGER No.3
BATTERY CHARGER No.1
54
52
51
53
74-D2
72-25
72-22
72-12
72-15
74-D1
D20R
72-20
D20L
D10L
72-10
D10R
72-20TB
72-10TB
72-28
72-26
72-21
72-38
72-37
72-17
72-18
74-PA
INVERTER No. 2
INVERTER No. 4
INVERTER No. 3
INVERTER No. 1
SHUNT TRIP
72-02
72-01
Y-40 PREFERRED AC
Y-30 PREFERRED AC
Y-20 PREFERRED AC
Y-10 PREFERRED AC
1200 AMP
300 AMP
D21A
D11A
BATTERY #2
BATTERY #1
27-D2
27-D1
SYSTEM DESIGN
PREFERRED AC
INSTRUMENT AC
DC
COMPONENT DESI
STATION BATTERY
BATTERY CHARGE
PREF. AC INVERTE
BYPASS REGULAT
INST. AC BUS TRAI
SWITCH (Y-50)
Interlocks
KIRK KEY INTERLC
TRIPS
SHUNT TRIP DEVIC
TS 3.8.4
TS 3.8.5
TS 3.8.6
TS 3.8.7
TS 3.8.8
TS 3.8.9
TS 3.8.10
ORM 3.7.
EK - 05
PREFERRED AC
125 VD
125
UNDERVO
LEGEND
BUS TIE INTERLOCKED-ALT FEED
START UP POWER FEED
D/G 2400 AC
125 VDC
120 VAC
NORMAL 2400 AC SAFETY RELATED-MAY BE LOAD SHED
480 VAC SAFETY RELATED-MAY BE LOAD SHED

11. **OPEN** 480V AC Feeder breaker on the MCC/Lighting Panel associated with Datalogger Battery Charger that was removed from service.

Battery Charger	Ganged Breaker	Location
ED-206	25/27/29	EL-03, NE Turbine Deck
ED-207	1/3/ 5	EL-04, SW CSR

7.8.2 Station Battery Chargers 1, 2, 3, and 4

NOTE:	If battery discharge due to extended loss of vital AC power, then Electrical Maintenance assistance will likely be required.

NOTE:	Only one battery charger on each DC Bus (No 1 and No 2) should be in service at any one time.

NOTE:	Battery chargers shall normally be paired as follows: 1 (ED-15) and 2 (ED-16) or 3 (ED-17) and 4 (ED-18). This does not apply when the Plant is in Modes 5 and 6.

a. **ALIGN** a charger onto a discharged battery as follows:

1. **ENSURE CLOSED** (ON) 480 VAC supply breaker for charger being placed in service:

Battery Charger	480 VAC Supply Breaker	Description
#1 (ED-15)	52-146	Station Battery Charger #1 ED-15
#2 (ED-16)	52-225	Station Battery Charger #2 ED-16
#3 (ED-17)	52-285	Station Battery Charger #3 ED-17
#4 (ED-18)	52-186	Station Battery Charger #4 ED-18

2. CLOSE (ON) AC Input breaker on battery charger.

Battery Charger	AC Input Breaker	Description
#1 (ED-15)	52-146A	Station Battery Charger No 1 ED-15 AC Input
#2 (ED-16)	52-225A	Station Battery Charger No 2 ED-16 AC Input
#3 (ED-17)	52-285A	Station Battery Charger No 3 ED-17 AC Input
#4 (ED-18)	52-186A	Station Battery Charger No 4 ED-18 AC Input

3. CLOSE (ON) DC Output breaker on battery charger.

Battery Charger	DC Output Breaker	Description
#1 (ED-15)	72-15A	Station Battery Charger No 1 ED-15 DC Output
#2 (ED-16)	72-25A	Station Battery Charger No 2 ED-16 DC Output
#3 (ED-17)	72-12A	Station Battery Charger No 3 ED-17 DC Output
#4 (ED-18)	72-22A	Station Battery Charger No 4 ED-18 DC Output

4. RESET battery charger alarms by PRESSING Alarm Reset pushbutton on applicable charger as necessary.

5. CLOSE (ON) 125 VDC breaker for battery charger being placed in service.

Battery Charger	125 VDC Breaker	Description
#1 (ED-15)	72-15	Battery Charger No 1 ED-15
#2 (ED-16)	72-25	Battery Charger No 2 ED-16
#3 (ED-17)	72-12	Battery Charger No 3 ED-17
#4 (ED-18)	72-22	Battery Charger No 4 ED-18

3.3.1 Battery Chargers ED-15, ED-16, ED-17, ED-18

The battery chargers are the solid state type. They furnish single-phase ungrounded power (via inverters) for reactor safety circuits and for nuclear and other critical instrumentation, and DC for battery charging and other miscellaneous loads. Two charging voltages are provided, one for floating and one for equalizing the battery. Each battery charger is rated to provide a continuous output of 200 amperes and is provided with a feature to limit output current at 200 amperes ±5%. Because of this feature, should a short circuit develop on the battery charger output, total contribution of fault current from the charger would be limited to 200 amperes ±5%.

Each of the two chargers on a bus is supplied from a separate motor control center. Each motor control center is fed by a separate engineered safeguard channel. EB-01, (ultimately Diesel Generator 1-1) supplies Charger ED-15 on Buses ED-10L/ED-10R, and Charger ED-18 on Buses ED-20L/ED-20R; while EB-02 (ultimately Diesel Generator 1-2) supplies Charger ED-17 on Buses ED-10L/ED-10R, and Charger ED-16 on Buses ED-20L/ED-20R (Reference 32). To remove the possibility of a common mode failure affecting both engineered safeguard channels, administrative controls are in place to ensure that when the reactor is not in cold shutdown, only one charger per bus is in service (Reference 42).

The 1972 FSAR identified that a fully discharged battery can be recharged in less than four hours by using both chargers in parallel. The current FSAR identifies that a fully discharged battery can be recharged in less than thirteen hours by using both chargers in parallel. The difference being the replacement of the 664 A-H battery with an 1739 A-H battery, the previously omitted inclusion of the final 5% recharge time and limiting the charger output to 180 A (in place of the 200 A rating) as tested in Technical Specifications. Presently, the time required to recharge the fully discharged 1739 A-H station batteries is 11.4 hours (Reference 81).

During a post-accident condition with simultaneous loss of off-site power one battery charger will be powered automatically from its emergency diesel generator in 10 seconds. After the initial 10 seconds of emergency operation, in which the station battery carries the full load, one battery charger will then provide the majority load current with the station battery providing the remaining current requirement. The battery chargers will supply 800 A-H to each bus over the four-hour period (200A rated X 4 hours). With the present 125V DC system loads of 832 A-H left channel, and 904 A-H right channel, each battery charger is sufficient in size to provide the majority load current requirement and upon return to normal conditions, provide all load current while recharging the station batteries in reasonable time. This statement is supported by using the four hour load profile from Reference 43 as the 125V DC System emergency load profile. The four hour profile with no load shed is used because no load shedding occurs without a "loss of all AC" event. For Left Channel total system load is 832 A-H of which 800 A-H is provided by the battery charger, leaving approximately 32 A-H removed from ED-01. For Right Channel the total system load is 904 A-H of which 800 A-H is provided by the charger, leaving approximately 104 A-H removed from ED-02.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064.K2.02	
	Importance Rating	2.8	

K/A Statement: Knowledge of bus power supplies of the following: Fuel oil pumps

Proposed Question:

Fuel Oil Transfer Pump P-18B is normally energized from __(1)__ and can NOT be energized by__(2)__ in an emergency situation.

A. (1) MCC-1
(2) DG 1-1

B. (1) MCC-1
(2) DG 1-2

C. (1) MCC-8
(2) DG 1-1

D. (1) MCC-8
(2) DG 1-2

Proposed Answer: B

Explanation (Optional):

A. Incorrect, MCC-1 normally supplies power to P-18B, while MCC-8 supplies power to P-18A. P-18B can only be powered from offsite power or from DG 1-1. DG 1-2 can supply power to P-18A, but not P-18B. DG 1-1 can supply power to P-18B if necessary.
B. Correct, see choice A.
C. Incorrect, see choice A.
D. Incorrect, see choice A.

Technical Reference(s): TS Bases 3.8.3, PL-EDG "Emergency Diesel Generator Lesson Plan", E-4 Sheet 1&2

(Attach if not previously provided, including version/revision number) ________

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

Question Source: Bank # ________

	Modified Bank #	________ (Note changes or attach parent)
	New	___X___
Question History:	Last NRC Exam	__________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

5) The system automatically transfers fuel oil to a day tank via the P-18A pump (pump P-18B is operated manually only).

   a) A low level switch opens the solenoid isolation valve to the day tank and starts the P-18A pump.

   b) A high level switch on the day tank will close the solenoid isolation valve and stop pump P-18A when the tank is full.

EO 6 List the power supplies for the following Emergency Diesel Generator system components:

- ***Fuel Oil Transfer Pumps***

   c) Pump P-18A is powered by 480V MCC-8.

   d) Pump P-18B is powered by 480V MCC-1.

VA-39

6) Fuel oil is gravity-fed through a level actuated solenoid valve from the day tank to its respective belly tank.

7) The engine driven Fuel Oil Booster Pump (P-209A, P-209B) supplies fuel from the belly tank to the fuel injection pumps

8) The system normal operating pressure is between 40 psig and 60 psig.

   a) Measured in the engine to bedplate day tank fuel return line.

BASES

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel, Lube Oil, and Starting Air

BASES

BACKGROUND

The Diesel Generators (DGs) are provided with a storage subsystem having a required fuel oil inventory sufficient to operate one diesel for a period of 7 days, while the DG is supplying maximum post-accident loads. The fuel oil storage subsystem is comprised of the Fuel Oil Storage Tank and a fuel oil day tank. This onsite fuel oil capacity is sufficient to operate the DG for longer than the time to replenish the onsite supply from offsite sources.

Fuel oil is transferred from the Fuel Oil Storage Tank to either day tank by either of two Fuel Transfer Systems. The fuel oil transfer system which includes fuel transfer pump P-18A can be powered by offsite power, or by either DG. However, the fuel oil transfer system which includes fuel transfer pump P-18B can only be powered by offsite power, or by DG 1-1.

Examination Outline Cross-Reference:

		RO	SRO
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064.A3.01	
	Importance Rating	4.1	

K/A Statement: Ability to monitor automatic operation of the ED/G system, including: Automatic start of compressor and ED/G

Proposed Question:

The Diesel Generator Starting Air Compressor(s) auto-start at what pressure?

A. 245 psig

B. 235 psig

C. 220 psig

D. 215 psig

Proposed Answer: B

Explanation (Optional):

A. Incorrect, 245 psig is the normal Air Start Tank pressure, maintained by the Starting Air compressors.
B. Correct, at 235 psig, D/G starting air compressors automatically start on low Air Start Tank pressure
C. Incorrect, at 220 psig, the D/G Trouble alarm comes in due to low Air Start Tank pressure.
D. Incorrect, at 215 psig, the D/G becomes inoperable due to insufficient Starting Air pressure.

Technical Reference(s): DBD-5.01
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ___________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__

55.43 _____

Comments:

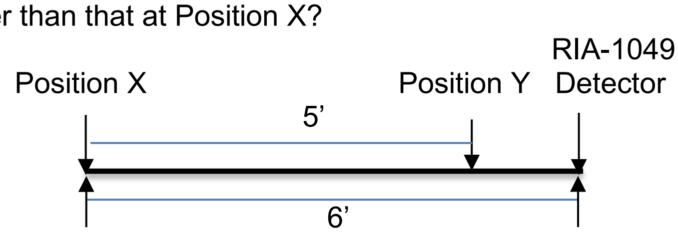
The air compressor has a working pressure of 250 psi (Reference 51). Normal air storage tank pressure is in the region of 245 psi and minimum pressure for engine cranking is 75 psi (Reference182 and 183). It takes approximately 134 minutes for C-3A or C-3B to recharge a pair of depleted tanks to 235 psi (Reference 217 & 261). The pressure in the tank controls the compressor start/stop switch. If the pressure in the tank drops to a setpoint in the region of 235 psig, then the compressor automatically starts. At 12.7 CFM C-3A will recharge T-31A and T-31B from 200 psi to 235 psi in approximately 20 minutes, similarly C-3B recharges T-31C and T-31D in approximately 20 minutes (Reference 217 & 261). Low pressure in an air storage tank will actuate an alarm in the Control Room at approximately220 psig. The tanks are equipped with automatic drain traps to keep system moisture levels low in order to minimize corrosion problems within the system.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073.K5.02	
	Importance Rating	2.5	

K/A Statement: Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Radiation intensity changes with source distance.

Proposed Question:

A hot spot has been identified six (6) feet upstream of RIA-1049, Radwaste Discharge Monitor, at Position X. Assume that during a release, the hot spot moved five (5) feet downstream towards the radiation monitor to Position Y. The change in radiation intensity seen at the radiation monitor detector, RIA-1049, with the hot spot at Position Y would be how many times greater than that at Position X?



A. 5

B. 6

C. 25

D. 36

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, the applicant applies a linear relationship between intensity and distance and does not account for the 1' between the detector and the final Position Y.
B. Incorrect, the applicant applies a linear relationship between intensity and distance.
C. Incorrect, the applicant misapplies the inverse square law and applies 5' as the final distance, rather than the correct 1'. This would be due to misinterpreting the figure.
D. Correct, due to the inverse square law, the count rate decreases by ¼ when the distance is doubled $(Ix/Iy)=(Xx/Xy)^2$, where I is intensity in cps and X is distance from the

Technical Reference(s):	GFES Lesson Plan N-RO-01-L-044-I, Radiation Protection
(Attach if not previously provided,	
including version/revision number)	

Proposed references to be provided to applicants during examination: ______None_______

Learning Objective: ________________________ (As available)

Question Source: Bank # _______

Modified Bank # ___X___ (Note changes or attach parent)

New _______

Question History: Last NRC Exam Palisades 2012

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _12__

55.43 _____

Comments:

Modified from Palisades 2012 NRC Exam. Modified stem to change scenario, the radiation monitor, and the particle distances from the monitor. The answer and one distractor were changed.

Palisades 2012 #50 Ref DOE-HDBK-1130-2008 Module 4
Palisades 2010 #51 Ref DOE-HDBK-1130-2008

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076.A2.02	
	Importance Rating	2.7	

K/A Statement: Ability to (a) predict the impacts of the following malfunctions or operations of the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Service water header pressure.

Proposed Question:

Given the following Plant conditions:

- The Plant is at 100% power.
- Today is January 12th.
- Service Water (SW) Pumps P-7A and P-7C are running.
- SW Pump P-7B is in Standby.
- Dilution Water Pumps P-40A and P-40B are running.

Frazil ice begins to develop on Traveling Screens F-4B and F-4C and it is noted that SW Header Pressures and SW Pump Bay Levels are lowering.

Which one of the following describes __(1)__ The impact of these conditions, and __(2)__ The correct action to take?

A. (1) SW Pump P-7B will start at 45 psig discharge pressure for P-7A or P-7C.
(2) Trip reactor if SW Pump Bay level lowers to 572'.

B. (1) SW Pump P-7B will start at 40 psig discharge pressure for P-7A or P-7C.
(2) Trip reactor if SW Pump Bay level lowers to 574'.

C. (1) SW Pump P-7B will start at 40 psig discharge pressure for P-7A or P-7C.
(2) Trip reactor if SW Pump Bay level lowers to 572'.

D. (1) SW Pump P-7B will start at 45 psig discharge pressure for P-7A or P-7C.
(2) Trip reactor if SW Pump Bay level lowers to 574'.

Proposed Answer: **C**

Explanation (Optional):

At 40 psig SW header pressure on the running SW pump(s), the standby SW pump will auto-start. At 45 psig, a "Non-Critical Service Water Low Pressure" alarm will annunciate, alerting the operators of the degrading conditions. Per AOP-35, the reactor is required to be tripped at 572' SW Bay Level. SW Bay Level of 574' is an entry criteria into AOP-35 and will also annunciate a "SW Pump Bay Low Level" alarm.

A. Incorrect, part 1 is incorrect. Part 2 is correct. See explanation.
B. Incorrect, part 1 is correct, part 2 is incorrect. See explanation.
C. Correct, see explanation.
D. Incorrect, part 1 and part 2 are both incorrect. See explanation.

Technical Reference(s): ARP-7, AOP-35
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ____None____

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam ___________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:
Question used from Palisades 2009 Audit Exam.

TITLE: AUXILIARY SYSTEMS SCHEME EK-11 (C-13)

37	43	49	55	61	67
38	44	50	56	62	68
39	45	51	57	63	69
40	46	52	58	64	70
41	47	53	59	65	71
42	48	54	60	66	72

SERVICE WATER PUMPS STANDBY PUMP RUNNING	
Sensor:	114-204 + 152-204/a, 144-205 + 152-205/a, or 144-103 + 152-103/a
Trip Setpoints:	Breaker closed on a pump selected for STANDBY
Alternate Indication:	None

AUTOMATIC FUNCTION:
- Service Water Pump selected for STANDBY starts on low discharge pressure (40 psig) on running Service Water Pump.

OPERATOR ACTION:
- **REFER TO** AOP-35.

FOLLOWUP ACTION:
- **INITIATE** Work Request for troubleshooting/repairs of the tripped Service Water Pump.
- **IMPLEMENT** any applicable Technical Specifications LCO 3.7.8 actions.

REFERENCES:
- AOP-35, "Loss of Service Water"
- Technical Specifications LCO 3.7.8

TITLE: AUXILIARY SYSTEMS SCHEME EK-11 (C-13)

1	7	13	19	25	31
2	8	14	20	26	32
3	9	15	21	27	33
4	10	16	22	28	34
5	11	17	23	29	35
6	12	18	24	30	36

SERVICE WATER PUMP BAY LO LEVEL	
Sensor:	LIA-1338, Service Water Bay Lo Level
Trip Setpoints:	Elevation 574' 3"
Alternate Indication:	North Bay Level - PPC points LIT_1336A and LIT_1336B South Bay Level - PPC points LIT-1337A and LIT-1337B

AUTOMATIC FUNCTION:
- None

NOTES:	• This alarm may annunciate when starting a dilution water pump and would not necessitate an automatic entry into AOP-35. • When icing conditions are evident, then potential exists for frazil ice accumulation at the Service Water Pump Basket Strainers, Fire Pump Suction Strainers, and Secondary Side components.

OPERATOR ACTION:
- **CHECK** North Bay level on PPC points LIT_1336A and LIT_1336B.
- **CHECK** South Bay level on PPC points LIT_1337A and LIT_1337B.
- **MONITOR** Service Water Bay levels.
- **CHECK** Service Water header pressure on PI-1318 and PI-1319.
  - o IF Service Water Bay level is lowering, THEN **TRIP** one operating Dilution Water Pump P-40A or P-40B.
  - o IF Service Water Bay level lowers to 574', THEN **TRIP** remaining operating Dilution Water Pump.
- **REFER TO** AOP-35.
- **REFER TO** Site Emergency Plan EI-1.

FOLLOW UP ACTION:
- **ENSURE** Traveling Screens are backwashing AND operating properly.
- **IMPLEMENT** any applicable Technical Specifications LCO 3.7.9 actions.

REFERENCES:
- EI-1, "Emergency Classification and Actions"
- AOP-35, "Loss of Service Water"
- Technical Specifications LCO 3.7.9

REACTOR AND EQUIPMENT TRIP CRITERIA

Reactor Trip

- Service Water Bay level lowers to 572'
- Operator actions are not maintaining either Critical Service Water Header Pressure greater than or equal to 42 psig
- Loss of Non-Critical Service Water as indicated by the following alarms:
  - EK-1165, "NON CRITICAL SERV WATER LO PRESS" (45 psig)

  AND

  - PPC Urgent Alarm, "EXC FIELD COLD AIR RTD-31" [T_EXCITER_31] (48°C)

    OR

  - PPC Urgent Alarm, "EXC DIODE COLD AIR RTD-32" [T_DIODE_32] (48°C)

    OR

  - EK-0259, "EXCITER COOLER HI TEMP" (50°C)

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078.K4.02	
	Importance Rating	3.2	

K/A Statement: Knowledge of IAS design feature(s) and/or interlocks which provide for the following: Cross-over to other air systems.

Proposed Question:

The Service Air system will isolate from the Instrument Air system when Instrument Air header pressure lowers to what setpoint?

A. 92 psig

B. 85 psig

C. 80 psig

D. 60 psig

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, this is the pressure when the standby IA compressor auto-starts.
B. Correct, the SA system isolates from the IA system by the automatic closure of CV-1212 Service Air Header Isol.
C. Incorrect, this is the Service Air low pressure alarm setpoint.
D. Incorrect, this is the pressure at which certain valves (specifically Aux Feedwater) will be supplied with Nitrogen backup.

Technical Reference(s): AOP-37, DBD-1.05
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

LOSS OF INSTRUMENT AIR

USER ALERT

Read each step of the procedure prior to performing that step. Since the cause and nature of the abnormal condition cannot be predicted, performance of steps out of sequence may be necessary.

1.0 PURPOSE

This procedure provides operator actions that must be accomplished subsequent to a loss of instrument air. Entry into this procedure due to receipt of EK-1103, "Service Air Lo Pressure" or EK-1102, "Instrument Air Lo Pressure" indicates that the operating air compressor(s) is (are) not capable of meeting the system demand.

2.0 ENTRY CONDITIONS

- EK-1101, "CONTAINMENT INSTR AIR LO PRESS" (85 psig)
- EK-1102, "INSTRUMENT AIR LO PRESS" (85 psig)
- EK-1103, "SERVICE AIR LO PRESS" (80 psig)

3.0 EXIT CONDITIONS

- The diagnosis of a Loss of Instrument Air is NOT confirmed

OR

- All applicable steps of this procedure have been completed

4.0 AUTOMATIC ACTIONS

- Standby Instrument Air Compressor(s) starts at 92 psig
- CV-1212, Service Air Header Isol closes at 85 psig in the Instrument Air Header

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103.A2.03	
	Importance Rating	3.5	

K/A Statement: 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: Phase A and B isolation.

Proposed Question:

Given the following conditions:

- The Plant tripped in response to a Loss of Coolant Accident.
- EOP-1.0, “Standard Post-Trip Actions,” is in progress.
- Pressurizer pressure is 1565 psia and lowering.
- Pressurizer level is off-scale low.
- Containment pressure is 4.1 psig and slowly rising.
- Containment Area Rad Monitors indicate:
  - RIA-1805 – 8 R/hr
  - RIA-1806 – 9 R/hr
  - RIA-1807 – 8 R/hr
  - RIA-1808 – 9 R/hr
- Alarm EK-1342, “Safety Injection Initiated,” has annunciated.
- Alarm EK-1126, “CIS Initiated,” has NOT annunciated.

Which one of the following actions is required to be performed based on the above conditions?

A. Ensure operating all Containment Air Cooler ‘A’ and ‘B’ fans.

B. Push left and right “Injection Initiate” pushbuttons.

C. Close both Steam Generator Main Steam Isolation Valves.

D. Ensure open all Containment Air Cooler inlet and outlet valves.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, this is the action to take if containment temperature is > 125°F when a SIAS is not present. Only the Containment Air Cooler ‘A’ fans are started when containment pressure > 4.0 psig.

B. Incorrect, EOP-1.0 requires validating the SIAS initiated alarm is in OR, if the alarm is <u>not</u> in, pushing the left and right pushbuttons to actuate SI when PZR pressure is less than 1605 psia. In this case, a valid SIAS has occurred, as evidenced by the SI Initiated

alarm that is locked in and manually actuating SIAS would not result in any additional action(s).

C. Correct, this action is required when containment pressure setpoint for containment isolation is exceeded (4.0 psig).
D. Incorrect, there is an associated action with containment air coolers when containment pressure is > 0.85 psig, but the action is to ensure open all outlet valves only not inlet valves; air cooler #4 inlet valve will automatically close on a safety injection signal.

Technical Reference(s): EOP-1.0, ARP-7, ARP-8
(Attach if not previously provided, ________________
including version/revision number) ________________

Proposed references to be provided to applicants during examination: ____None____

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam Palisades 2012
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 _10__
55.43 _____

Comments:
Question modified from Palisades 2012 Exam. Modified question stem to provide individual rad monitor readings, used a high containment pressure rather than high containment radiation, and required applicants to determine whether or not a valid Safety Injection occurred. Replaced two distractors and changed overall answer.

7. Containment Isolation

A. The specific plant parameters that will cause an automatic containment isolation are:

1) Containment High Pressure, initiates ≥ 3.7 psig ≤ 4.3 psig

2) Containment High Radiation of 10 R/hr on RIA-1805, 1806, 1807, 1808

3) Refueling Monitors RIA-2316 and RIA 2317

a) Refueling Radiation Monitors are placed in service by placing the Cutout Switches for RIA-2316 & 2317 in the "IN" position using the key switches on panel C-11 rear.

B. EOP Supplement 6 Checklist contains all Containment Isolation Valves for CHR and CHP.

C. Containment High Pressure Logic

1) 8 pressure sensing bellows, each of which operates two independently adjustable micro-switches

2) The switches are set as follows:

a) Pressure Switches 1801, 1802, 1803, 1804

1. 1 contact opens @ ~ 3.4 psig (94.1 inches)

2. 1 contact opens @ ~ 4 psig (110.7 inches)

b) Pressure Switches 1801A, 1802A, 1803A, 1804A

1. Both contacts close > 4 psig

c) To get a CHP Left Channel Signal you must have a 2/4 logic using Pressure Switches 1801, 1802A, 1803, 1804A.

d) To get a CHP Right Channel Signal you must have 2/4 from Pressure Switches 1801A, 1802, 1803A, 1804.

e) Use E-17 logic print, Sheet 6, to show the result of various pressure switch combinations.

3) Equipment Actuation due to CHP

a) CHP Actuation Tables given for LEFT and RIGHT Channel

b) Some equipment actuated from both left and right channels.

c) Relays which actuate equipment are dependent upon power

1. Y-40 for right channel

2. Y-10 for left channel

d) SIAS actuation places the Containment Spray Pumps in "STANDBY".

1. If CHP is Reset, but NOT SIAS, then if another CHP occurs due to high Containment pressure, the Containment Spray Pumps will NOT automatically start.

2. Pushing Right Channel CHP Reset only resets the Right Channel, it DOES NOT RESET the SIAS channel that the CHP actuated.

3. Pushing Left Channel CHP Reset only resets the Left Channel, it DOES NOT RESET the SIAS channel that the CHP actuated.

D. Containment High Radiation Logic

1) Use E-17 Logic Print, Sheet 7, to show the results of various radiation monitor combinations.

a) Pushing either High Rad Initiate P/Bs (CHR-L CS or CHR-R CS) will result in BOTH Left and Right CHR Actuation.

b) Any combination of the RIA-1805, 1806, 1807 or 1808 in a 2 out of 4 coincidence will result in a Left and Right Channel Actuation

5.0 OPERATOR ACTIONS

ACTIONS\EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **VERIFY** PCS Pressure Control. • PZR pressure between 1650 and 2185 psia • PZR pressure trending toward 2010 and 2100 psia	5.1 IF PPCS is malfunctioning, THEN **PERFORM** the following: a. **OPERATE** PPCS in manual. b. **OPERATE** PZR heaters and PZR spray to maintain PZR pressure within the limits of EOP Supplement 1. 5.2 IF PZR pressure is less than 1605 psia, THEN **ENSURE** SIAS is initiated: a. EK-1342, "SAFETY INJ INITIATED," in alarm OR: 1) **PUSH** left and right INJECTION INITIATE pushbuttons on EC-13. • PB1-1 • PB1-2 b. All available HPSI and LPSI pumps operating with the associated loop isolation valves open. 5.3 IF PZR pressure is less than 1300 psia, THEN **STOP** PCPs as needed to establish one PCP operating in each loop. 5.4 IF PZR pressure is less than minimum PCP operation limits of EOP Supplement 1, THEN **STOP** all PCPs.

5.0 OPERATOR ACTIONS

ACTIONS\EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. **VERIFY** Containment Isolation:	
a. Containment pressure less than 0.85 psig. • PIA-1814 • PIA-1815	a.1 IF Containment pressure is greater than or equal to 4.0 psig, THEN **ENSURE ACTUATED** Containment Isolation: 1) EK-1126, "CIS INITIATED," in alarm OR: a) **PUSH** left and right HIGH RADIATION INITIATE pushbuttons on EC-13. • CHRL-CS • CHRR-CS 2) **ENSURE CLOSED** the following: • MSIVs: ○ CV-0510 ('A' S/G) ○ CV-0501 ('B' S/G) • Main Feed Reg Valves: ○ CV-0701 ('A' S/G) ○ CV-0703 ('B' S/G) • Bypass Feed Reg Valves: ○ CV-0735 ('A' S/G) ○ CV-0734 ('B' S/G) • CCW Isolation Valves: ○ CV-0910 (KEY: 337) ○ CV-0911 (KEY: 338) ○ CV-0940 (KEY: 336)
(Continue)	(Continue)

5.0 OPERATOR ACTIONS

<u>ACTIONS\EXPECTED RESPONSE</u>	<u>RESPONSE NOT OBTAINED</u>
8. **(Continued)**	
b. Containment Area Monitor alarms clear and not rising. • RIA-1805 • RIA-1806 • RIA-1807 • RIA-1808	b.1 <u>IF</u> Containment radiation level is greater than 1 x 10^1 R/hr on any Containment Area Monitor, <u>THEN</u> **ENSURE ACTUATED** Containment Isolation: 1) EK-1126, "CIS INITIATED," in alarm <u>OR</u>: a) **PUSH** left and right HIGH RADIATION INITIATE pushbuttons on EC-13. • CHRL-CS • CHRR-CS 2) **CORROBORATE** Containment Area Monitor readings by comparing to Containment High Range Monitor readings. • RIA-2321 • RIA-2322
c. Condenser Off Gas Monitor RIA-0631 alarm clear and not rising.	
d. Main Steam Line Monitor alarms clear and not rising. • RIA-2323 • RIA-2324	

5.0 OPERATOR ACTIONS

ACTIONS\EXPECTED RESPONSE

9. **VERIFY** Containment Atmosphere:

   a. Containment temperature less than 125°F.

RESPONSE NOT OBTAINED

a.1 IF SIAS is NOT present, THEN **ENSURE OPERATING** all available Containment Air Cooler fans.

- V-1A and V-1B
- V-2A and V-2B
- V-3A and V-3B
- V-4A and V-4B

a.2 **ENSURE OPEN** Containment Air Cooler high capacity outlet valves as Service Water System capacity permits:

- CV-0867
- CV-0861
- CV-0864
- CV-0873

(Continue)

(Continue)

5.0 OPERATOR ACTIONS

ACTIONS\EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
9. (Continued)	**(Continued)**
b. Containment pressure less than 0.85 psig. • PIA-1814 • PIA-1815	b.1 IF Containment pressure is greater than or equal to 4.0 psig, THEN **PERFORM** the following: 1) **ENSURE OPERATING** all available Containment Air Cooler 'A' fans. • V-1A • V-2A • V-3A • V-4A 2) **ENSURE OPEN** all available Containment Spray Valves. 3) **ENSURE OPERATING** all available Containment Spray Pumps. 4) **ENSURE** at least minimal acceptable spray flow: • 1 pump 1425 gpm • ≥ 2 pumps 2850 gpm

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001.K1.03	
	Importance Rating	3.4	

K/A Statement: Knowledge of the physical connections and/or cause effect relationships between the CRDS and the following systems: CRDM.

Proposed Question:

Which of the following describes how control rods function on a reactor trip signal?

A. The magnetic clutches and CRDM motors all de-energize, allowing the control rods to fall into the core.

B. The magnetic clutches and CRDM motors all energize, allowing the control rods to be driven into the core.

C. The CRDM motors are energized, driving the control rods into the core. If the rod does not move, the magnetic clutches are de-energized, allowing the rods to fall into the core.

D. The magnetic clutches de-energize, allowing the control rods to fall into the core. The CRDM motors energize, so that if a rod does not drop, the rod is driven into the core.

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, the CRDM motors are energized and the motor will rotate in the downward direction, forcing the rods into the core if the rods do not drop due to the de-energizing of the magnetic clutches.
B. Incorrect, de-energizing the magnetic clutch drops the control rod into the reactor by allowing the interconnecting pinion gears and drive shaft to rotate freely as the rod free falls.
C. Incorrect, if the rod does not drop due to de-energizing the magnetic clutches, then the CRDM motors and brakes energize, rotating downward and forcing the rods into the core.
D. Correct, de-energizing the magnetic clutch drops the control rod into the reactor by allowing the interconnecting pinion gears and drive shaft to rotate freely as the rod free falls. If the rod does not drop, the CRDM motors and brakes energize, rotating downward and forcing the rods into the core.

Technical Reference(s): PL-CRD Control Rod Drive System Lesson Plan
(Attach if not previously provided, __________
including version/revision number) __________

Proposed references to be provided to applicants during examination: ______None________

Learning Objective: ________________________ (As available)

Question Source:

Bank # _______

Modified Bank # _______ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:

Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.41 __6__

55.43 _____

Comments:

Lesson Content	Instructor Notes

OBJ 3: Describe the design feature and interlock that provides for automatic rundown of control rods (1-41) after a Reactor trip.

7. When a reactor trip signal is present, the CRDM motors and the brakes will energize and the motor will rotate in the downward direction, forcing the control rod into the core. This would be used in case the rod does not drop for some reason; the motor can push the rod down by driving through an anti-reverse clutch. The motor cannot drive the rod up under these conditions since the anti-reverse clutch will transmit torque only in the drive down direction. This would be necessary in the situation if gravity were not sufficient such as if the control rod were to get stuck due to excessive friction or possibly a LOCA in the reactor head region with large flow velocities in the upward direction.

OBJ 2: Describe the operational design of the Control Rod Drive Magnetic Clutch.

CRD-VA-18, 19, 20 – Magnetic Clutch

9. Magnetic Clutch
   a) Magnetic clutch, when energized, connects the motor to the vertical drive shaft to allow raising and lowering of:
      1. Shutdown rods (Groups A and B)
      2. Regulating rods (Groups 1, 2, 3 and 4)
   b) When de-energized, the drive shaft is disconnected from the motor and brake, allowing the shutdown and regulating rods to fall into the core by gravity.
   c) Part-length control rods **DO NOT** have magnetic clutches, therefore they do NOT trip.

d) The magnetic clutch has an anti-reverse feature. The anti-reverse feature functions when the clutch is de-energized. Its function is to ensure a control rod is forced down into the core by the motor if necessary, in the event of a reactor trip. It also prevents upward forces from forcing the rod out of the core, during this time also.

   1. The anti-reverse feature of the clutch has "pawl" like mechanisms, which allow rotation only in the downward direction once the clutch is disengaged.

   2. Additionally, once the control rod has reached the bottom after a reactor trip, a limit switch (which we will learn more about later) de-energizes the motor and the brake, resulting in the brake becoming engaged. Then the anti-reverse feature along with the brake prevents any outward motion of the control rod.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	014.K3.02	
	Importance Rating	2.5	

K/A Statement: Knowledge of the effect that a loss or malfunction of the RPIS will have on the following: Plant computer.

Proposed Question:

Control rod position will NOT be available from the Palisades Plant Computer (PPC) if, at a minimum, which of the following is/are lost:

Note: Primary Indication Position (PIP), Secondary Position Indication (SPI).

A. ONLY the PIP is lost.

B. ONLY the SPI is lost.

C. BOTH the PIP and SPI are lost.

D. ONLY the “RODMON” program is lost.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, both PIP and SPI can provide rod position to the PPC. While losing the PIP would result in the loss of a large amount of control rod monitoring capabilities, the SPI node would still be able to provide adequate indication of Control Rod position.
B. Incorrect, both PIP and SPI can provide rod position to the PPC. By losing the SPI only, the following indications are lost: 1) Secondary Control Rod Position Indication, 2) redundant PPDIL and PDIL alarms on the PPC, 3) redundant 4” and 8” deviation alarms on the PPC.
C. Correct, both PIP and SPI would need to be lost to lose PPC indication for rod position.
D. Incorrect, both PIP and SPI can provide rod position to the PPC. “RODMON” uses inputs from PIP both SPI. "RODMON" will use the synchro rod positions (since they are more accurate) and the Loop 2 ΔT (from PIP) as long as they are valid. If they go invalid, “RODMON” will swap over to using the reed switch rod positions (SPI inputs) and/or Loop 1 ΔT (SPI input).

Technical Reference(s): PL-CRD Control Rod Drive System Lesson Plan, SOP-6
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: ______None______

Learning Objective: ________________________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __6__

55.43 _____

Comments:

Lesson Content	Instructor Notes

PIP (PRIMARY INDICATION POSITION)

A. Palisades Plant Computer (PPC) Interface (PIP). The PIP software does calculations related to the Control Rods on the basis of analog and digital inputs and, as a result of these calculations, generates digital outputs and provides analog and digital data to the PPC host system.

1. Analog Inputs:
   - 45 synchro positions
   - Loop 2 delta temp
2. Digital Inputs:
   - 45 rotary switch positions
   - Rod drop timing start
3. Analog Outputs:
   - 45 rod position (in inches) to host
   - Loop 2 delta temperature "core power"
   - Rod drop timing data
4. Digital Outputs:
   - Rod Control Relays
   - Annunciators
   - Digital Rod displays

CRD-VA-75 – PIP Processor – Target Rod

B. The PIP will put "-199.9" in place of a rod position when it detects a bad synchrotransmitter to digital converter.

1. Signal from synchro not good
2. Switch contacts not made up or more than one made up

C. Group Target Rod Processing

1. The PIP software determines the 7 group target rod selections given the input from the Rod Control Selector Switches.

2. The target rod positions are provided internally to the PPC for use in Manual Sequencing (MS) and in determining the Power Dependent Insertion Limit (PDIL), and Pre-Power Dependent Insertion Limit (PPDIL).
3. If the target rod position is determined to be "bad", PPC processing is suspended for the programs using the “bad” group target rod.
   a) No contacts or more than one contact closed on switch

CRD-VA-76 – PIP Processor – Upper Rod Stop

D. Upper Rod Stop Processing
   1. The PIP software determines the state of the 45 sets of upper rod stop (URS) contacts.
   2. This processing is suspended for any rod with an invalid position (bad synchrotransmitter).
   3. The URS contacts stay "as is" if either the PIP loses power or if the synchrotransmitter fails.
   4. The URS contacts are used in the following:
      a) Core Matrix light logic
      b) Manual Sequential (MS) RAISE logic
      c) Manual Group (MG) regulating and power shaping rod RAISE logic

CRD-VA-77 – PIP Processor – Lower Rod Stop

   5. Lower Rod Stop Processing
      a) The PIP software determines the state of the 25 lower rod stop (LRS) contacts for the rods in the regulating and power shaping rods.
      b) This processing is suspended for any rod with an invalid position (bad synchrotransmitter).
      c) The LRS contacts are used in the following:
         1) Manual Sequential (MS) LOWER logic
         2) Manual Group (MG) LOWER logic

CRD-VA-78 – PIP Processor – Upper/Lower Sequential Permissive

   6. Upper Sequential Permissive Processing

a) The PIP software determines the state of the 3 upper sequential permissive (USP) contacts based on the position of the group target rods for regulating groups 1, 2, and 3.

b) The USP contacts are used in Manual Sequential (MS) RAISE logic.

7. Lower Sequential Permissive Processing

a) The PIP software determines the state of the 3 lower sequential permissive (LSP) contacts based on the position of the group target rods for regulating groups 2, 3, and 4.

b) The LSP contacts are used Manual Sequential (MS) LOWER logic.

CRD-VA-79 – PIP Processor – 4 and 8 inch Deviation

8. Four Inch Deviation Processing

a) The PIP software determines the state of the 7 four inch group deviation contacts

b) If the highest rod in a group is more than 4 inches from the lowest rod in the group, the four-inch deviation contact for that group is opened.

c) If the deviation is less than 4 inches minus the deadband, the contact is closed.

d) The contacts are used to initiate the annunciator EK-0911 "Rod Position 4 inches Deviation" and to illuminate the appropriate group deviation light on C-02.

9. Eight Inch Deviation Processing

a) The PIP software determines the state of the 7 eight inch group deviation contacts

b) If the highest rod in a group is more than 8 inches from the lowest rod in the group, the eight-inch deviation contact for that group is opened.

c) If the deviation is less than 8 inches minus the deadband, the contact is closed.

d) The contacts are used to initiate the annunciator EK-0912 "Rod Position 8 inches Deviation" and to illuminate the appropriate group deviation light on C-02.

CRD-VA-80 – PIP Processor – Out of Sequence

10. Out of Sequence Processing

a) The PIP software determines the state of the out of sequence (OOS) contact based on the positions of the group target rods for the four regulating groups. This processing is intended to limit the ways in which the operators can move the control rods of the regulating groups.

b) If any of the regulating group target rods have bad synchrotransmitters, no OOS processing is completed between adjacent rod groups.

c) If an OOS condition is detected, annunciator EK-0916 "Control Rods Out-Of-Sequence" is received.

CRD-VA-81 – PIP Processor - PDIL

11. Power Dependent Insertion Limit (PDIL) Processing

a) The PIP software calculates power dependent insertion limits (PDIL) for each regulating group and determines the state of four PDIL contacts based on the position of the group target rod.

b) If the core power signal is invalid, the PDIL alarm for regulating group 1 is alarmed (opened) to annunciate this fact.

c) Core power is generated using the TDY_0200 analog input.

d) If a PDIL condition is detected, annunciator EK-0924 (Group 1), EK-0930 (Group 2), EK-0936 (Group 3), or EK-0942 (Group 4) will be received.

CRD-VA-82 – PIP Processor - PPDIL

12. Pre-Power Dependent Insertion Limit (PPDIL) Processing

a) The PIP software calculates the four pre-power dependent insertion limits (PPDIL) and determines the state of the four PPDIL contacts based on the position of the group target rod for the corresponding four regulating groups.

b) The calculations for PPDIL utilize the fact that PDIL has already been calculated. Since the PPDIL curves are offset by a fixed amount from the PDIL curves using the same core power estimate, the PPDIL can be quickly calculated by adding an offset to the PDIL. In addition, the PPDIL is clamped at a maximum value.

c) If a PPDIL condition is detected, annunciator EK-0923 (Group 1), EK-0929 (Group 2), EK-0935 (Group 3), or EK-0941 (Group 4) will be received.

CRD-VA-83 – PIP Processor – Watchdog Timer/Rod Drop Timing

13. Watchdog Timer Processing

a) The PIP software must also maintain the watchdog timer, which is implemented using a programmable delay timer card. If the software for any reason fails to update this watchdog timer or power is lost, the watchdog will cause EK-0918, PIP Trouble, alarm to annunciate.

14. Rod Drop Timing

a) This mode allows the operators to acquire a drop time profile of a rod under test. During this mode, all normal PIP activities are suspended (except for watchdog timer) and the PIP system is used exclusively for acquiring the profile of the rod drop. Even normal rod position indications will not change during the test.

D. Secondary Position Indication

1. The reed switches measure control rod position by use of control rod actuated magnetic reed switches. They transmit rod height signals to the secondary position indication and rod matrix light display.

CRD-VA-48 – SPI

2. An assembly containing a number of series resistors to form a voltage divider network with reed switches (approximately 2 inches apart) connected at each junction is attached to the control rod extension housing. A voltage is applied to the network; output voltage depends on which reed switches are closed in the voltage divider. A magnet on top of the control rod extension will close the reed switches as the control rod moves. Overlap between adjacent reed switches is provided. The output is a voltage directly proportional to control rod position.
3. Output voltage is provided to the Secondary Position Indication (SPI) Node, which interprets the output voltage and provides position indication to the PPC.

1. EK-0971, “SPI TROUBLE”
    a) The PPC SPI Node resets a “Watchdog Timer” every couple of seconds. A failure of the SPI Node will cause the “Watchdog Timer” to not be reset, which will bring in the alarm.
    b) If the SPI Node is lost the following Control Rod Drive System functions are affected:
        1) The Secondary Control Rod Position Indication is lost.
        2) The redundant channel of the Control Rod Out of Sequence alarm, which alarms on the PPC, is lost.
        3) The redundant PPDIL and PDIL alarms, which alarm on the PPC, are lost.
        4) The redundant 4 inch and 8 inch deviation alarms, which alarm on the PPC, are lost.
        5) The PPC status page and Control Rod page can be checked to help validate this alarm.

10. EK-0918, “PIP TROUIBLE”

CRD-VA-114 – EK-0918, PIP Trouble

    a) If the PIP is not functioning and does not reset its “Watchdog Timer”, this alarm will come in.

b) If the PIP is not functioning, all the Control Rod functions performed by the PIP will be lost. See the System Interrelationships section for more detail on the impact of a loss of the PIP.

c) Check the PPC status page as well as the Control Rod page on the PPC to validate this alarm.

A. On the PPC a program called “RODMON” is performing all the same calculations as the PIP; however, it does NOT interface with rod control, annunciators or the LEDs.

1. "RODMON" will use the synchro rod positions (since they are more accurate) and the Loop 2 ΔT (from PIP) as long as they are valid. If they go invalid, “RODMON” will swap over to using the reed switch rod positions (SPI inputs) and/or Loop 1 ΔT (SPI input).

2. “RODMON” will determine and display the following and will provide an audible PPC alarm for those conditions marked with an *:

   a) 45 validated rod positions

   b) 7 validated Group Target Rod (GTR)

   c) Upper Rod Stop (URS)

   d) Lower Rod Stop (LRS).

   e) Shutdown Rod Insertion (SRI)*

   f) Upper Sequential Permissive (USP)

   g) Lower Sequential Permissive (LSP)

   h) Four Inch Deviation

   i) Eight Inch Deviation*

   j) Out of Sequence (OOS)*

   k) Power Dependent Insertion Limit*

   l) Pre-power Dependent Insertion Limit (PPDIL)

Examination Outline Cross-Reference:

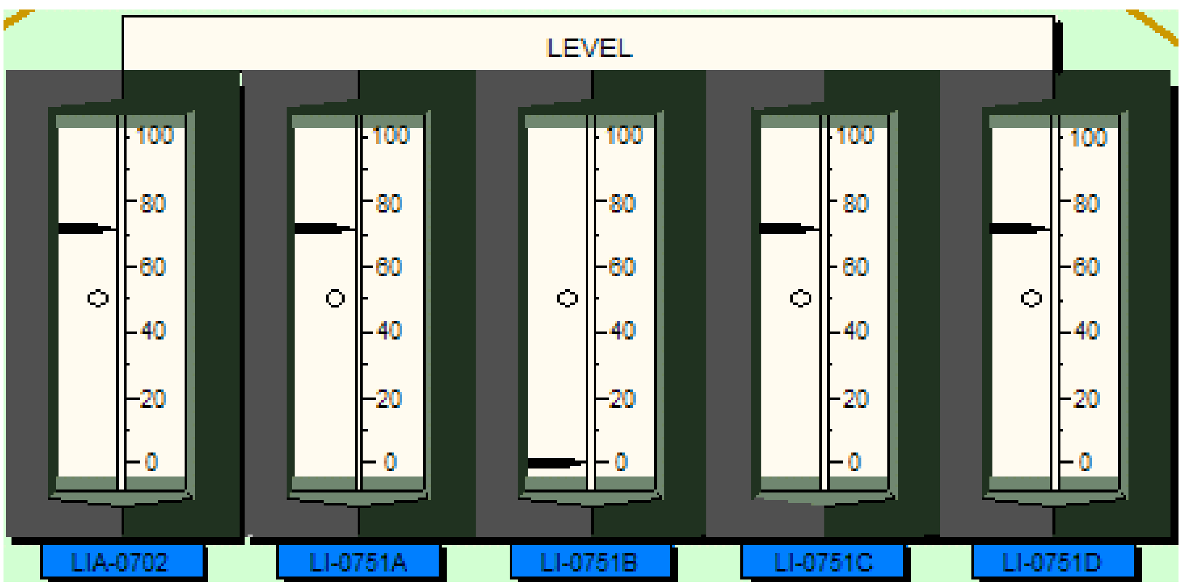
		RO	SRO
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	016.K5.01	
	Importance Rating	2.7	

K/A Statement: Knowledge of the operational implication of the following concepts as they apply to NNIS: Separation of control and protection circuits.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power.
- RPS Channel B for 'A' Steam Generator Low Level is BYPASSED due to a failure of LI-0751B, Steam Generator E-50A Low Level Indicator.
- Refer to the below graphic of 'A' Steam Generator level instrumentation.



Which one of the following additional instrument failures will result in a Reactor trip? (Assume no operator action.)

A. LI-0751A, Steam Generator E-50A Low Level Indicator, fails LOW.

B. LI-0751A, Steam Generator E-50A Low Level Indicator, fails HIGH.

C. LIA-0702, Steam Generator E-50A Level Alarm Indicator, fails LOW.

D. LIA-0702, Steam Generator E-50A Level Alarm Indicator, fails HIGH.

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, there are now 2 channels that feed the RPS that exceed a setpoint, however, a trip will not be processed because RPS channel 'B' is bypassed.
B. Incorrect, there are now 2 channels that feed the RPS that exceed a setpoint, however, RPS channel 'B' is bypassed and there is no RPS trip for high S/G level.
C. Incorrect, there are now 2 channels that feed the RPS that exceed a setpoint, however, RPS channel 'B' is bypassed and LIA-0702 does not provide an input to RPS.
D. Correct, if LIA-0702 fails high, the feed regulating valve associated with 'A' S/G would close which would cause an actual low level condition (a high level override from LIA-0702 closes CV-0701 'A' FRV). An RPS trip would be generated from the remaining 3 channels that are not bypassed.

Technical Reference(s): ARP-5, FSAR 7.2, page 7.2-2, 7.2-9
(Attach if not previously provided, __________
including version/revision number) __________

Proposed references to be provided to applicants during examination: None

Learning Objective: __________ (As available)

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

Question History: Last NRC Exam Palisades 2007
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

TITLE: PRIMARY COOLANT PUMP STEAM GENERATOR AND ROD DRIVES SCHEME EK-09 (C-12)

37	43	49	55	61	67
38	44	50	56	62	68
39	45	51	57	63	69
40	46	52	58	64	70
41	47	53	59	65	71
42	48	54	60	66	72

STEAM GEN E-50A HI LEVEL	
Sensor:	LIA-0702, Steam Generator E-50A Level Alarm Ind
Trip Setpoints:	84.7%
Alternate Indication:	Steam Generator level indications on EC-12

AUTOMATIC FUNCTION:
- High Level Override from LIA-0702 closes CV-0701, E-50A Feed Regulating Valve.

OPERATOR ACTION:
- CHECK Steam Generator level.
- IF Steam Generator level is high, THEN REFER TO AOP-3.
- IF Steam Generator level(s) are dropping, THEN:
  - o TRIP Reactor (a fault exists in High Level Override circuit).
  - o REFER TO AOP-3.

FOLLOW UP ACTION:
- INITIATE Work Request for troubleshooting/repairs as required.

REFERENCES:
- AOP-3, "Main Feedwater Transients"

Finally, a set of annunciators (non-class 1E) are also located on the above cabinets for operator convenience. An extension of the RPS is housed in an additional panel, also in the control room. This panel contains:

Four (4) Thermal Margin Monitors

Four (4) Reactor Power Calibration and Indication Assemblies

Another panel in the control room (C12) contains:

Four (4) pressure switch alarms (dual output each, one for PORV logic, one for ATWS logic)

7.2.2 DESIGN BASES

The RPS is designed under the following bases to assure adequate protection for the reactor core:

1. Instrumentation and controls for this Plant conform to the provisions of the General Design Criteria as indicated in Section 5.1 and to IEEE 279-1971.

2. No single component failure will prevent safety action.

3. Four independent measurement channels complete with sensors, sensor power supply units, amplifiers and trip units are provided for each safety parameter with the exception of loss of load and rate trips.

4. The channels are provided with a high degree of independence by separate connection of the sensors to the process systems and of the channels to preferred power supply buses. Separate raceways are used to ensure independence from cable faults.

5. The four normal measurement channels provide trip signals to four independent trip paths.

6. A trip signal from any two-out-of-four protective channels causes a reactor trip.

7. When one of the four channels is taken out of service for maintenance, the protective system logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel assumes a tripped condition, which results in a one-out-of-three channel logic.

8. The protective system ac power is supplied from four separate buses.

A key-operated bypass switch ("Zero Power Mode Bypass" switch, see Subsection 7.2.5.2) allows this trip to be bypassed at low power level. The trip bypass is automatically reset above 10^{-4}% full power.

7.2.3.6 Loss of Load

A reactor trip will automatically be initiated after a turbine trip occurs. A turbine low auto stop oil condition occurs with all types of turbine trips. The reactor trip will be initiated when the turbine auto stop oil pressure decreases, causing contacts in the auto stop oil pressure switches to close and, via two out of three logic, energize two turbine trip auxiliary relays (see Figure 7-14, Sheet 2).

Each relay will provide a reactor trip signal to two of four protective system channels.

The loss of load reactor trip is an anticipatory trip which is not required to protect the reactor since the primary trip is high primary system pressure. As such, its measuring channels components are not required to be Class 1E and its circuits need not meet IEEE 279-1971. This trip is automatically bypassed when three of four power range safety channels indicate power is below 17% full power (see Subsection 7.2.5.2).

Isolation of Nonclass 1E turbine trip circuits from the Class 1E Reactor Protective System is provided by the turbine trip relays (see Subsection 7.2.9).

7.2.3.7 Low Steam Generator Water Level

Low steam generator downcomer water levels will cause a loss-of-heat-removal capability from the Primary Coolant System.

A reactor trip signal is initiated by two-out-of-four logic from four independent downcomer level differential pressure transmitters on each steam generator. A 25.9% narrow range minimum trip setting assures that the heat transfer surface (tubes) are covered with water when the reactor is critical. The 25.9% level corresponds to the location of the feed ring. Pretrip alarms are actuated to provide for annunciation of approach to reactor trip conditions.

7.2.3.8 Low Steam Generator Pressure

A reactor trip on low steam generator secondary pressure is provided to protect against excessively high steam flow caused by a steam line break. The trip set point is ≥ 500 psia.

An abnormally high main steam flow from either steam generator will cause the secondary pressure to drop rapidly.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	029.A1.02	
	Importance Rating	3.4	

K/A Statement: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Containment Purge System controls including: Radiation levels.

Proposed Question:

Given the following conditions:

- Core offload has just begun during a refueling outage.
- The Air Space Purge Fan, V-46 is running with the Containment Purge Supply Valves (CV-1813 and CV-1814) open, per SOP-24, "Ventilation and Air Conditioning System."
- Containment background radiation is 15 mR/hr.

The following readings were just noted on the Containment Refueling Radiation Monitors:

- RIA-2316: 98 mR/hr
- RIA-2317: 76 mR/hr

With the given conditions, what is the status of the Containment Purge system and why?

A. In-progress; the setpoint of 80 mR/hr has not been reached on 2/2 Containment Refueling Monitors.

B. In-progress; the setpoint of 80 mR/hr above background radiation has not been reached on 2/2 Containment Refueling Monitors.

C. Isolated; the setpoint of 80 mR/hr above background radiation has been reached on 1/2 Containment Refueling Monitors.

D. Isolated; the setpoint of 80 mR/hr has been reached on 1/2 Containment Refueling Monitors.

Proposed Answer: **C**

Explanation (Optional):

The applicant must both the setpoint and coincidence for a Containment High Radiation (CHR) condition during refueling operations. A CHR during refueling operations occurs when 1 of 2 Containment Refueling Monitors exceeds 80 mR/hr above the background radiation level. In this case, only one rad monitor would need to exceed 90 mR/hr to actuate a CHR, which RIA-2316 has.

A. Incorrect, applicant assumes both an incorrect setpoint and an incorrect coincidence for actuation.
B. Incorrect, applicant assumes an incorrect coincidence for actuation.
C. Correct, see explanation above.
D. Incorrect, applicant assumes an incorrect setpoint for actuation.

Technical Reference(s):	PL-RMS Radiation Monitoring System Lesson Plan
(Attach if not previously provided,	___
including version/revision number)	___

Proposed references to be provided to applicants during examination: ___None___

Learning Objective: ___ (As available)

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

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	5
	55.43	___

Comments:

Containment Refueling Monitors RIA-2316/2317

The monitors are energized and calibrated prior to starting core alterations; they are de-energized at the conclusion of Mode 6 activities.

Each channel has a RIA in the Control Room; the RIAs are identical to the typical digital ratemeters

The Containment Refueling Monitors provide inputs to the Containment Isolation logic during refueling operations (containment isolation logic is described later in Section **Error! Reference source not found.**, **Error! Reference source not found.**.)

RE-2316 and RE-2317 are removed and stored after Mode 6 operations.

Each monitor has a key switch located on the back of Panel C-11.

VA-1 RIA-2316 Key Switch RF-1

The key switches, RF-1 and RF-2, require Keys 54 and 55 for channels RIA-2316 and 2317, respectively.

When the key switch is in the "IN" position, the associated monitor inputs to a 1/2 high alarm logic to initiate containment isolation.

The keys are removable only in "OUT" position.

The high alarm setpoint for RIA-2316 and RIA-2317 is 80 mR/hr above background.

Containment High Radiation (CHR)

A CHR signal is generated by the following area radiation monitors:

2/4 Containment Radiation Monitors RIA-1805/1806/1807/1808 High (Trip 2)

Setpoint 10 R/hr

1/4 channels High (Trip 2) actuates Control Room alarm EK-1363, "CONTAINMENT HI RADIATION."

1/2 Containment Refueling Monitors RIA-2316/2317 High with associated keyswitch in "IN" position.

Setpoint 80 mR/hr above background

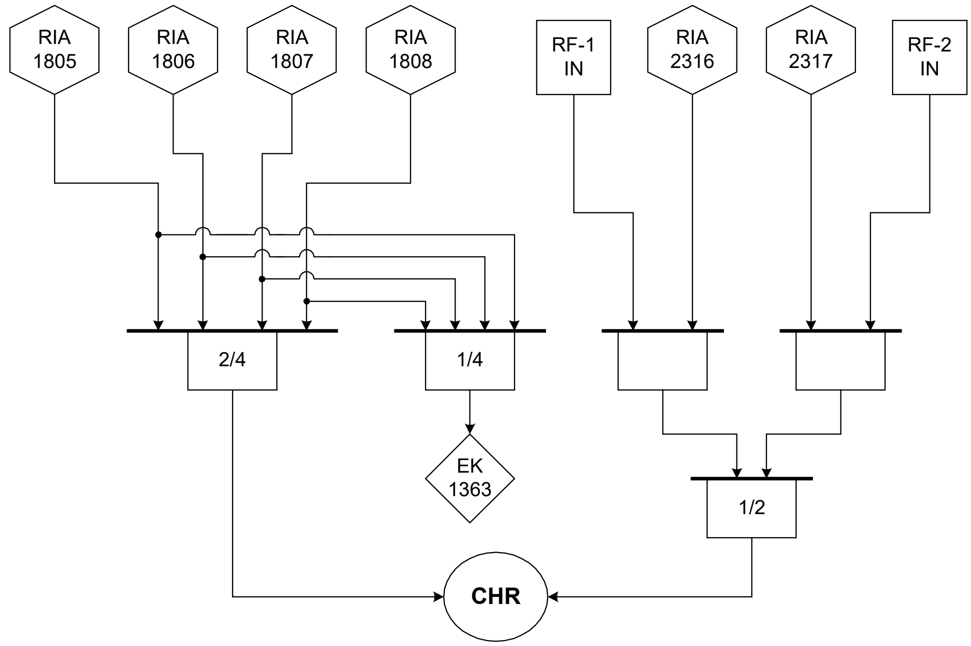


Figure 1. CHR Logic

The CHR signal initiates the following:
Actuates Containment Isolation
Containment Isolation actuated from CHR does not close CCW supply to containment valves (these are closed on Containment High Pressure signal).
Trips Air Room Purge Fan V-46.
Actuates Control Room HVAC Emergency System (Trains A and B)
Disables automatic operation of Safeguards Rooms Sump Pumps (P-72A, 72B, 73A, 73B)
The pumps can be operated in manual mode.
Actuates Control Room alarm EK-1363, “CONTAINMENT HI RADIATION” only when initiated by RIA-1805/1806/1807/1808.
Actuates Control Room alarm EK-1126, “CIS INITIATED.”
c. CHR Signals must be manually reset at Panel C-13.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035.K6.01	
	Importance Rating	3.2	

K/A Statement: Knowledge of the effect of a loss or malfunction on the following will have on the S/Gs: MSIVs.

Proposed Question:

Given the following conditions:

- The Plant is at 24% power.
- The Main Generator is synchronized to the grid.
- A single Main Steam Isolation Valve closes on a spurious signal.

Assuming the reactor does NOT trip, which ONE of the following correctly describes the <u>initial</u> response of S/G Pressure and Level in the affected loop?

	<u>S/G Pressure</u>	<u>S/G Level</u>
A.	Rises	Rises
B.	Lowers	Rises
C.	Lowers	Lowers
D.	Rises	Lowers

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, part 1 is correct, part 2 is incorrect. The applicant believes SG level will rise due to the loss of steam flow while maintaining feedwater flow. This is an incorrect initial response which does not take into account the SG shrink/swell effect.
B. Incorrect, this is an incorrect initial response which does not take into account the SG shrink/swell effect. The applicant believes heat is still removed from the SG or SG pressure spikes causing an ASD to open to lower pressure. This is not the initial response. Also, the applicant believes SG level will rise do to the loss of steam flow while maintaining feedwater flow.
C. Incorrect, part 1 is incorrect, part 2 is correct. The applicant believes heat is still remove from the SG or SG pressure spikes causing an ASD to open to lower pressure. This is NOT the initial response.
D. Correct, SG pressure initially rises due to heat no longer being removed from the SG. With this pressure increase, the SG level will lower (shrink) due to the saturation

pressure rising.

Technical Reference(s): EOP-2.0 Basis
(Attach if not previously provided, __________
including version/revision number) __________

Proposed references to be provided to applicants during examination: None

Learning Objective: __________ (As available)

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

Question History: Last NRC Exam Turkey Point 2011
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

As a result of the Reactor trip initiation, the control rods will be rapidly inserted. Steam flow to the Main Turbine will be terminated and the Main Generator output breakers will open. A rapid decrease in Reactor power and a negative startup rate will be observed. This rapid decrease is followed by a decrease in indicated power (approximately -1/3 decades per minute) until the subcritical multiplication level is reached. Indicated power will stabilize at the subcritical multiplication level and decrease slowly over a period of hours.

Initially, Feedwater temperature decreases sharply due to the loss of steam heating to the Feedwater heaters or due to actuation of Auxiliary Feedwater. Heat from the PCS is absorbed by the cooler Feedwater supplied to the Steam Generators (S/Gs). At power, there is a large differential between PCS T_{AVE} and average S/G temperature. Following the trip of the Reactor and Turbine, the heat transfer rate from the PCS to the S/G decreases to decay heat removal and the PCS to S/G ΔT decreases to a few degrees. As a new equilibrium is achieved, the combined effect of the cooler Feedwater and the S/G heating up to an average temperature closer to PCS temperature results in a net heat extraction from the PCS. Loop differentials between hot and cold leg temperatures will drop to less than ten degrees and PCS T_{AVE} will decrease to 532°F controlled by the Turbine Bypass System.

For an uncomplicated Reactor trip, it is expected that the Reactor Vessel will remain full. The subcooled margin in the PCS loops is typically 50°F or higher, and Reactor Vessel Upper Head (RVUH) subcooling margins can be significantly lower than that for the PCS loops but still high enough to prevent voids from forming. At steady state conditions, the upper head region is about 1°F cooler than the core exit temperature and, therefore, the subcooled margin of the RVUH is essentially equal to that of the hot leg. Under transient conditions, with PCPs running, there is a time lag between the change in the core exit temperature and the change in RVUH temperature to approximately the same temperature.

Pressurizer pressure and level will initially decrease due to the lowering of PCS temperature and resultant inventory shrinkage. However, this effect will usually be tempered by operation of Pressurizer heaters and Charging Pumps to restore level to the programmed hot zero power band.

S/G pressure will usually increase. Since heat is being removed from the PCS but not from the S/G (except for the cooling from the Feedwater), the S/G heats up to decrease PCS to S/G differential temperature. S/G pressure increases as temperature increases. As S/G pressure increases, the Turbine Bypass Valve and/or Atmospheric Steam Dump Valves will usually open to control S/G pressure at hot standby pressure (which is above normal 100% power S/G pressure).

After a Reactor trip the S/G level decreases rapidly. This is explained as follows. Steam generator level is inferred from the S/G downcomer level. During normal 100% power operation, each S/G has a recirculation ratio of approximately 4 to 1 (ratio of water returning to the downcomer from the dryers and separators to Feedwater entering the downcomer). This accounts for a major portion of the water level entering the downcomer. When steam flow is stopped by the Turbine trip, recirculation stops. The reduced flowrate into the downcomer results in reduced head losses through the downcomer and up the riser section. The downcomer water level, and thus the S/G indicated level, both drop. This drop in level will occur even before the Feedwater system automatically readjusts.

Plant operators should not overreact to this lowered level in the S/Gs. Excessive feeding of the S/Gs with cooler Feedwater to recover level results in PCS temperatures being driven down below the desired no load value. This could cause Pressurizer level to fall to a point where the Pressurizer is drained. PCS pressure will then drop until the safety injection system is actuated. This complicates the recovery from a simple Reactor trip considerably.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041.A4.06	
	Importance Rating	2.9	

K/A Statement: Ability to manually operate and/or monitor in the control room: Atmospheric relief valve controllers.

Proposed Question:

Given the following conditions:

- The Plant is at 3% power.
- Main Turbine trip testing was completed and all Main Turbine protective relays have NOT been reset.
- PCS temperature is 532°F.
- HIC-0780A, Steam Dump Control, is in AUTO.
- PIC-0511, Steam Bypass Pressure Controller, is in MANUAL with 0% demand.

If reactivity is added to the core to cause PCS temperature to rise, at what PCS temperature will steam flow stop the temperature rise?

A. 532°F

B. 535°F

C. 540°F

D. 545°F

Proposed Answer: C

Explanation (Optional):

A. Incorrect, applicant incorrectly assumed the TBV will use the steam pressure signal to maintain its setpoint of 900 psia (saturation temperature of 532°F). This signal is overriden by the controller being in MANUAL.
B. Incorrect, applicant incorrectly assumed the ADV would open at an incorrect temperature of 535°F. At 535°F, on a decreasing temperature, the modulate signal is removed and open ADVs will close, to allow for the TBV to control at its 900 psia main steam header pressure setpoint.
C. Correct, the ADVs and TBV will modulate open at 540°F to control temperature. The ADVs will open at 540°F, the deadband value for the valves to open on increasing temperature from HIC-0780A. This 8°F deadband for the ADVs and TBV to open is to allow the TBV to open and maintain main steam header pressure at 900 psia, however the TBV will not open with PIC-0511 in MANUAL at 0%. The output from HIC-0780A to

the TBV and the ADVs remains enabled by not resetting the 386/AST relay (turbine trip signal still present).

D. Incorrect, the applicant incorrectly assumed the ADVs and/or the TBV will not open and that temperature will increase until pressure causes the first bank of safeties to open (approximately 1000 psia; saturation temperature of 545°F).

Technical Reference(s): PL-MSS Main Steam System
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ______None______

Learning Objective: ________________________ (As available)

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

Question History: Last NRC Exam Palisades 1999
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Modified question stem to provide information about main turbine testing and turbine trip relays have not been reset. Changed correct answer and changed one distractor.

Lesson Content	Instructor Notes

1. HIC-0780A, Steam Dump Controller inputs

   a. T_{AVE} from the T_{AVE}/T_{REF} Controller

      T_{AVE}, - T_{REF} Calculators TYT-0100 and TYT-0200 provide the T_{AVE} signal input to the steam dump controller.

   b. Both modes (Modulate and Quick Open) of operation require a turbine trip signal to cause the ADVs to open.

      1) 386 AST relay arms the quick open function.

      2) 386X1 AST relay arms the modulate function.

   c. Upon a Turbine trip and receipt of the 386 AST relay actuation, contacts close in the ADV circuitry. When these contacts close, the controller then uses T_{AVE} to open/close the ADVs.

2. Quick Open Mode

   a. When a turbine trip causes the 386 AST relay to energize, a quick open signal is generated.

   b. If T_{AVE} is ≥ 556.9°F, the steam dump control relay (SDCR) is energized and closes contacts to align the quick open air supply solenoids to the ADV valve actuators and the TBV to open the valves fully.

   c. The ADVs and TBV will stay full open until T_{AVE} is less than 556.9°F.

   d. When T_{AVE} lowers to less than 556.9°F the SDCR will de-energize and remove the quick open function.

   e. The modulating mode will then control the ADVs.

3. Modulate Mode

   a. When a turbine trip causes the 386X1 AST relay to energize, a contact is closed to arm Steam Dump Controller HIC-0780A.

   b. Steam Dump Controller HIC-0780A will modulate the ADVs and TBV based on a T_{AVE} – T_{REF} Error Signal.

   c. The error signal is developed by comparing actual T_{AVE} to 532°F (Reference No-load Value).

   d. The control system will modulate the ADVs and TBV from full open when T_{AVE} is at 556.9°F (25°F error) to full closed at 535°F (3°F error).

   e. For increasing T_{AVE} , the control system will modulate the ADVs and TBV from full closed when T_{AVE} is at 540°F (8°F error) to full open at 556.9°F (25°F error).

f. The 3°F error offset on decreasing T_{AVE} and the 8°F error offset on increasing T_{ave} allows Turbine Bypass Valve CV-0511 and Controller PIC-0511 to control T_{AVE}.

B. Turbine Bypass Valve CV-0511

1. The 6 inch, air operated, automatically actuated turbine bypass valve (TBV) has a capacity of approximately 4.5% of rated steam flow (508,000 lbm/hr).

2. The turbine bypass provides reactor decay heat removal following reactor shutdown and provides PCS cooldown capability.

3. The turbine bypass valve may be manually operated from the Control Room.

4. The TBV has a red (open) and a green (closed) position indicating lamps on C-01.

a. As the TBV is opened, the GREEN closed light stays illuminated until PIC-0511, TBV Controller, output signal shows 25% open. At 25% output signal from the controller until the TBV is full open, both the RED and GREEN light stay illuminated. When the TBV is full open the GREEN light goes out.

b. This is different from the ADVs as the ADV valve position lights are both off when the ADVs are in an intermediate valve position.

5. The TBV is normally lined up to control in automatic from either the Steam Dump controller signal or main steam header pressure signal, whichever is greater.

6. CV-0511 is controlled by PIC-0511, which receives its signal from PT-0510.

a. PT-0510 is located on the steam line to the STOP valves.

C. PIC-0511 Turbine Bypass Valve Controller Operation

1. Controller is a typical Yokogawa controller and similar to the ADV Controller with respect to controller features.

2. Auto – Controls steam header pressure at selected set point (900 psia).

Set point is Operator selected. The controller will maintain ±5 psia of chosen set point.

3. Manual - Operator controls signal output to the valve using the slide bar.

Valve opens based upon output signal.

4. Input is steam header pressure from PT-0510.

5. Alarm indication (Same as ADV controller)

a. Yellow light in solid indicates a process input/output failure or other internal problems.

b. Yellow light flashing indicates a low voltage condition in the controller battery. This should not affect operation as long as power is available to the controller. Controller program will be lost if the controller loses power with a flashing yellow light.

c. Red light indicates a controller computer failure. Controller should fail to last good value but may not be held for long. Manual control may be available.

6. PIC-0511 controls CV-0511 to maintain the steam pressure setpoint.

a. Normally 900 psia (T_{AVE} at 532°F)

b. At 5 psi greater than the setpoint (905 psia, 532.6°F), CV-0511 will be full open.

c. CV-0511 will be fully closed at 5 psi less than the setpoint (895 psia, 531.3°F) on PIC-0511.

d. The TBV scale is 800 psia to 1000 psia.

Since S/G pressure is approximately 770 psia at full power, I&C has set the out of range alarm function (Yellow alarm light in solid) below expected values for full power operation to prevent the yellow light from being illuminated all the time at full power conditions.

e. The TBV pressure control function DOES NOT require a turbine trip (e.g. does not require 386AST relay actuation).

7. PM-0511 auctioneers the signals from PIC-0511 and HIC-0780A, taking the larger of the two signals.

a. Therefore, in addition to the pressure control input, CV-0511 receives a modulate open signal from steam dump controller HIC-0780A through PM-0511.

Example: If the Atmospheric Dump Valves are being operated in manual using HIC-0780A, a signal will also be sent to PM-0711. If this signal is greater than the signal from PT-0510, the TBV will open.

b. This is the same as the signal received by the ADVs from the T_{AVE} Computer. When the TBV opens as a result of input from HIC-0780A, the output meter on PIC-0511 will show "zero" output. Note also that the TBV can open from HIC-0780A when PIC-0511 is in the manual mode.

c. The 386X1 AST turbine trip relay must be energized to receive this signal.

d. Modulate signal is removed when T_{AVE} is lowered to 535°F (+3°F) and will be provided when T_{AVE} is greater than 540°F (+8°F). The TBV should already be full open due to the pressure signal if T_{AVE} is at 540°F (962.8 psia).

e. CV-0511 also receives the same 'quick opening' signal as the ADVs.

1) T_{AVE} at 556.9°F and turbine trip via the 386 AST turbine trip relay

2) Opens SV-0589B and closes SV-0589C to align the 'quick open' air supply and close the modulate air supply.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	____
	Group #	2	____
	K/A #	056.G2.1.23	
	Importance Rating	3.9	____

K/A Statement: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question:

Given the following conditions:

- The Plant is at 35% power during a power escalation.
- P-10A, Heater Drain Pump (the first heater drain pump) was just started.
- BOTH Condensate Pumps are operating.
- ONE Main Feedwater Pump is operating.
- One of the operating Condensate Pumps trips.

Which of the following describes the impact on the Condensate System Recirculation Valve, CV-0730, and what must the operator do?

CV-0730 will throttle in the . . .

A. OPEN direction and direct more flow to feedwater trains. Monitor Heater Drain Pump for normal operation.

B. CLOSED direction and direct more flow to feedwater trains. Monitor Heater Drain Pump for normal operation.

C. OPEN direction and direct more flow to the Main Condenser Hotwell. Align alternate Gland Seal Exhauster to maintain vacuum.

D. CLOSED direction and direct more flow to the Main Condenser Hotwell. Align alternate cooling to Air Ejector Condenser to maintain vacuum.

Proposed Answer: B

Explanation (Optional):
CV-0730 modulates on a flow signal from FC-0730 to maintain a minimum flow of 6800 gpm (1600 gpm through the gland seal condenser and 5200 gpm through the air ejector condenser). At low power (i.e. <25%, the valve is full open to provide a flow path for the Condensate pumps). At approximately 35-40% power, the valve should be full closed to ensure adequate cooling flow and adequate NPSH for the Feedwater Pumps. At 30% power, in this case, the valve would be partially open. If a condensate pump were to trip, the valve would close in order to maintain its minimum flow requirement and support FW pump NPSH.

A. Incorrect, CV-0730 would throttle closed. Throttling open would not allow more flow to the FW pumps, instead directing more flow to the hotwell.
B. Correct, see explanation.
C. Incorrect, CV-0730 would throttle closed.
D. Incorrect, while CV-0730 throttling closed is correct, doing so would not result in more flow to the hotwell and would allow more flow to the FW pumps.

Technical Reference(s): PL-CDFW Main Condenser, Condensate and Feedwater
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam Palisades 2003
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments:

f. Condensate Recirc Control Valve CV-0730

1) 12" valve modulates on a flow signal from FC-0730 to maintain minimum flow of 6800 gpm, 1600 gpm through the gland seal condenser and 5200 gpm through the air ejector condenser.

2) Also provides a flow path for the Condensate Pumps during plant start up and shutdown.

3) Red and green position indicating lights on C-01

4) Fails closed on loss of air.

5) On a down power, CV-0730 should start opening at approximately 25% power.

On a power escalation, CV-0730 should be full closed at 35 to 40% power.Have class review Gary Katt memo from 1998, Attachment 1 of this lesson plan.

ATTACHMENT 1: GKatt Memo – Condensate Recirculation Valve CV-0730

Author: Gary Katt at CPC-PA1
Date: 2/3/98 11:50
Subject: Condensate Recirculation Valve CV-0730

CV-0730 is designed to maintain ~6800 gpm through the condensate system to provide adequate flow path for the condensate pumps and adequate cooling for the gland steam and air ejector condensers. There is some concern whether CV-0730 will open (due to a previous problem during a forced outage), and what to do if it won't open.

1. FC-0730 was completely overhauled during the last forced outage, and was working properly.
2. The valve should start to open at ~25% power. Monitoring the valve as power is lowered from 25% will enable Operations to determine if the controller is operating properly. If problems develop, this should give I&C time to fix the controller prior to causing problems with the secondary side.
3. The minimum required flow for each condensate pump is approximately 1400 gpm. The pumps should not be operated if there is no flow path available.
4. CV-0730 can be manually failed open by closing a small metal flapper in FC-0730. This flapper is located inside the panel below the flow indicator. Moving this small flapper tight against the adjacent nozzle will result in a full open air signal being sent to CV-0730 causing the valve to open. This flapper would have to be secured in place to keep CV-0730 in the full open position.
5. Alternate flow paths are available if the valve is stuck closed and all other means to fix it have failed:

As long as a feedwater pump is still operating there is no concern for CV-0730 failing to open as the feedwater pump recirculation valve will be open maintaining an adequate flow path for the condensate pumps. If the feed pumps are tripped then the feed pump recirculation valves could be failed open to maintain the desired flow path.

Caution:
Another flow path if the feed pumps are tripped would be to recirculate the condensate back to the condenser through the E-6A/B recirculation line.

Ensure Main Feed Pumps Aux Oil Pump is running prior to establishing flow through an idle feed pump. Refer to SOP-11, "To Recirculate Condensate/Feedwater System Using P-2A or P-2B.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	071.K4.04	
	Importance Rating	2.7	

K/A Statement: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Isolation of waste gas release tanks.

Proposed Question:

A waste gas release is in progress from Waste Gas Decay Tank T-68A. Waste Gas Radiation Monitor, RIA-1113, spikes above the HIGH alarm setpoint. As a result, the release is automatically terminated by which combination of the following actions:

1. CV-1113, Waste Gas Surge Tank to Stack, closes
2. CV-1119A, T-68A Discharge Control Valve, closes
3. CV-1123, Waste Gas Decay Tank to Stack, closes
4. The operating Main Exhaust fan, V-6A or V-6B, trips

A. 1 and 2

B. 2 and 3

C. 1 and 3

D. 2 and 4

Proposed Answer: **C**

Explanation (Optional):
On a high radiation condition sensed by RIA-1113, valves CV-1113 and CV-1123 will close, isolating the waste gas decay tank and waste gas surge tank from the vent stack. CV-1119A will not automatically close on the high radiation condition, it must be manually closed. A Main Exhaust Fan is required to be operating during a release, and if a fan were to trip during the release, the release would have to be manually secured. The tripping of V-6A or V-6B will not isolate the release.

A. Incorrect, see explanation.
B. Incorrect, see explanation.
C. Correct, see explanation.
D. Incorrect, see explanation.

Technical Reference(s): PL-RMS Radiation Monitoring System Lesson Plan, M-211 Sheets 2 and 3

(Attach if not previously provided, ______________________________

including version/revision number) ____

Proposed references to be provided to applicants during examination: ______None_______

Learning Objective: ________________________ (As available)

Question Source: Bank # _______

Modified Bank # _______ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__

55.43 _____

Comments:

RIA-1113, Waste Gas Radiation Monitor	
VA-66 RIA-1113 Purpose	
Purpose	
Digital channel RIA-1113 monitors radioactive waste gas discharges from the waste gas decay tanks to the stack.	
It is designed to detect higher-than-expected radiation levels in the waste gas release and terminate the release upon detection of such.	
VA-67 RIA-1113 Flow Path	
Flow Path	
RE-1113 is a full flow in-line detector that monitors the waste gases that are released from the waste gas decay tanks.	
Effluent from the detector outlet goes to the stack.	
OBJ 12 Describe the consequences of operating the Radiation Monitoring System with Waste Gas Monitor RE-1113 improperly isolated in accordance with SOP-38.	
Waste Gas Surge Tank Relief Valve RV-1111 discharges through RE-1113 to the stack. If RE-1113 is improperly isolated, then RV-1111 will also be isolated and thereby disabled. See Figure 1 below.	
*When isolating RE-1113, bypass valve MV-WG118 must be opened **before closing either** RE-1113 inlet isolation valve MV-WG117 or RE-1113 outlet isolation valve MV-WG119.*	
Failure to perform the alignment as specified will isolate the RV-1111 release path and disable the WGST overpressure protection.	
Generally, RE-1113 would be isolated only for maintenance needs.	

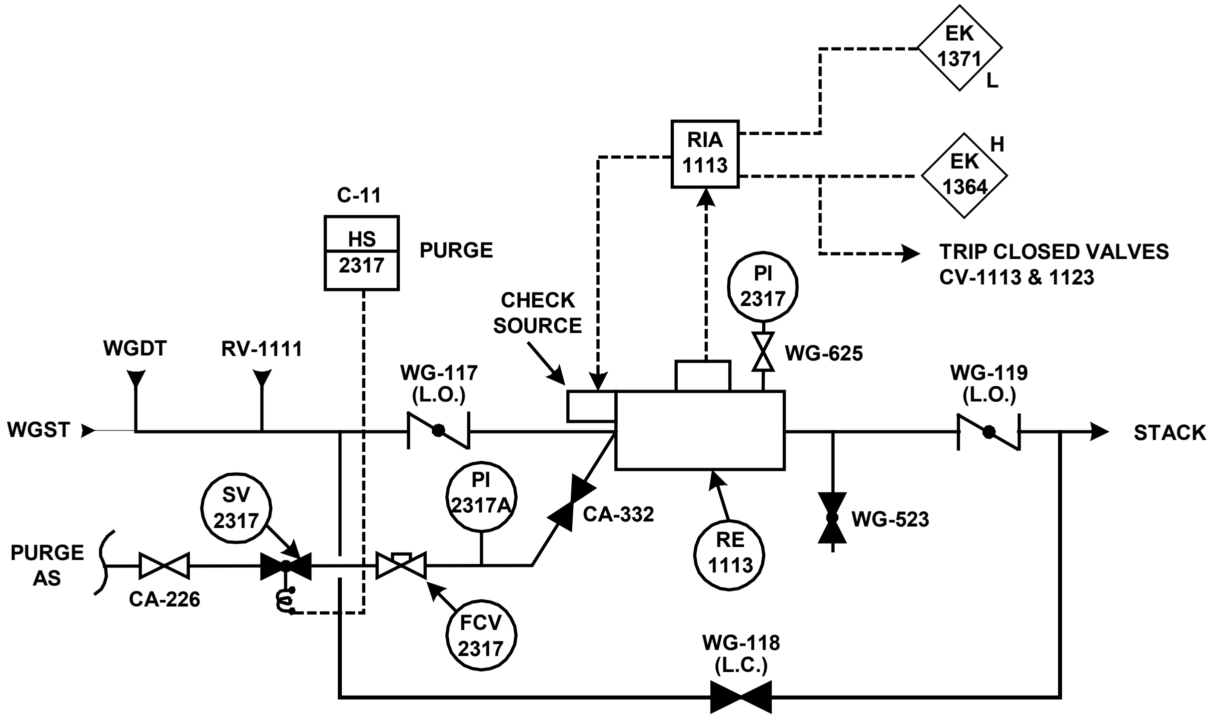


Figure 1. RIA-1113 Flow Path

Alarms and Setpoints	
RIA-1113 HIGH alarm activates Control Room annunciator EK-1364, "GASEOUS WASTE MONITORING HI RADIATION."	
The HIGH alarm setpoint is variable and is calculated specifically for each waste gas release.	Addressed in System Operations Section.
RIA-1113 low signal output actuates Control Room annunciator EK-1371, "RADIATION MONITOR SYSTEM CKT FAILURE."	
VA-68 RIA-1113 Auto Actions	
Automatic Actuations	
On a HIGH alarm, RIA-1113 automatically closes waste gas discharge valves CV-1113 and CV-1123, thereby terminating the waste gas release.	

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072.A2.02	
	Importance Rating	2.8	

K/A Statement: Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

Proposed Question:

The Plant is at 100% power. Containment Radiation Monitor, RIA-1805, has recently experienced multiple high spikes and has been removed from service by I&C Technicians, who have removed the fuse for the monitor.

An operator accidently tests Containment Radiation Monitor RIA-1806, per SOP-39, “Area Radiation Monitoring System,” rather than RIA-1805 which has been removed from service.

Containment Radiation Monitors RIA-1807 and RIA-1808 remain unaffected.

Given the conditions above, does a Containment Isolation Signal result and what is the correct action?

A. Containment Isolation occurs; Enter AOP-31, “Spurious Containment Isolation.”

B. Containment Isolation occurs; Enter EOP-1.0, “Standard Post-Trip Actions.”

C. Containment Isolation does NOT occur; attempt to reset RIA-1806 per SOP-39, “Area Radiation Monitoring System.”

D. Containment Isolation does NOT occur; Initiate Work Request to repair.

Proposed Answer: A

Explanation (Optional):

A. Correct, a containment isolation occurs on Containment High Radiation (CHR), as the coincidence for a CHR is 1/3 with RIA-1805 removed from service (fuse pulled). Testing the rad monitor RIA-1806 places the monitor in tripped condition (Trip 2) and initiates internal self-check. The self-check applies a voltage to the channel circuitry that corresponds to 10^3 R/hr, which trips the channel (exceeding the 10 R/hr trip setpoint for a CHR).
B. Incorrect, the applicant incorrectly believes a reactor trip is required. A reactor trip is only required per AOP-31 if the containment isolation is caused by a high containment pressure condition.
C. Incorrect, the applicant incorrectly believes a containment isolation does not occur and

does not believe that removing a monitor from service will allow a 1/3 coincidence to actuate a CHR. This action would be correct if a monitor were to not reset during a test.

D. Incorrect, the applicant incorrectly believes a containment isolation does not occur and does not believe that removing a monitor from service will allow a 1/3 coincidence to actuate a CHR. This action is a correct action per ARP-8 (#63) if a single monitor were to fail.

Technical Reference(s): SOP-39, PL-RMS Radiation Monitoring System, AOP-31, ARP-8

(Attach if not previously provided, including version/revision number) ________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ________________ (As available)

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

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

7.4.2 To Place In Operation

a. **REFER TO** Attachment 3, Checklist CL 39, "Area Monitors System Checklist."

b. **CHECK** operate light illuminated.

c. IF operate light NOT illuminated, THEN **REFER TO** Attachment 2, "System Malfunctions and Troubleshooting."

NOTE:	To reset Containment Radiation Monitors (RIAs -1805, 1806, 1807, and 1808) High Alarms the TRIP 1 and TRIP 2 lights, on the effected monitors, will need to be depressed.

d. **RESET** all alarms.

e. IF operate light still NOT illuminated, THEN **DECLARE** the associated monitor inoperable. Refer to Attachment 2, "System Malfunctions and Troubleshooting."

7.4.3 To Test

a. **CHECK** other three monitors not tripped.

NOTE:	The following step will trip the monitor.

b. **PLACE** Selector Switch to CHECK position.

c. **VERIFY** monitor indicates approximately 10^3 R/hr.

d. **RELEASE** Selector Switch.

e. **RESET** all alarms.

4.2 TO REMOVE A CONTAINMENT AREA MONITOR FROM SERVICE

NOTE:	The following alarms may be expected when performing the following: • EK-1363 "Containment High Radiation" • EK-1371 "Radiation Monitor System Circuit Failure"

a. CONTACT I&C to remove fuse (F1) on the back of Containment Area Monitor being removed from service. This will provide 1/3 CHR actuation logic.

b. IF one of the Containment AreaMonitors is inoperable AND 2/3 CHR actuation logic is desired, THEN a temporary modification is required. REFER TO Entergy Procedure EN-DC-136, "Temporary Modifications."

Containment Radiation Monitors RIA-1805/1806/1807/1808.
The Containment Radiation Monitors are analog channels.
VA-1 RIA-1805/1806/1807/1808
Each channel has a RIA in the Control Room; the four RIAs are identical and include the following controls and indications:
Analog meter
Three-position function selector switch
Trip 1, Trip 2, and Operate lamps/pushbuttons
Analog meter – provides measured radiation field indication
Vertical, single-scale meter
Meter spans six decades, from 10^{-2} to 10^{4} R/hr
Three-position function selector switch
"CHECK" – Places monitor in tripped condition (Trip 2) and initiates internal self-check.

The self-check applies a voltage to the channel circuitry that corresponds to 10^3 R/hr, which trips the channel.
The function selector switch spring returns to "OPERATE" from this position.
"OPERATE" – Places monitor in service.
"TRIP ADJ" – Causes the channel output to go full-scale high.
*This switch position should **NOT** be used by Operations personnel*
Pushbutton lamps:
Operate (Green)
When illuminated, indicates that the monitor is in service.
If not illuminated, then the monitor is not in service, or may have failed low.
Trip 1 (Alert – Yellow)
Lamp illuminates when measured radiation meets or exceeds the alert alarm setpoint.
Alert alarm locks in and yellow lamp remains illuminated until alarm is manually reset by pressing the Trip 1 button
Alert setpoint is approximately 1 R/hr.
Trip 2 (High – Red)
Lamp illuminates when measured radiation meets or exceeds the high alarm setpoint.
High alarm locks in and red lamp remains illuminated until alarm is manually reset by pressing the Trip 2 button
Actuates Control Room alarm EK-1363, "CONTAINMENT HI RADIATION."
High alarm setpoint is approximately 10 R/hr.
*The Trip 2 bistable provides an input to the Containment Isolation logic (described later in Section **Error! Reference source not found.**, **Error! Reference source not found.**.)*

TITLE: SAFEGUARDS SAFETY INJECTION AND ISOLATION SCHEME EK-13 (EC-13)

37	43	49	55	61	67	73
38	44	50	56	62	68	74
39	45	51	57	63	69	75
40	46	52	58	64	70	76
41	47	53	59	65	71	77
42	48	54	60	66	72	78

CONTAINMENT HI RADIATION	
Sensor:	RIAX-3/1805, RIAX-3/1806, RIAX-3/1807, RIAX-3/1808
Trip Setpoints:	10 R/hr
Alternate Indication:	Containment Rad Monitors RIA-1805/1806/1807/1808 indications

AUTOMATIC FUNCTION:
- CIAS on 2 of 4 coincidence.

OPERATOR ACTION:
- REFER TO appropriate EOP based on additional indications.
- IF actual Containment Radiation is greater than or equal to 10 R/hr on any single Containment Area Monitor, THEN **VERIFY** Containment Isolation signal initiated (Window EK-1126 lit) OR **PUSH** left or right High Radiation Initiate pushbutton on EC-13, DBA, Shutdown, & Misc Services Control Pnl.

FOLLOW UP ACTION:
- IF inadvertent Containment Isolation results, THEN GO TO AOP-31.
- IF alarm caused by a single inoperable channel, THEN:
  - o **INITIATE** Work Request.
  - o **IMPLEMENT** any applicable Technical Specifications LCO 3.3.3, LCO 3.3.4 actions.
- REFER TO EI-1.
- CONSIDER sampling PCS to perform failed fuel analysis.

REFERENCES:
- AOP-31, "Spurious Containment Isolation"
- Technical Specifications LCO 3.3.3, LCO 3.3.4
- EI-1, "Emergency Classification and Actions"

SPURIOUS CONTAINMENT ISOLATION

USER ALERT

Read each step of the procedure prior to performing that step. Since the cause and nature of the abnormal condition cannot be predicted, performance of steps out of sequence may be necessary.

1.0 PURPOSE

Provide operator actions that must be accomplished subsequent to a spurious containment isolation.

2.0 ENTRY CONDITIONS

- EK-1126, "CIS INITIATED" with Containment pressure and radiation levels normal.

3.0 EXIT CONDITIONS

- The diagnosis of a spurious containment isolation is NOT confirmed.

OR

- All applicable steps of this procedure have been completed.

4.0 AUTOMATIC ACTIONS

- Containment Isolation Valves close
- Safety Injection initiated (CHP only)
- Containment Spray initiated (CHP only)
- Reactor trip (CHP only)

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	086.A3.02	
	Importance Rating	2.9	

K/A Statement: Ability to monitor automatic operation of the Fire Protection System including: Actuation of the FPS.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power.
- Multiple fire alarms were just received.
- Fire Protection header pressure lowered to 88 psig prior to stabilizing at 105 psig.

With NO operator action, what is the status of the Fire Protection pumps?

	Fire Jockey Pump P-13	Electric Fire Pump P-9A	Diesel Fire Pump P-9B	Diesel Fire Pump P-41
A.	ON	OFF	OFF	OFF
B.	OFF	ON	OFF	OFF
C.	OFF	ON	ON	OFF
D.	ON	ON	ON	ON

Proposed Answer: **B**

Explanation (Optional):

The Fire Jockey Pump (P-13) is rated for 50 gpm at 115 psi; it is designed to handle small leaks and maintain fire header pressure. The electric Fire Pump (P-9A) auto-starts when header pressure drops to 98 psig. The two diesel Fire Pumps (P-9B and P-41) auto-start on low header pressure of 83 psig and 68 psig, respectively. Other than the Jockey Pump, all fire pumps need to be manually secured upon a low pressure auto-start. In this scenario, header pressure is not maintained with the Jockey Pump and the Electric Fire Pump will start upon reaching the auto-start setpoint of 98 psig. The Fire Header Lo Pressure alarm comes in at 95 psig, indicating that the Electric Fire Pump should have auto-started prior to reaching the alarm. The Diesel Fire Pumps P-9B and P-41 will not start since the low header pressure setpoints are not reached. As pressure recovers above the auto-start setpoints, the Electric Fire Pump remains running until manually secured.

A. Incorrect, the Electric Fire Pump will start. The applicant must understand the auto-start

setpoints of each FP pump and if the pumps must be manually secured or secure automatically.

B. Correct, see explanation.
C. Incorrect, the Diesel Fire Pump P-9B will not start.
D. Incorrect, the Diesel Fire Pumps P-9B and P-41 will not start.

Technical Reference(s): DBD-1.10, AOP-40
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

8.1.2 Fire Jockey Pump

The Fire Jockey Pump (P-13) is a 7 ½ hp, 480 V ac vertical turbine pump. Its purpose is to maintain the system pressure. A local switch at the pump has a "Hand," "Auto," and "Off" position. In auto, the pump is controlled by Panel C-36. The pump is rated for 50 gpm at approximately 115 psi.

8.1.3 Electric Fire Pump

The Electric Fire Pump (P-9A) is a three-stage vertical turbine pump designed to deliver 1500 gpm at 125 psig net. It is driven by a 150 hp, 480 V ac motor which is supplied by the LC-13 bus. The LC-13 bus is supplied by the 1-C bus. The electric fire pump is started automatically by pressure switch (PS-1311) when system pressure drops to 98 psig. The motor can be started manually by a push button at the Control Panel (C-36).

8.1.4 Diesel Fire Pumps

The Diesel Fire Pumps (P-9B and P-41) are three-stage vertical turbine pumps that deliver 1500 gpm at 125 psig. Both are driven by Cummins Model NH-220-1F, six cylinder, 150 hp, 1760 rpm diesels through a Johnson Right Angle Drive. Each has a day tank of 275 gallons of #2 diesel fuel (located outside of the area). The king Knight Control Panels (C-37 and C-137) have a selector switch with the following positions: "Auto," "Off," "Manual A," "Manual B," and "Test." When the automatic position is selected, the P-9B Diesel will start when fire header pressure is equal to 83 psig by pressure switch (PS-1310). The P-41 Diesel Pump will start by sensing a pressure equal to 68 psig by pressure switch (PS-5350). Pumps may be started manually from the Control Room.

When the manual positions are selected, either Battery A or B is chosen to power the starter and a manual start push button is used. When in the test position, the diesels will automatically start by normal sequence as if low pressure were sensed on low header pressure or a manual start button was pushed on the appropriate panel.

TITLE: AUXILIARY SYSTEMS SCHEME EK-11 (C-13)

1	7	13	19	25	31
2	8	14	20	26	32
3	9	15	21	27	33
4	10	16	22	28	34
5	11	17	23	29	35
6	12	18	24	30	36

FIRE SYSTEM HEADER LO PRESS	
Sensor:	PS-1312, Low Low Pressure Fire System Main Header
Trip Setpoints:	95 psig
Alternate Indication:	Local fire header pressure

AUTOMATIC FUNCTION:
- P-9A, Motor Driven Fire Pump is auto started by PS-1311 at 98 psig.
- P-9B, Diesel Fire Pump is auto started by PS-1310 at 83 psig.
- P-41, Diesel Fire Pump is auto started by PS-5350 at 68 psig.

OPERATOR ACTION:
- **CHECK** system pressure recorder.
- **RESTART** P-13, Jockey Pump OR P-9A as necessary to return system pressure to normal.
- IF an unscheduled Fire Pump Start alarm is received, THEN **DISPATCH** an NPO to the Diesel Generator Rooms, 1D Switchgear Room, and Cable Spreading Room to check for flooding.

FOLLOW UP ACTION:
- **CHECK** for excessive demand (ie, tripped sprays or fire hoses in use). Long term operation of firewater pumps at less than 60 psig discharge pressure may result in pump damage.
- WHEN excess demand is located and corrected, THEN **RESTORE** Fire System lineup to normal per SOP-21.

REFERENCES:
- SOP-21, "Fire Protection System"

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.1.25	
Importance Rating	3.9	

K/A Statement: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question:

Given the following conditions:

- The Plant has experienced a small break LOCA and has implemented EOP-4.0, "Loss of Coolant Accident Recovery."
- Pressurizer Level indicates 60% on LIC-0101B.
- Pressurizer pressure is 1500 psia.
- Containment temperature is 205ºF.

What is the actual Pressurizer level?

A. 50%

B. 56%

C. 82%

D. 86%

Proposed Answer: **A**

Explanation (Optional):

A. Correct, at 205ºF, the applicant must use page 1 of Supplement 9 to determine the corrected pressurizer indicated level, which is 54% (60% indicated minus 6% error). At 54% pressurizer corrected indicated level and 1500 psia, the actual pressurizer level is 50%.
B. Incorrect, the applicant used the correct hot calibrated (Supplement 9), but did not account for error due to containment temperature.
C. Incorrect, the applicant used the cold calibrated (Supplement 10).
D. Incorrect, the applicant used the cold calibrated (Supplement 10) and did not account for error due to containment temperature.

Technical Reference(s): EOP-4.0, EOP Supplement 9, EOP Supplement 10
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: EOP Supplement 9, 10

Learning Objective: ______________________ (As available)

Question Source: Bank # ________

Modified Bank # ________ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10_

55.43 ______

Comments:

INSTRUCTIONS

<u>NOTE</u>: Use ANY of the following to determine Average of Qualified CETs:

- PPC point "KCETA" (Average of Qualified CETs)
- PPC Incore Qualified CET Map (PPC page 313)
- Manual calculation. Refer to SOP-34, "Palisades Plant Computer (PPC) System."

© 26. **VERIFY** SI Pump throttling criteria are satisfied by ALL of the following:

a. Based on the Average of Qualified CETs, PCS subcooling meets ONE of the following:

- At least 25°F subcooled for non-degraded Containment conditions
- Greater than the minimum subcooling curve on EOP Supplement 1 for degraded Containment conditions

b. Corrected PZR level is greater than 20% (40% for degraded Containment) and controlled. REFER TO EOP Supplements 9 and 10.

(continue)

CONTINGENCY ACTIONS

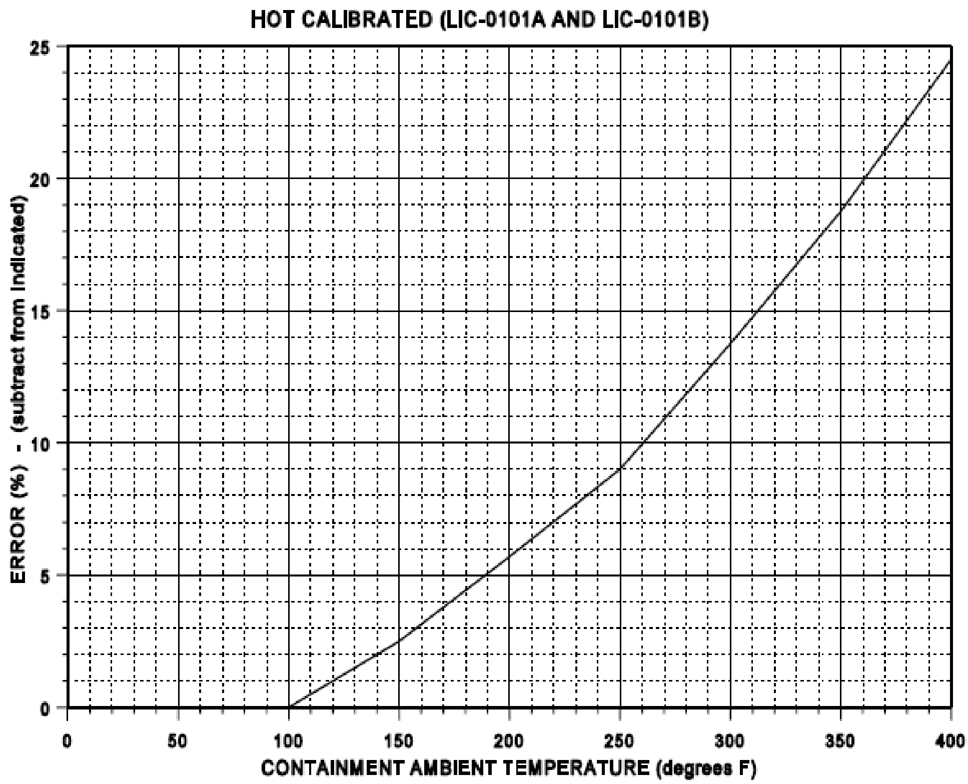
26.1 <u>IF</u> ANY of the SI Pump throttling criteria can NOT be maintained, <u>THEN</u> **RAISE** HPSI flow <u>AND</u> **START** HPSI Pumps as necessary.

PUMP	VALVE	
	NUMBER	DESCRIPTION
Train 1		
P-66B	MO-3009	HPSI Train 1 to Loop 1B
	MO-3011	HPSI Train 1 to Loop 2A
	MO-3007	HPSI Train 1 to Loop 1A
	MO-3013	HPSI Train 1 to Loop 2B
Train 2		
P-66A	MO-3066	HPSI Train 2 to Loop 1B
	MO-3064	HPSI Train 2 to Loop 2A
	MO-3068	HPSI Train 2 to Loop 1A
	MO-3062	HPSI Train 2 to Loop 2B

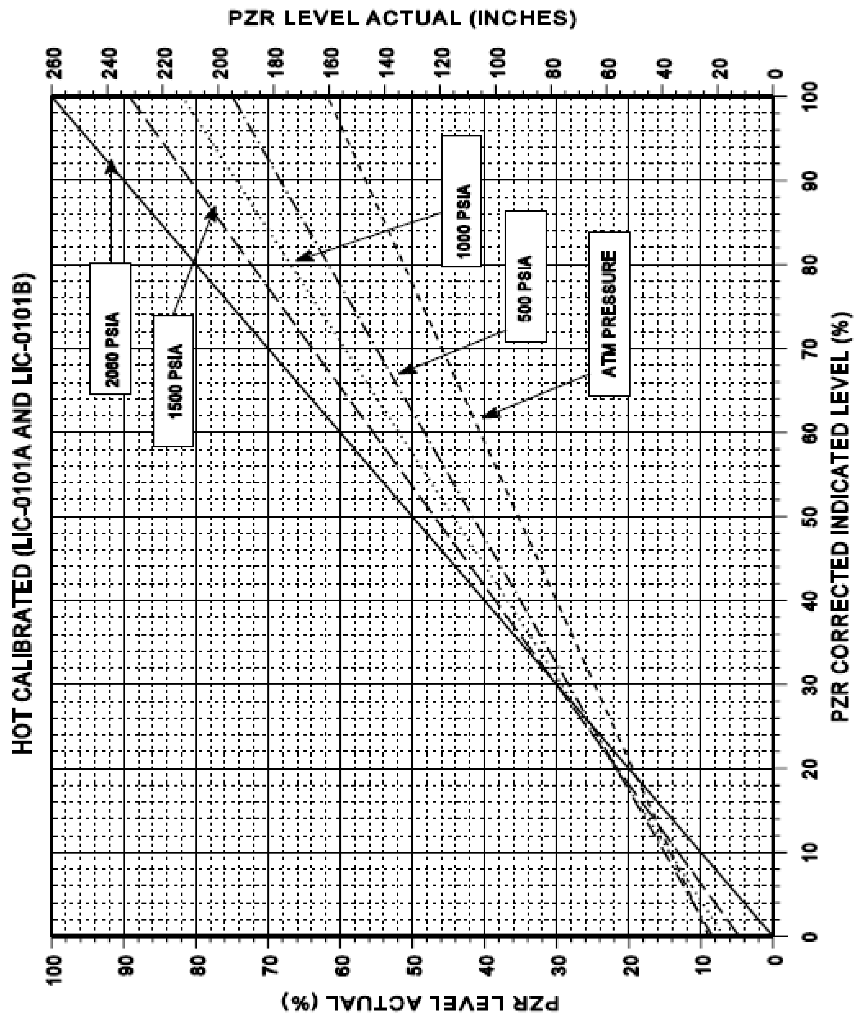
PALISADES NUCLEAR PLANT
EMERGENCY OPERATING
PROCEDURE

Proc No	EOP Supplement
Supplement	9
Revision	5
Page	1 of 2

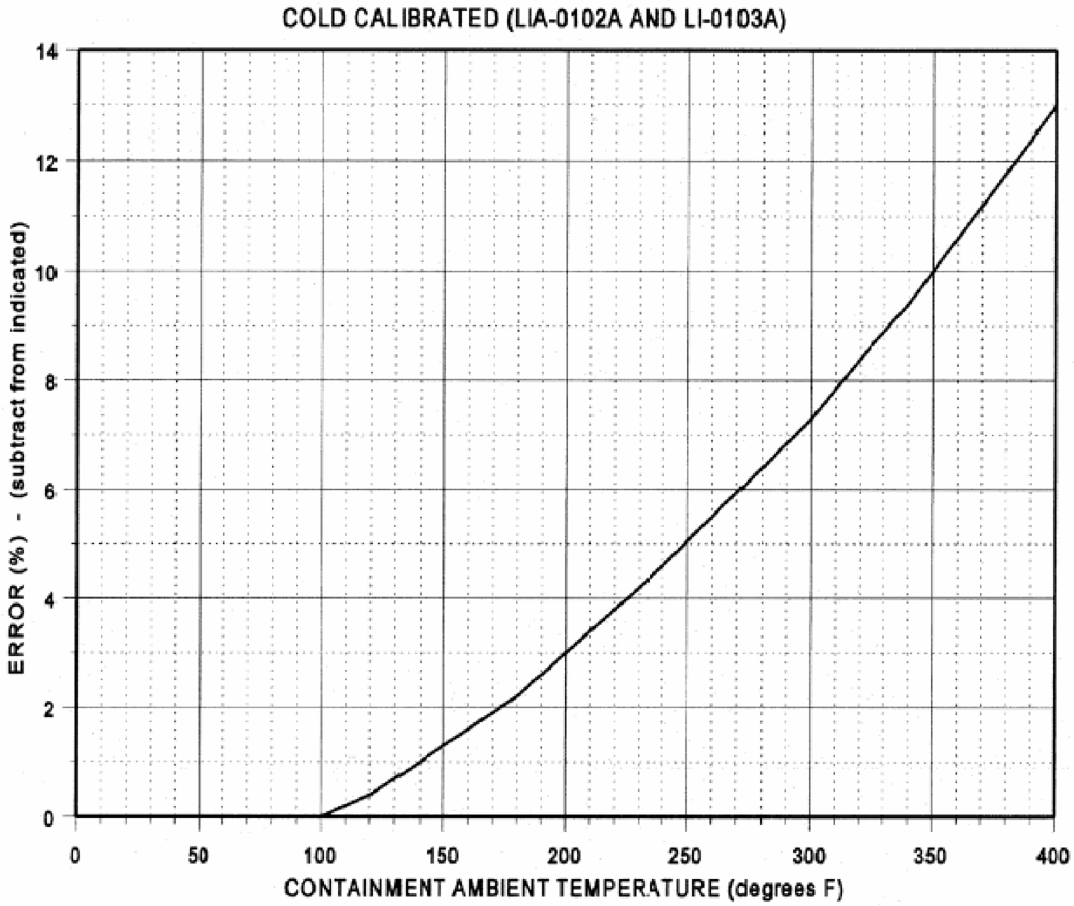
TITLE: PZR Level Correction - Hot Cal



TITLE: PZR Level Correction - Hot Cal

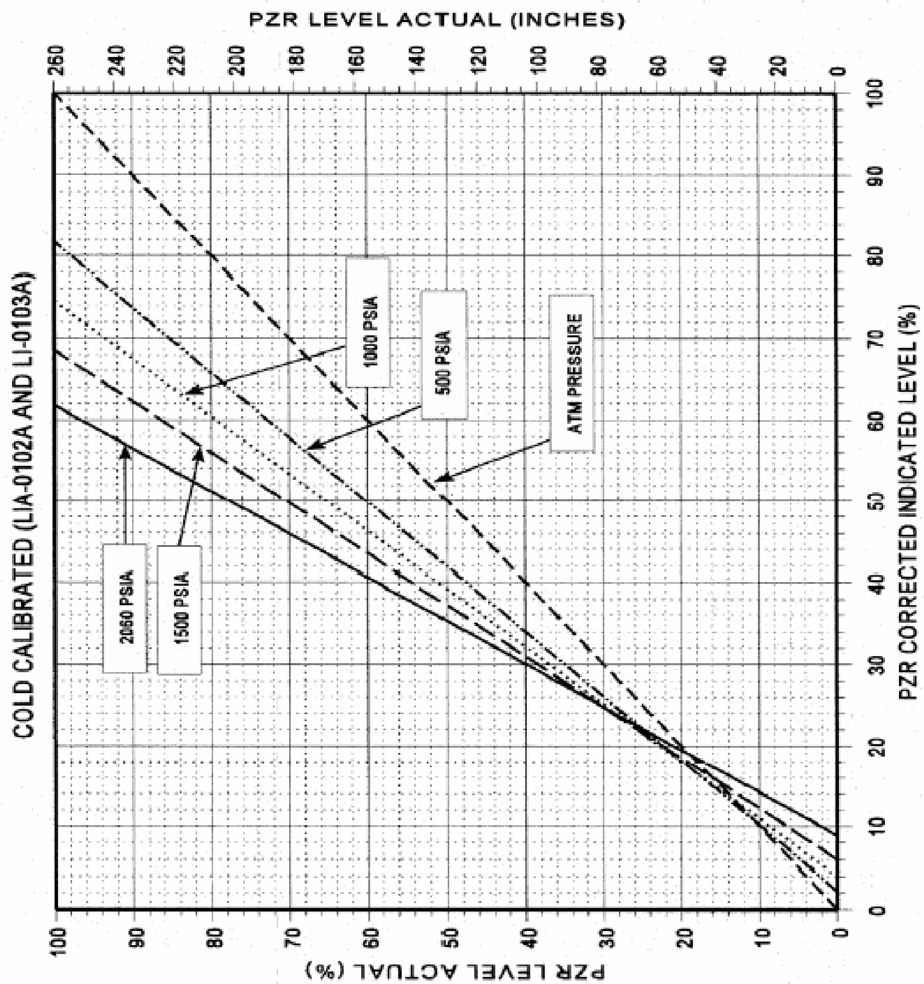


NOTE: Page 1 of this supplement should be used to obtain the corrected indicated level prior to using this figure.



PZR Level – Cold Cal

PZR Level – Cold Cal



NOTE: Page 1 of this Supplement should be used to obtain the corrected indicated level prior to using this figure.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.1.30	
	Importance Rating	4.4	

K/A Statement: Ability to and operate components, including local controls.

Proposed Question:

Component Cooling Water (CCW) has been lost to Containment for greater than 10 minutes. Per AOP-36, "Loss of Component Cooling," why is CCW flow manually re-initiated and where is the preferred restoration performed from?

Assume access to all plant areas is possible, all plant conditions are stable, and CCW flow restoration is desired.

A. Manual flow is re-initiated to prevent thermal shock and possible equipment damage. This is performed from inside Containment 590' level, using PCP & CRDM return isolation valves.

B. Manual flow is re-initiated to prevent a possible low system pressure auto start on a standby CCW pump. This is performed from inside Containment 590' level, using PCP & CRDM return isolation valves.

C. Manual flow is re-initiated to prevent a possible low system pressure auto start on a standby CCW pump. This is performed from inside the CCW Pump Room, 590' level, using the CCW Return from Containment isolation (MV-CC713).

D. Manual flow is re-initiated to prevent thermal shock and possible equipment damage. This is performed from inside the CCW Pump Room, 590' level, using the CCW Return from Containment isolation (MV-CC713).

Proposed Answer: **A**

Explanation (Optional):

A. Correct, flow is manually re-initiated to allow a controlled restoration of flow to components in Containment, to prevent thermal shock and potential equipment damage when cooling water flow is restored. Per AOP-36, with access to containment, flow is to be restored using the PCP and CRDM return isolation valves.
B. Incorrect, see Choice A, flow is slowly restored to minimize the potential for thermal shock to the system and equipment. Doing this will provide the operators better system pressure control.
C. Incorrect, see Choice A, flow is slowly restored to minimize the potential for thermal shock to the system and equipment. Doing this will provide the operators better system

pressure control. The applicant believes containment access is not available or not prudent and restoration should be performed from the CCW Pump room.

D. Incorrect, see Choice A. The applicant believes containment access is not available or not prudent and restoration should be performed from the CCW Pump room.

Technical Reference(s): AOP-36 and AOP-36 bases
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 12
55.43 ______

Comments:

Question modified from Palisades 2005 NRC Exam. Modified all distractors to better comply with procedural validity. Stem changed to ask for the preferred restoration method.

(ATTACHMENT 2) COMPONENT COOLING WATER RESTORATION TO CONTAINMENT

Description:

This attachment provides guidance for restoring CCW to Containment in a controlled manner and provides actions to be taken if CCW flow to the Containment has been lost for greater than 10 minutes and restoration of CCW flow to the Containment is desired. The intent of these actions is to allow a controlled restoration of flow to components in the Containment, to prevent thermal shock when cooling water flow is restored.

Steps are provided for restoration of flow to the Containment if a Containment entry is possible or if Containment entry is not possible.

Training Emphasis:

NONE

LOSS OF COMPONENT COOLING

COMPONENT COOLING WATER RESTORATION TO CONTAINMENT

1. IF CCW flow to Containment has been lost for greater than 10 minutes, AND restoration is desired, THEN:

   a. **ENSURE CLOSED** CCW Isolation Valves:

   - CV-0910 HS-0910 KEY: 337
   - CV-0911 HS-0911 KEY: 338
   - CV-0940 HS-0940 KEY: 336

   b. **VERIFY** EK-1172, "COMPONENT CLG SURGE TANK T-3 HI-LO LEVEL," clear.

   c. **ENSURE** at least one CCW Pump operating.

NOTE: Step 2 will restore CCW with containment entry possible (preferred).
Step 3 will restore CCW with containment entry not possible.

2. IF restoring CCW and Containment entry is possible, THEN:

   a. **CLOSE** AND **OPEN** one turn, the following valves:

   - MV-CC110, PCP P-50A CCW Clg Return
   - MV-CC112, PCP P-50B CCW Clg Return
   - MV-CC114, PCP P-50C CCW Clg Return
   - MV-CC116, PCP P-50D CCW Clg Return
   - MV-CC108, CRDM Cooling Return

   LOCATION: Containment 590' Level, near VHX-4, Containment Air Cooler

COMPONENT COOLING WATER RESTORATION TO CONTAINMENT

b. IF Containment pressure is less than 4 psig, THEN **RESET** Containment High Pressure Circuits by pushing the left and right high pressure reset pushbuttons.

- CHPL - RESET
- CHPR - RESET

Location: Panel EC-13

c. IF CHP is RESET, THEN **PLACE** CCW Isolation Valves Handswitches to AUTO:

- HS-0910, Comp Clg Water Inlet to Cntmt Switch KEY: 337
- HS-0911, Comp Cool Water Discharge From Contain Sw KEY: 338
- HS-0940, Comp Cool Water Discharge From Contain Sw KEY: 336

d. IF Containment pressure is greater than 4 psig and less than 35 psig, or CHP did not reset, THEN **PLACE** CCW Isolation Valves Handswitches to OPEN:

- HS-0910, Comp Clg Water Inlet to Cntmt Switch KEY: 337
- HS-0911, Comp Cool Water Discharge From Contain Sw KEY: 338
- HS-0940, Comp Cool Water Discharge From Contain Sw KEY: 336

e. **MONITOR** CRDM parameters as appropriate.

f. **MONITOR** Primary Coolant Pump parameters as appropriate.

g. **VERIFY** CCW cooled components temperatures have stabilized.

COMPONENT COOLING WATER RESTORATION TO CONTAINMENT

h. **FULLY OPEN** the following valves:

- MV-CC110, PCP P-50A CCW Clg Return
- MV-CC112, PCP P-50B CCW Clg Return
- MV-CC114, PCP P-50C CCW Clg Return
- MV-CC116, PCP P-50D CCW Clg Return
- MV-CC108, CRDM Cooling Return

LOCATION: Containment 590' Level, Near VHX-4, Containment Air Cooler

3. IF restoring CCW and Containment entry NOT possible, THEN:

a. **UNLOCK** AND **CLOSE** MV-CC713, CCW from Containment.

LOCATION: CCW Pump Room, 590' Level.

b. IF Containment pressure is less than 4 psig, THEN **RESET** Containment High Pressure Circuits by pushing the left and right high pressure reset pushbuttons.

- CHPL - RESET
- CHPR - RESET

Location: Panel EC-13

c. IF CHP is RESET, THEN **PLACE** CCW Isolation Valves Handswitches to AUTO:

- HS-0910, Comp Clg Water Inlet to Cntmt Switch KEY: 337
- HS-0911, Comp Cool Water Discharge From Contain Sw KEY: 338
- HS-0940, Comp Cool Water Discharge From Contain Sw KEY: 336

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	_____
	Group #	_____	_____
	K/A #	G2.1.44	
	Importance Rating	3.9	_____

K/A Statement: Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supported instrumentation.

Proposed Question:

The Plant is in the middle of core offload during a refueling outage. In accordance with GOP-11, "Refueling Operations and Fuel Handling," and LCO 3.7.12, "Fuel Handling Area Ventilation System," core alterations, movement of irradiated fuel, or cask movements are required to be suspended if which of the following Fuel Handling Area Ventilation System alignments is true:

A. Only V-8/B, Fuel Handling Exhaust Fan, is operating.

B. V-7, Fuel Handling Area Supply Fan, is operating.

C. Both V-70A/B, Fuel Handling Area Exhaust Fans, OFF.

D. V-69, Fuel Handling Area Supply Fan, OFF.

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, per GOP-11, no more than one, V-8A or V-8B, can be operating to comply with LCO 3.7.12.
B. Correct, per GOP-11, V-7 must be OFF.
C. Incorrect, per GOP-11, V-70A/B must be OFF.
D. Incorrect, per GOP-11, V-69 must be OFF.

Technical Reference(s): GOP-11
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

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

	New	___X___
Question History:	Last NRC Exam	

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 _10__	
	55.43 _____	

Comments:

5.6.3 If operational CAMs are not available:

a. Continuous air sampling of the affected area shall be performed when refueling operations are in progress.

b. Samples shall be analyzed for airborne activity at two-hour intervals.

c. RIA-1817 and RIA-5712 may be used as additional indications of airborne radioactivity levels. One or more Containment Air Coolers must be running for RIA-1817 to draw a valid sample and the selector switch must be positioned to the Recirculation Fans.

5.6.4 If core alterations or movement of irradiated fuel in Containment is in progress, RIA-2316 and RIA-2317 are required to be operable.

a. If less than the above radiation monitors are operable, refer to Technical Specifications LCO 3.3.6 for required actions.

5.7 FUEL HANDLING VENTILATION REQUIREMENTS (CAP047528)

5.7.1 When Technical Specifications LCO 3.7.12, "Fuel Handling Area Ventilation System," is applicable, the following system alignment is required:

- No more than one V-8A/B, Fuel Handling Exhaust Fan operating
- V-7, Fuel Handling Area Supply Fan OFF
- V-70A/B, Fuel Handling Area Exhaust Fans OFF
- V-69, Fuel Handling Area Supply Fan OFF

5.7.2 If fuel handling ventilation changes are required when Technical Specifications LCO 3.7.12, "Fuel Handling Area Ventilation System," is applicable, ensure that core alterations, movement of irradiated fuel or cask movement is suspended while changing the alignment to avoid unintended entry into LCO 3.7.12 required actions.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.2.12	
	Importance Rating	3.0	

K/A Statement: Knowledge of surveillance procedures.

Proposed Question:

Diesel Generator (DG) 1-1 is running for a monthly surveillance test per MO-7A-1, "Emergency Diesel Generator 1-1." While raising load on DG 1-1 to gather two-hour load limit data, the NCO incorrectly stabilized load at 2790 kW.

For DG 1-1 to be below the two-hour load limit, which one of the following is the least amount of load that must be reduced?

A. 50 kW

B. 100 kW

C. 150 kW

D. 300 kW

Proposed Answer: **A**

Explanation (Optional):

A. Correct, the two-hour load limit is 2750 kW. Reducing DG load by 50 kW would place DG loading at 2740 kW, under the maximum two-hour load limit.
B. Incorrect, the applicant is applying the maximum value of the DBA load band of 2700 kW.
C. Incorrect, the applicant is applying the minimum value of the DBA load band of 2650 kW.
D. Incorrect, the applicant is applying the maximum continuous load limit of 2500 kW.

Technical Reference(s): MO-7A-1, SOP-22, PL-EDG Emergency Diesel Generators Lesson Plan

(Attach if not previously provided, including version/revision number) ____________

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ________

Modified Bank # ___X___ (Note changes or attach parent)
New _______

Question History: Last NRC Exam Palisades 2010
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:

Question modified from Palisades 2010 NRC Exam. Modified stem to ask for 2-hr load limit rather than continuous load limit. Changed distractors to accommodate the change in the stem.

NOTE: Actual load readings, local and remote, should be taken simultaneously to verify meter calibrations.

5. **RECORD** engine performance data at 1.0 MW and 2.0 MW.

LOAD (MW)	ACTUAL LOAD		7R RACK READING (mm)	MANIFOLD PRESSURE PI-EAMP-1
	C-04 PANEL	EC-22 PANEL		
1.0				
2.0				

d. **REPEAT** Steps a through c until Diesel Generator 1-1 reaches full load at 2.40 MW (2.30 to 2.50 MW).

e. WHEN Diesel Generator 1-1 reaches full load, THEN **RECORD** time below and in Step 5.9.1a.

Time Diesel Generator 1-1 reached full load: ____________

NOTE: Raising or lowering generator load requires only a small governor adjustment. Small adjustment and observing system response will limit overshoot or undershoot of desired band.

5.8.2 **PERFORM** the following to obtain Diesel Generator 1-1 DBA load data:

a. **RAISE** Diesel Generator 1-1 load to between 2650 KW and 2700 KW using the G1-1/GSL, D/G 1-1 Governor Set Point Switch on EC-22 panel AND **RECORD** time.

Time load is 2650 to 2700 KW:_____________

NOTE: Slight power excursions outside of the band will not invalidate the test.

b. **STABILIZE** load AND **OPERATE** the Diesel Generator 1-1 for at least 15 minutes.

p. WHEN synchroscope nears "1200" hours on meter, THEN **CLOSE** the appropriate Generator Breaker:

D/G 1-1
152-107, D/G 1-1 to Bus 1C

D/G 1-2
152-213, D/G 1-2 to Bus 1D

q. **CHECK CLOSED** the Generator Breaker AND **ADD** approximately 50 KW to the generator with the generator Governor Setpoint switch.

r. **TURN OFF** Synchroscope.

CAUTION

Each D/G is limited to a 2500 KW continuous load rating and a 2750 KW two-hour load rating.

s. **ADJUST** D/G load as follows:

1. **RAISE** OR **LOWER** D/G load with the Governor Set Point switch.

2. WHEN loading the D/G, THEN **RAISE** load in approximately 500 KW increments. **ALLOW** Engine to run at each power level for approximately five minutes before raising load.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.2.35	
	Importance Rating	3.6	

K/A Statement: Ability to determine Technical Specification Mode of Operation.

Proposed Question:

Given the following plant conditions:

- The current time is 0400.
- Reactor power is 0%.
- All control rods are fully inserted.
- PCS average temperature is 320°F.
- PCS cooldown rate is 37°F/hr.

Assuming PCS cooldown rate remains stable, what MODE, as defined by Technical Specifications, will the Plant be in at 0800?

A. MODE 3

B. MODE 4

C. MODE 5

D. MODE 6

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, the Plant is in Mode 3 as of 0400, but will not be in Mode 3 at 0800 given the conditions.
B. Incorrect, the Plant will pass through Mode 4 during the cooldown between 0400 and 0800, however, the Plant will exit Mode 4 prior to 0800.
C. Correct, after 4 hours of a 37°F/hr cooldown rate, PCS temperature will be 172°F. This would place the Plant in Mode 5.
D. Incorrect, no information was provided for head tensioning. Mode 6 requires one or more head bolts less than fully tensioned.

Technical Reference(s): Technical Specifications, Section 1.1
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: ______None______

Learning Objective: ________________________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10_

55.43 _____

Comments:

1.1 Definitions

LEAKAGE	a. Identified LEAKAGE (continued) 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known not to interfere with the operation of leakage detection systems and not to be pressure boundary LEAKAGE; and 3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator to the Secondary System (primary to secondary LEAKAGE). b. Unidentified LEAKAGE All LEAKAGE (except Primary Coolant Pump seal leakoff) that is not identified LEAKAGE; c. Pressure Boundary LEAKAGE LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an PCS component body, pipe wall, or vessel wall.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average primary coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 300
4	Hot Shutdown[b]	< 0.99	NA	$300 > T_{ave} > 200$
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
	Tier #	3	
	Group #		
	K/A #	G2.3.4	
	Importance Rating	3.2	

K/A Statement: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question:

Given the following conditions:

- The Plant automatically tripped and Safety Injection actuated due to a Large Break LOCA that occurred inside containment.
- LPSI Pump P-67B suction piping was damaged from a water hammer event.
- All attempts to isolate the leak from the Control Room have been unsuccessful.
- An NPO was attempting to isolate the leak locally when he slipped and was injured. The NPO remains in the area and needs assistance.
- Another NPO, stating that he fully understands the potential health risks, has volunteered to find the injured NPO and bring them to a low dose area.

What is the maximum allowed Total Dose (TEDE) exposure the Emergency Plant Manager can authorize the NPO to receive while performing this task?

A. 5 REM.

B. 10 REM.

C. 25 REM.

D. No upper limit for TEDE exposure.

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, this limit is the 10CFR20 annual limit for the whole body.
B. Incorrect, this limit applies to the protection of property.
C. Incorrect, this limit applies to life-saving or protection of large populations, not on voluntary basis.
D. Correct, no upper TEDE limit applies to life-saving or protection of large populations only on a voluntary basis to persons who are fully aware of the risks involved.

Technical Reference(s): Site Emergency Plan SEP

(Attach if not previously provided, including version/revision number) ______

Proposed references to be provided to applicants during examination: ____None____

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 __12__
55.43 ______

Comments:

6.5 AID TO AFFECTED PERSONNEL

6.5.1 Emergency Personnel Exposure Criteria

Although an emergency situation transcends the normal requirements for limiting exposure, there are suggested levels of exposure acceptable in emergencies. Even under these conditions, every reasonable effort to minimize exposure must be made and personnel must be provided with appropriate monitoring devices. Three categories of risk versus benefit must be considered:

a. Saving of human life and reduction of injury.

b. Protection of health and safety of the public.

c. Protection of property.

In order to avoid restricting actions that may be necessary to save lives, it shall be left to the judgment of the individual to determine the amount of exposure that he will accept to perform an emergency action that will result in the saving of human life. Emergency team members are instructed in radiation effects and the risks involved for emergency doses. Basic guidelines provided to emergency team members are the EPA recommendations contained in Table 6-3. These exposures must be authorized by the Emergency Plant Manager (with the exception of life-saving efforts) based on the recommendation of the TSC Rad Coordinator.

The Radiation Protection Procedures shall be followed. In the event emergency exposure limits are approved, the same administrative methods for dose control shall be used with the higher emergency exposure limits.

Once the emergency condition has been mitigated, steps shall be taken to recover from the incident. All actions from this point shall be preplanned in order to limit exposures. Normal exposure limits will be used, areas will be controlled, and exposure of personnel documented.

TABLE 6-3
GUIDANCE ON DOSE LIMITS FOR WORKERS PERFORMING EMERGENCY SERVICES

Dose Limit[a] (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	lifesaving or protection of large populations	only on a voluntary basis to persons fully aware of the risks involved

[a]Sum of external effective dose equivalent and committed effective dose equivalent to nonpregnant adults from exposure and intake during an emergency situation. 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 from an incident, except those received in unrestricted areas by members of the public during the intermediate phase of the incident.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	_____
	Group #	_____	_____
	K/A #	G2.3.14	
	Importance Rating	3.4	_____

K/A Statement: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question:

A welding contractor arrived on site today (9/24/16) and will be performing weld overlay work on the Reactor Head during the upcoming outage three weeks from now. The contractor workers' radiation exposure history in the last year is given as follows:

- 200 mR whole body from a medical procedure three weeks ago.
- 300 mR TEDE while at Monticello Nuclear Plant from 12/3/15 to 12/6/15
- 500 mR TEDE while at Perry Nuclear Plant from 3/6/16 to 3/13/16
- 800 mR TEDE while at Palisades Nuclear Plant from 5/1/16 to 5/14/16
- The contract worker has provided completed NRC Form 5's for each quarter for the past 2 years.

Assuming no dose extensions have been authorized for the worker beyond the Annual Entergy Administrative Dose Guideline (ADG), which one of the following values is the maximum amount of whole body radiation the worker can receive at Palisades during the upcoming refueling outage and not exceed the Annual ADG?

A. 200 mR.

B. 400 mR.

C. 700 mR.

D. 1200 mR.

Proposed Answer: **C**

Explanation (Optional):
Per EN-RP-201-004, the ADG is a company-imposed occupational dose guideline used for the purposes of maintaining doses below the regulatory dose limits established for 10CFR Part 20. As this is occupational dose only, the 200 mR from the medical procedure does not count towards the ADG. Additionally, the 300 mR from working at Monticello does not count as it was during the last calendar year. Only the current calendar year accumulated dose counts towards the ADG (500 mR + 800 mR = 1300 mR). Therefore, 700 mR of margin exists to reach the ADG limit.

A. Incorrect, the applicant believes that the medical exposure counts as well as the prior

year exposure from Monticello.

B. Incorrect, the applicant understands that the medical exposure does not count, but incorrectly counts the dose obtained from working at Monticello, which occurred in the prior year.
C. Correct, the occupational dose accumulated in within the calendar year counts. (2000-(500+800) = 700 mR margin
D. Incorrect, the applicant believes the ADG is specific to dose accumulated within the Entergy fleet.

Technical Reference(s): EN-RP-201
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: ______None_______

Learning Objective: ________________________ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _12__
55.43 _____

Comments:

5.3, continued

[2] Maximum Annual Administrative Guidelines

- TEDE = 4.5 rem
- LDE = 12 rem
- SDE, WB = 40 rem
- SDE, ME = 40 rem
- Declared Pregnant Woman (DPW) TEDE = 50 mrem/month, 450 mrem/gestation period.
- Minors TEDE – Minors are not allowed access to RCAs.
- Unmonitored individual TEDE = 100 mrem/year
- Members of the Public TEDE = 100 mrem/year

[3] Routine Annual Administrative Guidelines

- TEDE = The lesser of:

  2000 mrem per year <u>OR</u>

  5000 mrem – (1250 mrem x UQ per year)

  <u>Where</u> UQ = the number of undocumented quarters for the current year

  (EXCEPT when Lifetime TEDE is greater than or equal to the individuals age x 1 rem in which case the annual TEDE guideline will be set to 1 rem.)

- LDE = 12 rem
- SDE, WB = 40 rem
- SDE, ME = 40 rem
- TODE = 40 rem

5.3[3], continued

- Lifetime greater than age = 1000 mrem onsite TEDE up to 2000 mrem TEDE for year.
- Declared Pregnant Woman TEDE = 50 mrem/month, 400 mrem/gestation period
- Minors TEDE – Minors are not allowed access to RCAs.
- Unmonitored Individuals TEDE = 50 mrem/month, 100 mrem/year
- Members of the General Public TEDE = 50 mrem

5.4 EXTENDING ADMINISTRATIVE DOSE GUIDELINES (ADG)

[1] Prior to dose extension, requesting supervisor should:

(a) Evaluate dose equalization in the department, **AND**

(b) Check other personnel qualifications to perform tasks, **AND**

(c) Check other means to reduce dose.

[2] Obtain verification of the worker's current year exposure prior to allowing a worker to exceed 2000 mrem TEDE for the year. Any of the following may be used for verification:

(a) An NRC Form 5 or equivalent provided by either the worker or the licensee(s) providing monitoring for each monitoring period, **OR**

(b) An NRC Form 4 or equivalent signed by the person, **OR**

(c) Electronic, telephone or facsimile transfer of exposure data provided by the licensee(s) providing the monitoring.

5.4, continued

[3] Extend a Radiation Workers' administrative TEDE ADG to the guidelines described in the following table, after obtaining the indicated approvals.

NOTE

Responsible individuals may be designated to authorize dose extensions.

Exposure Guideline	Requirements	Authorizations (Note)
Greater than 2000 mrem and less than or equal to 3000 mrem per year	No undocumented quarters in the current year	Individual's supervisor recommends RP Manager approves
Greater than 3000 mrem and less than or equal to 4000 mrem per year	No undocumented quarters in the current year	Individual's supervisor recommends Radiation Protection Manager approves Plant General Manager approves
Greater than 4000 mrem and less than 4500 mrem per year for Radiation Workers. Greater than 400 mrem but less than or equal to 450 mrem /gestation period	No undocumented quarters in the current year	Radiation Protection Manager approves Plant General Manager approves Site Vice President approves
Greater than 1000 mrem and less than or equal to 2000 mrem for individuals whose lifetime exposure greater than or equal to 1000 mrem * n where n = age	No undocumented quarters in the current year	Individual's Supervisor recommends Radiation Protection Manager approves

Examination Outline Cross-Reference:

		RO	SRO
	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.4.2	
	Importance Rating	3.9	

K/A Statement: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question:

Which of the following conditions would result in raising the pressure setpoint portion of the Thermal Margin/Low Pressure reactor trip?

A. Operating with Group 4 rods inserted 4" into the core instead of full out.

B. Operating with power at 95% instead of 100%.

C. Operating with T_{ave} 2°F above program.

D. Operating with pressurizer pressure 15 psia below normal.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, the applicant incorrectly believes that this will cause ASI to be more positive, but actually lowers temperature in the top of the core, thus allowing a greater margin to DNB.
B. Incorrect, the applicant incorrectly believes that this will result in a lower inlet temperature to the core, but at a lower power less heat is added so the likelihood of DNB is lowered.
C. Correct, Operating at an elevated temperature places the plant closer to DNB conditions. This is actually determined by T_{cold} temperatures which will be higher with a higher Tave and the same power level.
D. Incorrect, the applicant incorrectly believes that a lower pressure affects the trip setpoint, but actually affects the margin to trip.

Technical Reference(s): LCO 3.3.1, PL-RPS Reactor Protection System Lesson Plan

(Attach if not previously provided, including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam Palisades 1999

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__5__
	55.43	_____

Comments:

Lesson Content	Instructor Notes

a. *Protection Against Single Failures*

(1) Field Instrument Failure

OBJ 28	Predict how the following conditions will impact operation of the Reactor Protective System: high failure of monitored parameter input signal

(a) Failure in the high direction

i. If the trip unit is designed to trip on a high value of the measured plant parameter, then a field instrument failure in the high direction will cause the trip unit to trip.

- This will not cause a reactor trip because the logic requires two channels to reach the set point, and only one channel is in the trip condition.
- This will not prevent a reactor trip because three operable redundant instruments remain in service, and any two of the operable channels going into trip will cause a reactor trip.
- The trip logic is changed from 2/4 to 1/3, unless the failed instrument's trip unit is bypassed.
    - If the failed instrument's trip unit is bypassed, then the trip logic is changed to 2/3.
- This will not cause a reactor trip because the trip unit has not actuated a trip signal in any of the three associated logic matrices.
- This will not prevent a reactor trip because three operable redundant instruments remain in service, and any two of the operable channels going into trip will cause a reactor trip.
    - Low failure of a pressurizer pressure instrument will prevent the pressurizer high pressure trip unit from actuating on the affected channel, but will actuate the Thermal Margin/Low Pressure trip unit.

ii. If the trip unit is designed to trip on a low value of the plant parameter, then a field instrument failure in the high direction will prevent the trip unit from tripping.

- This will not cause a reactor trip because the trip unit has not actuated a trip signal in any of the three associated logic matrices.
- This will not prevent a reactor trip because three operable redundant instruments remain in service, and any two of the operable channels going into trip will cause a reactor trip.

Lesson Content	Instructor Notes

- High failure of a pressurizer pressure instrument will prevent the Thermal Margin/Low Pressure trip unit from actuating on the affected channel, but will actuate the pressurizer high pressure trip unit.

- The trip logic is changed from 2/4 to 2/3, unless the failed instrument's trip unit is placed in the trip condition.
  - If the failed instrument's trip unit is placed in the trip condition, then the trip logic is changed to 1/3.

OBJ 28	Predict how the following conditions will impact operation of the Reactor Protective System: low failure of monitored parameter input signal

(b) Failure in the low direction

i. If the trip unit is designed to trip on a high value of the measured plant parameter, then a field instrument failure in the low direction will prevent the trip unit from tripping.

- This will not cause a reactor trip because the trip unit has not actuated a trip signal in any of the three associated logic matrices.

- This will not prevent a reactor trip because three operable redundant instruments remain in service, and any two of the operable channels going into trip will cause a reactor trip.
  - Low failure of a pressurizer pressure instrument will prevent the pressurizer high pressure trip unit from actuating on the affected channel, but will actuate the Thermal Margin/Low Pressure trip unit.

SLIDE 1	***Basis – Thermal Margin / Low Pressure (TM/LP) Trip***

(a) Thermal Margin / Low Pressure (TM/LP) Trip

i. ***The TM/LP trip is provided to prevent reactor operation when the DNBR is insufficient. The TM/LP trip protects against slow reactivity or temperature increases, and against pressure decreases.***

ii. ***The trip set points are derived from the core thermal limits through application of appropriate allowances for measurement uncertainties and processing errors. The allowances specifically account for instrument drift in both power and inlet temperatures, calorimetric power measurement, inlet temperature measurement, and primary system pressure measurement.***

iii. ***Other uncertainties including allowances for assembly power tilt, fuel pellet manufacturing tolerances, core flow measurement uncertainty and core bypass flow, inlet temperature measurement time delays, and ASI measurement, are included in the development of the TM/LP trip set point used in the accident analysis.***

Table 3.3.1-2 (page 1 of 1)
Thermal Margin/Low Pressure Trip Function Allowable Value

The Allowable Value for the Thermal Margin/Low Pressure Trip, P_{trip}, is the higher of two values, P_{min} and P_{var}, both in psia:

$P_{min} = 1750$
$P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9559$

Where:

$QA = -0.720(ASI) + 1.028;$	when $-0.628 \leq ASI < -0.100$
$QA = -0.333(ASI) + 1.067;$	when $-0.100 \leq ASI < +0.200$
$QA = +0.375(ASI) + 0.925;$	when $+0.200 \leq ASI \leq +0.565$
ASI = Measured ASI	when $Q \geq 0.0625$
ASI = 0.0	when $Q < 0.0625$
$QR_1 = 0.412(Q) + 0.588;$	when $Q \leq 1.0$
$QR_1 = Q;$	when $Q > 1.0$

Q = THERMAL POWER/RATED THERMAL POWER

T_{in} = Maximum primary coolant inlet temperature, in °F

ASI, T_{in}, and Q are the existing values as measured by the associated instrument channel.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.4.6	
	Importance Rating	3.7	

K/A Statement: Knowledge of EOP mitigation strategies.

Proposed Question:

The Plant has entered EOP-7.0, "Loss of All Feedwater Recovery." Both Steam Generator levels are at 10% narrow range and slowly decreasing. Other than attempting to re-establish feedwater, which of the following is performed to mitigate the event?

A. Cooldown the PCS T_{ave} to less than 525°F.

B. Maintain PCS T_{ave} less than 540°F.

C. Minimize PCS subcooling.

D. Secure ONLY one PCP in each loop.

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, a cooldown to 525°F-532°F is performed to allow feeding the S/Gs with AFW, however, since adequate shutdown margin may not be established at this point, 525°F is the bounding low temperature.
B. Correct, the S/G steaming strategy to maintain T_{ave} less than 540°F (the maximum post-trip temperature) is intended to maintain existing PCS temperature and prevent uncontrolled heatup; ensuring PCS heat removal is maintained throughout the event.
C. Incorrect, subcooling should be maximized in order to minimize potential for voiding and to provide sufficient margin for reestablishing HPSI flow if the minimum value cannot be maintained.
D. Incorrect, all PCPs are secured to minimize the heat input to the PCS.

Technical Reference(s): EOP-7.0
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

	Modified Bank #	________ (Note changes or attach parent)
	New	___X___

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__10__
	55.43	_____

Comments:

2.0 STRATEGY

Prior to implementing the actions provided in the Loss of All Feedwater procedure, the operator would have performed EOP-1.0, "Standard Post Trip Actions," and concluded that a loss of feedwater had occurred. In the Loss of All Feedwater procedure, the operator begins using the Safety Function Status Checks to verify the status of the safety functions and the correct procedure has been implemented. Loss of All Feedwater procedure actions provide instructions on regaining and maintaining PCS inventory control, PCS pressure control, and PCS heat removal.

The operator actions for a Loss of All Feedwater are directed at determining the cause of the Loss of All Feedwater, minimizing heat input to the PCS, conserving steam generator inventory, and regaining a source of feedwater to at least one steam generator. If this is not possible, the procedure provides explicit criteria and instructions for initiating Once-Through-Cooling. If a source of feedwater is regained, and a cooldown is necessary, operator actions for cooldown to shutdown cooling are provided.

4.0 OPERATOR ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS
© 1. **VERIFY** ANY of the following, at intervals of approximately fifteen minutes: a. Attachment 1, "Safety Function Status Check Sheet," acceptance criteria are satisfied. b. Corrective actions to restore Attachment 1, "Safety Function Status Check Sheet," acceptance criteria are effective.	1.1 **GO TO** ONE of the following: • EOP-1.0, "Standard Post-Trip Actions," Attachment 1, "Event Diagnostic Flowchart" <u>AND</u> **RE-DIAGNOSE** the event. • For events initiated from a lower mode, the EOP considered appropriate by the Shift Supervisor. • EOP-9.0, "Functional Recovery Procedure."
© 2. **REFER TO** the Site Emergency Plan <u>AND</u> **CLASSIFY** the event per EI-1, " Emergency Classification and Actions."	
3. **OPEN** the placekeeper <u>AND</u> **RECORD** the time of EOP entry.	
<u>NOTE</u>: Placing PZR heaters in AUTO or controlling heaters manually will be necessary for pressure control due to the loss of main sprays.	
4. **STOP** ALL PCPs.	

Maintaining PCS subcooling greater than or equal to **[minimum RCS subcooling]** ensures that the fluid surrounding the core is subcooled, and provides sufficient margin for reestablishing HPSI flow if the minimum value can not be maintained. Voids may exist in some parts of the PCS such as the Reactor Vessel head. In themselves, this is not a major problem, provided that the voids do not interfere with core heat removal.

STEP 4

NOTE: Placing PZR heaters in AUTO or controlling heaters manually will be necessary for pressure control due to the loss of main sprays.

4. **STOP** ALL PCPs.

CEN-152 LOAF Step 4:

4. Stop all RCPs.

Technical Basis:

A Loss of All Feedwater results in a reduction of the ability of the Steam Generators to remove heat from the PCS. Since natural circulation heat removal is adequate to remove the decay heat generated in the core, the PCPs are stopped to eliminate their heat input to the PCS.

Technical Basis:

The intent of this step is to ensure that PCS T_{AVE} is being controlled less than 540°F **[maximum expected post-trip temperature]**.

The operator is directed to ensure T_{AVE} is less than 540°F **[maximum expected post-trip temperature]** by controlled operation of ANY of the following:

- Turbine Bypass Valve
- Atmospheric Steam Dumps
- Hogging Air Ejector

If the Condenser is available, the Turbine Bypass Valve is the preferred method for controlling PCS heat removal. If the loss of power also caused a loss of Instrument Air header pressure, the Atmospheric Steam Dump Valves (ASDVs) must be used to control PCS temperature. The ASDVs have an automatic nitrogen backup that is immediately available for an effectively unlimited period.

If the ASDVs and Turbine bypass valve fail to operate to maintain temperature and the additional steaming paths do not provide adequate cooling, then T_{AVE} will rise to a **{maximum expected temperature while on S/G code safeties}**.

Initiation of a controlled S/G steaming path via the Hogging Air Ejector requires system alignments using EOP Supplement 23.

T_{AVE} can be controlled at any point less than 540°F **[maximum expected post-trip temperature]**. Due to plant conditions at the entry to this EOP, T_{AVE} may initially be well below the expected post-trip band.

For the event, S/G steaming strategy is intended to maintain existing PCS temperature and prevent uncontrolled heatup.

This step ensures PCS heat removal and is continuously applicable.

Training Emphasis:

None

STEP 18

18. IF P-8C AFW Pump is operating AND ALL of the following conditions exist:

- Unable to provide adequate AFW flow

- S/G pressures are greater than 900 psia

THEN **LOWER** T_{AVE} to between 525°F and 532°F using Turbine Bypass Valve or Atmospheric Dump Valves.

CEN-152 LOAF Step:

None

Technical Basis:

A loss of feedwater event is the bounding condition for the Auxiliary Feedwater System. For P-8A and P-8B the required flow is 280 gpm (140 to each) at 985 psig to both steam generators or 280 gpm at 985 psig to one steam generator. For Pump P-8C, the required flow for the loss of feedwater event is 280 gpm (140 to each) at 900 psia to both steam generators or 280 gpm at 900 psia to one steam generator. Operation of the turbine bypass system or atmospheric dump valves is required to depressurize to 900 psia (T_{AVE} 525°F to 532°F). The preceding flowrates will remove decay heat and pump heat from four operating primary coolant pumps. A minimum T_{AVE} of 525°F was specified since adequate shutdown margin may not be established yet or emergency boration in progress at this point for going below 525°F.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.4.11	
	Importance Rating	3.4	

K/A Statement: Knowledge of abnormal condition procedures.

Proposed Question:

Which of the following combination of conditions meets the reactor trip criteria per AOP-35, “Loss of Service Water”?

1. Annunciator EK-1124, “Traveling Screen Hi DP” in alarm.
2. Service Water Bay level is 574’
3. Annunciator EK-1165, “Non Critical Serv Water Lo Press” in alarm
4. EK-0259, “Exciter Cooler Hi Temp” in alarm.

A. 1 and 2

B. 1 and 3

C. 2 and 3

D. 3 and 4

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, traveling screen high ΔP alarms are not reactor trip criteria. If the alarm is in and bay level is dropping, the Dilution Water Pump(s) must be secured and reactor power reduced. Service Water Bay level has reactor trip criteria, however, the level is 572’, which is not met in this case.
B. Incorrect, traveling screen high ΔP alarms are not reactor trip criteria. If the alarm is in and bay level is dropping, the Dilution Water Pump(s) must be secured and reactor power reduced.
C. Incorrect, Service Water Bay level has reactor trip criteria, however, the level is 572’, which is not met in this case.
D. Correct, Non-Critical Service Water pressure low combined with high exciter temperatures meets reactor trip criteria per AOP-35.

Technical Reference(s): AOP-35
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ___________________________ (As available)

Question Source: Bank # ________

Modified Bank # ________ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __10__

55.43 ______

Comments:

REACTOR AND EQUIPMENT TRIP CRITERIA

Reactor Trip

- Service Water Bay level lowers to 572'
- Operator actions are not maintaining either Critical Service Water Header Pressure greater than or equal to 42 psig
- Loss of Non-Critical Service Water as indicated by the following alarms:
  - EK-1165, "NON CRITICAL SERV WATER LO PRESS" (45 psig)

  AND

  - PPC Urgent Alarm, "EXC FIELD COLD AIR RTD-31" [T_EXCITER_31] (48°C)

    OR

  - PPC Urgent Alarm, "EXC DIODE COLD AIR RTD-32" [T_DIODE_32] (48°C)

    OR

  - EK-0259, "EXCITER COOLER HI TEMP" (50°C)

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	2
	K/A #	009.EA2.34	
	Importance Rating	_____	4.2

K/A Statement: Ability to determine or interpret the following as they apply to a small break LOCA: Conditions for throttling or stopping HPI.

Proposed Question:

A small break Loss of Coolant Accident (LOCA) has occurred. The crew is at Step 23 of EOP-4.0, "Loss of Coolant Accident Recovery," SI Pump throttling.

The following plant conditions are observed:

- Containment Pressure is 3.4 psig and slowly rising.
- Containment temperature is 130°F and slow rising.
- PCS Pressure is 1400 psia and rising.
- Average CET temperature is 543°F and lowering.
- PCS subcooling exceeds the minimum required and is rising.
- Actual S/G levels are 64% and stable for both S/Gs.
- Actual Pressurizer level is 26% and slowly rising.
- RVLMS channels indicate 4 Red lights and 4 Green lights.
- ONLY HPSI Pump P-66A is running.

For the given conditions:

The required action in response to SI pump throttling criteria is to __(1)__. The basis of the action to throttle SI pump flow is to __(2)__?

A. (1) RAISE HPSI flow AND START HPSI Pumps as necessary
(2) Reduce potential for PCS overpressurization

B. (1) THROTTLE HPSI flow OR STOP one HPSI Pump at a time
(2) Reduce potential for PCS overpressurization

C. (1) RAISE HPSI flow AND START HPSI Pumps as necessary
(2) Reduce potential for PCS overcooling

D. (1) THROTTLE HPSI flow OR STOP one HPSI Pump at a time
(2) Reduce potential for PCS overcooling

Proposed Answer: **A**

Explanation (Optional):

The purpose of throttling SI pump flows is to reduce the potential for PCS overpressurization and have better control of PZR pressure and level via normal methods. The applicant could misunderstand the basis for this action by believing the influx of cold, borated water in excess of the minimum required could cause an overcooling concern.

SI Pump throttling criteria is met if ALL of the following are met:

1) Average of QCET is at least 25°F subcooled or greater than the minimum subcooling curve on EOP Supplement 1 (degraded containment conditions)
2) Corrected PZR level is greater than 20% (40% for degraded containment) and controlled.
3) At least one S/G is available for PCS heat removal with corrected level being maintained or being restored to between 60% and 70%.
4) Operable RVLMS channels indicate greater than 102 inches above the bottom of fuel alignment plate (621'8")

A. Correct, SI pump throttling criteria is not met. Containment is degraded in this case, as containment pressure is greater than 3 psig. With degraded containment, pressurizer level must be greater than 40% and it is not. Saturation temperature at 1400 psia is 587°F, therefore, subcooling is 34°F. S/G level and RVLMS indicated level are both satisfactory. Perform RNO action.
B. Incorrect, SI pump throttling criteria is not met, see choice A for explanation.
C. Incorrect, SI pump throttling criteria is not met, see choice A for explanation. The purpose of SI pump throttling is to reduce the possibility for PCS overpressurization.
D. Incorrect, while SI Pump throttling criteria is not met, the purpose of SI pump throttling is to reduce the possibility for PCS overpressurization.

Technical Reference(s): EOP-4.0, EOP-4.0 Basis____________________
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____
55.43 __5__

Comments:

This meets SRO criteria 10CFR55.43 (b)5 since the applicant must assess plant conditions and determine the required procedural action in response to those conditions. The applicant must also know the basis behind the procedural action.

STEP 23

NOTE: Use ANY of the following to determine Average of Qualified CETs:

- PPC point "KCETA" (Average of Qualified CETs)
- PPC Incore Qualified CET Map (PPC page 313)
- Manual calculation. Refer to SOP-34, "Plant Process Computer (PPC) System."

© 23. **VERIFY** SI Pump throttling criteria are satisfied by ALL of the following:

a. Based on the Average of Qualified CETs, PCS subcooling meets ONE of the following:

- At least 25°F subcooled for non-degraded Containment conditions
- Greater than the minimum subcooling curve on EOP Supplement 1 for degraded Containment conditions

b. Corrected PZR level is greater than 20% (40% for degraded Containment) and controlled. **REFER TO** EOP Supplements 9 and 10.

23.1. IF ANY of the SI Pump throttling criteria can NOT be maintained, THEN **RAISE** HPSI flow AND **START** HPSI Pumps as necessary.

PUMP	VALVE NUMBER	VALVE DESCRIPTION
Train 1		
P-66B	MO-3009	HPSI Train 1 to Loop 1B
	MO-3011	HPSI Train 1 to Loop 2A
	MO-3007	HPSI Train 1 to Loop 1A
	MO-3013	HPSI Train 1 to Loop 2B
Train 2		
P-66A	MO-3066	HPSI Train 2 to Loop 1B
	MO-3064	HPSI Train 2 to Loop 2A
	MO-3068	HPSI Train 2 to Loop 1A
	MO-3062	HPSI Train 2 to Loop 2B

c. At least one S/G is available for PCS heat removal with corrected level being maintained or being restored to between 60% and 70%. **REFER TO** EOP Supplement 11.

d. Operable RVLMS channels indicate greater than 102 inches above the bottom of fuel alignment plate (621' 8").

<u>CEN-152 LOCA Steps 18 and 19</u>:

18. IF HPSI pumps are operating, **AND ALL** of the following conditions are satisfied:

- RCS subcooling is greater than or equal to [minimum RCS subcooling]

- Pressurizer level is greater than [minimum level for inventory control] and **NOT** lowering

- At least one steam generator is available for RCS heat removal with level being maintained or restored to [normal control band].

- Reactor vessel level is greater than [top of the hot leg nozzles]

THEN <u>throttle</u> HPSI flow or <u>stop</u> **ONE** HPSI pump at a time.

19. **IF ANY** of the HPSI throttle criteria can **NOT** be maintained, **THEN**:

    a. Raise HPSI flow.

    b. Start HPSI pumps as necessary.

Technical Basis:

The intent of this step is to establish the conditions that must be met before SI flow can be reduced during an event.

For some events, the SI pumps will run continuously for a long period of time while PCS inventory, pressure, and heat removal control are being regained. In some cases, control of these three safety functions is not regained during the event and the SI pumps run for the duration of the event recovery. Throttling of SI is expected when the SIAS actuation was spurious, the leak rate can be accommodated by SI and Charging flow when PCS pressure lowers, or when a leak is isolated.

The following SI Pump throttling criteria are provided:

a. PCS subcooling is greater than or equal to **[minimum RCS subcooling]** or greater than the minimum subcooling curve for degraded Containment conditions in EOP Supplement 1. Representative CET temperature (from PPC data or manual calculation using OA-108) is used due to the location of the CETs. When natural circulation is in progress, the CETs provide the best indication of fluid conditions adjacent to the core. The CETs do not rely on loop flow (as do the loop RTDs) for detecting fluid conditions adjacent to the core. With no flow in the loops, the loop RTDs may not provide adequate indication of core fluid conditions.

   Maintaining PCS subcooling greater than or equal to **[minimum RCS subcooling]** ensures that the fluid surrounding the core is subcooled, and provides sufficient margin for reestablishing HPSI flow if the minimum value can not be maintained. Voids may exist in some parts of the PCS such as the Reactor Vessel head. In themselves, this is not a major problem, provided that the voids do not interfere with core heat removal.

b. PZR level being controlled greater than **[minimum level for inventory control]** coexisting with PCS subcooling greater than or equal to **[minimum RCS subcooling]** indicates that adequate PCS inventory exists to allow throttling or stopping SI flow.

- Reactor vessel level is greater than [top of the hot leg nozzles]

THEN throttle HPSI flow or stop **ONE** HPSI pump at a time.

Technical Basis:

The intent of this step is to ensure that HPSI flow is reduced or HPSI Pumps are stopped if SI Pump throttling criteria are met.

If HPSI Pumps and Charging Pumps were started by SIAS sequencing, then this step is used to reduce the chances of overpressurizing the PCS and low temperature stressing of the Reactor Vessel.

If all of the SI Pump throttling criteria are met, the operator may either throttle the HPSI injection valves or stop HPSI pumps as necessary. The operator should recognize that when HPSI flow is throttled, the core and PCS may heat up until S/G heat removal (PCS heat removal) can be increased by the operator. Depending upon how long the adjustment takes, the PCS fluid could expand appreciably, thereby further complicating the event recovery.

If HPSI Pumps and Charging Pumps were started by SIAS, then this step is used to reduce the potential for overpressurizing the PCS.

Training Emphasis

The normal throttling process should be to first throttle one pumps discharge valves or secure the first pump. The second pumps discharge valves should then be throttled and finally the second pump is secured. When throttling HPSI flow, the operator should attempt to maintain balanced flow to each loop.

Associated Notes, Cautions, Warnings:

None

Deviations from EPG:

This step uses the plant specific method for determining the necessity for throttling HPSI flow or stopping HPSI Pumps and is consistent with CEN-152 intent.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	1
	K/A #	000038.G2.1.20	
	Importance Rating	_____	4.6

K/A Statement: Ability to interpret and execute procedure steps.

Proposed Question:

Given the following:

- EOP-5.0, "Steam Generator Tube Rupture Recovery," has been implemented due to a Steam Generator Tube Rupture in progress on the 'A' S/G.
- 'A' S/G has been isolated per EOP-5.0.
- 'A' S/G pressure is 840 psia.
- 'B' S/G pressure is 560 psia.
- Pressurizer pressure is 930 psia.
- Pressurizer level is 42%.
- PCS temperature is 508°F.

For these conditions, which of the following actions, regarding Pressurizer pressure, will the CRS direct in accordance with EOP-5.0?

A. Raise Pressurizer pressure to re-establish adequate subcooling margin

B. Raise Pressurizer pressure to prevent backflow dilution of the PCS

C. Lower Pressurizer pressure to ensure Main Steam Safety Valves remain closed

D. Lower Pressurizer pressure to minimize leakage into the 'A' S/G from the PCS

Proposed Answer: D

Explanation (Optional):

Pressurizer pressure shall be maintained per the following criteria of EOP-5.0 Step 17 (continuously applicable):

- Less than 940 psia
- Within the limits of EOP Supplement 1
- Preferably within 50 psid of the isolated S/G pressure

A. Incorrect, subcooling margin is adequate at approximately 33°F. EOP Supplement 1 requires a minimum of 25°F subcooling.
B. Incorrect, raising Pressurizer pressure would prevent backflow to the PCS; however, per the EOP basis document, the amount of dilution would not jeopardize shutdown margin
C. Incorrect, Pressurizer pressure is maintained less than 940 psia in order to prevent the

MSSVs from opening.

D. Correct, the Pressurizer pressure and ruptured S/G pressure should be within 50 psid of each other. Maintaining PCS pressure approximately equal to or less than the affected S/G pressure allows for minimum from the PCS into the S/G.

Technical Reference(s):	EOP-5.0, EOP-5.0 Basis, EOP Supplement 1
(Attach if not previously provided,	___
including version/revision number)	___

Proposed references to be provided to applicants during examination: None

Learning Objective: ___ (As available)

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

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	___
	55.43	5

Comments:

This meets SRO-only criteria 10CFR55.43 (b)5 as the applicant must assess plant conditions and then direct procedural action to mitigate the event/transient as well as understanding the basis of why the action is being taken.

INSTRUCTIONS

© 17. **DEPRESSURIZE** the PCS by performing ALL of the following:

a. **MAINTAIN** PZR pressure within ALL of the following criteria:

- Less than 940 psia
- Within the limits of EOP Supplement 1
- Preferably within 50 psid of the isolated S/G pressure

b. **OPERATE** Main or Auxiliary Spray valves.

c. IF SI pump throttling criteria are met, THEN **PERFORM** ANY of the following:

1) **THROTTLE** HPSI flow

2) **CONTROL** charging and letdown flow

CONTINGENCY ACTIONS

17.1 IF PZR pressure can NOT be lowered and maintained within ALL of the following criteria:

- Less than 940 psia
- Within the limits of EOP Supplement 1,
- Preferably within 50 psid of the isolated S/G pressure

THEN **OPERATE** the PORVs as follows:

a. **ENSURE OPEN** PORV Isolation Valves. Refer to SOP-1B, "Primary Coolant System - Cooldown," Attachment 6.

b. **MAINTAIN** PZR pressure within limits of EOP Supplement 1.

c. **PLACE** BOTH of the following PORV LTOP enable keyswitches to ENABLE:

- HS-0105A (KEY: 1)
- HS-0105B (KEY: 4)

d. **CYCLE** one PORV using the handswitch to OPEN and AUTO or CLOSE as necessary to attain the desired pressure.

- PRV-1042B
- PRV-1043B

CEN-152 SGTR Step 9:

*9. Depressurize the RCS:

a. Maintain pressurizer pressure within **ALL** of the following criteria:

- Less than [lowest MSSV setpoint]
- [Approximately equal] to the most affected steam generator pressure
- [Post Accident PT curves]
- [RCP NPSH limits]

b. Operate main or auxiliary pressurizer spray.

c. **IF** HPSI throttle criteria are met, **THEN**:

1) Control charging and letdown flow.

2) Throttle HPSI flow as necessary.

9.1 [**IF** pressurizer pressure can **NOT** be lowered and maintained within the specified criteria, **THEN** operate the [PORV(s) or pressurizer vent(s)].

Technical Basis:

The intent of this step is to minimize the transfer of inventory from and to the PCS, to prevent lifting the Main Steam Safety Valves and establish control of PCS pressure.

The procedural goals associated with PCS Pressure Control are:

- Providing subcooling to support the core heat removal process,
- Avoiding overpressure situations for PTS and RT NDT considerations,
- Minimizing the pressure differential between the S/G and the PCS to minimize the leakage,
- Deliberately creating a primary to secondary differential pressure to establish backflow to control affected S/G level rise,

- Reduce S/G pressure/temperature,
- Controlling PCS pressure below the Main Steam Safety Valve (MSSV) lift pressure to prevent uncontrolled release of radioactivity to the environment.

Pressurizer pressure may be reduced by any of the following:

1. Operation of Pressurizer sprays and heaters.

2. Control of charging pumps, letdown and HPSI pumps (if HPSI Stop/Throttle criteria are met).

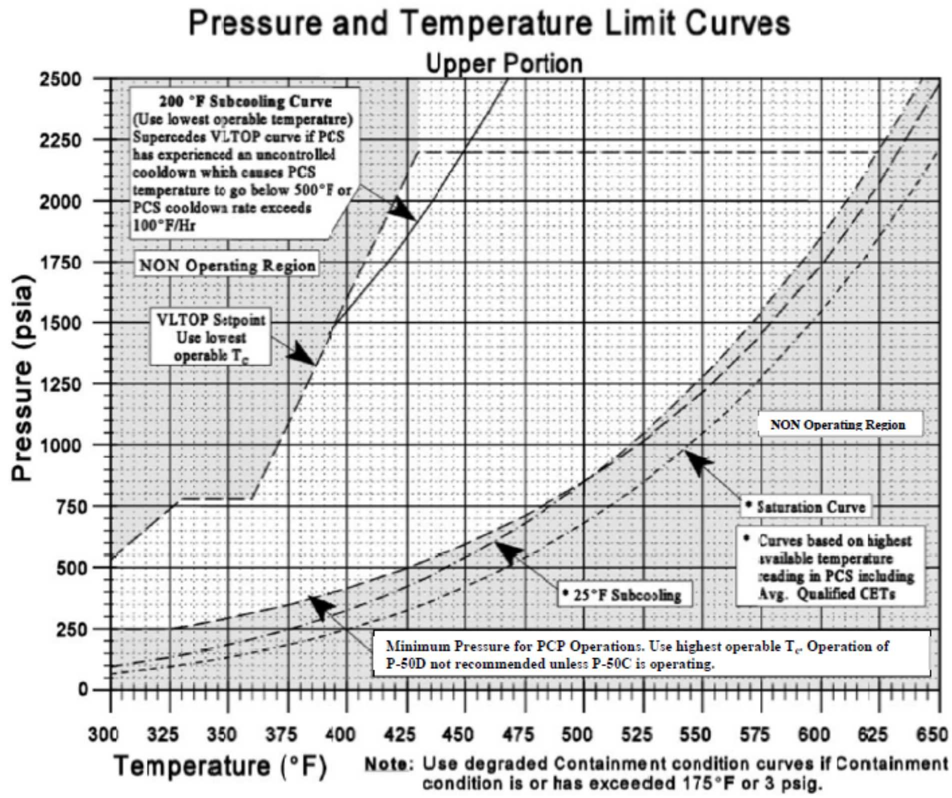
3. As a last resort, by operating the PORVs. Pressure control by this method requires close operator attention, because the resultant pressure decrease when the PORV(s) is opened can be dramatic. In addition, the operator must closely monitor PCS inventory control and pressure/ temperature conditions in the RDT/Containment while utilizing this method.

Maintaining the PCS pressure within the limits of the **[Post Accident PT curves]**, **[approximately equal]** to the isolated S/G pressure and below the **[lowest MSSV lift setpoint]** will minimize the loss of primary fluid to the secondary side and the possibility of overfilling the isolated S/G. This action will minimize the potential for release of radiation to the environment by minimizing PCS to S/G leakage. Maintaining PCS pressure approximately equal to or less than the affected S/G pressure allows for the backflow of secondary water into the PCS which provides several operational benefits.

These benefits include:

- S/G level can be maintained within the indicating range,
- Controlling S/G level, the probability of filling the main steam piping with water is greatly reduced,
- Use of the blowdown system for S/G level control can be minimized, thus minimizing contamination of the secondary,
- Depressurization of the isolated S/G can be performed without steaming to the condenser or to the atmosphere,
- Less secondary makeup water is required for the PCS cooldown.

Boron dilution of the PCS will occur due to unborated secondary water flowing through the tube rupture into the PCS. However, under most circumstances, this dilution will not threaten the maintenance of adequate shutdown margin.



PRESSURE AND TEMPERATURE LIMIT CURVES

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		1
K/A #	CE/E06.EA2.1	
Importance Rating		3.9

K/A Statement: Ability to determine and interpret the following as they apply to the (Loss of Feedwater): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question:

Given the following:

- The Plant has entered EOP-7.0,"Loss of All Feedwater Recovery."
- The Plant has experienced a loss of all instrument air.
- P-8A and P-8B, Auxiliary Feedwater Pumps, are NOT available.

Which one of the following procedures will the CRS use to feed the Steam Generators using P-8C, Auxiliary Feedwater Pump, for the above conditions?

A. EOP Supplement 19, "Alternate Auxiliary Feedwater Methods."

B. AOP-37, "Loss of Instrument Air."

C. SOP-12, "Feedwater System."

D. EOP Supplement 31, "Supply AFW Pumps from Alternate Sources."

Proposed Answer: A

Explanation (Optional):

A. Correct, EOP Supplement 19 contains directions for manually initiating AFW flow by manually operating AFW control valves without air.
B. Incorrect, the applicant may believe there is guidance in AOP-37 for manually operating the AFW control valves without instrument air.
C. Incorrect, the applicant may believe there is guidance in SOP-12 for manually operating the AFW control valves without instrument air.
D. Incorrect, the applicant may believe there is guidance in EOP supplement 31 for manually operating the AFW control valves, however, this procedure is used to establish a water source if T-2 is not available.

Technical Reference(s): EOP-7.0, EOP Supplement 19
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective:	____________________________ (As available)		
Question Source:	Bank #	___X___	
	Modified Bank #	_______	(Note changes or attach parent)
	New	_______	

Question History: Last NRC Exam Palisades 2010

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	_____
	55.43	__5__

Comments:

This question meets the criteria for an SRO-only question because the applicant must assess the facility conditions given in the stem and use those conditions to select the appropriate procedure to mitigate the consequences a loss of all feed water using AFW flow control valves in manual.

TITLE: LOSS OF ALL FEEDWATER RECOVERY

INSTRUCTIONS	CONTINGENCY ACTIONS
NOTE: Steps 9, 11 through 17 may be performed concurrently.	
9. IF Auxiliary Feedwater from T-2 is desired, THEN **RESTORE** Auxiliary Feedwater flow to at least one S/G by performing ANY of the following:	
a. **VERIFY** power is available to P-8A and P-8C.	a.1 **INITIATE** actions to restore power to P-8A and P-8C. Refer to AOP-8, "Loss of Bus 1C" and AOP-9, Loss of Bus 1D," respectively.
b. **INITIATE** actions to restore Auxiliary Feedwater locally. Refer to EOP Supplement 19.	
c. **ISOLATE** T-2 from the Hotwell. Refer to EOP Supplement 28, Step 2.0.	
10. IF ALL of the following conditions are met: • Main Feedwater from the Condenser will be used • T-2 is NOT available as a makeup source to the Hotwell THEN **ISOLATE** T-2 from the Hotwell. Refer to EOP Supplement 28, Step 2.0	

5.0 CONTROL OF AFW FLOW TO S/Gs USING CONTROL VALVES

TABLE 1

Pump	To S/G	Flow Control Valve	Control Room FIC	Panel C-150 HIC	Panel C-33 HIC	FI	AC Power to CV
P-8A/ P-8B	'A'	CV-0749	FIC-0749	HIC-0749C	HIC-0749	FI-0749A	Y10 BKR 14/ Y30 BKR 6
	'B'	CV-0727	FIC-0727	HIC-0727C	HIC-0727	FI-0727A	
P-8C	'A'	CV-0737A CV-0737	FIC-0737A	N/A	HIC-0737A	FI-0737	Y20 BKR 14
	'B'	CV-0736A CV-0736	FIC-0736A	N/A	HIC-0736A	FI-0736	

1. **CONTROL** flow to one or both S/Gs by throttling the associated AFW flow control valve identified in Table 1 from any of the following prioritized locations:
   a. Control Room
   b. C-150
   c. Panel C-33
   d. Locally at the selected valve as follows. Refer to Table 2. (Page 16)
      1) **MANUALLY OPERATE** the selected control valve handwheel to the FULL CLOSED position.
      2) **ISOLATE** the air supply to the selected control valve.
      3) **BLEED OFF** air pressure through the PCV/filter drain.
      4) **THROTTLE OPEN** the selected control valve with the handwheel to achieve the desired S/G flow rate.

NOTE: FI-0727F and FI-0749C are located in 590' CCW Room.

e. IF S/G level or flow indication for P-8A/B AFW Pumps is unavailable from Control Room, C-150, and C-33, THEN **PERFORM** the following:

1) **ALIGN** local gauge FI-0727F (AFW Flow to S/G E-50B) as follows:

a) **ENSURE CLOSED**:

- FI-0727F Lo Cal/Drain (I3)
- FI-0727F Hi Cal/Drain (I4)

b) **ENSURE OPEN** at least one (1) full turn FI-0727F Bypass (IB).

c) **SLOWLY OPEN** FI-0727F Lo Side Isol (I1).

d) **SLOWLY OPEN** FI-0727F Hi Side Isol (I2).

e) **SLOWLY CLOSE** FI-0727F Bypass (IB).

NOTE: Figure 4 shows the total AFW flow required to remove decay heat.

f) **REFER TO** Figure 4 (page 22) to determine required flow rate to maintain S/G level/remove decay heat.

2) **ALIGN** local gauge FI-0749C (AFW Flow to S/G E-50A) as follows:

a) **ENSURE CLOSED**:

- FI-0749C Lo Cal/Drain (I3)
- FI-0749C Hi Cal/Drain (I4)

b) **ENSURE OPEN** at least one (1) full turn FI-0749C Bypass (IB).

c) **SLOWLY OPEN** FI-0749C Lo Side Isolation (I1).

d) **SLOWLY OPEN** FI-0749C Hi Side Isolation (I2).

e) **SLOWLY CLOSE** FI-0749C Bypass (IB).

NOTE: Figure 4 shows the total AFW flow required to remove decay heat.

f) **REFER TO** Figure 4 (page 22) to determine required flow rate to maintain S/G level/remove decay heat.

NOTE: FI-0736A and FI-0737A are located in West Safeguards on the west wall.

f. IF S/G level or flow indication for P-8C AFW Pump is unavailable from Control Room, C-150, and C-33,
THEN **PERFORM** the following:

1) **ALIGN** local gauge FI-0736A (AFW Flow to S/G E-50B) as follows:

a) **ENSURE CLOSED**:

- FI-0736A Lo Cal/Drain (I3)
- FI-0736A Hi Cal/Drain (I4)

b) **ENSURE OPEN** at least one (1) full turn FI-0736A Bypass (IB).

c) **SLOWLY OPEN** FI-0736A Lo Side Isol (I1).

d) **SLOWLY OPEN** FI-0736A Hi Side Isol (I2).

e) **SLOWLY CLOSE** FI-0736A Bypass (IB).

NOTE: Figure 4 shows the total AFW flow required to remove decay heat.

f) **REFER TO** Figure 4 (Page 22) to determine required flow rate to maintain S/G level/remove decay heat.

2) **ALIGN** local gauge FI-0737A (AFW Flow to S/G E-50A) as follows:

a) **ENSURE CLOSED**:

- FI-0737A Lo Cal/Drain (I3)
- FI-0737A Hi Cal/Drain (I4)

b) **ENSURE OPEN** at least one (1) full turn FI-0737A Bypass (IB).

c) **SLOWLY OPEN** FI-0737A Lo Side Isol (I1).

d) **SLOWLY OPEN** FI-0737A Hi Side Isol (I2).

e) **SLOWLY CLOSE** FI-0737A Bypass (IB).

NOTE: Figure 4 shows the total AFW flow required to remove decay heat.

f) **REFER TO** Figure 4 (Page 22) to determine required flow rate to maintain S/G level/remove decay heat.

Alternate Auxiliary Feedwater Methods

TABLE 2

Control Valve	Air Isolation Valve	PCV/Filter	Location
CV-0727	MV-CA382	PCV-0727	Near S End of Hx - CCW Rm
CV-0749	MV-CA385	PCV-0749	Near Ctmt Wall SE Corner CCW Hx Rm
CV-0736A	MV-CA386	PCV-0736A	Near NW Corner - West Safeguards
CV-0737A	MV-CA387	PCV-0737A	Near South and West Wall - West Safeguards

Examination Outline Cross-Reference:

	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	1
	K/A #	055.EA2.02	
	Importance Rating	_____	4.6

K/A Statement: Ability to determine or interpret the following as they apply to a Station Blackout: RCS core cooling through natural circulation cooling to S/G cooling

Proposed Question:

At time 1330, a Station Blackout occurred.

The crew entered EOP-3.0, “Station Blackout Recovery,” with the following conditions:

- PCS pressure is 1905 psig and slowly lowering.
- P-8B AFW Pump tripped on overspeed and cannot be reset.
- Both S/G Levels are lowering.
- Loop T_{cold} temperatures are 495°F and lowering rapidly.
- Loop T_{hot} temperatures are 535°F and rising slowly.
- Qualified CET indicate 542°F and rising slowly.

Complete the following statements:

To ensure that the PCS is cooled by Natural Circulation, safety related 2400 VAC power must be restored from an Emergency Diesel Generator by time __(1)__ in accordance with the FSAR, Station Blackout Analysis.

To maintain Natural Circulation conditions for the temperatures given above, the CRS should direct the NCO to throttle the ADVs __(2)__.

A. (1) 1430
(2) OPEN

B. (1) 1430
(2) CLOSED

C. (1) 1730
(2) OPEN

D. (1) 1730
(2) CLOSED

Proposed Answer: **D**

Explanation (Optional):

15 minutes following the SBO, Natural Circulation is developing. T_{hot} and T_{cold} separate, but the delta T between should be no more than 50°F per Natural Circulation criteria. For the given conditions, the delta T is at 40°F with T_{hot} slowly rising. Since T_{cold} is lowering at an accelerated rate, the ADV's need to be throttled closed to reduce the cooldown rate in order to maintain loop delta T < 50°F.

A. Incorrect, see explanation and selection 'B'
B. Incorrect, Palisades is a DC coping unit and the FSAR SBO analysis assumes a 4 hour duration until AC power has to be restored. The applicant could chose the 2 hour time requirement as the station batteries are rated at 2 hours without performing the load shed, however, this is not in accordance with EOP-3.0. By performing the DC load shed, within the 30 minute time requirement per the FSAR, the 4160VAC safety bus power restoration requirement is expanded to 4 hours.
C. Incorrect, see explanation.
D. Correct, see explanation. FSAR SBO safety analysis takes credit for the operator action of establishing 4160VAC safety bus power within 4 hours.

Technical Reference(s): EOP-TCA 20, EOP-3.0____________________________
(Attach if not previously provided, ____
including version/revision number) ____

Proposed references to be provided to applicants during examination: _____None________

Learning Objective: _________________________ (As available)

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

Question History: Last NRC Exam ____________
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 _____
55.43 __5__

Comments:
This meets SRO criteria 10CFR55.43 (b)5, for the assessment and procedural selection to comply with the SBO safety analysis in the FSAR which has time requirements as part of the facility license to ensure that PCS Core Cooling is established and maintained through Natural Circulation.

TCA 20- Failure to close DG supply breakers

Description

This operator action is associated with recovery from failure of breakers or associated breaker control circuits to realign the power supply to the safety related buses 1C and 1D from the normal (off-site) power supply to the diesel generators following a LOOP. The power supply breakers to buses 1C and 1D from the safeguards bus are 152-105 and 152-203, and from the startup transformer are 152-106 and 152-202. These breakers must open (if they are closed) to permit the DG supply breakers 152-107 and 152-213 to close in order to supply buses 1C and 1D from their associated diesel generator. The failures considered for this recovery action are failure of breakers 152-105, 152-106, 152-202 or 152-203 to mechanically open, failure of breakers 152-107 or 152-213 to mechanically close or failures associated with any of the breaker control circuits. The operators must identify the cause of the failure, open or close (as necessary) the failed breaker or electrically disable the permissive logic.

Scenario:

1. A Loss Of Offsite Power event occurs.
2. Failure of both DGs to load on to their associated buses (SBO).
3. Turbine-driven auxiliary feedwater (AFW) pump receives starts.
4. Station batteries depleted at 4 hours.
5. Instrumentation and control are lost following battery depletion - turbine-driven pump fails.
6. Failure to recover an AC power source (off-site power or DG) leads to prevent core damage.

Latest Time: 4 hours

TIMING

Start Time

The time of the abnormal event. Usually t = 0 sec

Compelling Signal Time

This is the time the compelling signal is received in the control room. This is the time after occurrence of the abnormal event.

Compelling Signal Time: 0 minutes
Reference/Basis: Station Blackout event indicators as described in the compelling signals. This occurs immediately after the LOOP event.

Median Cognitive Response Time

This is the time operators respond to the compelling signal. It could be as early as the time of the compelling signal. The median response time is typically obtained from actual data (if available), plant simulator exercises, operator interviews, or from industry data (simulator exercises).

Median Cognitive Response Time: 15 minutes
Basis: Time to implement EOP-1.0 Actions and EOP-3.0 steps prior to the associated step.

Action Time

This is the time to perform actions after correct diagnosis of the compelling signal. This includes preparation time (e.g.,. donning protective clothing), travel time, manipulation time, etc.

Action Time: 95 minutes
Basis: See reference EA for a detailed listing of the actions included.

System Time

This is the latest time available to correctly perform the action and still avoid an undesirable state. The undesirable state is core damage (defined as peak clad temperature exceeding 2200°F) or containment failure. It could also be the latest time to perform an action that is dictated by the latest start time of subsequent actions. As an example, the system time for action #1 is the start time of action #2. The start time of action #2 is the latest time to begin the action to avoid an undesirable plant state.

System Time: 240 minutes
Basis: Time to station battery depletion

Compelling Signal

Station blackout conditions
Diesel generator trouble alarms
Indication of transmission line and startup or safeguards transformer voltage
Associated EOPs

The PRA estimates that there is approximately 85 minutes from the time a flow control valve fails in the full open position (after battery depletion) that the steam generator will overfill and water will enter the steam lines.

PRA reference - EA-PSA-DG-REC-03-14, Revision 0, "Calculation of Four Human Error Recovery Events to be Used in Loss of Offsite Power Scenarios"

ACTIONS\EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
22. IF Instrument Air Compressors are not available, THEN **VERIFY** the following:	
a. EK-1348, "N2 HEADER LO PRESS, is clear.	a.1 **REFER TO** ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (EC-13)."
b. EK-3531, "AUX FW CVs BACKUP N_2 LOW PRESSURE," is clear.	b.1 **REFER TO** ARP-24, "Cooling Towers Scheme EK-35 (EC-126)." • N2 Station 1 AND N2 Station 2 local pressure greater than 500 psig LOCATION: Station 1 - 590' Component Cooling Room Station 2 - 590' Turbine Building North of E-4A
23. **PERFORM** EOP Supplement 36.	
© 24. **VERIFY** natural circulation flow in at least one PCS loop by all of the following: • Core ΔT less than 50°F (Average of Qualified CETs minus T_C) • Loop T_Hs and Loop T_Cs constant or lowering • Average of Qualified CETs at least 25°F subcooled • Difference between Loop T_H and Average of Qualified CETs is less than or equal to 15°F	24.1 **ENSURE** proper control of S/G feeding and steaming rates.

© = Continuously applicable step

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	1
	K/A #	000056.G2.2.22	
	Importance Rating	_____	4.7

K/A Statement: Knowledge of limiting conditions for operations and safety limits.

Proposed Question:

Given the following:

- The Plant is in MODE 6 with core offload in progress.
- All 2400VAC busses are being supplied by Safeguards Transformer 1-1.
- Switchyard Rear "R" bus has been de-energized for maintenance.
- A Loss of Offsite Power occurs due to a Safeguards Transformer 1-1 fault and both Diesel Generators (DG) energize their respective 2400VAC busses.

For these conditions, which one of the following describes the Technical Specification 3.8.2 “AC Sources – Shutdown” implications for this event and why?

A. LCO 3.8.2 is met because Station Power Transformer 1-2 is a qualified offsite source in Mode 6.

B. LCO 3.8.2 is met because both DGs are operable and supplying the 2400 VAC safety-related busses.

C. LCO 3.8.2 must be entered because no required offsite source is available to supply the 2400 VAC safety-related busses.

D. LCO 3.8.2 must be entered because only one of two qualified offsite sources required to be operable in MODE 6 is available.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, although Station Power Transformer 1-2 is considered a qualified offsite source in Mode 5-6 or defueled, it is not considered available unless backfeed is actually aligned through the Main Transformer No 1 to be able to power 2400 VAC busses. One operable offsite circuit ensures that all required loads may be powered from offsite power. Any of the three offsite supplies, Safeguards Transformer 1-1, Station Power Transformer 1-2, or Startup Transformer 1-2 is acceptable as a qualified circuit.
B. Incorrect, LCO 3.8.2 requires at least one qualified offsite source and at least one DG operable.
C. Correct, there must be a minimum of one of three qualified offsite sources operable to supply the 2400 VAC busses.

D. Incorrect, the applicant believes that two offsite power sources are required to be operable in Mode 6, which is true for Modes 1-4.

Technical Reference(s): LCO 3.8.1, LCO 3.8.2 & Bases

(Attach if not previously provided, ______

including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # X (Note changes or attach parent)

New ______

Question History: Last NRC Exam Palisades 2008

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 2

Comments:

This question meets the criteria for an SRO-only question as the applicant must understand and apply Tech Spec conditions and actions in accordance with rules of application requirements and understand the bases for the spec.

Question modified from Palisades 2008 NRC Exam. Modified stem to change scenario to a Loss of Offsite Power with information that both D/Gs started to supply their respective busses. Additionally, information that Station Power Transformer 1-2 was capable of being used via Main Transformer backfeed was removed.

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown"; and

b. One Diesel Generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The required offsite circuit inoperable.	----------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. ---	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	A.2.3	Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND		
	A.2.4	Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. The required DG inoperable.	B.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	B.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	B.3	Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND		
	B.4	Initiate action to restore required DG to OPERABLE status.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The safety analyses do not explicitly address electrical power. They do, however, assume that various electrically powered and controlled equipment is available. Electrical power is necessary to terminate and mitigate the effects of many postulated events which could occur in MODES 5 and 6.

Analyzed events which might occur during MODES 5 and 6 are Loss of PCS inventory or Loss of PCS Flow, (which in MODES 5 and 6 would be grouped as a Loss of Shutdown Cooling event), and radioactive releases (Fuel Handling Accident, Cask Drop, Radioactive Gas Release, Etc.).

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed above MODE 5 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the primary coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced, and in minimal consequences.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO requires one offsite circuit to be OPERABLE. One OPERABLE offsite circuit ensures that all required loads may be powered from offsite power. Since only one offsite AC source is required, independence is not a criterion. Any of the three offsite supplies, Safeguards Transformer 1-1, Station Power Transformer 1-2, or Startup Transformer 1-2 is acceptable as a qualified circuit.

BASES

LCO (continued)

An OPERABLE DG, associated with a distribution subsystem required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit.

Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and loss of shutdown cooling).

The DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective 2400 V bus on detection of bus undervoltage, and accepting required loads. Proper "Normal Shutdown" loading sequence, and tripping of nonessential loads, is a required function for DG OPERABILITY. A Service Water Pump must be started soon after the DG to assure continued DG operability. The DBA loading sequence is not required to be OPERABLE since the Safety Injection Signal is disabled during MODES 5 and 6.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that equipment and instrumentation is available to:

a. Provide coolant inventory makeup,

b. Mitigate a fuel handling accident,

c. Mitigate shutdown events that can lead to core damage, and

d. Monitor and maintain the plant in a cold shutdown condition or refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODES 5 and 6. This LCO provides the necessary ACTIONS if the AC electrical power sources required by this LCO become unavailable during movement of irradiated fuel assemblies.

The AC source requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.1, "AC Sources - Operating."

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	1
	K/A #	000058.G2.2.37	
	Importance Rating	_____	4.6

K/A Statement: Ability to determine operability and/or availability of safety related equipment.

Proposed Question:

Electrical Maintenance has just completed the Monthly Battery Check surveillances ME-12A and ME-12B on both Station Batteries ED-01 and ED-02.

Given the following battery parameters:

	Station Battery ED-01	Station Battery ED-02
Electrolyte Level	1/2 inch below max	1/2 inch below max
Electrolyte Temp.	68°F	76°F
Float Voltage	2.11V	2.12V
Specific Gravity	1.205	1.192

Note – Battery cell parameters provided are representative of both the pilot cell and each connected cell.

Based on the given conditions, which of the following is the maximum time allowed before the Plant must be in Mode 3, based on the applicable Technical Specifications?

A. 7 hours

B. 8 hours

C. 30 hours

D. 31 hours

Proposed Answer: **A**

Explanation (Optional):
Station Battery ED-01 has low electrolyte temperature, which provides direct entry into LCO 3.8.6 Condition C, requiring ED-01 to be declared inoperable. With the ED-01 inoperable, LCO 3.8.4 Condition B must be applied, requiring restoration of the battery within 24 hours.

Station Battery ED-02 has low specific gravity, lower than the Category C limits of LCO 3.8.6, requiring entry into LCO 3.8.6 Condition C. Since the battery cell parameters are given (assumed that the pilot cell and each connected cell values are the same), Condition C must be entered since it is known that Category C limits for specific gravity are not met for ED-02. One is not allowed the 24 hours to verify this in Condition A as it is already known and given in the

question stem. This (LCO 3.8.6 Condition C) requires the battery to be immediately declared inoperable. With ED-02 inoperable, LCO 3.8.4 Condition B must be applied, however, now both batteries are inoperable and LCO 3.0.3 must be applied, requiring the plant to be in Mode 3 within 7 hours; the 1 hour of shutdown preparation time does not extent the maximum allowable time to reach Mode 3.

A. Correct, see explanation.
B. Incorrect, the applicant is directly entering LCO 3.0.3, but not appropriately applying the 1 hour shutdown preparation time allowance. The applicant could also be incorrectly using the LCO 3.8.6 Condition A 1 hour to verify battery ED-02 is not within Category C limits and then incorrectly applying LCO 3.8.4 Condition C, which would not be applied as both batteries are inoperable.
C. Incorrect, the applicant is incorrectly applying LCO 3.8.6 Condition A requirements to verify battery cell parameters within Category C limits within 24 hours. The battery cell parameters (in this case, a low specific gravity not within Category C limits) is known. However, in this case, the applicant also does not apply LCO 3.0.3 for 2 inoperable batteries and incorrectly enters LCO 3.8.4 Condition C, to be in Mode 3 within 6 hours (24 hours + 6 hours).
D. Incorrect, the applicant is incorrectly applying LCO 3.8.6 Condition A requirements to verify battery cell parameters within Category C limits within 24 hours. The battery cell parameters (in this case, a low specific gravity not within Category C limits) is known. The applicant is correctly applying LCO 3.0.3 time requirements (7 hours to be in Mode 3), but incorrectly applying LCO 3.8.6 requirements.

Technical Reference(s): LCO 3.8.6, LCO 3.8.4
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: TS 3.8.4, TS 3.8.6

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____
55.43 __5__

Comments:

This question meets the criteria for an SRO-only question as the applicant must apply TS Required Actions in accordance with rules of application requirement, of which are not specifically immediate or within one hour actions requirements.

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Left Train and Right Train batteries shall be within limits.

APPLICABILITY: When associated DC electrical power source(s) are required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Category A or B limits.	A.1 Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits. AND	1 hour
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits. AND	24 hours AND Once per 7 days thereafter
	A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 70°F. OR One or more batteries with one or more battery cell parameters not within Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	31 days
SR 3.8.6.2	Verify average electrolyte temperature of representative cells is ≥ 70°F.	31 days
SR 3.8.6.3	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days

Table 3.8.6-1 (page 1 of 1)
Battery Surveillance Requirements

PARAMETER	CATEGORY A: NORMAL LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: NORMAL LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[b][c]	≥ 1.205	≥ 1.200 <u>AND</u> Average of connected cells ≥ 1.205	Not more than 0.020 below average connected cells <u>AND</u> Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits.

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Left Train and Right Train DC electrical power sources shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required DC electrical power source battery charger inoperable.	A.1 Verify functional cross-connected battery charger is connected supplying power to the affected DC train. <u>AND</u> A.2 Restore required DC electrical power source battery charger to OPERABLE status.	2 hours 7 days
B. One required DC electrical power source battery inoperable.	B.1 Verify OPERABLE directly connected and functional cross-connected battery chargers are connected supplying power to the affected DC train. <u>AND</u> B.2 Restore required DC electrical power source battery to OPERABLE status.	2 hours 24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 125 V on float charge.	7 days
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify battery connection resistance is ≤ 50 μohm for inter-cell connections, ≤ 360 μohm for inter-rack connections, and ≤ 360 μohm for inter-tier connections.	92 days
SR 3.8.4.3	Inspect battery cells, cell plates, and racks for visual indication of physical damage or abnormal deterioration that could degrade battery performance.	12 months

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the plant shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the plant, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 31 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	_____	1
Group #	_____	2
K/A #	000033.AA2.10	
Importance Rating	_____	3.8

K/A Statement: Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Tech-Spec limits if both intermediate range channels have failed.

Proposed Question:

Given the following conditions:

- A Reactor start-up is in progress in Mode 3
- All shutdown rods and part-length rods are withdrawn
- Group 1 regulating rods remain fully inserted
- Source Range NI-1 indicates 500 cps
- Source Range NI-2 indicates 500 cps
- Wide Range NI-3 indicates $3x10^{-7}$%
- Wide Range NI-4 indicates $2x10^{-7}$%

What actions are required per Technical Specifications?

A. Enter Tech Spec 3.0.3 due to BOTH Wide Range NI channels inoperable.

B. Immediately stop all positive reactivity additions and restore BOTH Wide Range NI channels to OPERABLE status prior to MODE 2.

C. Immediately stop all positive reactivity additions and restore ONE Wide Range NI channel to OPERABLE status prior to MODE 2.

D. Remain in Mode 3 with power less than 1E-4% and High Startup Rate Trips bypassed.

Proposed Answer: **B**

Explanation (Optional):

The neutron flux monitoring channels consist of two combined source range/wide range channels, designated NI-1/3 and NI-2/4. The wide range portions, (NI-3 and NI-4) provide neutron flux power indication from < 1E-7% RTP to > 100% RTP. The source range portions, designated NI-1 and NI-2, provide source range indication over the range of 0.1 to 1E+5 cps.

A. Incorrect, in accordance with LCO 3.3.9, two neutron flux channels (NI-1/3 and NI-2/4) are required to be operable. LCO 3.3.9 has required action A.1 if no neutron flux monitoring channel is operable (the action is the same as one channel inoperable). In this case, NI-3 and NI-4 are not operable as they are indicating significantly lowered than expected for an approach to critical. Therefore, LCO 3.0.3 is not applicable due to the

NI-3 and NI-4 failures.

B. Correct, both Wide Range NIs are required in Mode 3.

C. Incorrect, both Wide Range NIs are required in Mode 3. The applicant could believe only one Wide Range NI is required to be operable in Mode 3.

D. Incorrect, the High Startup Rate Trip uses the Wide Range NIs and is automatically bypassed when power is <$1x10^{-4}$ % and is not applicable in this Mode. The applicant could believe that this trip is required in Mode 3.

Technical Reference(s): Tech Spec 3.3.1 and Bases, Tech Spec 3.3.9 and Bases, GOP-3

(Attach if not previously provided, including version/revision number) ____________________

Proposed references to be provided to applicants during examination: ____None_______

Learning Objective: ________________________ (As available)

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

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 _____
55.43 __5__

Comments:

This question meets the criteria for an SRO-only question as the applicant must apply TS Required Actions in accordance with rules of application requirement, which are not specifically immediate or within one hour actions requirements.

1. Excore – Source and Wide Range Nuclear Instruments

a. Source and Wide Ranges cover approximately 12 decades.

1) SR covers 6 decades from 0.1 to 10^5 cps (approximately 10^{-10} to 10^{-4} % power).

2) WR covers 10 1/3 decades from 10^{-8} to 200 % power.

3) When SR is ~ 3 cps, WR should read ~ $1x10^{-7}$ %.

GOP-3 Section 5.2

5.2 **NEUTRON FLUX MONITORS**

5.2.1 When Source Range Neutron Flux Monitors, NI-1 and NI-2, indicate 3.0 CPS, the Wide Range Neutron Flux Monitors, NI-3 and NI-4, should be responsive and indicate approximately $1x10^{-7}$% power.

5.2.2 When the Wide Range Neutron Flux Monitors, NI-3 and NI-4, begin to indicate, for each decade of Source Range change, the Wide Range Neutron Flux Monitors should change by approximately one decade.

GCL-3

MODE 3 ≥ 525°F TO MODE 2

		TIME	DATE	INITIAL
1.24	Precautions And Limitations of Entergy Procedure EN-RE-327, "PWR Startup Critical Predictions and Evaluation Process," and Attachment 5, "Critical Approach," of this procedure. **REVIEWED** by Licensed Operators who will perform critical approach.	______	______	______

NOTE: All four source/wide range neutron flux monitors shall be operable prior to performing critical approach to meet requirements of Technical Specifications LCO 3.3.1, LCO 3.3.7, LCO 3.3.8, and LCO 3.3.9.

		TIME	DATE	INITIAL
1.25	**RECORD** the following All-Rods-In (ARI) count rates and power levels:	______	______	______

a. Source Range NI-1 ______ CPS

b. Source Range NI-2 ______ CPS

c. Wide Range NI-3 ______ % Power

d. Wide Range NI-4 ______ % Power

GCL-3

MODE 3 ≥ 525°F TO MODE 2

			TIME	DATE	INITIAL
2.0	**WITHDRAWAL OF SHUTDOWN AND PART-LENGTH RODS**				
2.1	**ENSURE** the following:				
	a.	Shutdown Rod Group 'A' withdrawn. Refer to SOP-6, "Reactor Control System."	______	______	______
	b.	Shutdown Rod Group 'B' withdrawn. Refer to SOP-6, "Reactor Control System."	______	______	______
	c.	Part Length Rods withdrawn. Refer to SOP-6, "Reactor Control System."	______	______	______
2.2	**EVALUATE** Neutron Flux Monitors Operability:				
	a.	**ENSURE** Neutron Flux Monitors responding as expected per instrument indication/PPC data trends. Refer to procedure Steps 5.2.1 and 5.2.2.	______	______	______
	b.	IF requirements of GCL-3, Step 2.2a are NOT met or further operability confirmation is needed, THEN **CONTACT** Reactor Engineering.	______	______	______
	c.	**VERIFY** Wide Range NI-3 and NI-4 indicating at approximately $1x10^{-6}$% to $5x10^{-6}$%.	______	______	______

GCL-3

MODE 3 ≥ 525°F TO MODE 2

			TIME	DATE	INITIAL
2.3	IF GCL-3, Step 2.2c is NOT met, THEN:				
	a.	**REMOVE** affected Wide Range Neutron Flux Monitor from operation. Refer to SOP-35, "Neutron Monitoring System."	______	______	______
	b.	**NOTIFY** I&C to adjust the monitor to read between 1×10^{-6}% to 5×10^{-6}%.	______	______	______
	c.	**RETURN** affected Wide Range Neutron Flux Monitor to operation. Refer to SOP-35, "Neutron Monitoring System."	______	______	______

3.0 ESTABLISHING PRE-CRITICAL BORON CONCENTRATION

NOTE: Dilution to critical will typically be used for the initial startup for a new cycle and transient xenon startups.

			TIME	DATE	INITIAL
3.1	**CONTACT** Reactor Engineering for determination of critical approach method.				
	•	Dilution Rod Withdrawal	______	______	______
3.2	IF performing critical approach by dilution, THEN:				
	a.	**RECORD** Shutdown boron concentration plus 100 ppm. Refer to Technical Data Book, Figure 1.2, >= 525, 2%Δp curve, plus 100 ppm. = ________ ppm	______	______	______
	b.	**RECORD** All Rods Out (ARO) critical boron concentration plus 100 ppm. (Value will be provided by Reactor Engineering in a formal communication.) = ________ ppm	______	______	______

Table 3.3.1-1 (page 1 of 2)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable High Power Trip	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.8	≤ 15% RTP above current THERMAL POWER with a minimum of ≤ 30% RTP and a maximum of ≤ 109.4% RTP
2. High Startup Rate Trip[b]	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.8	NA
3. Low Primary Coolant System Flow Trip[c]	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 95%
4. Low Steam Generator A Level Trip	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 25.9% narrow range
5. Low Steam Generator B Level Trip	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 25.9% narrow range
6. Low Steam Generator A Pressure Trip[c]	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 500 psia
7. Low Steam Generator B Pressure Trip[c]	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 500 psia
8. High Pressurizer Pressure Trip	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≤ 2255 psia

(a) With more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

(b) Trip may be bypassed when Wide Range Power is < 1E-4% RTP or when THERMAL POWER is > 13% RTP.

(c) Trips may be bypassed when Wide Range Power is < 1E-4% RTP. Bypass shall be automatically removed when Wide Range Power is ≥ 1E-4% RTP.

B 3.3 INSTRUMENTATION

B 3.3.9 Neutron Flux Monitoring Channels

BASES

BACKGROUND	The neutron flux monitoring channels consist of two combined source range/wide range channels, designated NI-1/3 and NI-2/4. The wide range portions, (NI-3 and NI-4) provide neutron flux power indication from < 1E-7% RTP to > 100% RTP. The source range portions, designated NI-1 and NI-2, provide source range indication over the range of 0.1 to 1E+5 cps. This LCO addresses MODES 3, 4, and 5. In MODES 1 and 2, the neutron flux monitoring requirements are addressed by LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation." When the plant is shutdown, both neutron flux monitoring channels must be available to monitor neutron flux. If only one section of a neutron flux monitoring channel (source range or wide range) is functioning, the neutron flux monitoring channel may be considered OPERABLE if it is capable of detecting the existing reactor neutron flux. In this application, the RPS channels need not be OPERABLE since the reactor trip Function is not required. By monitoring neutron flux, loss of SDM caused by boron dilution can be detected as an increase in flux. Two channels must be OPERABLE to provide single failure protection and to facilitate detection of channel failure by providing CHANNEL CHECK capability.
APPLICABLE SAFETY ANALYSES	The neutron flux monitoring channels are necessary to monitor core reactivity changes. They are the primary means for detecting, and triggering operator actions to respond to, reactivity transients initiated from conditions in which the RPS is not required to be OPERABLE. The neutron flux monitoring channel's LCO requirements support compliance with 10 CFR 50, Appendix A, GDC 13 (Ref. 1). The FSAR, Chapters 7 and 14 (Refs. 2 and 3, respectively), describes the specific neutron flux monitoring channel features that are critical to comply with the GDC.

BASES

APPLICABLE SAFETY ANALYSIS (continued)	The OPERABILITY of neutron flux monitoring channels is necessary to meet the assumptions of the safety analyses and provide for the detection of reduced SDM. The neutron flux monitoring channels satisfy Criterion 4 of 10 CFR 50.36(c)(2).
LCO	The LCO on the neutron flux monitoring channels ensures that adequate information is available to verify core reactivity conditions while shut down. The safety function of these instruments is to detect changes in core reactivity such as might occur from an inadvertent boron dilution. Two neutron flux monitoring channels are required to be OPERABLE. If only one section of a neutron flux monitoring channel (source range or wide range) is functioning, the neutron flux monitoring channel may be considered OPERABLE if it is capable of detecting the existing reactor neutron flux. For example, with the source range count rate indicator functioning properly within its range, and in reasonable agreement with the other source range, a neutron flux monitor channel may be considered OPERABLE even though its wide range indicator is not functioning. The source range nuclear instrumentation channels, NI-1 and NI-2, provide neutron flux coverage extending an additional one to two decades below the wide range channels for use during refueling, when neutron flux may be extremely low. This LCO does not require OPERABILITY of the High Startup Rate Trip Function or the Zero Power Mode Bypass Removal Function. Those functions are addressed in LCO 3.3.1, RPS Instrumentation.

Examination Outline Cross-Reference:

	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	2
	K/A #	000061.G2.2.25	
	Importance Rating	_____	4.2

K/A Statement: Knowledge of the bases in Technical Specification limiting conditions for operations and safety limits.

Proposed Question:

Given the following conditions:

- The Plant is in Mode 4 coming out of a refueling outage.
- The Spent Fuel Pool is full of irradiated fuel.
- The Mode Change Checklist is complete with one discrepancy.
    - A channel check on Spent Fuel Pool Area Monitor, RIA-2313, was not performed during SHO-1, “Operator’s Shift Items Modes 1, 2, 3 and 4.”

The Plant Manager wants to continue the Plant heatup and transition into Mode 3. Can the transition to Mode 3 continue; why or why not?

A. Yes, the Plant can transition to Mode 3, provided no fuel is being moved within the SFP.

B. Yes, the Plant can transition to Mode 3, provided a risk assessment is performed.

C. Yes, the Plant can transition to Mode 3, as the monitor is not in the Mode of applicability.

D. No, the Plant must remain in Mode 4 until the monitor is restored.

Proposed Answer: **A**

Explanation (Optional):

A. Correct, ORM 3.7.16 requires both fuel pool area radiation monitors to be operable with fuel in the fuel pool area. Specification 3.0.4c is applicable to this specification, per ORM Table 3.17.6. As such, entry into a different mode with an inoperability provided the associated actions to be entered (restore in 72 hours in this case) do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed.
B. Incorrect, the plant can transition to Mode 3, as discussed in explanation A, however, the allowance to change modes is not due to performing a risk assessment. The applicant misunderstands Specification 3.0.4c in this case.
C. Incorrect, while the plant can transition to Mode 3, ORM 3.17.6 is always applicable with fuel in the fuel pool area. The applicant misunderstands Table 3.17.6.
D. Incorrect, specification 3.0.4c is applicable for the fuel pool area radiation monitors. The applicant misunderstands the application of Specification 3.0.4c.

Technical Reference(s):	ORM 3.17.6, ORM Specification 3.0.4 Bases
(Attach if not previously provided,	____________________________
including version/revision number)	____________________________

Proposed references to be provided to applicants during examination: ORM 3.17.6

Learning Objective: ________________________ (As available)

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

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	_____
	55.43	2

Comments:

This question meets SRO-only criteria as the applicant must understand and apply conditions in the Operational Requirements Manual and must also apply generic LCO requirements.

3.17.6 OTHER INSTRUMENTATION

SPECIFICATION The instrument channels listed in Table 3.17.6 shall be OPERABLE.

APPLICABILITY: As specified in Table 3.17.6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3. One or two SIRWT temperature channels inoperable.	3.a Provide alternate means of temperature monitoring.	7 days
4. One Main Feedwater flow channel inoperable.	4.a Provide alternate means of flow monitoring.	24 hours
5. One Main Feedwater temperature channel inoperable.	5.a Provide alternate means of temperature monitoring.	24 hours
6.1. One AFW Flow indicator for one or more flow paths inoperable.	6.1.a Determine the OPERABILITY of the associated AFW flow control valve; the requirements of Technical Specification LCO 3.7.5 may apply.	2 hours
6.2. Two AFW flow indicators for one flow path inoperable.	6.2.a Declare associated control valve inoperable; the requirements of Technical Specification LCO 3.7.5 apply.	Immediately
8. One Primary Safety Valve position indicator channel inoperable for one or more valves.	8.a Restore channels to OPERABLE status.	Prior to next MODE 3 entry from MODE 4.

3.17.6 OTHER INSTRUMENTATION (CONTINUED)

CONDITION	REQUIRED ACTION	COMPLETION TIME
9. One or two PORV position indicator channels inoperable for one or more valves.	9.a Restore channels to OPERABLE status.	Prior to next MODE 4 entry from MODE 5.
10. One PORV Block Valve position indicator channels inoperable for one or more valves.	10.a Restore channels to OPERABLE status. AND 10.b If the PORV path is required for LTOP or as a PCS vent, and the valve position lights are inoperable, verify PORV block valve is open.	Prior to next MODE 4 entry from MODE 5. Once per 12 hours
12.1. One Flux-ΔT power comparator channel inoperable.	12.1.a Restore channels to OPERABLE status.	Prior to next MODE 1 entry from MODE 2.
12.2. Two Flux-ΔT power comparator channels inoperable.	12.2.a Limit power to ≤ 70% RTP.	2 hours
15. Excore deviation alarm inoperable.	15.a Calculate T_q using the excore or incore readings.	Once per 12 hours.
16. One or two ASI alarm channels inoperable.	16.a Restore channels to OPERABLE status.	Prior to next MODE 4 entry from MODE 5.

3.17.6 OTHER INSTRUMENTATION (CONTINUED)

CONDITION	REQUIRED ACTION	COMPLETION TIME
19. One or two fuel pool area radiation monitors inoperable.	19.a Stop moving fuel within the fuel pool area until monitoring capability is restored.	Immediately
	<u>AND</u>	
	19.b Restore monitor to OPERABLE status or provide equivalent monitoring capability.	72 hours
21. Required Action and associated Completion Time of Conditions 3 through 19 not met. <u>OR</u> Less than two required Flux-ΔT power comparator channels OPERABLE. <u>OR</u> Less than two required ASI Alarm channels OPERABLE. <u>OR</u> Less than one required instrument channel per valve for Instruments 8, 9, or 10.	21.a Operations to determine compensatory measures, as required, and to assign or extend Completion Time.	12 hours
	21.b Convene an OSRC to review compensatory measures or extension of completion time.	10 business days

TESTING REQUIREMENTS: Refer to Table 4.17.6.

Table 3.17.6 (page 1 of 1)
Other Instrumentation

INSTRUMENT	APPLICABLE MODES	REQUIRED CHANNELS	MINIMUM OPERABLE CHANNELS
3. SIRW Tank Temperature	1,2,3	2[(a)]	0
4. Main Feedwater Flow	Above 15% RTP	1 per line	0
5. Main Feedwater Temperature	Above 15% RTP	1 per line	0
6. Auxiliary Feedwater Flow Indication	1,2,3, 4 when steam generator is relied upon for heat removal	2 per line	0
8. Primary Safety Valve Position Indication	1,2,3	2 per valve	1 per valve
9. PORV Position Indication	1,2,3, & 4, when PORV block valve is open or its position indication is inoperable	3 per valve	1 per valve
10. PORV Block Valve Position Indication	At all times, unless the PCS is depressurized and vented	2 per valve	1 per valve
12. Flux-ΔT Power Comparator	1	4	2
15. Excore Detector Deviation Alarm	Above 25% RTP	1	0
16. ASI Alarm	Above 25% RTP	4	2
19. Fuel Pool Area Radiation Monitor	When fuel is in fuel pool area	2[(a)]	0

(a) Specification 3.0.4c is applicable

3.17 OTHER INSTRUMENTATION SYSTEMS BASES (continued)

16. AXIAL SHAPE INDEX Alarm - The ASI Alarm Channel monitors the ASI using the Excore upper and lower detector signals as inputs and provides an alarm when ASI administrative limits are exceeded.

    This alarm is only functional above a nominal 15% indicated power when the High Startup Rate trip is bypassed. It uses the High Startup Rate Pre-Trip Unit to provide the alarm function, and shares the same alarm window. It is not required to be OPERABLE below 25% RTP.

    Action 3.17.6.16 - One Or Two ASI Alarm Channels Inoperable - The ASI alarm is one function of the Thermal Margin Monitor. Four channels are provided, but two are sufficient for ASI monitoring. If one or two channels are inoperable, they must be restored prior to the next MODE 4 entry from MODE 5.

19. Fuel Pool Area Radiation Monitor - The spent fuel pool is provided with two radiation monitors. These instruments provide warning of a release in the case of a fuel handling accident.

    A Note permits the use of the provisions of Specification 3.0.4c. This allowance permits entry into the applicable MODE/Specified Condition while relying on the ACTIONS.

    Action 3.17.6.19 - One Or Two Fuel Pool Area Monitors Inoperable - With one or two Fuel Pool Area Radiation Monitors inoperable, fuel movement in the spent fuel pool area must be stopped. The monitor must be restored to OPERABLE status or equivalent monitoring capability provided within 72 hours. The Fuel Pool is designed to be adequately subcritical even at zero ppm boron concentration. The specified 72 hours is adequate to repair the installed instrumentation or to provide other monitoring equipment without incurring undue risk of a criticality.

Specification 3.0.2

Upon discovery of a failure to meet a Specification, the Required Actions shall be met, except as provided in Specification 3.0.5.

If the Specification is met or is no longer applicable prior to expiration of the specified completion time(s), completion of the Required Action(s) is not required, unless otherwise stated.

Specification 3.0.3

When a Specification is not met and the associated ACTIONS and Completion Times are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, Operations to determine compensatory measures, as required, and to assign or extend Completion Time. Convene an OSRC within 10 business days to review compensatory measures or extension of completion time.

Specification 3.0.4

When a Specification is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time.

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant.

These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the Specification would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

Specification 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The Specification 3.0.4.b risk assessments do not have to be documented.

The ORM allows continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the Specification, the use of the Specification 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above.

Specification 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the Specification not met based on a Note in the Specification which states Specification 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	2
	K/A #	000076.G2.2.38	
	Importance Rating	_____	4.5

K/A Statement: Knowledge of conditions and limitations in the facility license.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power.
- At 0800 on 9/26/16, Surveillance Requirement 3.4.16.2 was complete.
- The initial dose equivalent I-131 sample result was 2 μCi/gm.

Assuming Dose Equivalent I-131 concentration were to continue to rise by 2 μCi/gm upon each subsequent 4 hour sample, and is not restored, what is <u>the latest time</u> the Plant is required to be in Mode 3 with T_{ave} < 500ºF, per LCO 3.4.16?

A. 1400 on 9/28/16

B. 1500 on 9/28/16

C. 1800 on 9/29/16

D. 1900 on 9/29/16

Proposed Answer: A

Explanation (Optional):

A. Correct, using a linear relationship of 2 μCi/gm rise upon each sample (every 4 hours per LCO 3.4.16 Required Action A.1, the Dose Equivalent I-131 (DE I-131) would reach 26 μCi/gm after 48 hours. LCO 3.4.16 Condition B is required to be entered if the Required Action and associated Completion Time of Condition A not met OR DE 1-131 $\geq$ 40 μCi/gm OR Gross specific activity of the primary coolant not within limit. In this case, the required action and associated completion time of Condition A is not met. Adding 6 hours to the Condition B entrance time, the latest time the plant would be required to be in Mode 3 with T_{ave} < 500ºF is 1400 on 9/28/16.

B. Incorrect, the applicant entered LCO 3.0.3 upon reaching the 48 hours of DE 1-131 exceeding 1.0 μCi/gm. LCO 3.0.3 allows an extra hour to prepare for plant shutdown, whereas LCO 3.4.16 Condition B does not.

C. Incorrect, the applicant waited to enter LCO 3.4.16 Condition B until DE 1-131 reached 40 μCi/gm (40 μCi /gm reached at 1200 on 9/29/16), rather than entering Condition B after 48 hours per Required Action A.2.

D. Incorrect, the applicant entered LCO 3.0.3 upon reaching 40 μCi/gm DE 1-131 rather

than entering Condition B.

Technical Reference(s): LCO 3.4.16
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: TS LCO 3.4.16

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ______
55.43 1

Comments:
This question meets the criteria for an SRO-only question as the applicant must apply TS Required Actions in accordance with rules of application requirements as well as the application of generic LCO requirements.

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.16 PCS Specific Activity

LCO 3.4.16 The specific activity of the primary coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with PCS average temperature (T_{ave}) $\geq$ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 µCi/gm.	----------------NOTE------------------ LCO 3.0.4.c is applicable. --	
	A.1 Verify DOSE EQUIVALENT I-131 < 40 µCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 ≥ 40 μCi/gm. <u>OR</u> Gross specific activity of the primary coolant not within limit.	B.1 Be in MODE 3 with T_{ave} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify primary coolant gross specific activity ≤ 100/Ē μCi/gm.	7 days

Need to include the next TS page which shows the TS SR 3.4.16.2 requirements - < or = 1.0 μCi/gm.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	2
	K/A #	CE/A11.AA2.02	
	Importance Rating	_____	3.4

K/A Statement: Ability to determine and interpret the following as they apply to the (RCS Overcooling): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question:

Given the following conditions:

- The Plant is in Mode 3 at Normal Operating Temperature and Pressure.
- A Main Steam Line break occurs on the 'A' Main Steam Line outside Containment, upstream of the MSIV.
- After 15 minutes, 'A' Steam Generator blows dry.
- Average of the Qualified Core Exit Thermocouples (QCET) lowers to 442°F
- T_{cold} lowers to 425°F.

With Primary Coolant Pumps in operation ___(1)___ should be used to monitor PCS cooldown limits. The Primary Coolant System cooldown limits ___(2)___ been exceeded.

(1)	(2)
A. QCET	have
B. QCET	have NOT
C. T_{cold}	have
D. T_{cold}	have NOT

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, part 1 is incorrect. T_{cold} should be used to monitor cooldown and not QCET. QCET is used when determining subcooling for PCP operation. The cooldown calculation is correct.
B. Incorrect, both parts are incorrect. T_{cold} should be used to monitor cooldown with PCPs running. Cooldown rate has been exceeded.
C. Correct, EOP-6 requires the operator to determine the cooldown rate during an ESDE per EOP Supplement 2. EOP Supplement 2 uses T_{cold}, as this would be the temperature

of the PCS water entering the reactor vessel. Per EOP-6.0, the operators are bounded by TS 3.4.3 values of 100°F/hr maximum cooldown. During this event, T_{cold} dropped greater than 100°F over 5 minutes (no-load T_{cold} is 532°F, but could be as high as 540°F due to ADV control) and the cooldown rate limit of 100°F/hr was exceeded.

D. Incorrect, the use of T_{cold} is correct, however, the cooldown has been exceeded. The applicant was using the average QCET value for the cooldown rate.

Technical Reference(s): EOP-6, TS 3.4.3
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:
55.41 ______
55.43 5

Comments:

This meets SRO criteria 10CFR55.43 (b)5 since the safety analysis in the FSAR and TSs have requirements as part of the facility license to ensure that PCS system cooldown rate limitations are maintained.

INSTRUCTIONS

© 23. **MAINTAIN** PCS pressure within the limits of EOP Supplement 1 by performing ANY of the following:

a. **CONTROL** the following:

- PZR heaters
- Main Spray
- Auxiliary Spray (Supplement 37)

b. IF SI Pump throttling criteria are met,
THEN **CONTROL** HPSI, Charging, and Letdown flows.

(continue)

CONTINGENCY ACTIONS

23.1 IF the PCS is oversubcooled
OR PZR pressure is greater than the maximum limits of EOP Supplement 1,
THEN **PERFORM** ANY of the following to restore subcooling or PCS pressure to within the appropriate limit:

a. **OPERATE** available S/G(s) to stop the cooldown
AND **STABILIZE** Qualified CET temperatures and Loop T_Cs.

b. **OPERATE** the following to lower PZR pressure within allowable limits:

- Main Spray
- Auxiliary Spray (Supplement 37)

c. IF SI Pump throttling criteria are met,
THEN **CONTROL** HPSI, Charging, and Letdown flows.

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

23. **(continued)**

(continued)

d. IF ALL of the following conditions are met:

- Above actions to lower PCS pressure are NOT effective
- PORVs are required to open to reduce PCS pressure
- PZR level is less than 85%

THEN **PERFORM** BOTH of the following:

1) **OPEN** PORV Isolation Valves:

MO-1042A
MO-1043A

CAUTION

Rupture of the Quench Tank rupture disk is likely during any sustained opening of PORVs. This would result in rising Containment atmosphere temperature and pressure. Quench Tank temperature and pressure should be monitored during PORV operation.

2) **CYCLE** the PORVs as necessary to maintain BOTH of the following:

(continue)

(continue)

INSTRUCTIONS	CONTINGENCY ACTIONS
23. **(continued)**	**(continued)**
	• PZR corrected level less than 85% (**REFER TO** EOP Supplements 9 and 10) • PZR pressure within the limits of EOP Supplement 1.
	3) IF ALL of the following PORV closing criteria are met: • PZR pressure is less than 2100 psia • PZR pressure is less than the maximum limits of EOP Supplement 1 • PORVs are NOT required open to reduce PZR pressure, THEN CLOSE the PORVs: • PRV-1042B • PRV-1043B
	4) IF the PORV closing criteria are met AND either PORV will NOT close, THEN CLOSE associated PORV Isolation Valve: • MO-1042A • MO-1043A
(continue)	**(continue)**

INSTRUCTIONS

23. **(continued)**

CONTINGENCY ACTIONS

(continued)

5) **ENSURE** started the following containment cooling fans:

   a) ALL available Containment Air Cooler 'A' fans for ALL available Containment Air Coolers.

   b) IF SIAS not present, THEN ALL available Containment Air Cooler 'B' fans for ALL available Containment Air Coolers.

6) IF ANY of the following conditions exist:

   - Containment pressure is greater than or equal to 4.0 psig.
   - Any operable CONTAINMENT Radiation Monitor rises to 1 x 10^1 R/hr,

   THEN **PERFORM** ALL of the following:

(continue)

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

23. (continued) | (continued)

a) **VERIFY** "CIS INITIATED" (EK-1126) is alarmed
OR
MANUALLY INITIATE CIS by pushing left or right HIGH RADIATION INITIATE pushbuttons on EC-13:

- CHRL-CS
- CHRR-CS

b) **VERIFY** Containment Isolation. Refer to EOP Supplement 6.

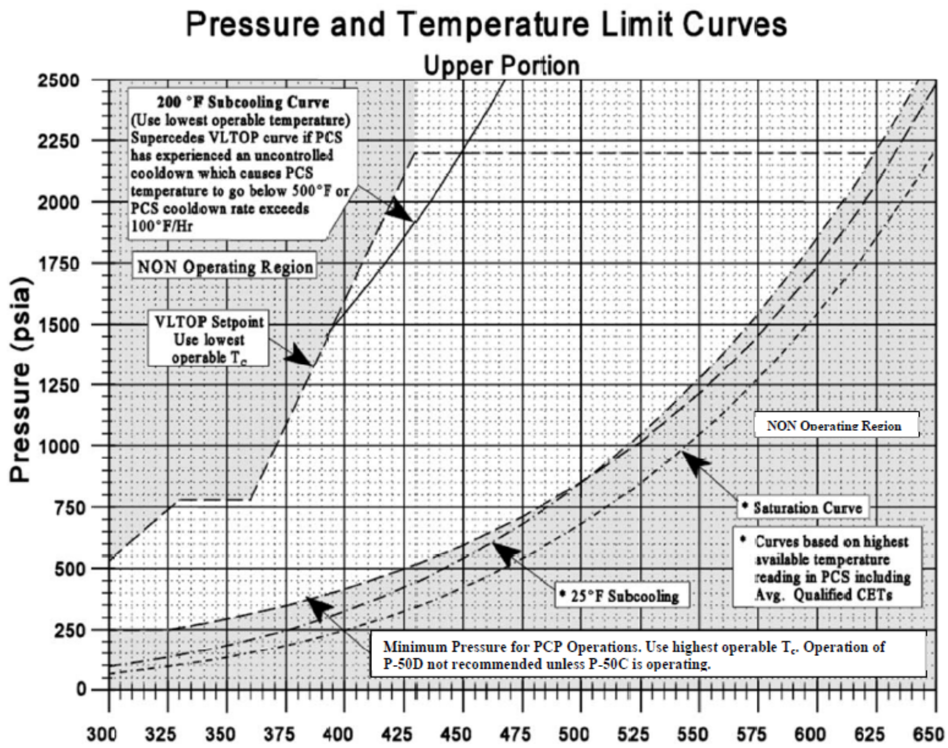
7) IF the Pressure Control safety function is still in jeopardy, THEN GO TO EOP-9.0.

23.2 IF PCS cooldown rate exceeds Technical Specification limits, THEN **PERFORM** ANY of the following to restore the cooldown rate to within Technical Specification limits:

a. **OPERATE** available S/G(s) to stop the cooldown AND **STABILIZE** Qualified CET temperatures and Loop T_Cs.

(continue) | (continue)

PRESSURE AND TEMPERATURE LIMIT CURVES



Note: Use degraded Containment condition curves if Containment condition is or has exceeded 175°F or 3 psig.

3.4 PRIMARY COOLANT SYSTEM (PCS)

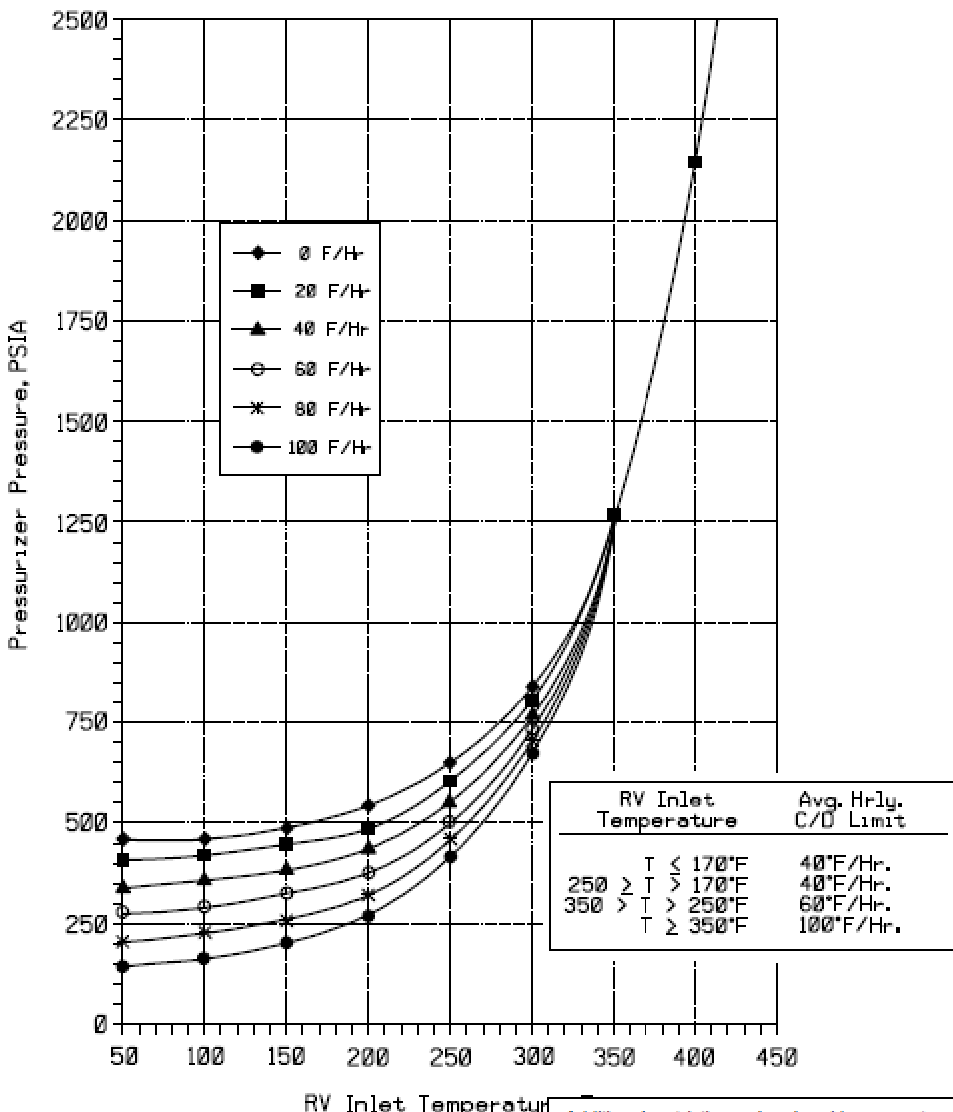
3.4.3 PCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 PCS pressure, PCS temperature, and PCS heatup and cooldown rates shall be maintained within the limits of Figure 3.4.3-1 and Figure 3.4.3-2.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Action A.2 shall be completed whenever this Condition is entered. -------------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine PCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with PCS pressure < 270 psia.	6 hours 36 hours



Additional restrictions when head is on reactor vessel:

1. Maintain average core exit temperature: 135°F ≥ T ≥ 110°F for ≥ 3 hours.
2. Following completion of item 1, maintain average hourly cooldown (C/D) limit of 20°F/hour based on core exit temperature indication.

Figure 3.4.3-2 (Page 1 of 1)
Pressure – Temperature Limits for Cooldown
Applicable up to 42.1 EFPY

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	1
	K/A #	010.G2.2.42	
	Importance Rating	_____	4.6

K/A Statement: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power
- Pressurizer Pressure indication, PIA-0102BLL, indicates 1500 psig
- The following alarms are in:
  - EK-0756, Pressurizer Safety Inj Signal 'B' Lo-Lo Press
  - EK-0601C, TM/LO Pressure Channel Trip
  - EK-0605C, TM/LO Pressure Channel Pre-Trip

Based on the given conditions, the CRS will enter which of the following Tech Specifications?

A. TS 3.3.1, Reactor Protection System (RPS) Instrumentation and TS 3.3.3, Engineered Safety Features (ESF) Instrumentation

B. TS 3.3.1, Reactor Protection System (RPS) Instrumentation and TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System

C. TS 3.3.3, Engineered Safety Features (ESF) Instrumentation and TS 3.3.7, Post Accident Monitoring (PAM) Instrumentation

D. TS 3.3.3, Engineered Safety Features (ESF) Instrumentation and TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System

Proposed Answer: **A**

Explanation (Optional):

A. Correct, the applicant should identify that a Pressurizer Pressure transmitter (PT-0102B) has failed low and provides inputs into RPS and SIS, given the alarms provided. The applicant should understand the TS implication and that TS 3.3.1 (RPS) and TS 3.3.3 (SIS) are applicable.
B. Incorrect, TS 3.3.7 is not applicable to this pressure transmitter. The applicant is confusing this pressure transmitter for one that provides input to WR pressure indication and LTOP (PT-0105A/B).
C. Incorrect, TS 3.3.8 is not applicable to this pressure transmitter. The applicant is confusing this pressure transmitter for one that provides input to the LTOP and SDC

Interlocks (PT-0104A/B).

D. Incorrect,

Technical Reference(s): TS 3.3.1, TS 3.3.3, TS 3.3.1 Bases

(Attach if not previously provided, including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source:

- Bank # ______
- Modified Bank # ______ (Note changes or attach parent)
- New X

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:

- Memory or Fundamental Knowledge ______
- Comprehension or Analysis X

10 CFR Part 55 Content:

- 55.41 ______
- 55.43 3

Comments:

This questions meets SRO-only criteria as the applicant must use Tech Spec Bases knowledge to apply given entry-level conditions to Tech Spec applicability.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

9. Thermal Margin/Low Pressure (TM/LP) Trip

The TM/LP trip is provided to prevent reactor operation when the DNBR is insufficient. The TM/LP trip protects against slow reactivity or temperature increases, and against pressure decreases.

The trip is initiated whenever the PCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher.

The TM/LP trip uses Q Power, ASI, pressurizer pressure, and cold leg temperature (T_c) as inputs.

Q Power is the higher of core THERMAL POWER (ΔT Power) or nuclear power. The ΔT power uses hot leg and cold leg RTDs as inputs. Nuclear power uses the power range excore channels as inputs. Both the ΔT and excore power signals have provisions for calibration by calorimetric calculations.

The ASI is calculated from the upper and lower power range excore detector signals, as explained in Section 1.1, "Definitions." The signal is corrected for the difference between the flux at the core periphery and the flux at the detectors.

The T_c value is the higher of the two cold leg signals.

The Low Pressurizer Pressure trip limit (P_{var})is calculated using the equations given in Table 3.3.1-2.

The calculated limit (P_{var}) is then compared to a fixed Low Pressurizer Pressure trip limit (P_{min}). The auctioneered highest of these signals becomes the trip limit (P_{trip}). P_{trip} is compared to the measured PCS pressure and a trip signal is generated when the measured pressure for that channel is less than or equal to P_{trip}. A pre-trip alarm is also generated when P is less than or equal to the pre-trip setting, P_{trip} + ΔP.

The TM/LP trip setpoint is a complex function of these inputs and represents a minimum acceptable PCS pressure for the existing temperature and power conditions. It is compared to actual PCS pressure in the TM/LP trip unit.

Table B 3.3.1-1 (page 1 of 1)
Instruments Affecting Multiple Specifications

Required Instrument Channels	Affected Specifications
Nuclear Instrumentation	
Source Range NI-1/3, Count Rate Indication @ C-150 Panel	3.3.8 (#1)
Source Range NI-1/3 & 2/4, Count Rate Signal	3.3.9 & 3.9.2
Wide Range NI-1/3 & 2/4, Flux Level 10^{-4} Bypass	3.3.1 (#3, 6, 7, 9, & 12)
Wide Range NI-1/3 & 2/4, Startup Rate	3.3.1 (#2)
Wide Range NI-1/3 & 2/4, Flux Level Indication @EC-06 Panel for 3.3.7	3.3.7 (#3) & 3.3.9
Power Range NI-5, 6, 7, & 8, Tq	3.2.1 & 3.2.3
Power Range NI-5, 6, 7, & 8, Q Power	3.3.1 (#1 & 9)
Power Range NI-5, 6, 7, & 8, ASI	3.3.1 (#9) & 3.2.1 & 3.2.4
Power Range NI-5, 6, 7, & 8, Loss of Load/High Startup Rate Bypass	3.3.1 (#2 & 10)
PCS T-Cold Instruments	
TT-0112CA, Temperature Signal (SPI ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112CA & 0122CA, Temperature Signal (C-150)	3.3.8 (#6 & 7)
TT-0122CB, Temperature Signal (PIP ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112CA & 0122CB, Temperature Signal (LTOP)	3.4.12.b.1
TT-0112CC & 0122CD (PTR-0112 & 0122) Temperature Indication	3.3.7 (#2)
TT-0112 & 0122 CC & CD, Temperature Signal (SMM)	3.3.7 (#5)
TT-0112 & 0122 CA, CB, CC, & CD, Temperature Signal (Q Power & TMM)	3.3.1 (#1 & 9) & 3.4.1.b
PCS T-Hot Instruments	
TT-0112HA, Temperature Signal (SPI ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112HA & 0122HA, Temperature Signal (C-150)	3.3.8 (#4 & 5)
TT-0122HB, Temperature Signal (PIP ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112 & 0122 HC & HD, Temperature Signal (SMM)	3.3.7 (#5)
TT-0112HC & 0122HD (PTR-0112 & 0122) Temperature Indication	3.3.7 (#1)
TT-0112 & 0122 HA, HB, HC, & HD, Temperature Signal (Q Power & TMM)	3.3.1 (#1 & 9)
Thermal Margin Monitors	
PY-0102A, B, C, & D	3.3.1 (#1 & 9)
Pressurizer Pressure Instruments	
PT-0102A, B, C, & D, Pressure Signal (RPS & SIS)	3.3.1 (#8 & 9) & 3.3.3 (#1.a & 7a)
PT-0104A & B, Pressure Signal (LTOP & SDC Interlock)	3.4.12.b.1 & 3.4.14
PT-0105A & B, Pressure Signal (WR Indication & LTOP)	3.3.7 (#5) & 3.4.12.b.1
PI-0110, Pressure Indication @ C-150 Panel	3.3.8 (#2)
SG Level Instruments	
LT-0751 & 0752 A, B, C, & D, Level Signal (RPS & AFAS)	3.3.1 (#4 & 5) & 3.3.3 (#4.a & 4.b)
LI-0757 & 0758 A & B, Wide Range Level Indication	3.3.7 (#11 & 12)
LI-0757C & 0758C, Wide Range Level Indication @ C-150 Panel	3.3.8 (#10 & 11)
SG Pressure Instruments	
PT-0751 & 0752 A, B, C, & D, Pressure Signal (RPS & SG Isolation) PT-0751A and PT-0752A Pressure Signal (C-150/150A)	3.3.1 (#6 & 7) & 3.3.3 (#2a, 2b, 7b, 7c) 3.3.8 (#8 & 9)
PIC-0751 & 0752 C & D, Pressure Indication	3.3.7 (#13 & 14)
PI-0751E & 0752E, Pressure Indication @ C-150 Panel	3.3.8 (#8 & 9)
Containment Pressure Instruments	
PS-1801, 1802, 1803, & 1804, Switch Output (RPS)	3.3.1 (#11)
PS-1801, 1802A, 1803, & 1804A, Switch Output (ESF)	3.3.3 (#5.a)
PS-1801A, 1802, 1803A, & 1804, Switch Output (ESF)	3.3.3 (#5.b)

Note: The information provided in this table is intended for use as an aid to distinguish those instrument channels which provide more than one required function and to describe which specifications they affect. The information in this table should not be taken as inclusive for all instruments nor affected specifications.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

The ESF Actuation Functions are generated by comparing a single measurement to a fixed bistable setpoint. The ESF Actuation Functions utilize the following input instrumentation:

- Safety Injection Signal (SIS)

  The Safety Injection Signal can be generated by any of three inputs: Pressurizer Low Pressure, Containment High Pressure, or Manual Actuation. Manual Actuation is addressed by LCO 3.3.4; Containment High Pressure is discussed below. Four instruments (channels A through D), monitor Pressurizer Pressure to develop the SIS actuation. Each of these instrument channels has two individually adjustable ESF bistable trip devices, one for the bypass removal circuit (discussed below) and one for SIS. Each ESF bistable trip device actuates two auxiliary relays, one for each actuation train. The output contacts from these auxiliary relays form the logic circuits addressed in LCO 3.3.4. The instrument channels associated with each Pressurizer Low Pressure SIS actuation bistable include the pressure measurement loop, the SIS actuation bistable, and the two auxiliary relays associated with that bistable. The bistables associated with automatic removal of the Pressurizer Low Pressure Bypass are discussed under Function 7.a, below.

- Low Steam Generator Pressure Signal (SGLP)

  There are two separate Low Steam Generator Pressure signals, one for each steam generator. For each steam generator, four instruments (channels A through D) monitor pressure to develop the SGLP actuation. Each of these instrument channels has two individually adjustable ESF bistable trip devices, one for the bypass removal circuit (discussed below) and one for SGLP. Each Steam SGLP bistable trip device actuates an auxiliary relay. The output contacts from these auxiliary relays form the SGLP logic circuits addressed in LCO 3.3.4. The instrument channels associated with each Steam Generator Low Pressure Signal bistable include the pressure measurement loop, the SGLP actuation bistable, and the auxiliary relay associated with that bistable. The bistables associated with automatic removal of the SGLP Bypass are discussed under Function 7.a, below.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	1
	K/A #	012.A2.03	
	Importance Rating	_____	3.7

K/A Statement: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Incorrect channel bypassing.

Proposed Question:

Given the following conditions:

- The Plant is at 100% power
- Source/Wide Range NI-2/4 is being removed from operation
- The NCO incorrectly performed the following on the RPS channels associated with NI-2/4:
  - D RPS channel in bypass
  - C RPS channel in trip

What actions should the NCO have taken and what procedure should be used to remove the Source/Wide Range NI from operation?

A. ‘C’ channel should have been bypassed and ‘D’ tripped. SOP-36, Reactor Protective System and Anticipated Transient Without Scram (ATWS) System.

B. ‘B’ channel should have been bypassed and ‘D’ tripped. SOP-36, Reactor Protective System and Anticipated Transient Without Scram (ATWS) System.

C. ‘C’ channel should have been bypassed and ‘D’ tripped. SOP-35, Neutron Monitoring System.

D. ‘B’ channel should have been bypassed and ‘D’ tripped. SOP-35, Neutron Monitoring System.

Proposed Answer: D

Explanation (Optional):

A. Incorrect, ‘C’ channel should only be tripped or bypassed when removing NI-1/3 from service. SOP-36 is not used to remove the NI from operation, but is used to bypass the channel.
B. Incorrect, SOP-36 is not used to remove the NI from operation, but is used to bypass the channel.
C. Incorrect, ‘C’ channel should only be tripped or bypassed when removing NI-1/3 from service. SOP-35 is the correct procedure.

D. Correct, when taking Source/Wide Range NI out of service, the NCO should use SOP-35 Section 7.1.2. This section requires the NCO to place in bypass one affected channel Hi SUR Trip unit AND in trip any other affected channel Hi SUR Trip unit. SOP-36 is used to bypass the channel.

Technical Reference(s): SOP-35 (section 7.1.2), SOP-36 (section 7.4)
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:
This question meets SRO-only criteria as the applicant must have detailed knowledge of a System Operating procedure.

NOTE:	NI-1/3 and NI-2/4 Source/Wide Range monitors will indicate a step change of approximately 0.04% power between 0.05% and 0.10% indicated power range. This is due to switching from counts mode to the campbelling mode of operation.

7.1.2 Removing a Source/Wide Range Nuclear Instrument from Operation

NOTE:	Unless required to be removed from service for maintenance, Source/Wide Range Nuclear Instruments shall be kept in an operable condition during Mode 1.

a. **REVIEW** the following Technical Specifications sections for operability requirements, as applicable:

- LCO 3.3.1
- LCO 3.3.9
- LCO 3.9.2
- LCO 3.3.1 Table 3.3.1-1 Function 2
- LCO 3.3.7 Table 3.3.7-1 Function 3
- LCO 3.3.8 Table 3.3.8-1 Function 1

b. **PLACE** in BYPASS one affected channel Hi SUR Trip unit AND in TRIP any other affected channel Hi SUR Trip unit. Refer to System Operating Procedure SOP-36, "Reactor Protective System and Anticipated Transient Without Scram (ATWS) System," Section 7.4.

WIDE RANGE NI	AFFECTED CHANNELS
NI-3A	"A" and "C"
NI-4A	"B" and "D"

c. WHEN channel has been tested and declared operable, THEN **PERFORM** the following:

1. **PERFORM**, at Shift Manager discretion, Attachment 2, "Checklist CL 35, Neutron Monitors System Checklist," for affected channel.

7.4 RPS TRIP UNITS

7.4.1 General Information

a. RPS Trip Units are bypassed using the bypass keyswitch for the desired RPS Trip Unit. The key is captured in the bypass keyswitch when the key has been turned to the bypass position.

b. Each RPS Trip Unit has a unique lock cylinder; however, the same cylinder for each respective RPS Trip Unit is used on all four RPS channels.

c. IF an RPS Trip Unit is to be bypassed and the associated preferred AC bus is deenergized, THEN the yellow bypass lamp will NOT light when the bypass key is inserted and turned 90° clockwise.

d. S/G Low Level and Low Pressure Trip Units exist for EACH S/G. Close attention is required to ensure the correct RPS Trip Unit is selected.

e. IF an RPS Trip Unit is demonstrating anomalous behavior such that it needs to be bypassed OR if maintenance is to be performed on an RPS Trip Unit, THEN bypass ALL of the RPS Trip Units for the associated RPS channel.

7.4.2 To Bypass an RPS Trip Unit

NOTE:	The procedure is not required to be in hand, nor is placekeeping required to perform this section.

a. **REVIEW** requirements of Technical Specifications LCO 3.3.1.

b. **PERFORM** the following to bypass desired RPS Trip Unit:

   1. **ENSURE** the RPS Bypass Key is the correct key for the RPS Trip Unit to be bypassed.

   2. **INSERT** bypass key above affected RPS Trip Unit.

   3. **TURN** key 90° clockwise.

   4. **VERIFY** the yellow light above the bypass keyswitch is ON.

   5. **RECORD** evolution in the Operations Log unless logged in the applicable procedure.

c. WHEN bypass is no longer required, THEN:

1. **TURN** affected RPS Trip Unit bypass key 90° counterclockwise.

2. **VERIFY** the yellow light above the bypass keyswitch is OFF.

3. **REMOVE** key AND **RETURN** to Shift Manager.

4. **RECORD** evolution in the Operations Log unless logged in the applicable procedure.

7.4.3 To Place an RPS Trip Unit in the Tripped Condition

a. **REVIEW** requirements of Technical Specifications LCO 3.3.1.

b. **PLACE** desired RPS Trip Unit in tripped condition as follows:

1. **LOOSEN** both hold down screws on desired RPS Trip Unit (screws have "capture" feature and will not come all the way out).

2. **PULL** RPS Trip Unit out at least two inches, using the handle on the front of the unit.

3. **ENSURE** disengaged the connections on back of RPS Trip Unit.

4. **RECORD** evolution in the Operations Log unless logged in the applicable procedure.

c. WHEN RPS Trip Unit is no longer required to be tripped, THEN:

1. **INSERT** RPS Trip Unit flush with panel front.

2. **TIGHTEN** hold down screws.

3. **RESET** RPS Trip Unit

4. **VERIFY** RPS trip light is OFF.

5. **RECORD** evolution in the Operations Log unless logged in the applicable procedure.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	1
	K/A #	059.A2.04	
	Importance Rating	_____	3.4

K/A Statement: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feeding a dry S/G

Proposed Question:

Given the following conditions:

- EOP-7.0, "Loss of All Feedwater," is in progress.
- 'A' Steam Generator level: -75%.
- 'B' Steam Generator level: -95%.
- Both 'A' and 'B' S/G pressures: 410 psia.
- The crew is preparing to use the Condensate Pumps for feeding the Steam Generators.
- Feed Regulating Bypass Valves (CV-0734, CV-0735) have been opened to 20%.
- Feed pump discharge pressure 500 psia.

Which of the following describes:

1) Whether the resulting amount of feed flow will be acceptable, and
2) Gives the bases for subsequent selection of procedure actions?

A. 1) Feed flow to <u>both</u> S/G's will be acceptable.
   2) Both S/G's may be fed <u>at this rate</u> until levels are restored to between 60% - 70%.

B. 1) Feed flow to <u>neither</u> S/G will be acceptable.
   2) Once-through-cooling must be initiated as the heat sink is lost.

C. 1) Feed flow to 'A' S/G will be acceptable.
   2) Feed flow to 'B' S/G must be <u>reduced</u> to avoid the potential for significant S/G tube bundle damage.

D. 1) Feed flow to 'A' S/G will be acceptable.
   2) Feed flow to 'B' S/G must be <u>raised</u> to raise level above -84% to avoid the need to initiate once-through-cooling.

Proposed Answer: **C**

Explanation (Optional):

A S/G is considered dry at less than -125% and feedwater flow to a dry S/G could cause significant tube bundle damage. With S/G level less than -84%, feedwater flow should be limited to less than 300 gpm.

EOP Supplement 41 must be utilized to determine the expected FW flow compared to the differential pressure. In this case, with the feed regulating bypass valves 20% open and a 90 psid differential pressure, the expected feed flow is approximately 345 gpm.

A. Incorrect, the applicant believes the feed flow to both S/Gs is adequate, while feed flow to the 'B' S/G must be reduced to less than 300 gpm in order to protect the tube bundles from potential damage. S/G level of 60-70% is normal operational level and feed to that desired level should only be to S/Gs with a level greater than -84%.
B. Incorrect, the applicant believes the once-through-cooling (OTC) criterion are met and the heat sink is lost. OTC must be initiated if any of the following conditions are met: 1) Both S/Gs levels are below -84% and are not being restored or 2) T_c rises uncontrollably 5°F or greater. See explanation and choice A for explanation on whether feed flow is adequate.
C. Correct, see explanation and choice A.
D. Incorrect, feed flow to 'A' S/G is acceptable but with 'B' S/G less than -84% it is not desirable to raise feed flow, as the limit for this condition is 300 gpm.

Technical Reference(s): EOP-7.0 and basis, EOP Supplement 41
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: EOP Supplement 41

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:
This question meets SRO-only criteria as the SRO must use procedures to determine procedurally required actions.

Loss of All Feed (Steps 37 through 49)

INSTRUCTIONS | CONTINGENCY ACTIONS

CAUTION

Feedwater flow restoration to a dry S/G (level less than -125%) may cause significant S/G tube bundle damage.

Limit feed flow to less than 300 gpm for any S/G with level less than -84%.

NOTE: IF Auxiliary or Main Feedwater is NOT expected to be restored in a timely manner, THEN a plant cooldown should be considered to ensure adequate S/G inventory to allow depressurization of the S/Gs for feeding with the Condensate Pump.

© 41. **EVALUATE** availability of S/G inventory replenishment by ANY of the following methods listed in order of preference:

a. Auxiliary Feedwater from T-2.

b. Main Feedwater from the Condenser.

(Continue)

CEN-152 LOAF Step 6:

*6. Replenish inventory in at least one steam generator by restoring **ANY** of the following:

- Auxiliary Feedwater flow
- Main Feedwater flow

6.1 **IF** high pressure feedwater sources can **NOT** be restored, **THEN**:

a. Depressurize the Steam Generator(s).

b. Establish an alternate, low pressure feedwater source to at least one steam generator.

Technical Basis:

The intent of the step is to direct the operators to evaluate available Feedwater sources for use in supplying water to the S/Gs. The sources are listed in the following order based on their preference of use:

1. Auxiliary Feedwater from T-2
2. Main Feedwater from the Condenser
3. Condensate Pump
4. Auxiliary Feedwater from Service Water via P-8C
5. Auxiliary Feedwater from Fire Water via P-8A or P-8B

Auxiliary Feedwater is the preferred source as this is the expected, normal source of Feedwater post-trip.

Main Feedwater is preferred as an alternative to Auxiliary Feedwater because it is the normal Feedwater source. It is, however, very difficult to control S/G level using Main Feedwater at hot zero power due to the high differential pressure between the Main Feedwater Pumps and the S/Gs and the large amount of flow that the pumps put out.

Use of the Condensate Pumps is a viable alternative, however it will require depressurizing the S/Gs to less than the shutoff head of the Condensate Pumps. Power history and existing S/G inventory will dictate the amount of time available to cooldown and depressurize to feed with the Condensate Pumps. An assessment of plant status and equipment availability will indicate if this is a viable option.

Auxiliary Feedwater from Service Water or Fire Water is available but NOT preferred due to contamination of the S/Gs caused by using water from Lake Michigan. A decision must be made as to the relative consequences of not feeding the S/Gs and possible effects of feeding with lake water.

Associated Notes, Cautions, Warnings:

The caution alerts the operator that Feedwater flow restoration to a "Dry" S/G **{SG level for considering a S/G "Dry"}** may cause significant S/G tube bundle damage.

Because of this, feed flow is limited to less than 300 gpm (reference FSAR 4.3.4) whenever the S/G level is less than the **[SG level for initiation of OTC]**

The note on Feedwater informs the operator that if Auxiliary or Main Feedwater is not expected to be restored in a timely manner, then a plant cooldown should be considered to ensure adequate S/G inventory to allow depressurization of the S/Gs for feeding with the Condensate Pump.

Deviations from EPG:

The CEN-152 step to restore Feedwater from one of the listed sources is split apart to first assess equipment availability and then to select the step for implementation of the selected feedwater source.

Justification for Deviation:

The steps necessary to establish alternate Feedwater supplies are extensive. For clarity and ease of use, the steps are written to stand alone, with an assessment step performed first. In this manner the operator assesses plant status and equipment availability and then can quickly scan the first sentence of the following steps to select the desired feedwater source.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	1
	K/A #	063.G2.1.7	
	Importance Rating	_____	4.7

K/A Statement: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question:

Given the following Plant conditions:

- PCS temperature is 420ºF
- LTOP System is armed in LTOP MODE
- Charging Pump P-55A is in operation
- Letdown is in service

Then, the following occurs:

- 125 VDC Panel ED-11-1 de-energizes due to a fault
- EK-0702, "RELIEF VALVE 2006 DISCH HI TEMP," annunciates
- The NCO attempts to close CV-2003, CV-2004, and CV-2005, Letdown Orifice Stop Valves, in accordance with AOP-17, "Loss of 125V DC Panel(s)," but CV-2003 does NOT close

To address these conditions, the CRS will . . .

A. Direct the crew to re-establish Charging and Letdown flow per SOP-2A, "Chemical and Volume Control."

B. Implement AOP-23, "Primary Coolant Leak."

C. Direct the crew to bypass the CVCS purification demineralizers per SOP-2B, "Chemical and Volume Control System - Purification and Chemical Injection."

D. Implement AOP-31, "Spurious Containment Isolation."

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, re-establishing charging and letdown would not address the problem (a lifted relief valve causing a PCS leak).
B. Correct, a loss of D-11-1 causes CV-2009 (letdown containment isolation) to close; however, letdown is still flowing from PCS into the letdown line upstream of CV-2009. Pressure upstream of CV-2009 will remain high with a failure of CV-2003 to close,

causing RV-2006 to lift due to the higher letdown pressure.

C. Incorrect, the applicant believes that excessive letdown flow (and higher temperature) is the result of the DC loss.
D. Incorrect, CV-2009 fails closed on a loss of ED-11-1. The reasoning it closed was not due to a spurious containment isolation signal, but rather a loss of DC control power. AOP-31 will not correct the underlying cause of the failure of CV-2003 to close.

Technical Reference(s): AOP-23, AOP-17, ARP-4
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam Palisades 2010
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

This meets SRO-only criteria 10CFR55.43 (b)5 as the applicant must assess plant conditions and then select a procedure to mitigate the event/transient.

Bank question was modified to accommodate current Palisades' terminology and procedures.

LOSS OF BUS ED-10R

TABLE 2: COMBINED EFFECTS FROM LOSS OF ED-11-1 AND ED-11-2		
COMPONENTS AFFECTED	**EFFECT FROM LOSS OF POWER**	**ALTERNATE COMPONENTS/COMMENTS**
Charging and Letdown		
P-55C, 'C' Charging Pump	P-55C breaker Fails As Is.	Can manually operate breaker at load center.
CV-2009, Letdown Containment Isolation	Fail closed, position indication lost.	Close CV-2003, CV-2004, & CV-2005. Manually control charging from Control Room.
CV-2153, BA Blender M-51 Boric Acid Inlet Control	Fail closed, position indication lost.	Manual makeup. Refer to SOP-2A, section titled, "To Provide Blended Boric Acid When CV-2155, Boric Acid Blender Outlet Control Valve, CV-2153, Boric Acid Makeup Control Valve, and CV-2165, Primary Makeup Water Makeup Control Valve, are Not Operational."
CV-2155, Make-up Stop		
CV-2165, Boric Acid Blender M-51 PMU Inlet Control		
Primary Coolant System		
PRV-1042B, Power Operated Relief Valve	Fail closed, position indication lost.	PRV-1043B still available. Refer to TS LCO 3.4.11 and LCO 3.4.12.
PRV-1067, Reactor Head Vent	Fail closed, position indication lost.	Alternate valves PRV-1068, PRV-1070, PRV-1072, and vent to Quench Tank still available.
PRV-1069, Pressurizer Vent		
PRV-1071, Rx/PZR Vent to Containment Atmosphere		
CV-2083, PCP P-50A, B, C, & D Controlled Bleedoff	Fail closed, position indication lost.	Verify open CV-2191, Pri Coolant Pp Controlled Bleedoff Stop.
Safety Injection		
CV-3030, Cont Sump Isol to West Safeguards Pumps	Fail closed, position indication lost.	Right Train of SIS equipment still available. Refer to Technical Specifications LCO 3.5.2 and LCO 3.5.3.
CV-3031, SIRW Tank T-58 Outlet Isolation	Fail open, position indication lost.	
CV-3018, HPSI Pump P-66B to HPSI Train 2 (MZ-22)	Fail closed, position indication lost.	Valves fail in normal position. Can not crosstie HPSI Trains 1 and 2.
CV-3059, HPSI Pump P-66B to HPSI Train 1 (MZ-23)	Fail open, position indication lost.	

PRIMARY COOLANT LEAK

USER ALERT

Read each step of the procedure prior to performing that step. Since the cause and nature of the abnormal condition cannot be predicted, performance of steps out of sequence may be necessary.

1.0 PURPOSE

This procedure is written for PCS unidentified leakage greater than 0.15 gpm when the Shutdown Cooling System is not initially in service. When the Shutdown Cooling System is in service, AOP-30, "Loss of Shutdown Cooling," is the correct procedure to implement.

2.0 ENTRY CONDITIONS

- PCS leak rate greater than 0.15 gpm unidentified calculated in accordance with DWO-1, "Operator's Daily/Weekly Items Modes 1, 2, 3, and 4"

- Symptom of PCS leak in conjunction with any of the following:

  - Unexplained drop in VCT level
  - Unexplained rise in Containment pressure, temperature, humidity, radiation and/or sump level
  - Unexplained Charging and Letdown Flow Mismatch
  - EK-0201, "CONT GAMMA RIA-2321 HIGH" (400 R/hr)
  - EK-0202, "CONT GAMMA RIA-2322 HIGH" (400 R/hr)
  - EK-0213, "CONT GAMMA RIA-2321 ALERT" (40 R/hr)
  - EK-0214, "CONT GAMMA RIA-2322 ALERT" (40 R/hr)
  - EK-0702, "RELIEF VALVE 2006 DISCH HI TEMP" (200°F)
  - EK-0731, "QUENCH TANK HI TEMP" (150°F)
  - EK-0732, "QUENCH TANK HI PRESS" (10 psig)
  - EK-0733, "QUENCH TANK HI-LO LEVEL" (HI-79%, LO-67%)
  - EK-0743, "PRESSURIZER PWR OPERATED RELIEF VALVE DISCHARGE HI TEMP" (180°F)
  - EK-0744, "PRESSURIZER SAFETY VALVE RV-1039 DISCH HI TEMP" (180°F)
  - EK-0745, "PRESSURIZER SAFETY VALVE RV-1040 DISCH HI TEMP" (180°F)

TITLE: PRIMARY SYSTEM VOLUME LEVEL PRESSURE
SCHEME EK-07 (C-12)

1	7	13	19	25	31
2	8	14	20	26	32
3	9	15	21	27	33
4	10	16	22	28	34
5	11	17	23	29	35
6	12	18	24	30	36

RELIEF VALVE 2006 DISCH HI TEMP	
Sensor:	TIA-0202, Outlet Temp Ind Alarm RV-2006, Relief Valve
Trip Setpoints:	200°F
Alternate Indication:	Quench Tank pressure/level

AUTOMATIC FUNCTION:

- None

OPERATOR ACTION:

NOTE:	RV-2006, Letdown Heat Exch E-58 Inlet Safety Relief, setpoint 600 psig.

- IF Letdown in service, THEN **CHECK** PIC-0202, Intermediate Pressure Letdown Controller, controlling at approximately 460 psig OR **MANUALLY CONTROL**.
  - WHEN PIC-0202 is in MANUAL, THEN **ENSURE CLOSED** associated handswitch (HS-2003, HS-2004, HS-2005) for any Letdown Orifice Stop Valve that is not OPEN.
- IF letdown cannot be restored, THEN:
  - **CLOSE** CV-2003, Letdown Orifice Stop Valve.
  - **CLOSE** CV-2004, Letdown Orifice Stop Valve.
  - **CLOSE** CV-2005, Letdown Orifice Stop Valve.

FOLLOW UP ACTION:

- IF RV-2006 does NOT reseat, THEN:
  - **IMPLEMENT** AOP-23, if entry conditions are met.
  - **IMPLEMENT** EI-1 if applicable.
- **IMPLEMENT** any applicable Technical Specifications LCO 3.4.13 actions.

REFERENCES:

- Technical Specifications LCO 3.4.13
- EI-1, "Emergency Classification and Actions"
- AOP-23, "Primary Coolant Leak"

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	103.G2.4.41	
Importance Rating		4.6

K/A Statement: Knowledge of the emergency action level thresholds and classifications.

Proposed Question:

Given the following conditions:

- A large break LOCA occurred 30 minutes ago.
- Bus 1C is faulted and cannot be energized.
- P-54A CS Pump tripped on overcurrent.
- Containment pressure 60 psia and stable.
- Containment Hydrogen concentration 5% and stable.
- Containment Radiation is 17,000 R/hr as read on RIA-2321, Containment High Range Radiation Monitor.

Does a Loss or Potential Loss of the Containment Fission Product Barrier currently exist and if so why?

A. Yes, a Potential Loss of Containment condition exists due to conditions described in subcategory C.6

B. Yes, a Potential Loss of Containment condition exists due to conditions described in subcategory B.3

C. Yes, a Potential Loss of Containment condition exists due to conditions described in subcategory B.5

D. No, a Potential Loss of Containment does not currently exist

Proposed Answer: C

Explanation (Optional):

A. Incorrect, containment radiation is 17,000 R/hr on the containment high range monitors, whereas the potential loss of containment value is $\geq$ 20,000 R/hr, as defined in subcategory C.6.
B. Incorrect, containment pressure is 60 psia (~45 psig), less than the potential loss of containment value of $\geq$ 55 psig.
C. Correct, with a loss of Bus 1C, CS Pumps P-54B and P-54C are lost. Since P-54A tripped on overcurrent, one full train of containment cooling (TS 3.6.6 Bases) is not in operation as required by subcategory B.5.
D. Incorrect, See explanation in 'C' above.

Technical Reference(s): SEP Supplement 1
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: SEP Supplement 1
Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam Palisades 2005
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ______
55.43 __5__

Comments:

This question meets SRO-only criteria as the applicant must assess plant conditions and then determine the Emergency Action Level threshold.

Modified stem to include an electrical fault and additional (and modified) initial conditions. Modified potential answers and changed correct answer.

PALISADES NUCLEAR PLANT
SITE EMERGENCY PLAN – Supplement 1

TITLE: SITE EMERGENCY PLAN
SUPPLEMENT 1 - EAL WALL CHARTS

Approved:	THHiginbotham	/	8/26/15
	Procedure Sponsor		Date

Process Applicability Exclusion	☒

New Procedure/Revision Summary:

Rev 3, Editorial Correction

Specific Changes:

Revised Note 6 in Supplement 1, Pages 1 & 2. Removed references to an after-normal business hours telephone number for the National Earthquake Information Center (NEIC). That number is no longer in service. (DRN-15-00827)

F Fission Product Barriers	FG1.1 (1, 2, 3, 4)	FS1.1 (1, 2, 3, 4)	FA1.1 (1, 2, 3, 4)	FU1.1 (1, 2, 3, 4)
	Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1)	Loss or potential loss of **any** two barriers (Table F-1)	**Any** loss or **any** potential loss of either Fuel Clad or PCS (Table F-1)	**Any** loss or **any** potential loss of Containment (Table F-1)

Table F-1 Fission Product Barrier Matrix

	Fuel Clad Barrier (FC)		Primary Coolant System Barrier (PCS)		Containment Barrier (CNMT)	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A Core Cooling / Heat Removal	1. CET readings > 1,200°F based on average of qualified CETs	1. CET readings > 700°F based on average of qualified CETs 2. **Any** PCS/Core Heat Removal safety function criterion in Table F-2 is met for ≥ 15 min. (Note 4)	None	1. Once Through Core Cooling flow established 2. **Any** PCS/Core Heat Removal safety function criterion in Table F-2 is met for ≥ 15 min. (Note 4) 3. PCS cooldown rate > 100°F/hr **AND** PZR pressure > maximum limits of EOP Supplement 1	None	1. CET readings **cannot** be restored < 1,200°F based on average of qualified CETs within 15 min. 2. CET readings **cannot** be restored < 700°F based on average of qualified CETs **AND** Reactor vessel water level cannot be restored > 614 ft 0 in. (Sensor #8 RVLMS green light) within 15 min.
B Inventory	None	3. Reactor Vessel level <614 ft 0 in. (Sensor #8 RVLMS red light)	1. PCS leak rate > available makeup capacity as indicated by PCS subcooling < 25°F based on average of qualified CETs 2. RUPTURED SG results in an ECCS (SIAS) actuation	4. PCS leak rate > 50 gpm	1. A containment pressure rise followed by a rapid unexplained drop in containment pressure 2. Containment pressure or sump level response **not** consistent with LOCA conditions 3. RUPTURED SG is also FAULTED outside of containment 4. Primary-to-secondary leakrate > 10 gpm **AND** UNISOLABLE steam release from affected SG to the environment	3. Containment pressure ≥ 55 psig and rising 4. Containment hydrogen concentration ≥ 6% 5. Containment pressure > 4 psig with < one full train of containment cooling systems operating
C Radiation / Coolant Activity	2. Containment High Range Radiation Monitor readings > 2,000 R/hr as indicated on RIA-2321 and/or RIA-2322 3. Failed Fuel Survey Point dose rate value for primary coolant > 1 R/hr	None	3. Containment High Range Radiation Monitor readings > 200 R/hr as indicated on RIA-2321 and/or RIA-2322	None	None	6. Containment High Range Radiation Monitor readings > 20,000 R/hr as indicated on RIA-2321 and/or RIA-2322
D Isolation Status	None	None	None	None	5. Failure of **all** valves in **any one** line to close **AND** Direct downstream pathway to the environment exists after containment isolation signal	None
E Judgment	4. **Any** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	4. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	4. **Any** condition in the opinion of the Emergency Director that indicates loss of the PCS barrier	5. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PCS barrier	6. **Any** condition in the opinion of the Emergency Director that indicates loss of the containment barrier	7. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the containment barrier

Notes

1. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL AS1.2/AG1.2). Do **not** delay declaration awaiting dose assessment results

   The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time
2. The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
3. If loss of water level in the refueling pathway occurs while in Mode 5, 6 or DEF, consider classification under EALs CU3.1, CU3.2 or CU3.3
4. The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.
5. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should **not** be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.
6. The NEIC can be contacted by going to the USGS NEIC website: http://earthquake.usgs.gov/eqcenter/ or alternatively by calling (303) 273-8500. Select option #1 and then option #2 and inform the analyst you wish to confirm recent seismic activity in the vicinity of Palisades Power Plant. Provide the analyst with the following PLP coordinates: 42° 19' 22" north latitude, 86° 18' 52" west longitude.
7. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.
8. 1-3 D/G or backfeed can be used to avoid entry into SG1.1 only if one of the power sources is supplying power to a safeguards bus within the 4 hour time period allowed.

Table F-2 PCS/Core Heat Removal Safety Function Criteria
• Both SG levels ≤ -84% • Both PCS loop delta Ts ≥ 10°F (forced circulation) OR Core delta T ≥ 50°F (natural circulation) • PCS subcooling < 25°F based on average of qualified CETs

EAL Identifier

XXX.X

Category (A, H, E, S, F, C)

Emergency classification (G, S, A, U)

Subcategory number (1 if no subcategory)

Sequential number within subcategory/classification

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Air Cooler systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Main Steam Line Break (MSLB) or a large break Loss of Coolant Accident (LOCA). The Containment Spray and Containment Air Cooler systems are designed to the requirements of the Palisades Nuclear Plant design criteria (Ref. 1).

The Containment Air Cooler System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The systems are arranged with two spray pumps powered from one diesel generator, and with one spray pump and three air cooler fans powered from the other diesel generator. The Containment Spray System was originally designed to be redundant to the Containment Air Coolers (CACs) and fans. These systems were originally designed such that either two containment spray pumps or three CACs could limit containment pressure to less than design. However, the current safety analyses take credit for one containment spray pump when evaluating cases with three CACs, and no air cooler fans in cases with two spray pumps and both Main Steam Isolation Valve (MSIV) bypass valves closed. If an MSIV bypass valve is open, 2 service water pumps and 2 CACs are also required to be OPERABLE in addition to the 2 spray pumps for containment heat removal.

To address this dependency between the Containment Spray System and the Containment Air Cooler System the title of this Specification is “Containment Cooling Systems,” and includes both systems. The LCO is written in terms of trains of containment cooling. One train of containment cooling is associated with Diesel Generator 1-1 and includes Containment Spray Pumps P-54B and P-54C, Containment Spray Valve CV-3001 and the associated spray header. The other train of containment cooling is associated with Diesel Generator 1-2 and includes Containment Spray Pump P-54A, Containment Spray Valve CV-3002 and the associated spray header, and CACs VHX-1, VHX-2, and VHX-3 and their associated safety related fans, V-1A, V-2A, and V-3A.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	2
	K/A #	015.A2.01	
	Importance Rating	_____	3.9

K/A Statement: Ability to (a) predict the impacts on the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss of erratic operation

Proposed Question:

Given the following conditions:

- The Plant is in MODE 3 at normal operating pressure and temperature.
- The PCS is being diluted in preparation for critical approach two shifts from now.
- The Wide Range flux level plasma indication for Source/Wide Range NI-1/3 on Panel C-06 begins to act erratically and is determined to be unreliable.
- The NI-1/3 analog count rates on Panel C-02 are indicating normal.
- NI-1/3 indications on Panel C-02 channel check to NI-2/4 indications.

Given these conditions, which one of the following identifies if LCO 3.3.9, Neutron Flux Monitoring Channels, and LCO 3.3.7, Post Accident Monitoring Instrumentation, are satisfied and identifies the minimum ACTIONS, if any, that are required?

A. LCO 3.3.9 is met, LCO 3.3.7 is met. No ACTIONS are required.

B. LCO 3.3.9 is NOT met, LCO 3.3.7 is met. Immediately stop dilution of the PCS and perform a SDM verification within 4 hours.

C. LCO 3.3.9 is met, LCO 3.3.7 is NOT met. Restore NI-1/3 plasma indication to OPERABLE status within 30 days.

D. LCO 3.3.9 is met, LCO 3.3.7 is NOT met. Restore NI-1/3 plasma indication to OPERABLE status within 7 days.

Proposed Answer: C

Explanation (Optional):

A. Incorrect, the applicant believes that plasma indication is not a PAM instrument.
B. Incorrect, the applicant believes that a channel of Neutron Flux Monitoring is lost.
C. Correct, the plasma display is used to satisfy LCO 3.3.7, per SHO-1. Both channels must be within ½ decade of one another to be considered operable.
D. Incorrect, the applicant believes that LCO action 3.3.7.C applies, which is incorrect, as there is only one function affected (the Wide Range). The applicant believes the Wide Range and Source Range are both affected.

Technical Reference(s): LCO 3.3.7, LCO 3.3.1 Bases Table B 3.3.1-1, LCO 3.3.9, LCO 3.3.9 Bases, SHO-1

(Attach if not previously provided, including version/revision number) ____________________

Proposed references to be provided to applicants during examination: LCO 3.3.7, LCO 3.3.9

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam Palisades 2009

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____
55.43 __2__

Comments:

This question meets the SRO-only criteria as the applicant must apply knowledge from Tech Spec bases information to the Tech Spec to determine the required actions to take.

3.3 INSTRUMENTATION

3.3.7 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.7 The PAM instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (Not Used)			
E. Required Action and associated Completion Time of Condition C not met.	E.1	Enter the Condition referenced in Table 3.3.7-1 for the channel.	Immediately
F. As required by Required Action E.1 and referenced in Table 3.3.7-1.	F.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	F.2	Be in MODE 4.	30 hours
G. As required by Required Action E.1 and referenced in Table 3.3.7-1.	G.1	Initiate action in accordance with Specification 5.6.6.	Immediately

Table B 3.3.1-1 (page 1 of 1)
Instruments Affecting Multiple Specifications

Required Instrument Channels	Affected Specifications
Nuclear Instrumentation	
Source Range NI-1/3, Count Rate Indication @ C-150 Panel	3.3.8 (#1)
Source Range NI-1/3 & 2/4, Count Rate Signal	3.3.9 & 3.9.2
Wide Range NI-1/3 & 2/4, Flux Level 10^{-4} Bypass	3.3.1 (#3, 6, 7, 9, & 12)
Wide Range NI-1/3 & 2/4, Startup Rate	3.3.1 (#2)
Wide Range NI-1/3 & 2/4, Flux Level Indication @EC-06 Panel for 3.3.7	3.3.7 (#3) & 3.3.9
Power Range NI-5, 6, 7, & 8, Tq	3.2.1 & 3.2.3
Power Range NI-5, 6, 7, & 8, Q Power	3.3.1 (#1 & 9)
Power Range NI-5, 6, 7, & 8, ASI	3.3.1 (#9) & 3.2.1 & 3.2.4
Power Range NI-5, 6, 7, & 8, Loss of Load/High Startup Rate Bypass	3.3.1 (#2 & 10)
PCS T-Cold Instruments	
TT-0112CA, Temperature Signal (SPI ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112CA & 0122CA, Temperature Signal (C-150)	3.3.8 (#6 & 7)
TT-0122CB, Temperature Signal (PIP ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112CA & 0122CB, Temperature Signal (LTOP)	3.4.12.b.1
TT-0112CC & 0122CD (PTR-0112 & 0122) Temperature Indication	3.3.7 (#2)
TT-0112 & 0122 CC & CD, Temperature Signal (SMM)	3.3.7 (#5)
TT-0112 & 0122 CA, CB, CC, & CD, Temperature Signal (Q Power & TMM)	3.3.1 (#1 & 9) & 3.4.1.b
PCS T-Hot Instruments	
TT-0112HA, Temperature Signal (SPI ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112HA & 0122HA, Temperature Signal (C-150)	3.3.8 (#4 & 5)
TT-0122HB, Temperature Signal (PIP ΔT Power for PDIL Alarm Circuit)	3.1.6
TT-0112 & 0122 HC & HD, Temperature Signal (SMM)	3.3.7 (#5)
TT-0112HC & 0122HD (PTR-0112 & 0122) Temperature Indication	3.3.7 (#1)
TT-0112 & 0122 HA, HB, HC, & HD, Temperature Signal (Q Power & TMM)	3.3.1 (#1 & 9)
Thermal Margin Monitors	
PY-0102A, B, C, & D	3.3.1 (#1 & 9)
Pressurizer Pressure Instruments	
PT-0102A, B, C, & D, Pressure Signal (RPS & SIS)	3.3.1 (#8 & 9) & 3.3.3 (#1.a & 7a)
PT-0104A & B, Pressure Signal (LTOP & SDC Interlock)	3.4.12.b.1 & 3.4.14
PT-0105A & B, Pressure Signal (WR Indication & LTOP)	3.3.7 (#5) & 3.4.12.b.1
PI-0110, Pressure Indication @ C-150 Panel	3.3.8 (#2)
SG Level Instruments	
LT-0751 & 0752 A, B, C, & D, Level Signal (RPS & AFAS)	3.3.1 (#4 & 5) & 3.3.3 (#4.a & 4.b)
LI-0757 & 0758 A & B, Wide Range Level Indication	3.3.7 (#11 & 12)
LI-0757C & 0758C, Wide Range Level Indication @ C-150 Panel	3.3.8 (#10 & 11)
SG Pressure Instruments	
PT-0751 & 0752 A, B, C, & D, Pressure Signal (RPS & SG Isolation) PT-0751A and PT-0752A Pressure Signal (C-150/150A)	3.3.1 (#6 & 7) & 3.3.3 (#2a, 2b, 7b, 7c) 3.3.8 (#8 & 9)
PIC-0751 & 0752 C & D, Pressure Indication	3.3.7 (#13 & 14)
PI-0751E & 0752E, Pressure Indication @ C-150 Panel	3.3.8 (#8 & 9)
Containment Pressure Instruments	
PS-1801, 1802, 1803, & 1804, Switch Output (RPS)	3.3.1 (#11)
PS-1801, 1802A, 1803, & 1804A, Switch Output (ESF)	3.3.3 (#5.a)
PS-1801A, 1802, 1803A, & 1804, Switch Output (ESF)	3.3.3 (#5.b)

Note: The information provided in this table is intended for use as an aid to distinguish those instrument channels which provide more than one required function and to describe which specifications they affect. The information in this table should not be taken as inclusive for all instruments nor affected specifications.

Table 3.3.7-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION E.1
1.	Primary Coolant System Hot Leg Temperature (wide range)	2	F
2.	Primary Coolant System Cold Leg Temperature (wide range)	2	F
3.	Wide Range Neutron Flux	2	F
4.	Containment Floor Water Level (wide range)	2	F
5.	Subcooled Margin Monitor	2	F
6.	Pressurizer Level (wide range)	2	F
7.	(Deleted)		
8.	Condensate Storage Tank Level	2	F
9.	Primary Coolant System Pressure (wide range)	2	F
10.	Containment Pressure (wide range)	2	F
11.	Steam Generator A Water Level (wide range)	2	F
12.	Steam Generator B Water Level (wide range)	2	F
13.	Steam Generator A Pressure	2	F
14.	Steam Generator B Pressure	2	F
15.	Containment Isolation Valve Position	1 per valve[a]	F
16.	Core Exit Temperature - Quadrant 1	4	F
17.	Core Exit Temperature - Quadrant 2	4	F
18.	Core Exit Temperature - Quadrant 3	4	F
19.	Core Exit Temperature - Quadrant 4	4	F
20.	Reactor Vessel Water Level	2	G
21.	Containment Area Radiation (high range)	2	G

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

3.3 INSTRUMENTATION

3.3.9 Neutron Flux Monitoring Channels

LCO 3.3.9 Two channels of neutron flux monitoring instrumentation shall be OPERABLE.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channel(s) inoperable.	A.1 Suspend all operations involving positive reactivity additions. AND	Immediately
	A.2 Perform SDM verification in accordance with SR 3.1.1.1.	4 hours AND Once per 12 hours thereafter

BASES

APPLICABLE SAFETY ANALYSIS (continued)	The OPERABILITY of neutron flux monitoring channels is necessary to meet the assumptions of the safety analyses and provide for the detection of reduced SDM. The neutron flux monitoring channels satisfy Criterion 4 of 10 CFR 50.36(c)(2).
LCO	The LCO on the neutron flux monitoring channels ensures that adequate information is available to verify core reactivity conditions while shut down. The safety function of these instruments is to detect changes in core reactivity such as might occur from an inadvertent boron dilution. Two neutron flux monitoring channels are required to be OPERABLE. If only one section of a neutron flux monitoring channel (source range or wide range) is functioning, the neutron flux monitoring channel may be considered OPERABLE if it is capable of detecting the existing reactor neutron flux. For example, with the source range count rate indicator functioning properly within its range, and in reasonable agreement with the other source range, a neutron flux monitor channel may be considered OPERABLE even though its wide range indicator is not functioning. The source range nuclear instrumentation channels, NI-1 and NI-2, provide neutron flux coverage extending an additional one to two decades below the wide range channels for use during refueling, when neutron flux may be extremely low. This LCO does not require OPERABILITY of the High Startup Rate Trip Function or the Zero Power Mode Bypass Removal Function. Those functions are addressed in LCO 3.3.1, RPS Instrumentation.
APPLICABILITY	In MODES 3, 4, and 5, neutron flux monitoring channels must be OPERABLE to monitor core power for reactivity changes. In MODES 1 and 2, neutron flux monitoring channels are addressed as part of the RPS in LCO 3.3.1. The requirements for source range neutron flux monitoring in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation."

5.1.6 Wide Range Nuclear Instrumentation (NIs)

(Technical Specifications LCO 3.3.1, Table 3.3.1-1 Item 2, SR 3.3.1.1, LCO 3.3.7, Table 3.3.7-1 Item 3, SR 3.3.7.1, LCO 3.3.9, SR 3.3.9.1)

(Applicability: Required in all Modes)

NOTE:	Wide Range indication NI-1/3A and NI-2/4A at EC-06 (plasma display) shall be utilized to satisfy LCO 3.3.7.

a. **CHECK** both channels for agreement within 1½ decades of each other.

(Applicability: Required in Modes 3 and 4)

b. Per Technical Specification LCO 3.3.9, for an inoperable Neutron Monitoring channel (Source and Wide Range) **PERFORM** the following:

1. **IMMEDIATELY SUSPEND** all operations involving positive reactivity additions.

2. **CHECK** Shutdown Margin within 4 hours and then every 12 hours.

3. To document verification, **INITIAL** the space for the inoperable Neutron Monitor AND **NOTE** condition in the Comments section.

5.1.7 Quadrant Power Tilt (T_q)

(Technical Specifications LCO 3.2.3, SR 3.2.3.1)

(Applicability: Required in Mode 1 with Thermal Power greater than 25% of Rated Thermal Power)

a. **CHECK** "Channel Deviation Level 1 (5%)" red indicating lights for NI Channels 5, 6, 7, and 8 are NOT lit and Annunciator EK-06C3, "CHANNEL DEVIATION LEVEL 1 (5%)," NOT alarming.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	______	2
	Group #	______	2
	K/A #	034.K4.02	
	Importance Rating	______	3.3

K/A Statement: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Fuel movement.

Proposed Question:

A new fuel bundle is being lowered into the Reactor Core during core reload, when suddenly the Refueling Machine Hoist movement stops. What interlock is responsible for the Refueling Machine's response to the conditions?

A. Extreme hoist travel limit interlock

B. Cable slack interlock

C. Grapple open hoist interlock

D. Hoist load bypass interlock

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, the extreme hoist travel limit interlock is an up limit only. This would not be reached while lowering a fuel bundle. Incorrect, the extreme hoist travel interlock is only to stop the hoist on rising motion if the upper limit does not stop the machine. This interlock will shut off the RFM with no indication as to why.
B. Correct, the Refueling Machine hoist is stopped during lowering on hoist underload or cable slack.
C. Incorrect, the hoist cannot be raised passed the UGOZ (Upper Grapple Operating Zone) with an open grapple.
D. Incorrect, the hoist load bypass is used when lowering an empty grabble.

Technical Reference(s): PL-IOTD Refueling Operations Lesson Plan
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source: Bank # _______

	Modified Bank #	________ (Note changes or attach parent)
	New	___X___

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	_____
	55.43	__6__

Comments:

This question meets SRO-only criteria as fuel movements and operation of the Refueling Machine must be observed and approved by an SRO.

C.2.f. RFM Hoist operations.

1) Hoist load bypass
   a) Must use hoist load bypass when lowering empty grapple
2) Electronic load cell used for sensing:
   a) Under load
   b) Over load
   c) Cable slack
3) Weighing zones:
   a) Control Rod or fuel plus grapple
   b) Control Rod or fuel plus grapple plus hoist
   c) Exception: RFM computer adjusts for weight of hoist box when hoist begins to pick up fuel assembly and hoist when fuel assembly is in the hoist box.
4) Interlocks
   a) The computer stops hoist movement on:
      (1) Under or overload. Hoist over load/under load setpoints are based upon recommendations by the vendor and include expected maximum and minimum fuel weights to the end of plant life to eliminate the need for an equipment modification and reduce the number of setpoint changes.
      (2) There is an extreme hoist up limit that shuts the RFM and SFHM down.
         (a) If you are raising the hoist and for some reason it does not stop at up limit, the extreme travel limit shuts the machine down with no other messages to tell you what happened. This has tripped up some Operators in the past.
         (b) Commonly encountered fuel handling difficulties tells you how to deal with this.
   b) Bridge or trolley energized stops hoist movement
   c) Up motion stops:
      (1) hoist at up limit (green light)
      (2) Hoist Overload (computer screen and digital display)
      (3) Grapple open leaving UGOZ. (Orange light out for UGOZ and orange grapple open light on)
   d) Down motion stops:
      (1) Hoist under load (computer screen)
      (2) Cable slack (orange light)

5) Hoist not at “UP’ limit and in either the core slow zone or the high speed zone.

   a) In the core clear or high speed zone, may operate the bridge and trolley with the hoist not at up limit with an empty grapple

   b) May move B/T by pushing the BTI pushbutton. This is the way minor adjustments are made to the B/T position to get fuel into the applicable location occasionally and is addressed procedurally.

6) Fuel hoist overload and underload stops.

   a) Point out that SFHM has only fuel assembly or control rod over/underloads. RFM has hoist box + fuel or CR over/underloads since the RFM computer has to take into account the added weight of the hoist box for over/underloads.

7) Core area hoist speed restrictions. Hoist automatically slows down in transition zones to avoid overload (underload) while accepting (shedding) weight of hoist box.

8) Grapple operate zone interlocks.

9) Cable slack

Example: After dropping off a fuel bundle into the Core (opening the grapple) an attempt is made to raise the hoist. The machine shuts off at the upper grapple operate zone and will not rise further.

Determination: Grapple open hoist interlock is in effect. Therefore, closing the grapple will allow continued operation.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	2
	K/A #	071.G2.1.32	
	Importance Rating	_____	3.8

K/A Statement: Ability to explain and apply all system limits and precautions.

Proposed Question:

A liquid release is being prepared for release per SOP-17B, “Dirty Radioactive Waste System.” Which of the following statements is correct if no Dilution Water pumps are operating at the start of the release?

A. The release cannot occur with no Dilution Water pumps operating.

B. CV-1054, Treated Waste Discharge valve, cannot open without the installation of a Temporary Modification.

C. RIA-1049, Liquid Radwaste Monitor, cannot exceed 20,000 cpm greater than background.

D. Only one Service Water pump shall be operating during the release.

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, a release can occur, however it must be sampled, calculated, and lined up using independent verifications.
B. Correct, per the Note on Step 7.10.a.7 of SOP-17B, CV-1054 is interlocked with the Dilution Water pump breakers, such that the interlock must be defeated before it can be opened.
C. Incorrect, with no Dilution Water pumps operating, RIA-1049 cannot exceed 10,000 cpm greater than background. The applicant is confusing the maximum rad monitor setpoints with only one Dilution Water pump running (20,000 cpm greater than background).
D. Incorrect, an additional dilution requirement applies during such situation where there shall be at least one additional Service Water Pump either operating or available with the pump in Standby. The applicant is misunderstanding the additional dilution requirements established.

Technical Reference(s): SOP-17B
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: None

Learning Objective:	____________________ (As available)		
Question Source:	Bank #	_______	
	Modified Bank #	_______	(Note changes or attach parent)
	New	___X___	

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	_____
	55.43	__5__

Comments:

This meets SRO-only criteria 10CFR55.43 (b)5 as the applicant must assess plant conditions and apply procedural requirements, including precautions and limitations, from a System Operating Procedure. The applicant must also know the basis behind the procedural action.

5.0 PRECAUTIONS AND LIMITATIONS

5.1 GENERAL

5.1.1 Due to the possibility of vent line blockage, normally vented tanks containing boric acid shall not be pressurized.

5.1.2 The containment sump should be drained prior to reaching 585.350'. The sump level shall not be allowed to exceed 585.400'.

5.1.3 The F-59 and F-62 filter housings no longer contain filtration media and will not be used for filtering purposes.

5.1.4 Complete draining of T-76N/S, Controlled Chem Lab Drain Tanks, may result in a release of radioactive gases to the Auxiliary Building.

5.1.5 T-100, Spent Resin Storage Tank, should be drained prior to any water or resin transfer operation involving T-100.

5.1.6 Entergy Procedure EN-IS-123, "Electrical Safety," requirements shall be followed when operating electrical components.

5.1.7 If the Dirty Waste Drain Tank Vent becomes clogged, then filling the tank greater than 50% may cause Auxiliary Building Floor drains to back up and reduced flow into the DWDTs as the level in the tanks rise due to backpressure.

5.1.8 The following additional dilution requirements apply during the time a Temporary Modification is installed and the Dilution Water Pump Interlock that closes CV-1054 is disabled:

- The Batch Card supplied by Chemistry shall call for no more than one service water pump.
- There shall be at least one additional Service Water Pump either operating or available with the pump in STANDBY.

6.0 INITIAL CONDITIONS

None

7.10 TO DISCHARGE LIQUID WASTE VIA RADWASTE MONITOR

NOTE:	I&C Maintenance support may be required if venting of FT-1050 or FT-1051 is necessary.

a. The following requirements apply for releases to the environs:

1. All releases from the Liquid Radwaste System shall be authorized by Chemistry via a completed Form CH 6.21-3, "Release Order," and authorized by the Shift Manager.

2. The Liquid Radwaste Monitor RIA-1049 shall be source checked prior to the release and documented on the release order. (Palisades Offsite Dose Calculation Manual, Appendix A, Table C-2, Item 1.a.)

3. IF RIA-1049 is out of service, a batch may be released if two independent sample analyses, release calculations, and valve lineup verifications are obtained. (Palisades Offsite Dose Calculation Manual, Appendix A, Table C-1, Action 1.)

4. The Flow Rate Controller to be used for the Batch release shall be channel checked by verifying indication of flow during release on the release order. (Palisades Offsite Dose Calculation Manual, Appendix A, Table C-2, Item 3.a.) Release rate shall be checked periodically to ensure limit is not exceeded.

5. The Liquid Radwaste Monitor RIA-1049 high alarm shall be set per SOP-37, "Process Liquid Monitor System," prior to the release to ensure requirements of Palisades Offsite Dose Calculation Manual, Appendix A, Section III.F are met.

6. RIA-1049 alarm setpoint should be set at least 1.5 times the existing readout (background) in CPM <u>AND</u> not to exceed the following:

   - If at least one Dilution Water Pump is operating, then the maximum alarm setpoint is 20,000 cpm + background.

   - If no Dilution Water Pumps are in service and at least one Service Water Pump is operating, the maximum alarm setpoint is 10,000 cpm + background.

7. The tank levels shall be within 1% as when the sample was taken or, if the level is higher than the above criteria then the tank shall be reanalyzed and a new permit generated.

NOTE:	If conducting a release with no dilution water pumps, CV-1054, Treated Waste Discharge valve is interlocked with the Dilution Water pump breakers, such that the interlock must be defeated before it can be opened.

8. The correct number of Dilution Water and/or Service Water Pumps shall be operating as indicated on the release order.

9. Required documentation on the release order shall be completed for each release.

10. **ENSURE OPEN** MV-RW160, Disch to Mix Basin CV-1054 Outlet (located 590' elev Turbine Building, SW of M-18 along west wall).

11. Ensure one of the following lineups is complete, per the applicable section, for the liquid release:

    - 7.4.8, "To Discharge Miscellaneous Waste Distillate Tank(s) Via the Radwaste Monitor to the Mixing Basin"

    - 7.6.2, "To Discharge Utility Water Tank T-91 Via Radwaste Monitor"

    - 7.9.4, "To Pump Filtered Waste to Discharge Via Radwaste Monitor"

b. **CLOSE** bypass valves for the following:

Flow Transmitter	Bypass Valve
Location: Filtered Waste Pump Room	
FT-1050 Flow From Treated Waste Tanks	FT-1050 Bypass IB
FT-1051 Flow From Treated Waste Tanks	FT-1051 Bypass IB

c. **RESET** RIA-1049 at C-40 Panel using reset pushbutton labeled "RIA-1049 RESET."

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	3
	Group #	_____	_____
	K/A #	G2.1.4	
	Importance Rating	_____	3.8

K/A Statement: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.

Proposed Question:

Three SROs have completed proficiency watchstanding in order to maintain Active License status per 10CFR55.53(e). The following table depicts the position, date and hours stood for each watch by three SROs since the beginning of the year:
<u>Assume today is May 16, 2016.</u>

SRO #1	Position	Hours stood
1/7/16	CRS	12
1/19/16	SM	12
2/8/16	SM	12
2/18/16	CRS	12
3/22/16	CRS	12
3/28/16	CRS	12
4/7/16	STA	8
4/22/16	CRS	12
5/13/16	SM	12

SRO #2	Position	Hours stood
1/7/16	CRS	12
1/14/16	CRS	12
2/3/16	CRS	12
2/16/16	CRS	12
2/19/16	STA	12
2/28/16	STA	8
3/8/16	CRS	12
3/19/16	STA	8
4/4/16	CRS	12

SRO #3	Position	Hours stood
1/9/16	CRS	12
2/9/16	CRS	12
2/18/16	CRS	12
3/22/16	CRS	12
3/28/16	STA	12
4/9/16	CRS	12
4/16/16	CRS	12
5/1/16	CRS	12
5/6/16	CRS	12

Which of the following statements is correct regarding the SROs’ active license status?

A. Only SRO #1 failed to maintain active license status

B. Only SRO #2 failed to maintain active license status

C. Only SRO #3 failed to maintain active license status

D. ALL SROs successfully maintained active license status

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect, SRO #1 has 6 full 12-hour watches within the first quarter at either the CRS or SM position.
B. Incorrect, SRO #2 has 5 full 12-hour watches within the first quarter at the CRS position.
C. Correct, SRO #3 only has 4 full 12-hour watches within the first quarter at the CRS

position.

D. Incorrect, SRO #3 does not meet the 10CFR55.53.(e) requirements. See Choice C for explanation.

Technical Reference(s): 10CFR55.53.(e), Palisades Admin 4.00
(Attach if not previously provided, ______
including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments:
This question meets the criteria for an SRO-only question as the applicant must understand the License maintenance requirements specifically for the SRO job function.

b. For individuals to maintain an Active License, they must meet the following requirements:

NOTE:	To be credited for proficiency by standing five (5) twelve-hour shifts, the shifts stood must be full/normal twelve-hour shifts (standing an eight-hour shift and working four hours of overtime before or after the normal eight-hour shift does not count for proficiency).

1. Stand a minimum of seven (7) 8 hour shifts or five (5) 12 hour shifts per calendar quarter in one of the appropriate watch positions below in order to maintain proficiency.

   (a) SROs: Shift Manager, Control Room Supervisor

   (b) NCOs: NCO - Reactor, NCO - Turbine

2. Participate in the Operations Continuing Training Program.

c. Requirement (1) above shall be monitored quarterly by the Operations Scheduling Coordinator. Failure to meet requirement (1) results in the individuals license becoming Inactive and shall be logged in the Operations Log. The Operations Scheduling Coordinator will also notify the Training Department to remove the individual's qualifications. Training shall be notified when the individual's license becomes active and the active status shall be logged in the Operations Log. Failure to meet requirement (2) above is controlled by the Palisades LOR Training Program Description.

d. If an individual fails a training cycle quiz/exam or simulator evaluation mode scenario, then the Manger, Shift Operations Training or his/her designate shall notify Operations Management and the Control Room. Control Room personnel shall log in the Operations log that the individual is not allowed to perform license duties until further notice. Operations Management shall remove that individual from work activities requiring their qualification until completion of a remediation plan. Control Room personnel shall log in the Operations log when the individual is allowed to perform license duties.

e. Inactive Licenses are those held by individuals who have not maintained Proficiency, or who do not routinely perform duties on Shift but are required by Technical Specifications (or designated by Plant Management) to have a license and participate in Continuing Training. Inactive License holders shall not manipulate the controls of the facility or direct licensed activities of Licensed Operators on shift.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	3
	Group #	_____	_____
	K/A #	G2.1.37	
	Importance Rating	_____	4.6

K/A Statement: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question:

The design bases event for Limiting Condition of Operation 3.1.1, “Shutdown Margin,” is:

A. Positive reactivity addition resulting from a Rod Ejection event at beginning of core life from 0% power conditions.

B. Positive reactivity addition resulting from a Rod Ejection event at end of core life from 100% power conditions.

C. Excessive cooldown resulting from a Main Steam Line Break at beginning of core life from 100% power conditions.

D. Excessive cooldown resulting from a Main Steam Line Break at end of core life from 0% power conditions

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, PDILs are based on Rod Ejection event.
B. Incorrect, PDILs are based on Rod Ejection event.
C. Incorrect, MTC less negative during beginning of life conditions. At Hot Full Power, there is less mass in the S/G for boil-off, as evident by a lower S/G pressure.
D. Correct, from Hot Zero Power, there is more mass in the SG as evident by a higher S/G pressure. End of cycle conditions have the most negative MTC. Therefore, this reactivity transient is the most severe

Technical Reference(s): LCO 3.1.1 bases
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________________ (As available)

Question Source: Bank # X

Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam St Lucie 2015

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 __6__

Comments:

This question meets SRO-only criteria as the applicant is required to understand the Tech Spec bases as it applies to plant conditions.

BASES

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption that the control rod of highest reactivity worth is fully withdrawn following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (Departure from Nucleate Boiling Ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 280 cal/gm energy deposition for the control rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a Main Steam Line Break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected Steam Generator (SG), and consequently the PCS. This results in a reduction of the primary coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The most limiting MSLB with respect to potential fuel damage is a guillotine break of a main steam line initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating PCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against an inadvertent boron dilution; (Ref. 3) and an uncontrolled control rod bank withdrawal from subcritical conditions (Ref. 5).

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	3
	Group #	_____	_____
	K/A #	G2.2.11	
	Importance Rating	_____	3.3

K/A Statement: Knowledge of the process for controlling temporary design changes.

Proposed Question:

An emergency temporary modification can be implemented in the event of an imminent threat to the safety or reliability of the plant due to an unforeseen plant event:

When implementing an emergency temporary modification, which of the following are required?

1. The Shift Manager, with the concurrence of the Engineering Director, or designee, may direct the installation or removal of a Temporary Modification to the plant without approved controlling documentation, as long as the Temporary Modification does not adversely affect nuclear safety.
2. As soon as conditions permit, the Operations Manager and the Systems & Components Manager, or their designee, shall be verbally notified of the modification and a Condition Report shall be initiated by Maintenance.
3. A Temporary Modification or a permanent Engineering Change shall be completed within 7 calendar days after installation.
4. The Responsible Engineer should coordinate with other Departments (i.e., the Systems & Components Engineer, Operations, Maintenance, Training, Planner and Installer) to ensure they are cognizant of the change and have provided appropriate input.
5. If the Temp Mod is a Comp action, a separate CR will be written by the Shift Manager to track the comp measure.

A. 1, 2 and 3 only

B. 2, 3 and 4 only

C. 2, 4 and 5 only

D. 1, 3 and 5 only

Proposed Answer: **D**

Explanation (Optional):

EN-DC-136, Temporary Modifications, contains this requirement in section 5.3

5.3 EMERGENCY TEMPORARY MODIFICATION IMPLEMENTATION
[1] In the event of an imminent threat to the safety or reliability of the plant due to an unforeseen plant event:
(a) The Shift Manager, with the concurrence of the Engineering Director, or designee, may direct the installation or removal of a Temporary Modification to the plant on an "emergency" basis without approved controlling documentation, as long as the Temporary Modification does not adversely affect nuclear safety.
(b) As soon as conditions permit, the Operations Manager and the Systems & Components Manager or their designee shall be verbally notified of the "emergency" modification and a Condition Report shall be initiated by Engineering. The CR issued shall be used to track the installation of the Emergency Temporary Modification. Following installation, removal of the Emergency Temporary Modification shall follow the applicable steps of this procedure.
(c) **IF** the Temporary Modification is also a compensatory measure (operational), **THEN** the Shift Manager will ensure that a Condition Report is issued to track the compensatory measure. This is a separate CR from step 5.3.(1)(b).
(d) A Temporary Modification or a permanent Engineering Change shall be completed within 7 calendar days after installation.

A. Incorrect, #2 is wrong due to the CR is to be initiated by Engineering not Maintenance, and #5 is also required.
B. Incorrect, #2 is wrong due to the CR is to be initiated by Engineering not Maintenance, and #4 is incorrect
C. Incorrect, #2 is wrong due to the CR is to be initiated by Engineering not Maintenance and #4 is Incorrect
D. Correct, see explanation

Technical Reference(s): EN-DC-136
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: ___None___

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam Grand Gulf 2015
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 __3__

Comments:

This question meets SRO-only criteria as the SRO must have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

5.3 EMERGENCY TEMPORARY MODIFICATION IMPLEMENTATION

[1] In the event of an imminent threat to the safety or reliability of the plant due to an unforeseen plant event:

(a) The Shift Manager, with the concurrence of the Engineering Director, or designee, may direct the installation or removal of a Temporary Modification to the plant on an "emergency" basis without approved controlling documentation, as long as the Temporary Modification does not adversely affect nuclear safety.

(b) As soon as conditions permit, the Operations Manager and the Systems & Components Manager or their designee shall be verbally notified of the "emergency" modification and a Condition Report shall be initiated by Engineering. The CR issued shall be used to track the installation of the Emergency Temporary Modification. Following installation, removal of the Emergency Temporary Modification shall follow the applicable steps of this procedure.

(c) **IF** the Temporary Modification is also a compensatory measure (operational), **THEN** the Shift Manager will ensure that a Condition Report is issued to track the compensatory measure. This is a separate CR from step 5.3.(1)(b).

(d) A Temporary Modification or a permanent Engineering Change shall be completed within 7 calendar days after installation.

(1) Emergency Temporary Modifications that are removed prior to completion of the associated Engineering Change, should be documented with a Temporary Modification Evaluation (TMEV) EC subtype.

(2) Emergency Temporary Modifications that are not removed prior to the completion of the associated Engineering Change shall follow the applicable steps noted in this procedure such as work order development, scheduling and implementation actions in a timely manner as required.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	____	3
	Group #	____	____
	K/A #	G2.2.39	
	Importance Rating	____	4.5

K/A Statement: Knowledge of less than or equal to one hour technical specification action statements for systems.

Proposed Question:

Given the following conditions:

- The Plant is in MODE 5 during a forced outage.
- PCS temperature is at 180°F on Shutdown Cooling (SDC).
- SDC is in-service using P-67A LPSI Pump at a flowrate of 3000 gpm.
- Primary Coolant Pumps (PCPs) P-50A and P-50D are operating.
- Both S/G levels are 65%.
- P-8C, Auxiliary Feedwater Pump, is removed from service for maintenance.

Then, the following simultaneously occur:

- 2400VAC Bus 1C de-energizes due to a fault on the bus
- P-50D, Primary Coolant Pump, trips

Given these current conditions, which one of the following identifies whether the LCO for LCO 3.4.7, "PCS Loops - MODE 5, Loops Filled," is currently satisfied and identifies the minimum action, if any, required to satisfy the LCO without reliance on any action statement?

A. LCO 3.4.7 is met; no action statements are required to be entered.

B. LCO 3.4.7 is NOT met; starting P-50B PCP will allow all action statements to be exited.

C. LCO 3.4.7 is NOT met; restoring P-8C and making it available for operation will allow all action statements to be exited.

D. LCO 3.4.7 is NOT met; starting P-50C PCP will allow all action statements to be exited.

Proposed Answer: **C**

Explanation (Optional):

LCO 3.4.7 requires one SDC train operable and in operation with $\geq$ 2810 gpm flow through the reactor core and either:
A. One additional SDC train shall be operable, or
B. The secondary side water level of each S/G shall be $\geq$ -84%.

A. Incorrect, the applicant does not recall the requirement to have AFW available in order to take credit for an operable heat sink.
B. Incorrect, the applicant does not recall the requirement to have AFW available in order to take credit for an operable heat sink and believes that restoring 2 PCPs will allow action statement to be exited.
C. Correct, additional requirements are necessary for a S/G to replace a standby SDC train, per requirement B in the explanation. A S/G can act as a heat sink via natural circulation if the S/G has the minimum water level specified in SR 3.4.7.2, the S/G is operable, the S/G has available method of feedwater addition and a controllable path for steam release, and the ability to pressurize and control pressure in the PCS. In this case, AFW Pump P-8C would satisfy that requirement.
D. Incorrect, the applicant does not recall the requirement to have AFW available in order to take credit for an operable heat sink and the belief that restoring 2 PCPs will allow action statement to be exited.

Technical Reference(s): LCO 3.4.7 and bases
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ____None_________

Learning Objective: _________________________ (As available)

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____
55.43 __2__

Comments:

This exam question meets the criteria for an SRO-only criteria as the applicant must apply specific knowledge from the Tech Spec Bases to given plant conditions and then determine if the LCO is met.

Question used from Palisades 2010 Audit Exam.

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.7 PCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One Shutdown Cooling (SDC) train shall be OPERABLE and in operation with $\geq$ 2810 gpm flow through the reactor core, and either:

a. One additional SDC train shall be OPERABLE; or

b. The secondary side water level of each Steam Generator (SG) shall be $\geq$ -84%.

-----NOTES-----

1. The SDC pump of the train in operation may not be in operation for $\leq$ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause reduction of the PCS boron concentration; and

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

2. Both SDC trains may be inoperable for up to 2 hours for surveillance testing or maintenance provided:

   a. One SDC train is providing the required flow through the reactor core;

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

   c. Each SG secondary side water level is $\geq$ -84%.

3. Forced circulation (starting the first primary coolant pump) shall not be initiated unless one of the following conditions is met:

   a. SG secondary temperature is equal to or less than the reactor inlet temperature (T_c);

   b. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is $\leq$ 10°F/hour; or

   c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is $\leq$ 57%.

4. Primary coolant pumps P-50A and P-50B shall not be operated simultaneously.

5. All SDC trains may not be in operation during planned heatup to MODE 4 when at least one PCS loop is in operation.

APPLICABILITY: MODE 5 with PCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SDC train inoperable. AND Any SG with secondary side water level not within limit.	A.1 Initiate action to restore a second SDC train to OPERABLE status. OR A.2 Initiate action to restore SG secondary side water levels to within limits.	Immediately Immediately
B. Two SDC trains inoperable. OR SDC flow through the reactor core not within limits.	B.1 Suspend all operations involving reduction in PCS boron concentration. AND B.2 Initiate action to restore one SDC train to OPERABLE status and operation with $\geq$ 2810 gpm flow through the reactor core.	Immediately Immediately

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.7 PCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the PCS loops filled, the primary function of the primary coolant is the removal of decay heat and transfer this heat either to the Steam Generator (SG) secondary side coolant via natural circulation (Ref. 1) or the Shutdown Cooling (SDC) heat exchangers. While the principal means for decay heat removal is via the SDC System, the SGs via natural circulation are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary side water. If heatup of the PCS were to continue, the contained inventory of the SGs would be available to remove decay heat by producing steam. As long as the SG secondary side water is at a lower temperature than the primary coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the primary coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with PCS loops filled, the SDC trains are the principal means for decay heat removal. The number of trains in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC train for decay heat removal and transport. The flow provided by one SDC train is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for decay heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an SDC train that must be OPERABLE and in operation. The second path can be another OPERABLE SDC train, or through the SGs, via natural circulation each having an adequate water level. "Loops filled" means the PCS loops are not blocked by dams and totally filled with coolant.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The boron concentration must be uniform throughout the PCS volume to prevent stratification of primary coolant at lower boron concentrations which could result in a reactivity insertion. Sufficient mixing of the primary coolant is assured if one SDC pump is in operation. PCS circulation is considered in the determination of the time available for mitigation of the inadvertent boron dilution event. By imposing a minimum flow through the reactor core of 2810 gpm, sufficient time is provided for the operator to terminate a boron dilution under asymmetric flow conditions.

PCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2).

LCO

The purpose of this LCO is to require one SDC train be OPERABLE and in operation with either an additional SDC train OPERABLE or the secondary side water level of each SG ≥ -84%. SDC in operation with a flow through the reactor core ≥ 2810 gpm, provides enough flow to remove the decay heat from the core with forced circulation and provide sufficient mixing of the soluble boric acid. The second SDC train is normally maintained OPERABLE as a backup to the operating SDC train to provide redundant paths for decay heat removal. However, if the standby SDC train is not OPERABLE, a sufficient alternate method to provide redundant paths for decay heat removal is two SGs with their secondary side water levels ≥ -84%. Should the operating SDC train fail, the SGs could be used to remove the decay heat via natural circulation.

A SDC train may be considered OPERABLE (but not necessarily in operation) during re-alignment to, and when it is re-aligned for, LPSI service or for testing, if it is capable of being (locally or remotely) realigned to the SDC mode of operation and is not otherwise inoperable. Since SDC is a manually initiated system, it need not be considered inoperable solely because some additional manual valve realignments must be made in addition to the normal initiation actions. Because of the dual functions of the components that comprise the LPSI and shutdown cooling systems, the LPSI alignment may be preferred.

Note 1 permits all SDC pumps to not be in operation ≤ 1 hour per 8 hour period. The Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least

BASES

LCO (continued)

10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the PCS without the SDC pumps depends on the core decay heat load and the length of time that the pumps are stopped.

As decay heat diminishes, the effects on PCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the primary coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (Pressure and Temperature (P/T) limits or Low Temperature Overpressure Protection (LTOP) limits) must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit.

In MODE 5 with loops filled, it is sometimes necessary to stop all SDC forced circulation. This is permitted to change operation from one SDC train to the other, perform surveillance or startup testing, perform the transition to and from the SDC, or to avoid operation below the PCP minimum net positive suction head limit. The time period is acceptable because natural circulation is acceptable for decay heat removal, the primary coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows both SDC trains to be inoperable for a period of up to 2 hours provided that one SDC train is in operation providing the required flow, the core outlet temperature is at least 10°F below the corresponding saturation temperature, and each SG secondary water level is ≥ 84%. This permits periodic surveillance tests or maintenance to be performed on the inoperable trains during the only time when such evolutions are safe and possible.

Note 3 requires that one of the following conditions be satisfied before forced circulation (starting the first PCP) may be started:

a. SG secondary temperature is equal to or less than the reactor inlet temperature (T_c);

b. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is ≤ 10°F/hour; or

c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

BASES

LCO (continued)

Satisfying any of the above conditions will preclude a large pressure surge in the PCS when the PCP is started. Energy additions from the steam generators could occur if a PCP was started when the steam generator secondary temperature is significantly above the PCS temperature. The maximum pressurizer level at which credit is taken for having a bubble (57%, which provides about 700 cubic feet of steam space) is based on engineering judgement and verified by LTOP analysis. This level provides the same steam volume to dampen pressure transients as would be available at full power.

Note 4 specifies a limitation on the simultaneous operation of primary coolant pumps P-50A and P-50B which allows the pressure limits in LCO 3.4.3, "PCS Pressure and Temperature Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection System," to be higher than they would be without this limit.

Note 5 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting SDC trains to not be in operation when at least one PCP is in operation. This Note provides for the transition to MODE 4 where a PCP is permitted to be in operation and replaces the PCS circulation function provided by the SDC trains.

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. The two SDC heat exchangers operate as a single ~~heat exchanger comprised of two partial capacity~~ unit~~s~~. A s~~ingl~~eparate OPERABLE SDC heat exchanger ~~may be credited to either~~is required for each OPERABLE SDC train~~ or to both OPERABLE SDC trains simultaneously, provided 100% of the heat removal requirements can be demonstrated with the single OPERABLE SDC heat exchanger~~. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

An SG can perform as a heat sink via natural circulation when:

a. SG has the minimum water level specified in SR 3.4.7.2.

b. SG is OPERABLE.

c. SG has available method of feedwater addition and a controllable path for steam release.

d. Ability to pressurize and control pressure in the PCS.

If both SGs do not meet the above provisions, then LCO 3.4.7 item b (i.e. the secondary side water level of each SG shall be ≥ -84%) is not met.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	3
	Group #	_____	_____
	K/A #	G2.3.6	
	Importance Rating	_____	3.8

K/A Statement: Ability to approve release permits.

Proposed Question:

Which of the following personnel have the ability to approve a Containment Purge release permit in accordance with Procedure No CH 6.27, Containment Purge?

A. Shift Engineer and Heath Physics Technician

B. Shift Manager and RETS Analyst

C. Duty Station Manager and RETS Analyst

D. Shift Engineer and Chemistry Technician

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect, an HP Technician cannot be designated to approve the release per CH 6.27.
B. Correct, per CH 6.27, only the following individuals (or designees) can approve a Containment Purge release: Shift Manager and RETS Analyst, Chemistry Supervision, or designees.
C. Incorrect, the Duty Station Manager cannot be designated to approve the release per CH 6.27.
D. Incorrect, a Chemistry Technician cannot be designated to approve the release per CH 6.27.

Technical Reference(s): CH 6.27
(Attach if not previously provided, ____________________
including version/revision number) ____________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____

55.43 __4__

Comments:

This questions meets SRO-only criteria as the applicant must use procedure knowledge outside to perform a function that is only performed by an SRO licensed individual.

2.2 REFERENCE DOCUMENTS

2.2.1 Entergy Procedure EN-RP-131, "Air Sampling"

2.2.2 Entergy Procedure EN-AD-103, "Document Control and Records Management Programs"

2.2.3 Technical Specifications LCO 3.6.3, "Containment Isolation Valves"

2.2.4 Palisades Offsite Dose Calculation Manual (ODCM)

3.0 PREREQUISITES

3.1 Samples of containment atmosphere, gasses, particulates, and iodines, excluding tritium, shall have been obtained within 24 hours of purge initiation.

3.2 Containment Air Monitor RIA-1817 shall have shown no substantial increase in activity levels since the sample was obtained. (Containment air cooler recirculation fans shall have been operating during sampling and RIA-1817 evaluated.)

3.3 Technical Specifications LCO 3.6.3 requires purge exhaust valves and air room supply valves to be isolated in Modes 1, 2, 3, and 4.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Containment purge should only be authorized after approval of both Shift Manager and RETS Analyst, Chemistry Supervision or designees.

4.2 Both occupational and environmental radiation exposures must be kept as low as is reasonably achievable. Alternative to environmental release are decay via holdup in containment and internal cleanup via HEPA-Charcoal filtration. Both decay and internal cleanup shall be utilized, as available, to reduce environmental release. However, decay is a viable option only if early containment entry is not required.

4.3 Do not release waste gas decay tanks during the first 24 hours of a Containment purge UNLESS it has been verified that the sum of all release fractions (E_K) from all sources going to the stack is less than 1.0 via Attachment 1.

FORM CH 6.27-1
CONTAINMENT PURGE

Batch No____________________ Containment R_k Fraction = ____________ (R_k < 1.0)

RIA-1817 reading at time of CTMT air samples ______________ cpm

RIA-1817 reading prior to CTMT purge initiation ______________ cpm
(Must be < 20% higher than reading at time of CTMT air sample)

Release Authorization

Verification that total R_k at the stack is < 1: ________________________ Date/Time: ______/______

RETS Analyst: ________________________________ Date: ____________ Time: ____________

Shift Manager: ________________________________ Date: ____________ Time: ____________

Purge Initiated Date: ____________ Time: ____________

Purge Secured Date: ____________ Time: ____________

Release Review

Shift Manager: ________________________________ Date: ____________

RETS Analyst: ________________________________ Date: ____________

Purge Waived

The Containment Purge Card waived per provisions of Step 5.3.3a, b, or c

RETS Analyst: ________________________________ Date: ____________

Air concentration for tritium calculation is as follows:

Gross cpm ____________ CPM

Background cpm ____________ CPM

Net cpm ____________ CPM

Divided by counter efficiency ____________ DPM

Divided by 2.22E6 ____________ uCi

Divided by sample volume, typically 5 mls ____________ uCi/cc

Times volume (mls) of water in gas wash bottle ____________ uCi

Divided by flow (cc) though gas wash bottle ____________ uCi/cc

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	3
	Group #	_____	_____
	K/A #	G2.4.16	
	Importance Rating	_____	4.4

K/A Statement: Knowledge of EOP implementation hierarchy and coordination with other support procedures.

Proposed Question:

Given the following:

- The Plant is at 100% power.
- Emergency Diesel Generator 1-2 out of service.

Then, the following occurs:

- The Plant is manually tripped due to indications of a Steam Generator Tube Rupture.
- EOP-5.0, “Steam Generator Tube Rupture Recovery,” is in-progress.
- A valid Safety Injection Actuation Signal is received.
- Bus 1D de-energizes due to a ground/overcurrent fault.
- Pressurizer level is currently 33% and slowly rising.

Which one of the following procedures provides the most expeditious method of restoring available Pressurizer Heater capability for the above plant conditions?

A. AOP-9, “Loss of Bus 1D.”

B. AOP-10, “Loss of Bus 1E.”

C. EOP Supplement 29, “Restore Busses 1C, 1D, 1E Power From Offsite Source.”

D. SOP-30, “Station Power.”

Proposed Answer: **D**

Explanation (Optional):

A. Incorrect, AOP-9 contains directions for transfer heater power from Bus 1E to Bus 1C but this could take up to 5 hours (AOP-9 refers the operator to SOP-30 when restoring pressurizer heaters).
B. Incorrect, AOP-10 Attachment 1 “Emergency Feed of PZR Heater Transformer #15 from Bus 1C” is directed per EOP-5.0 Step 42 since Bus 1D and 1E are not energized, but this will require an extended amount of time compared to SOP-30. (Bus 1E undergoes a load shed upon a Safety Injection Signal to ensure adequate voltage levels on the ESF buses when the D/Gs are called on to supply power.)
C. Incorrect, EOP Supplement 29 directs starting DG 1-3 but this DG can only supply 1C or 1D bus.
D. Correct, SOP-30 contains a section for restoring Bus 1E following an SIAS and EOP-5.0

directs the use of this procedure, this action does not account for Bus 1D being de-energized. This is the purpose of the RNO of this step.

Technical Reference(s): EOP-5.0, AOP-9, AOP-10, SOP-30, EOP Supplement 29
(Attach if not previously provided, ______________________________
including version/revision number) ______________________________
Proposed references to be provided to applicants during examination: ______None______

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam Palisades 2009
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____
55.43 __5__

Comments:

This meets SRO-only criteria as the applicant must assess plant conditions and apply procedural requirements/steps outside of immediate actions and/or entry conditions.

Question modified from Palisades 2009 NRC Exam. Modified question to accommodate current Palisades terminology. Changed correct answer and one distractor.

INSTRUCTIONS

42. IF ANY of the following AC or DC buses are NOT energized,
THEN **RESTORE** power to the affected buses. Refer to the following applicable procedure:

BUS	PROCEDURE
1C or 1D	EOP Supplement 29
1E with No SIAS	EOP Supplement 29
1E with SIAS	SOP-30
Y10	AOP-12, "Loss of Preferred AC Bus EY-10"
Y20	AOP-13, "Loss of Preferred AC Bus EY-20"
Y30	AOP-14, "Loss of Preferred AC Bus EY-30"
Y40	AOP-15, "Loss of Preferred AC Bus EY-40"
Y01	AOP-16, "Loss of Instrument AC Bus EY-01"
Any DC Bus	AOP-17, "Loss of 125V DC Panel(s)"

CONTINGENCY ACTIONS

42.1 IF Bus 1D and Bus 1E are NOT energized,
THEN as resources permit, **PROVIDE** power to PZR Heaters from Bus 1C. Refer to AOP-10, "Loss of Bus 1E," attachment titled "Emergency Feed of PZR Heater Transformer #15 From Bus 1C."

ACTIONS\EXPECTED RESPONSE | **RESPONSE NOT OBTAINED**

10. (Continued)

c. **ENSURE CLOSED** the following breakers:

1) 152-201, Sta Pwr Transformer 12 and 20.

2) 52-2006, MCC-2 480V Feeder Bkr.

3) 52-1201, MCC-8 480V Feeder Bkr.

d. **ENSURE CLOSED** 152-211, 1D to PZR Heater Xfmr Ex-16. Refer to SOP-30, "Station Power," Section 7.5, "Pressurizer Heater Bus 15 and 16."

11. **REFER TO** Technical Specifications to determine LCO requirements:

Modes 1, 2, 3, and 4

- LCO 3.8.1
- LCO 3.8.9

Modes 5 and 6, and during movement of irradiated fuel assemblies

- LCO 3.8.2
- LCO 3.8.10

12. **ENSURE** all applicable steps have been completed.

a. **EXIT** this procedure.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	3
	Group #	_____	_____
	K/A #	G2.4.29	
	Importance Rating	_____	4.4

K/A Statement: Knowledge of the emergency plan.

Proposed Question:

Complete the following statements regarding the Emergency Plan:

An Assembly is the process of gathering personnel in designated areas __(1)__ following the declaration of an __(2)__ or higher.

A. (1) Outside the protected area
(2) Alert

B. (1) Inside the protected area
(2) Alert

C. (1) Outside the protected area
(2) Unusual Event

D. (1) Inside the protected area
(2) Unusual Event

Proposed Answer: **A**

Explanation (Optional):

A. Correct, an Assembly is performed at an Alert or higher and assembles personnel outside of the PA.
B. Incorrect, part 1 is incorrect, Accountability is performed inside the PA.
C. Incorrect, part 2 is incorrect, see Choice A.
D. Incorrect, both parts are incorrect, see Choice A

Technical Reference(s): EI-12.1
(Attach if not previously provided, ______________________
including version/revision number) ______________________

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source: Bank # _______

Modified Bank # ________ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ______
55.43 __1__

Comments:

This question meets SRO-only criteria as it is an SRO-only duty to direct the performance of an Assembly when required.

TITLE: PERSONNEL ACCOUNTABILITY AND ASSEMBLY

5.0 PROCEDURE

REFERENCE USE
• **Procedure and Procedure Precautions and Limitations are at the work location for reference.** • **Review and understand segments before performing any steps.** • **Signoff steps are completed, when included, before starting the next step.** • **Place keep in accordance with EN-HU-106, "Procedure and Work Instruction Use and Adherence."** • **Review the Procedure to verify segments have been completed.**

5.1 ACCOUNTABILITY

5.1.1 Accountability is the process of identifying all personnel **inside the Protected Area** following the declaration of an **Alert** or higher. Accountability should be completed in approximately 30 minutes.

5.1.2 Assembly Areas inside the Protected Area are:

a. Assembly Area I (Control Room)

b. Assembly Area II (Technical Support Center)

c. Assembly Area V (Operations Support Center)

d. Assembly Area VI (Men's Locker Room - Service Building)

e. Assembly Area VIII (Security Building, Administrative Area)

5.1.3 The Emergency Facility Leader or designee shall ensure accountability is completed in their Assembly Area.

OR

In the event the Proximity Card reader system is not available, manual accountability shall be completed and results delivered to Security. If radiological conditions prohibit hand carrying the list, it may be reported by faxing to Security or by telephone.

5.1.4 The Security Supervisor shall ensure the Accountability results are reported to the Emergency Plant Manager.

5.1.5 Plant employees and contractors should remain in the Assembly Area until:

a. Directed to return to their work area

OR

b. Directed to evacuate the site

5.2 ASSEMBLY

5.2.1 Assembly is the process of gathering personnel in designated areas **outside the Protected Area** following the declaration of an **Alert** or higher.

5.2.2 Assembly Areas outside the Protected Area are:

a. Assembly Area III (Training Building, 2nd Floor, Classroom H)

b. Assembly Area VII (Support Building Lunchroom)

5.2.3 Plant employees and contractors should remain in the Assembly Area until:

a. Directed to return to their work area

OR

b. Directed to evacuate the site

5.3 REVIEW OF ACCOUNTABILITY INSTRUCTIONS

At least quarterly, Emergency Planning should review the manual accountability instructions and make changes as needed.

6.0 ATTACHMENTS AND RECORDS

6.1 ATTACHMENTS

None